Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K1.05
	Importance Rating	3.6

Reactor Water Level Control

Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Reactor feedwater system

Proposed Question: #1

A plant startup is in progress with the following:

- Reactor Feed Pump Turbine (RFPT) A is in service.
- RFPT B is being started.

Which one of the following identifies the control to be manipulated to raise RFPT B speed when at 700 rpm and when at 3500 rpm, in accordance with OP-2A, Feedwater System?

When RFPT speed is at 700 rpm, manipulate RFPT B __(1)__ to raise speed. When RFPT is at 3500 rpm, manipulate RFPT B __(2)__ to raise speed.

	(1)	(2)
A.	FLOW CNTRL 06-84B	FLOW CNTRL 06-84B
B.	FLOW CNTRL 06-84B	MTR SPEED CHANGER
C.	MTR SPEED CHANGER	FLOW CNTRL 06-84B
D.	MTR SPEED CHANGER	MTR SPEED CHANGER

Proposed Answer: C

Explanation: At speeds less than 800 rpm, only the Motor Speed Changer (MSC) is capable of controlling. By 1800 rpm, the Motor Gear Unit (MGU) is in control. The MGU is controlled using RFPT B FLOW CNTRL 06-84B.

A. Incorrect – RFPT B FLOW CNTRL 06-84B will not work below 800 rpm. Plausible because this is the correct control during normal full power operation.
B. Incorrect – RFPT B FLOW CNTRL 06-84B will not work below 800 rpm. Plausible because this is the correct control during normal full power operation. Above 1800 rpm, RFPT B FLOW CNTRL 06-84B is used to control speed, with the Motor Speed Changer raised to the high stop. Plausible because this is the correct control at lower speed.
D. Incorrect – Above 1800 rpm, RFPT B FLOW CNTRL 06-84B is used to control speed, with the Motor Speed Changer raised to the high stop. Plausible because this is the correct control at lower speed.

Technical Reference(s): OP-2A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-06 1.05.a.7, 1.05.a.8

Question Source: Bank – 9/12 NRC #45

Question History: 9/12 NRC #45

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 #	215003 K1.02
	Importance Rating	3.6

IRM

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

Proposed Question: #2

A plant startup is in progress with the following:

- The Mode Switch is in START & HOT STBY.
- All Intermediate Range Monitor (IRM) detectors are fully inserted and indicating mid-scale on range 1.
- Then, a shorted contact causes IRM G detector to withdraw.

Which one of the following describes the resulting status of control rod blocks?

A. No rod block is received.

B. A rod block is received as soon as IRM G leaves the fully inserted position.

C. A rod block is received when IRM G exits the core region.

D. A rod block is received when IRM G indication lowers to below 2.5/125 of scale.

Proposed Answer: B

Explanation: With the Mode Switch in START & HOT STBY, a rod block is received if an IRM leaves the fully inserted position. The IRM downscale rod block is bypassed when an IRM is on range 1.

A. Incorrect – With the Mode Switch in START & HOT STBY, a rod block is received if an IRM leaves the fully inserted position. Plausible because this would be correct with the Mode Switch in RUN. Also plausible because SRMs are withdrawn during the startup.
C. Incorrect – The rod block is received earlier, as soon as the detector leaves the fully inserted position. Plausible because SRMs are withdrawn during the startup and while the IRM is in the core region it is still giving a signal proportional to core flux.
D. Incorrect – The IRM downscale rod block is bypassed when an IRM is on range 1. Also, the rod block is received earlier than a downscale would be detected. Plausible because IRMs do have a downscale rod block.

Technical Reference(s): ARP 09-5-2-02, OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.07.b

Question Source: Bank – 9/12 NRC #37

Question History: 9/12 NRC #37

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 #	218000 A4.01
	Importance Rating	4.4

ADS

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

Proposed Question: #3

The plant was operating at 100% power when a loss of coolant accident resulted in the following:

Time	Condition(s)/Event(s)
10 seconds	• Reactor water level is 59" and lowering. • ADS Normal/Override switches on Panel 09-4 are in the NORMAL position. • Drywell pressure is 2.5 psig and rising. • A failure prevented the auto start of **ALL** low pressure pumps. • Annunciator 09-4-1-28, ADS TIMERS ACTUATED, is received.
40 seconds	• Core Spray pump A is manually started and is running on minimum flow with a discharge pressure of 130 psig. • Reactor water level is 30" and slowly lowering.
144 seconds	• Reactor water level is 40" and slowly rising.
155 seconds	• Reactor water level is 60" and slowly rising. • Reactor pressure is 700 psig and slowly lowering.

Which one of the following describes the status of the ADS valves at time 190 seconds?

At time 190 seconds, the ADS valves are…

A. closed because the ADS timers reset before the valves opened.

B. closed because the low pressure ECCS pump running logic is not satisfied.

C. open but will automatically close if Reactor water level rises above 177".

D. open and will remain open until manual action is taken to reset the ADS signal.

Proposed Answer: D

Explanation: At 10 seconds, the ADS timers actuate due to Reactor water level being below 59.5” (with confirmatory level being below 177”). The timers count down for 134 seconds. By time 144 seconds, the timers will have timed out since Reactor water level is still below 59.5”. When the timers time out, with at least one Core Spray pump running and discharge pressure above 100 psig, the ADS valves open. The ADS valves remain open even if Reactor water level rises above the ADS setpoint. ADS must then be manually reset for the valves to close.

A. Incorrect – The ADS timers finished timing at approximately 144 seconds and Reactor water level was not above the ADS setpoint by this time. Plausible because the ADS timers would reset when Reactor water level goes above 59.5” if they were still timing.
B. Incorrect – A single Core Spray pump running with discharge pressure above 100 psig satisfies the low pressure ECCS pump running logic and allows the ADS valves to open at approximately time 144 seconds. Plausible because only one ECCS pumps is running and it has lower than normal discharge pressure.
C. Incorrect – Once the ADS timers finish timing, Reactor water level rising above the ADS setpoint will not close the ADS valves. Manual action must be taken to reset ADS in this situation. Plausible because the ADS timers will reset prior to timing out and 177” is part of the confirmatory logic.

Technical Reference(s): SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05.c.1

Question Source: Bank – 3/14 NRC #10

Question History: 3/14 NRC #10

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 #	263000 K2.01
	Importance Rating	3.1

DC Electrical Distribution

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

Proposed Question: #4

The plant is operating at 100% power when 125 VDC Bus A de-energizes due to a sustained electrical fault.

Which one of the following identifies a valve that is affected by this loss?

A. 23MOV-17, HPCI CST SUCT VLV

B. 13MOV-18, RCIC CST SUCT VLV

C. 23MOV-15, HPCI INBD STM SUPP VLV

D. 13MOV-15, RCIC INBD STM SUPP VLV

Proposed Answer: B

Explanation: 13MOV-18 is powered from 71BMCC-1, which is powered from 125 VDC Bus A.

A. Incorrect – 23MOV-17 is powered from 71BMCC-2. Plausible because this is the opposite division of DC power.
C. Incorrect – 23MOV-15 is powered from 480 VAC MCC-153. Plausible because the related 23MOV-16 is powered from the opposite division of DC power.
D. Incorrect – 13MOV-15 is powered from 480 VAC MCC-163. Plausible because the related 13MOV-16 is power from DC system A.

Technical Reference(s): OP-15, OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.04.c

Question Source: Bank - 2008 NRC #45

Question History: 2008 NRC #45

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 #	264000 K3.02
	Importance Rating	3.9

EDGs

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: A.C. electrical distribution

Proposed Question: #5

The plant was operating at 100% power when a loss of coolant accident and loss of all offsite power resulted in the following sequence of events:

Time (seconds)	Event(s)
0	Drywell pressure is 2.7 psig and rising. Reactor water level is 120” and lowering. Lines 3 and 4 de-energize.
10	Emergency Diesel Generator (EDG) C loads onto its respective emergency bus. The other three EDGs have locked out due to generator faults.
35	Reactor pressure is 220 psig and lowering.

Which one of the following describes the number of Residual Heat Removal (RHR) and Core Spray (CS) pumps injecting to the Reactor at time 45 seconds?

	Number of RHR Pumps Injecting	Number of CS Pumps Injecting
A.	Two	One
B.	Two	Zero
C.	One	One
D.	One	Zero

Proposed Answer: C

Explanation: All four EDGs have a start signal due to both high drywell pressure and undervoltage. The undervoltage signal also gives the EDGs a signal to load onto the emergency buses. With only one EDG loaded, one emergency bus is de-energized, resulting in two of four RHR pumps and one of two CS Pumps being unavailable. Additionally, one additional RHR pump on the energized emergency bus will not start due to lack of two EDGs tied to the bus. Therefore only one RHR pump and one CS pump have started. With Reactor pressure less than 450 psig, both systems will be injecting. The EDGs and ECCS systems are designed to ensure injection can be established before 45 seconds have elapsed.

A. Incorrect – Only one RHR pump will be injecting. Plausible because two RHR pumps have power available to them.
B. Incorrect – Only one RHR pump will be injecting. Plausible because two RHR pumps have power available to them. One CS pump will be injecting. Plausible because this would be correct if Reactor pressure were above CS injection capacity or if CS had a lockout feature similar to the RHR pumps in this situation.
D. Incorrect – One CS pump will be injecting. Plausible because this would be correct if Reactor pressure were above CS injection capacity or if CS had a lockout feature similar to the RHR pumps in this situation.

Technical Reference(s): SDLP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.10.a

Question Source: Bank – 9/12 NRC #51

Question History: 9/12 NRC #51

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 K3.01
	Importance Rating	3.4

Source Range Monitor

Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: RPS

Proposed Question: #6

A plant startup is in progress with the following:

- Special low-power physics testing is being conducted due to a new fuel design.
- The Mode Switch is in START & HOT STBY.
- The Source Range Monitor (SRM) Shorting Links are removed.
- Then, SRM C fails upscale.

Which one of the following describes the status of the Reactor Protection System (RPS)?

A. No scram signals are received.

B. A half-scram signal is received on RPS A only.

C. A half-scram signal is received on RPS B only.

D. A full scram signal is received.

Proposed Answer: D

Explanation: The SRM scram signal is only applicable with the associated RPS shorting links removed. With the shorting links removed, any SRM above 5 x 10^5 cps or INOP will cause a full scram.

A. Incorrect – A full scram is received. Plausible because this would be correct if the shorting links were installed, not removed.
B. Incorrect – A full scram is received. Plausible that a half scram would be received since only one SRM has failed. Plausible because this would be correct for an IRM A or C failure.
C. Incorrect – A full scram is received. Plausible that a half scram would be received since only one SRM has failed. Plausible because this would be correct for an IRM B or D failure. Plausible that RPS A would receive input from SRMs A and B, with RPS B receiving input from SRMs C and D.

Technical Reference(s): OP-16, SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.b.3, 1.05.c.e

Question Source: Modified Bank - 9/12 NRC #39

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Source Range Monitor

Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: RPS status

Proposed Question: #39

A Plant startup is in progress with the following:

- Special low-power physics testing is being conducted due to a new fuel design.
- The Mode Switch is in START & HOT STBY.
- The Source Range Monitor (SRM) Shorting Links are removed.
- Then, an inadvertent cold-water addition results in the following SRM indications:
    - SRM A: 75,000 cps
    - SRM B: 110,000 cps
    - SRM C: 90,000 cps
    - SRM D: 100,000 cps

Which one of the following describes the status of the Reactor Protection System (RPS)?

A. No scram signals have been received due to SRMs.

B. A half-scram signal has been received on RPS A due to SRMs.

C. A half-scram signal has been received on RPS B due to SRMs.

D. A full scram signal has been received due to SRMs.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K4.01
	Importance Rating	3.4

Component Cooling Water

Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump

Proposed Question: #7

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

- Reactor Building Closed Loop Cooling (RBCLC) pumps A and B are in service.
- RBCLC pump C is in standby.
- RBCLC discharge header pressure is 112 psig.
- Then, RBCLC pump A breaker trips on overcurrent.
- RBCLC discharge header pressure lowers to 87 psig.

Which one of the following describes the status of RBCLC pump C?

RBCLC pump C…

A. auto-starts due to low discharge header pressure.

B. auto-starts due to a RBCLC pump A breaker trip signal.

C. remains in standby until Operators manually start it.

D. remains in standby unless discharge header pressure lowers further.

Proposed Answer: B

Explanation: RBCLC pump C will auto-start if either another RBCLC pump breaker trips or header pressure lowers below 75 psig. Therefore, RBCLC pump C will auto-start due to RBCLC pump A breaker trip.

A. Incorrect – RBCLC pump C starts on a breaker trip signal, not on low discharge pressure. Plausible because it would start on low discharge pressure if pressure lowered further.
C. Incorrect – RBCLC pump C starts on a breaker trip signal. Plausible because many similar systems do not have auto-start features for pumps.
D. Incorrect – RBCLC pump C starts on a breaker trip signal. Plausible because pressure has not lowered to the low discharge pressure automatic start setpoint.

Technical Reference(s): OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.05.c.1

Question Source: Bank – 9/12 NRC #53

Question History: 9/12 NRC #53

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 #	215005 K4.02
Importance Rating	4.1

APRM / LPRM

Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals

Proposed Question: #8

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- An inadvertent reactivity addition occurs.
- APRMs indicate as follows:

APRM	Indication
A	5%
B	18%
C	8%
D	13%
E	10%
F	10%

Which one of the following describes the status of rod blocks and scram signals based on the APRMs?

A. NO rod block or scram signal is received.

B. A rod block is received, but NO scram signal is received.

C. A rod block and a half scram signal is received, but a full scram signal is NOT received.

D. A rod block and a full scram signal is received.

Proposed Answer: C

Explanation: With the Reactor Mode Switch out of RUN, APRMs cause a rod block if they exceed 12% and a half scram if they exceed 15%. Therefore, APRMs B and D are both causing a rod block. APRM B is also causing a half scram signal on RPS B. Since no other APRM has exceeded 15%, then a full scram signal is not received.

A. Incorrect – APRMs B and D are both causing a rod block. APRM B is also causing a half scram signal on RPS B. Plausible because if the Reactor Mode Switch were in RUN, then this would be the correct answer.
B. Incorrect – APRM B is also causing a half scram signal on RPS B. Plausible because APRM B is only slightly above the scram setpoint, and well below the scram setpoint if the Reactor Mode Switch were in RUN.
D. Incorrect – No full scram signal is received. Plausible because one APRM is above the scram setpoint and another one is above the rod block setpoint.

Technical Reference(s): ARPs 09-5-2-2, 09-5-2-44, 09-5-2-55

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.05.c.3.b

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 #	211000 K5.04
	Importance Rating	3.1

SLC

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Explosive valve operation

Proposed Question: #9

An ATWS has occurred with the following:

- MCC 162 has de-energized due to a fault.
- Standby Liquid Control (SLC) injection is being initiated from Panel 09-3.

Which one of the following describes the impact of the electrical fault on SLC injection capability?

A. Full design flow is still available through one pump and two squib valves.

B. Full design flow is still available through one pump and one squib valve.

C. Only 50% of design flow is available through one pump and two squib valves.

D. Only 50% of design flow is available through one pump and one squib valve.

Proposed Answer: B

Explanation: MCC 162 provides power for the following SLC components:

- SLC pump B
- SLC B squib valve
- SLC storage tank heater

With the loss of MCC 162, only SLC pump A and SLC A squib valve are available for injection. The SLC pumps are redundant, 100% capacity positive-displacement pumps that provide a flow rate of 50 gpm. The SLC squib valves are redundant, parallel valves each capable of passing the 50 gpm flow rate provided by the positive-displacement SLC pumps.

A. Incorrect – Only one squib valve is available. Plausible that the squib valve power source would be different than the pump power source, as it is on some other plants.
C. Incorrect – Only one squib valve is available. Plausible that the squib valve power source would be different than the pump power source, as it is on some other plants. Full flow is still available. Plausible because both squibs normally fire and allow flow in parallel.
D. Incorrect – Full flow is still available. Plausible because both squibs normally fire and allow flow in parallel.

Technical Reference(s): UFSAR Section 3.9, OP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11 1.05.a.2

Question Source: Bank - 9/12 NRC #34

Question History: 9/12 NRC #34

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 K5.13
	Importance Rating	2.9

Instrument Air

Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Filters

Proposed Question: #10

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

- Instrument Air Dryer 39AD-4A is in service.
- Instrument Air Dryer 39AD-4B is out of service for maintenance.

Then, the Instrument Air Dryer 39AD-4A after-filter ***completely*** clogs.

Which one of the following describes the response of Instrument Air and Scram Air header pressures with NO operator action?

A. Both Instrument Air and Scram Air header pressures remain stable due to automatic bypassing of the filter.

B. Instrument Air header pressure lowers and Scram Air header pressure remains stable.

C. Instrument Air header pressure remains stable and Scram Air header pressure lowers.

D. Both Instrument Air and Scram Air header pressures lower.

Proposed Answer: D

Explanation: With 39AD-4B out of service (and 39AD-3 normally out of service), only 39AD-4A passes Instrument Air on to either the Instrument Air and Scram Air headers based on system piping configuration. There is an alarm on high after-filter differential pressure, but no automatic bypass. Therefore, with the after-filter completely clogged, both Instrument Air and Scram Air header pressures lower.

A. Incorrect – Both Instrument Air and Scram Air header pressures lower. Plausible because there is an alarm on high after-filter differential pressure, and some Instrument Air systems do have automatic bypass features on similar faults.
B. Incorrect – Scram Air header pressure lowers. Plausible if the Scram Air header tapped off upstream of the dryer or after-filter. Plausible because the Scram Air header does have a separate filter.
C. Incorrect – Instrument Air header pressure lowers. Plausible if the Instrument Air header tapped off upstream of this dryer or after-filter and/or was served by a different dryer or after-filter. Also plausible because the Scram Air header does have a separate filter and there is another installed dryer (39AD-3).

Technical Reference(s): FM-39, ARP-09-6-2-37

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.a.7

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 #	262002 K6.01
	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.): A.C. electrical power

Proposed Question: #11

The plant is operating at 100% power when L-25 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact on the Uninterruptible Power Supply (UPS)?

The UPS…

A. transfers to the backup DC supply.

B. transfers to the secondary AC power source.

C. remains powered from the normal supply, but loses power to the rectifier.

D. remains powered from the normal supply, but loses the secondary AC power source.

Proposed Answer: D

Explanation: The normal AC supply to the UPS rectifier is MCC-262, from L-26. The backup DC supply to the UPS inverter is DC Bus A. The supply to the secondary AC power source (bypass transformer) is MCC-252, from L-25. Therefore, with the loss of L-25, the UPS stays on the normal AC supply, but loses power to the secondary AC power source (bypass transformer).

A. Incorrect – The UPS stays on the normal AC supply. Plausible because this would be correct for a loss of L-26.
B. Incorrect – The UPS stays on the normal AC supply. Plausible because this would be correct for a loss of L-26 and DC Bus A.
C. Incorrect – The rectifier remains powered from L-26. Plausible if the candidate believed the DC supply normally carried the UPS with the rectifier as the backup, and confused L-25 and L-26.

Technical Reference(s): OP-46B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71F 1.04.a

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 #	217000 K6.03
	Importance Rating	3.5

RCIC

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply

Proposed Question: #12

The plant has experienced a loss of coolant accident with the following:

- RCIC is injecting 400 gpm to the Reactor.
- CST level is 62" and slowly lowering.
- Torus water level is 10.5' and slowly lowering.

Which one of the following describes the effect of these conditions on the operation of RCIC?

RCIC is currently operating with suction from the…

A. CSTs. If CST level continues to lower, the RCIC suction path will automatically swap to the Torus.

B. Torus. If Torus level continues to lower, the RCIC suction path will automatically swap to the CSTs.

C. CSTs. If CST level continues to lower, the RCIC suction path will stay aligned to the CSTs and RCIC pump vortexing may become a concern.

D. Torus. If Torus level continues to lower, the RCIC suction path will stay aligned to the Torus and RCIC pump vortexing may become a concern.

Proposed Answer: A

Explanation: RCIC suction is normally aligned to the CSTs. RCIC suction swaps to the Torus when CST level drops <59.5”. Since CST level is still above 59.5”, RCIC suction is still aligned to the CST. Torus water level is below the EOP entry and Tech Spec low level and below the 10.75’ action level in EOP-4, but there is no interlock to prevent the suction swap from occurring based on low Torus water level. RCIC pump vortexing becomes a concern if Torus water level continues to lower below 5.7’.

B. Incorrect – Since CST level is still above 59.5”, RCIC suction is still aligned to the CST. Plausible because CST level is low and approaching the swap setpoint.
C. Incorrect – If CST level continues to lower, the RCIC suction path will automatically swap to the Torus. Plausible because Torus water level is also very low and vortexing may be a concern even after the suction swap.
D. Incorrect – Since CST level is still above 59.5”, RCIC suction is still aligned to the CST. Plausible because CST level is low and approaching the swap setpoint.

Technical Reference(s): OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.05.b.3

Question Source: Modified Bank – 9/14 NRC #12

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

RCIC

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply

Proposed Question: #12

The plant has experienced a loss of coolant accident with the following:

- RCIC is injecting 400 gpm to the Reactor.
- CST level is 57 inches and slowly lowering.
- Torus water level is 10.8 feet and slowly lowering.

Which one of the following describes the effect of these conditions on the operation of RCIC?

RCIC is currently operating with suction from the…

A. CSTs. If CST level continues to lower, the RCIC suction path will automatically swap to the Torus.

B. Torus. If Torus level continues to lower, the RCIC suction path will automatically swap to the CSTs.

C. CSTs. If CST level continues to lower, the RCIC suction path will stay aligned to the CSTs and RCIC pump vortexing may become a concern.

D. Torus. If Torus level continues to lower, the RCIC suction path will stay aligned to the Torus and RCIC pump vortexing may become a concern.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A1.05
	Importance Rating	4.1

HPCI

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: Suppression pool temperature: BWR-2,3,4

Proposed Question: #13

The plant is operating at 75% power with ST-4N, HPCI Quick-Start, In-service, and Transient Monitoring Test, in progress.

Which one of the following identifies the lowest Torus water temperature that, if exceeded, requires immediately securing the test, in accordance with ST-4N and Technical Specifications?

A. 95ºF

B. 100ºF

C. 105ºF

D. 110ºF

Proposed Answer: C

Explanation: TS 3.6.2.1 allows Torus water temperature to go as high as 105ºF during ST-4N (up from the normal limit of 95ºF). Once Torus water temperature exceeds 105ºF, TS 3.6.2.1 Condition C requires ST-4N to be immediately suspended.

A. Incorrect – 105ºF is the threshold temperature that requires immediately securing the test. Plausible because 95ºF is the normal limit in TS 3.6.2.1.
B. Incorrect – 105ºF is the threshold temperature that requires immediately securing the test. Plausible that ST-4N would have a requirement for securing the test that was greater than 95ºF but below the TS limit of 105ºF.
D. Incorrect – 105ºF is the threshold temperature that requires immediately securing the test. Plausible because 110ºF is an additional, higher limit in ST-4N and TS 3.6.2.1.

Technical Reference(s): ST-4N, Technical Specification 3.6.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.16

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 #	2
	Group #	1
	K/A #	205000 A1.02
	Importance Rating	3.3

Shutdown Cooling

Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: SDC/RHR pump flow

Proposed Question: #14

The plant is shutdown for a refueling outage with the following:

- RHR pump B is operating in the Shutdown Cooling (SDC) lineup.
- All other RHR pumps are secured.
- RHR pump B flow is 4000 gpm.

Which one of the following describes how RHR pump B flow should be adjusted, in accordance with OP-13D, RHR – Shutdown Cooling?

Flow should be…

A. lowered to prevent pump runout.

B. lowered to prevent valve flow erosion.

C. raised to prevent high pump vibration.

D. raised to prevent cycling of the minimum flow valve.

Proposed Answer: C

Explanation: OP-13D, RHR – Shutdown Cooling, Precaution C.2.4 states:

"Operation of any single RHR pump at flows less than 6500 gpm or operation of any two RHR pumps at flows less than 13000 gpm should be minimized to prevent high pump vibration."

A. Incorrect – 4000 gpm is below the minimum of 6500 gpm, therefore flow should be raised. Plausible because 4000 gpm is a large amount of flow well above the minimum flow valve setpoint. Also plausible because runout is a concern at high pump flows.
B. Incorrect – 4000 gpm is below the minimum of 6500 gpm, therefore flow should be raised. Plausible because 4000 gpm is a large amount of flow well above the minimum flow valve setpoint. Also plausible because higher flows would cause more erosion of the throttling valve.
D. Incorrect – The minimum flow valve cycles in the range of 1250-1450 gpm. At 4000 gpm, the minimum flow valve would be closed. Plausible because at even lower flow in other RHR operating modes, this could be of concern.

Technical Reference(s): OP-13D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13.d

Question Source: Bank – 9/12 NRC #29

Question History: 9/12 NRC #29

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 #	262001 A2.01
	Importance Rating	3.4

AC Electrical Distribution

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

Proposed Question: #15

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

- The Main Turbine trips.
- One minute later, the electrical distribution system is aligned as shown in the pictures on the following four (4) pages.

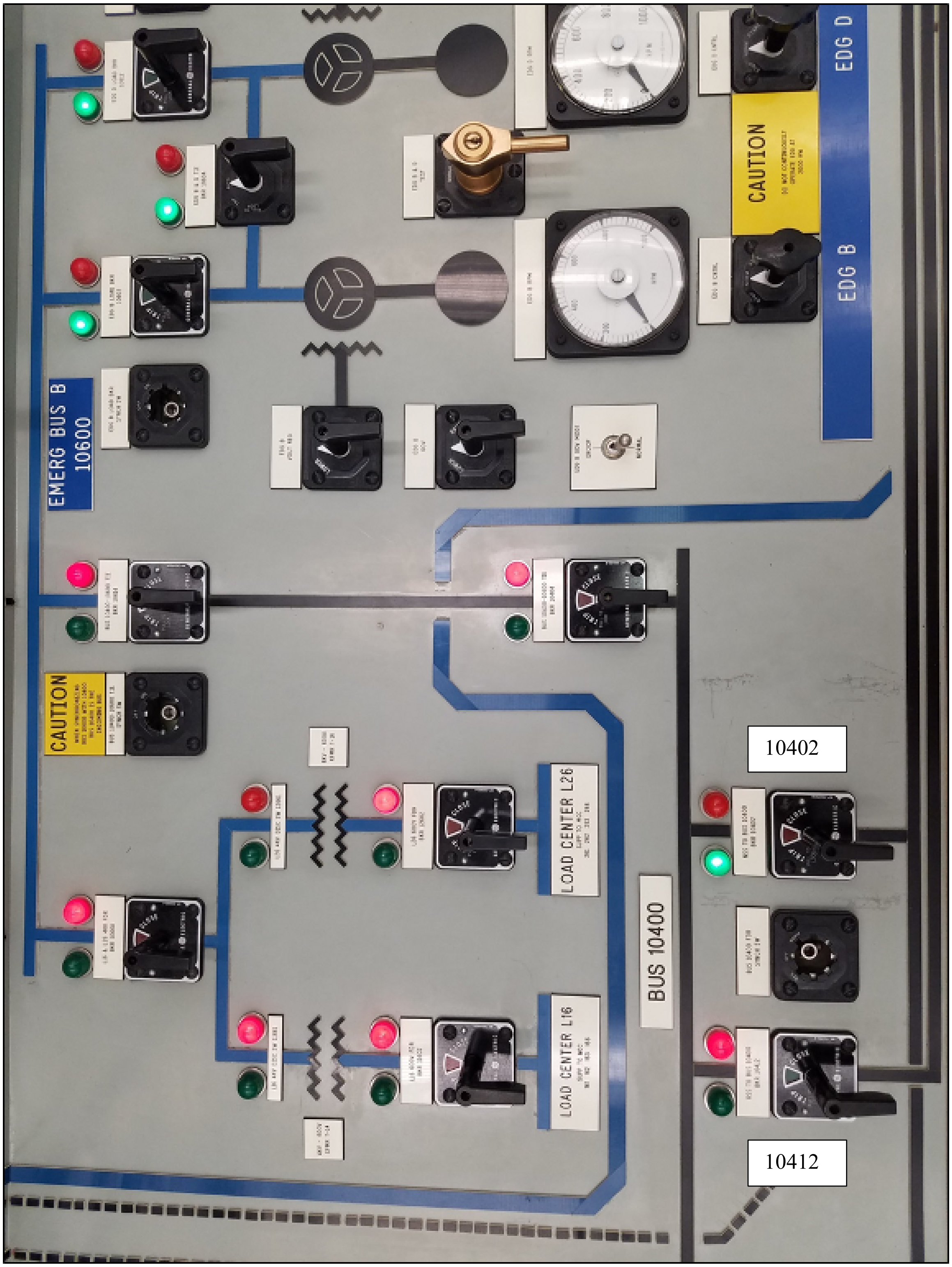
Which one of the following describes the required operator action, in accordance with the Abnormal Operating Procedures?

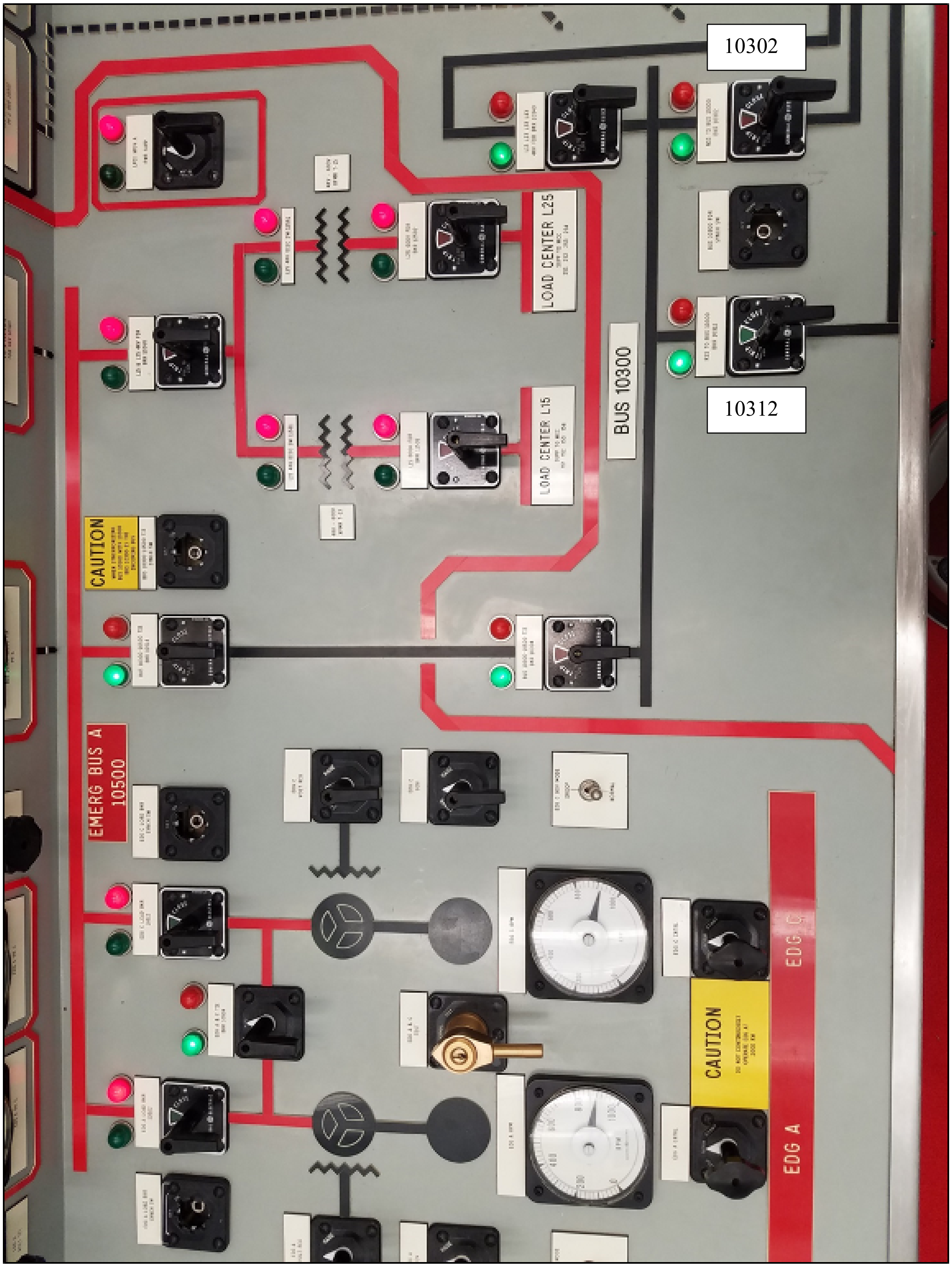
A. Close breaker 10312, BUS 10300 RESERVE SUPP BKR.

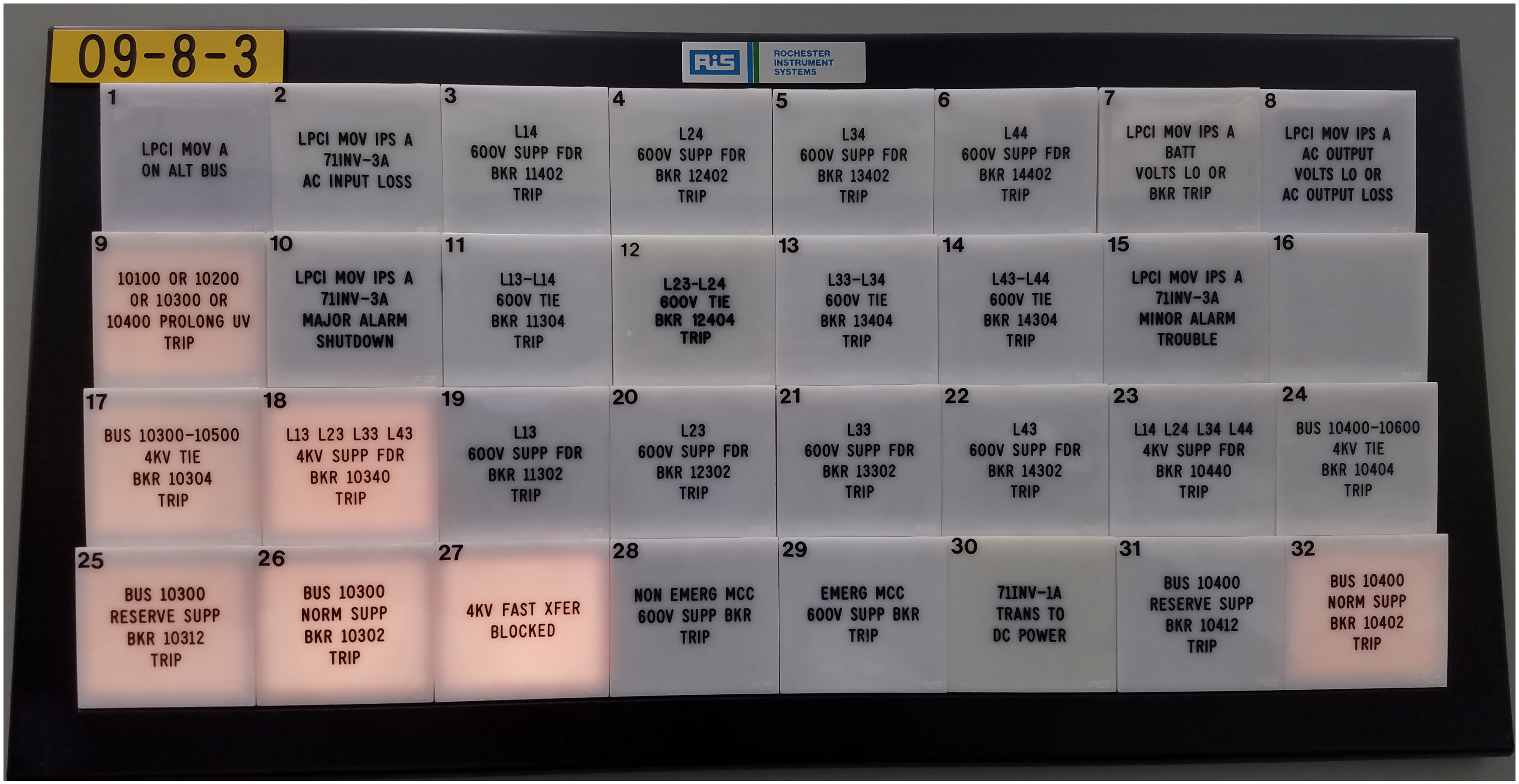
B. Close breaker 10302, BUS 10300 NORM SUPP BKR.

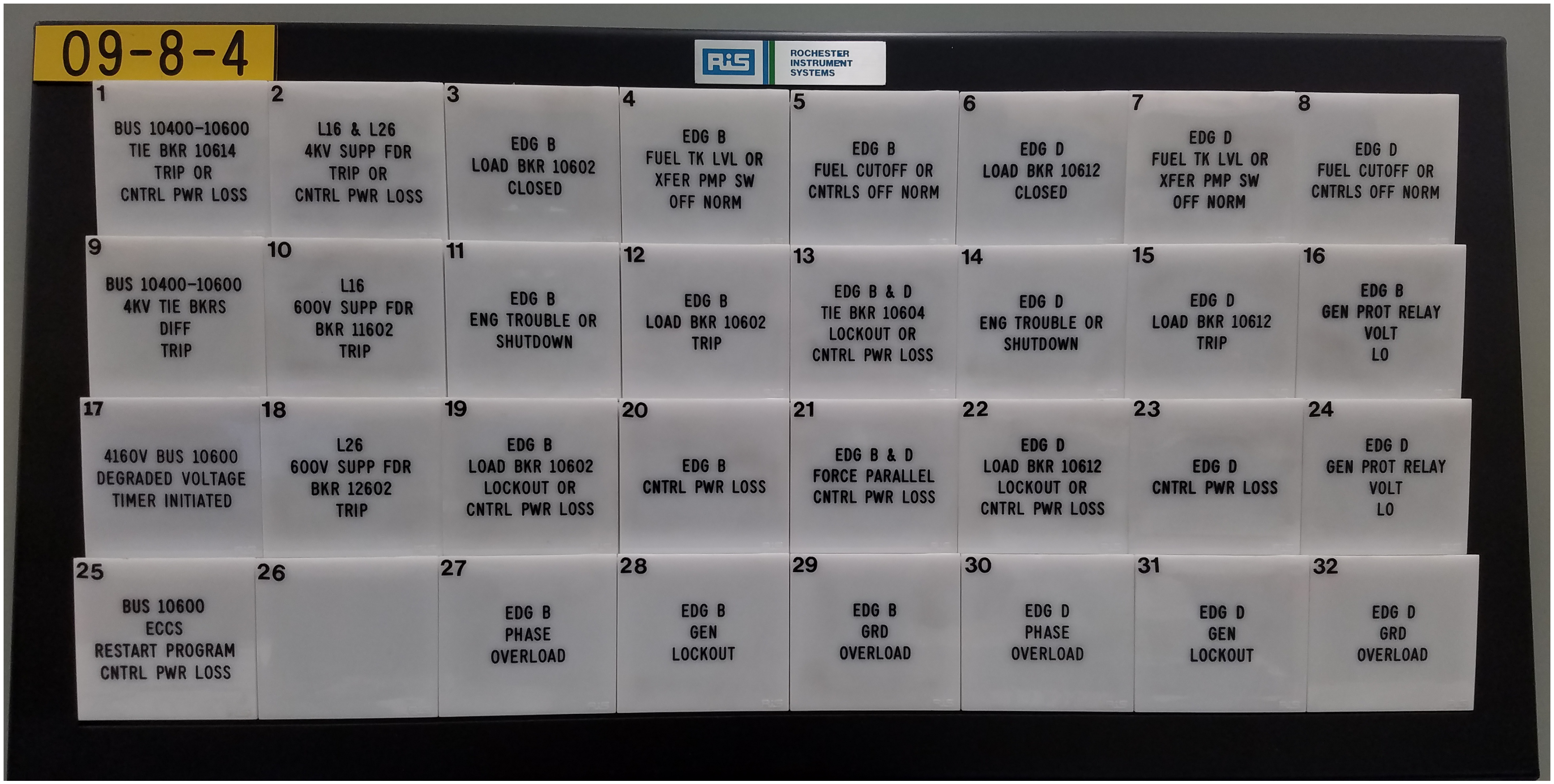
C. Cross-tie 10300 Bus L-Gear with 10400 Bus L-Gear.

D. Cross-tie 10300 Bus L-Gear with 10500 Bus L-Gear.


10402
10412
EMERG BUS B
10600
EDG D
EDG B
CAUTION
CAUTION
LOAD CENTER L26
LOAD CENTER L16
BUS 10400


10302
10312
BUS 10300
LOAD CENTER L25
LOAD CENTER L15
CAUTION
EMERG BUS A
10500
CAUTION
EDG C
EDG A


09-8-3
RIS ROCHESTER INSTRUMENT SYSTEMS
1 LPCI MOV A ON ALT BUS
2 LPCI MOV IPS A 71INV-3A AC INPUT LOSS
3 L14 600V SUPP FDR BKR 11402 TRIP
4 L24 600V SUPP FDR BKR 12402 TRIP
5 L34 600V SUPP FDR BKR 13402 TRIP
6 L44 600V SUPP FDR BKR 14402 TRIP
7 LPCI MOV IPS A BATT VOLTS LO OR BKR TRIP
8 LPCI MOV IPS A AC OUTPUT VOLTS LO OR AC OUTPUT LOSS
9 10100 OR 10200 OR 10300 OR 10400 PROLONG UV TRIP
10 LPCI MOV IPS A 71INV-3A MAJOR ALARM SHUTDOWN
11 L13-L14 600V TIE BKR 11304 TRIP
12 L23-L24 600V TIE BKR 12404 TRIP
13 L33-L34 600V TIE BKR 13404 TRIP
14 L43-L44 600V TIE BKR 14304 TRIP
15 LPCI MOV IPS A 71INV-3A MINOR ALARM TROUBLE
16
17 BUS 10300-10500 4KV TIE BKR 10304 TRIP
18 L13 L23 L33 L43 4KV SUPP FDR BKR 10340 TRIP
19 L13 600V SUPP FDR BKR 11302 TRIP
20 L23 600V SUPP FDR BKR 12302 TRIP
21 L33 600V SUPP FDR BKR 13302 TRIP
22 L43 600V SUPP FDR BKR 14302 TRIP
23 L14 L24 L34 L44 4KV SUPP FDR BKR 10440 TRIP
24 BUS 10400-10600 4KV TIE BKR 10404 TRIP
25 BUS 10300 RESERVE SUPP BKR 10312 TRIP
26 BUS 10300 NORM SUPP BKR 10302 TRIP
27 4KV FAST XFER BLOCKED
28 NON EMERG MCC 600V SUPP BKR TRIP
29 EMERG MCC 600V SUPP BKR TRIP
30 71INV-1A TRANS TO DC POWER
31 BUS 10400 RESERVE SUPP BKR 10412 TRIP
32 BUS 10400 NORM SUPP BKR 10402 TRIP


09-8-4
ROCHESTER INSTRUMENT SYSTEMS
1 BUS 10400-10600 TIE BKR 10614 TRIP OR CNTRL PWR LOSS
2 L16 & L26 4KV SUPP FDR TRIP OR CNTRL PWR LOSS
3 EDG B LOAD BKR 10602 CLOSED
4 EDG B FUEL TK LVL OR XFER PMP SW OFF NORM
5 EDG B FUEL CUTOFF OR CNTRLS OFF NORM
6 EDG D LOAD BKR 10612 CLOSED
7 EDG D FUEL TK LVL OR XFER PMP SW OFF NORM
8 EDG D FUEL CUTOFF OR CNTRLS OFF NORM
9 BUS 10400-10600 4KV TIE BKRS DIFF TRIP
10 L16 600V SUPP FDR BKR 11602 TRIP
11 EDG B ENG TROUBLE OR SHUTDOWN
12 EDG B LOAD BKR 10602 TRIP
13 EDG B & D TIE BKR 10604 LOCKOUT OR CNTRL PWR LOSS
14 EDG D ENG TROUBLE OR SHUTDOWN
15 EDG D LOAD BKR 10612 TRIP
16 EDG B GEN PROT RELAY VOLT LO
17 4160V BUS 10600 DEGRADED VOLTAGE TIMER INITIATED
18 L26 600V SUPP FDR BKR 12602 TRIP
19 EDG B LOAD BKR 10602 LOCKOUT OR CNTRL PWR LOSS
20 EDG B CNTRL PWR LOSS
21 EDG B & D FORCE PARALLEL CNTRL PWR LOSS
22 EDG D LOAD BKR 10612 LOCKOUT OR CNTRL PWR LOSS
23 EDG D CNTRL PWR LOSS
24 EDG D GEN PROT RELAY VOLT LO
25 BUS 10600 ECCS RESTART PROGRAM CNTRL PWR LOSS
26
27 EDG B PHASE OVERLOAD
28 EDG B GEN LOCKOUT
29 EDG B GRD OVERLOAD
30 EDG D PHASE OVERLOAD
31 EDG D GEN LOCKOUT
32 EDG D GRD OVERLOAD

Proposed Answer: C

Explanation: On the Main Turbine trip, breaker 10302, BUS 10300 NORM SUPP BKR, should have opened and breaker 10312, BUS 10300 RESERVE SUPP BKR, should have closed automatically on a fast transfer to maintain buses 10300 and 10500 energized from offsite power. The pictures show that breaker 10312 is open, an annunciator has been received indicating that it tripped, and the 10500 bus is powered from the EDGs. AOP-16, Loss of 10300 Bus, must be entered and the immediate actions require cross-tying 10300 Bus L Gear with 10400 Bus L Gear.

Note: While the answer choices do not directly test the first half of the K/A, they do require the candidate to predict the normal response of the AC electrical distribution system to a Main Turbine trip as an intermediate step in arriving at the correct answer (i.e. understand the integrated electrical plant response and what has gone wrong to then know what action to take).

A. Incorrect – Breaker 10312 has tripped and should not be re-closed. Plausible because this breaker should have automatically closed and, if the trip annunciator was not in, could be manually closed to back-up the failed automatic action.
B. Incorrect – Closing breaker 10302 is not correct because there is no power to it due to the Main Generator lockout that follows the Main Turbine trip. Plausible because this is the normal source of power to the 10300 bus prior to the Main Turbine trip.
D. Incorrect – Cross-tie is required to the 10400 bus L-Gear, not the 10500 bus L-Gear. Plausible because the 10500 bus is in the same division as the 10300 bus.

Technical Reference(s): AOP-1, ARP 09-8-3-25, AOP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.14.a

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 #	2
	Group #	1
	K/A #	203000 A2.04
	Importance Rating	3.5

RHR/LPCI: Injection Mode

Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. failures

Proposed Question: #16

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

- Annunciator 09-8-3-02, LPCI MOV IPS A 71INV-3A AC INPUT LOSS.
- An operator in the field verifies that breaker 71MCC-152-OE1 (71INV-3A LPCI Inverter Power Supply) has tripped.

Which one of the following describes the current ability of RHR loop A to inject in LPCI mode if a loss of coolant accident occurs, and the required operator action, in accordance with ARP 09-8-3-02?

The LPCI mode of RHR loop A is currently ___(1)___ if a loss of coolant accident occurs.

Place LPCI MOV A PWR SUPP switch in ___(2)___.

	(1)	(2)
A.	available	BYPASS
B.	available	ALT PULL TO LOCK
C.	NOT available	BYPASS
D.	NOT available	ALT PULL TO LOCK

Proposed Answer: B

Explanation: With the LPCI MOV A PWR SUPP switch initially in the normal position of NORMAL and loss of the AC input, the inverter is automatically supplied by the associated battery. Therefore, RHR loop A is still available for the LPCI mode if a LOCA were to occur. ARP 09-8-3-02 requires placing the LPCI MOV A PWR SUPP switch in ALT PULL TO LOCK to transfer to the alternate AC supply since the battery will otherwise discharge and LPCI will eventually become unavailable.

A. Incorrect – ARP 09-8-3-02 requires placing the LPCI MOV A PWR SUPP switch in ALT PULL TO LOCK. Plausible because this is an alternate switch position that is used to allow swap of the power supply following a LOCA.
C. Incorrect – RHR loop A is still available for the LPCI mode if a LOCA were to occur. Plausible because the normal supply to the MOV IPS is lost and manual action is required to be taken to swap to an alternate AC supply. ARP 09-8-3-02 requires placing the LPCI MOV A PWR SUPP switch in ALT PULL TO LOCK. Plausible because this is an alternate switch position that is used to allow swap of the power supply following a LOCA.
D. Incorrect – RHR loop A is still available for the LPCI mode if a LOCA were to occur. Plausible because the normal supply to the MOV IPS is lost and manual action is required to be taken to swap to an alternate AC supply.

Technical Reference(s): OP-43C, SDLP-10, ARP 09-8-3-02

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10

Question Source: Modified Bank – 2010 NRC #49

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

QUESTION	49.

The following describes the initial plant conditions:

- Reactor is at 100% power.
- LPCI independent power supply is in a normal lineup per OP-43C, "LPCI Independent Power Supply System".

Subsequently, the following occurs:

- Main battery breaker 1CB2 to LPCI Inverter 71INV-3B opens.
- A LOCA occurs. The reactor scrams and RPV pressure is rapidly lowering.

Which **ONE** of the choices below describes the correct operator response to address the opening of Breaker 1CB2 per OP-43C **AND** the reason for this response?

	Operator Response	Reason
A.	Place LPCI MOV B PWR SUPP switch in BYPASS	Prevent 1CB1 from opening on an Injection Signal. Power MOV bus from the inverter.
B.	Place LPCI MOV B PWR SUPP switch in ALT PULL TO LOCK	Prevent 1CB1 from opening on an Injection Signal. Power MOV bus from the inverter.
C.	Place LPCI MOV B PWR SUPP switch in BYPASS	Transfer MOV bus power to Alternate feed.
D.	Place LPCI MOV B PWR SUPP switch in ALT PULL TO LOCK	Transfer MOV bus power to Alternate feed.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A3.08
	Importance Rating	3.7

RPS

Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Recirculation pump trip

Proposed Question: #17

The plant 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 1170 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 Water Recirculation (RWR) pumps after this transient?

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

Proposed Answer: A

Explanation: Reactor pressure rising above 1153 psig actuates both the ARI and ATWS-RPT logic. The ARI logic energizes the ARI solenoids to vent the scram air header and cause an alternate Reactor scram method. The ATWS-RPT logic trips the RWR pumps. There is no time delay on the high pressure actuation of ARI and ATWS-RPT logic.

B. Incorrect – RWR pumps tripped due to ATWS-RPT logic actuation. Plausible because that Reactor pressure didn’t exceed the setpoint or didn’t stay high long enough.
C. Incorrect – ARI solenoids are energized due to ARI logic actuation. Plausible because that Reactor pressure didn’t exceed the setpoint or didn’t stay high long enough.
D. Incorrect – ARI solenoids are energized due to ARI logic actuation. RWR pumps tripped due to ATWS-RPT logic actuation. Plausible because that Reactor pressure didn’t exceed the setpoints or didn’t stay high long enough.

Technical Reference(s): OP-25, OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.05.c.8, SDLP-02H 1.05.c.2

Question Source: Bank – 3/14 NRC #50

Question History: 3/14 NRC #50

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

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

LPCS

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Valve operation

Proposed Question: #18

A loss of coolant accident has resulted in the following:

- Reactor water level is -40” and stable.
- Reactor pressure is 250 psig and slowly lowering.
- Emergency Depressurization has been performed.
- Torus water temperature is 105ºF and slowly rising.
- Core Spray is the only system available for injection into the Reactor.
- Core Spray pump A is running.
- Core Spray pump B has tripped.
- Core Spray pump A flow is 5000 gpm.
- Core Spray system A valves are aligned as shown in the picture on the next page.

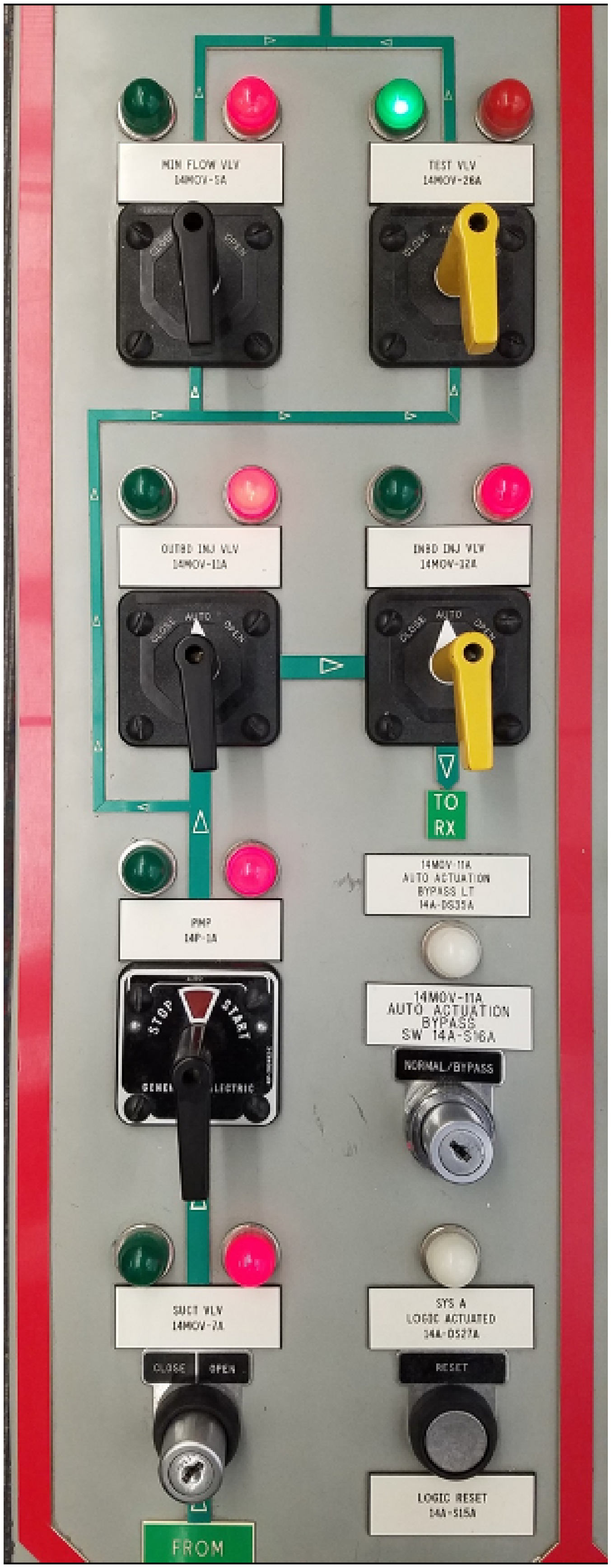
Which one of the following describes the status of Core Spray system A?

A. Core Spray system A is operating as designed.

B. The full flow test valve has NOT operated as designed.

C. The minimum flow valve has NOT operated as designed.

D. The inboard injection valve has NOT operated as designed.


MIN FLOW VLV
14MOV-5A
TEST VLV
14MOV-26A
OUTBD INJ VLV
14MOV-11A
INBD INJ VLV
14MOV-12A
TO
RX
14MOV-11A
AUTO ACTUATION
BYPASS LT
14A-DS35A
PMP
14P-1A
STOP
START
14MOV-11A
AUTO ACTUATION
BYPASS
SW 14A-S16A
NORMAL/BYPASS
SUCT VLV
14MOV-7A
CLOSE
OPEN
SYS A
LOGIC ACTUATED
14A-DS27A
RESET
LOGIC RESET
14A-S15A
FROM

Proposed Answer: C

Explanation: Since Reactor water level is less than 59.5” and Reactor pressure is less than 410 psig, Core Spray A should be automatically aligned for injection. The picture shows the injection valves full open, the full flow test valve closed, and the min flow valve open. With flow greater than 980 gpm, the minimum flow valve should be closed. This is diverting flow from the Reactor. With Reactor water level at -40” and stable with limited injection sources, this diverted water is significantly impacting the ability to provide adequate core cooling.

A. Incorrect – The min flow valve should be closed. Plausible because the min flow valve is normally open and does not close until <410 psig when the inboard injection valve opens.
B. Incorrect – The full flow test valve indicates closed, as it should be for plant conditions. Plausible if the candidate mis-identifies the full flow test valve or its position.
D. Incorrect – The inboard injection valve indicates open, as it should be for plant conditions. Plausible because above 410 psig, this valve would be required to be closed.

Technical Reference(s): OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.05.c.2

Question Source: Bank – 3/14 NRC #15

Question History: 3/14 NRC #15

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 #	261000 A4.03
	Importance Rating	3.0

SGTS

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

Proposed Question: #19

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

- A small coolant leak has developed in the Drywell.
- AOP-39, Loss of Coolant, has been entered.
- The CRS has directed venting the Torus per OP-37, Containment Atmosphere Dilution System.

Which one of the following describes when the Standby Gas Treatment (SBGT) fan is started and which control switch is used to start the fan, in accordance with OP-37 and OP-20, Standby Gas Treatment System?

The SBGT fan is started ___(1)___ opening the Torus vent valves.

The SBGT fan is started by manipulating the ___(2)___ control switch.

	(1)	(2)
A.	before	TRAIN A(B) INLET 01-125MOV-14A(B)
B.	before	TRAIN A(B) DISCH 01-125MOV-15A(B)
C.	after	TRAIN A(B) INLET 01-125MOV-14A(B)
D.	after	TRAIN A(B) DISCH 01-125MOV-15A(B)

Proposed Answer: A

Explanation: OP-37 requires starting the SBGT fan before opening the Torus vent valves. The SBGT fan is started by taking the TRAIN A(B) INLET 01-125MOV-14A(B) control switch to OPEN.

B. Incorrect – The SBGT fan is started by taking the TRAIN A(B) INLET 01-125MOV-14A(B) control switch to OPEN. Plausible because this is a similar control switch to the correct answer and this valve is also verified open when the fan starts.
C. Incorrect – OP-37 requires starting the SBGT fan before opening the Torus vent valves. Plausible because the Torus vent valves can be opened prior to starting the SBGT fan.
D. Incorrect – OP-37 requires starting the SBGT fan before opening the Torus vent valves. Plausible because the Torus vent valves can be opened prior to starting the SBGT fan. The SBGT fan is started by taking the TRAIN A(B) INLET 01-125MOV-14A(B) control switch to OPEN. Plausible because this is a similar control switch to the correct answer and this valve is also verified open when the fan starts.

Technical Reference(s): OP-37, OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B 1.05.a.1

Question Source: New

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 #	2
	Group #	1
	K/A #	223002 A4.05
	Importance Rating	2.5

PCIS/Nuclear Steam Supply Shutoff

Ability to manually operate and/or monitor in the control room: SPDS/ERIS/CRIDS/GDS: Plant-Specific

Proposed Question: #20

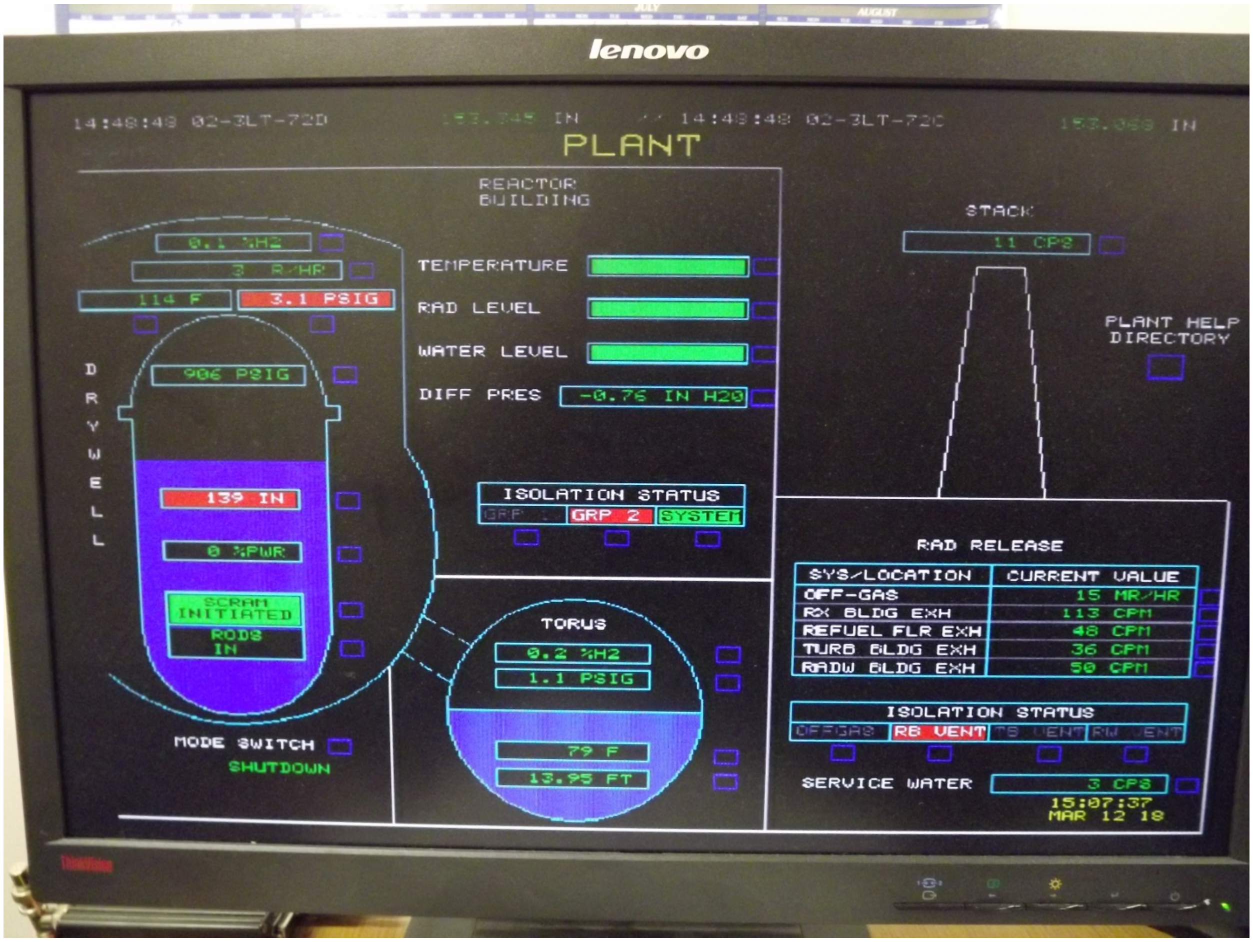
The plant has experienced a coolant leak in the Drywell with the following:

- AOP-39, Loss of Coolant, is being executed.
- The Reactor has been scrammed per AOP-1, Reactor Scram.
- Reactor water level got to a low of 75" following the scram before recovering.
- SPDS displays on the next four (4) pages show the status of various plant parameters.

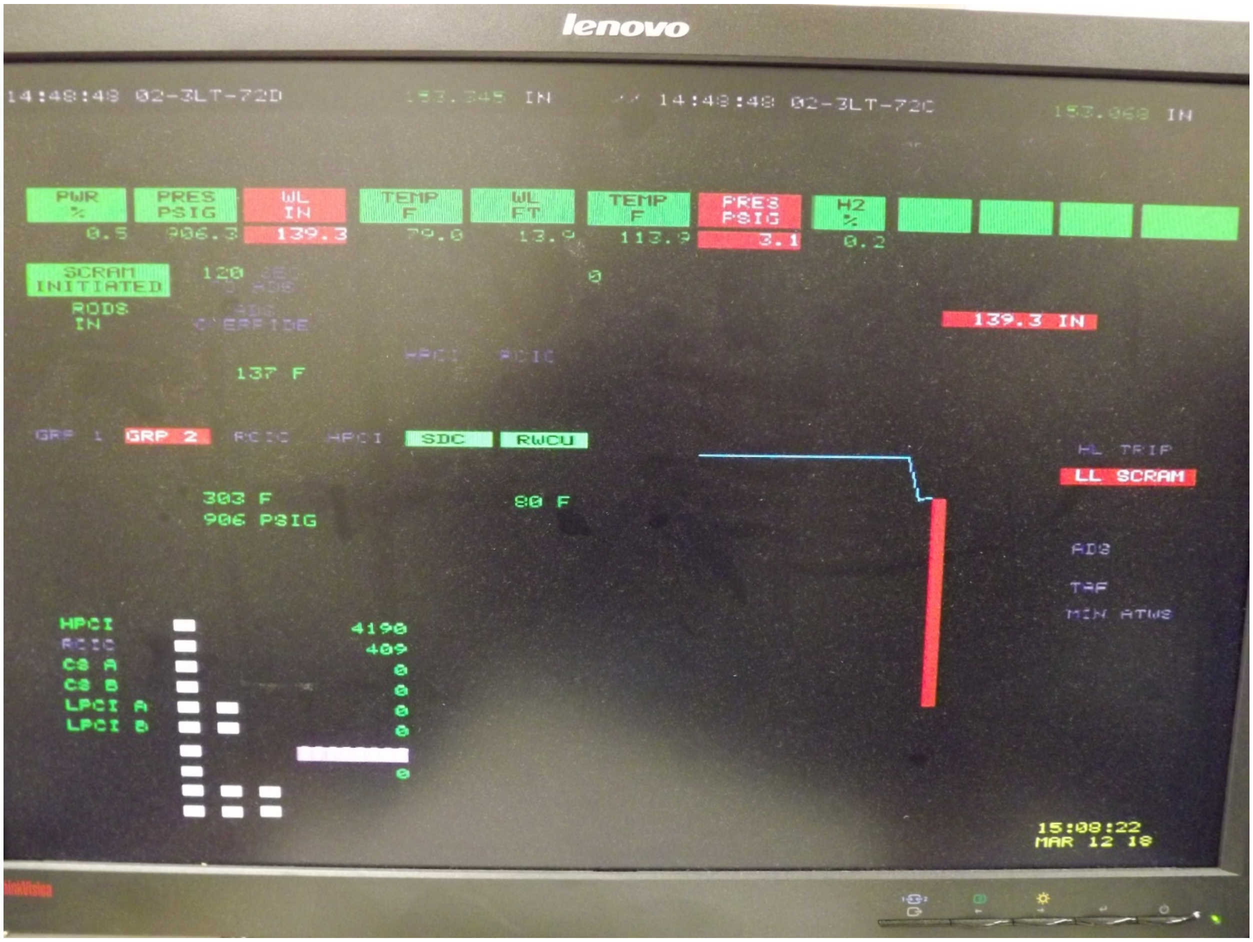
<u>Note:</u> The 3rd and 4th pages are duplicates of the displays shown on first two pages, but given in a different format to provide additional clarity.

Which one of the following describes the status of PCIS Groups I and II based on these SPDS displays?

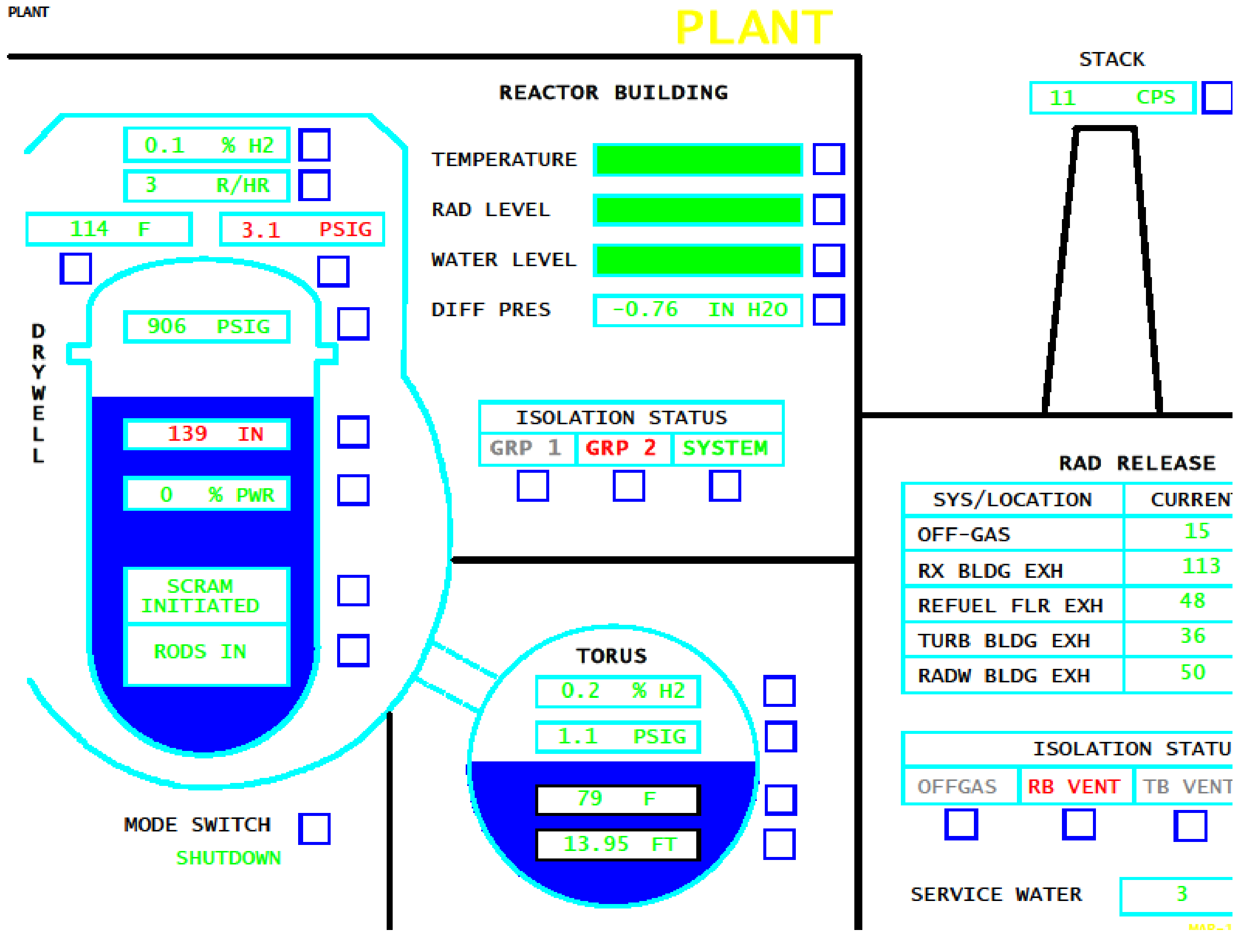
	Group I	Group II
A.	NO isolation is required.	An isolation has successfully occurred as required.
B.	NO isolation is required.	An isolation has failed to occur as required.
C.	An isolation has successfully occurred as required.	An isolation has successfully occurred as required.
D.	An isolation has successfully occurred as required.	An isolation has failed to occur as required.



14:48:48 02-3LT-72D
IN
14:48:48 02-3LT-72C
IN
PLANT
REACTOR BUILDING
STACK
11 CPS
0.1 %H2
3 R/HR
114 F
3.1 PSIG
TEMPERATURE
RAD LEVEL
WATER LEVEL
DIFF PRES
-0.76 IN H20
PLANT HELP DIRECTORY
DRYWELL
906 PSIG
139 IN
0 %PWR
SCRAM INITIATED
RODS IN
ISOLATION STATUS
GRP 2
SYSTEM
RAD RELEASE
SYS/LOCATION
CURRENT VALUE
OFF-GAS
15 MR/HR
RX BLDG EXH
113 CPM
REFUEL FLR EXH
48 CPM
TURB BLDG EXH
36 CPM
RADW BLDG EXH
50 CPM
TORUS
0.2 %H2
1.1 PSIG
79 F
13.95 FT
ISOLATION STATUS
RB VENT
MODE SWITCH
SHUTDOWN
SERVICE WATER
3 CPS
15:07:37
MAR 12 18


lenovo
14:48:48 02-3LT-72D
14:48:48 02-3LT-72C
PWR %
0.5
PRES PSIG
906.3
WL IN
139.3
TEMP F
79.0
WL FT
13.9
TEMP F
113.9
PRES PSIG
3.1
H2 %
0.2
SCRAM INITIATED
RODS IN
120
0
139.3 IN
137 F
GRP 2
SDC
RWCU
HL TRIP
LL SCRAM
303 F
906 PSIG
80 F
ADS
TAF
MIN ATWS
HPCI
RCIC
CS A
CS B
LPCI A
LPCI B
4190
409
15:08:22
MAR 12 18

PLANT


PLANT
REACTOR BUILDING
STACK
11 CPS
0.1 % H2
3 R/HR
114 F
3.1 PSIG
906 PSIG
DRYWELL
139 IN
0 % PWR
SCRAM INITIATED
RODS IN
TEMPERATURE
RAD LEVEL
WATER LEVEL
DIFF PRES
-0.76 IN H2O
ISOLATION STATUS
GRP 1
GRP 2
SYSTEM
RAD RELEASE
SYS/LOCATION
OFF-GAS 15
RX BLDG EXH 113
REFUEL FLR EXH 48
TURB BLDG EXH 36
RADW BLDG EXH 50
TORUS
0.2 % H2
1.1 PSIG
79 F
13.95 FT
ISOLATION STATU
OFFGAS
RB VENT
TB VENT
MODE SWITCH
SHUTDOWN
SERVICE WATER
3

RPV WATER LEVEL CONTROL (RCL)

RX	RPV		TORUS		DRYWELL		PC	RX BLDG			OFFSITE
PWR %	PRESS PSIG	WL IN	TEMP F	WL FT	TEMP F	PRESS PSIG	H2 %				
0.5	906.3	139.3	79.0	13.9	113.9	3.1	0.2	TEMP	RAD	WL	RAD

SCRAM INITIATED	120 SEC TO ADS
RODS IN	ADS OVERRIDE

VERTICAL RUN TEMP	137 F

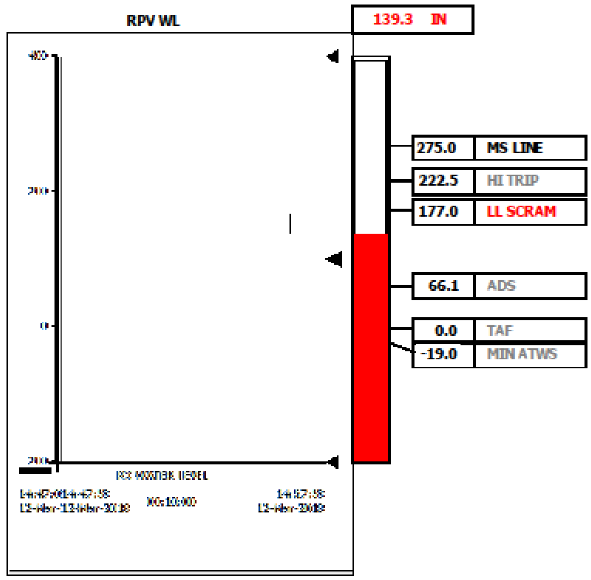
SRVs OPEN	0

SUCTN XFR TO TORUS	
HPCI	RCIC

ISOLATION STATUS					
GRP 1	GRP 2	RCIC	HPCI	SDC	RWCU

MARGIN TO RPVSAT			MARGIN TO BIIT		
RUN TEMP	303	F	TRS TEMP	80	F
RPV PRES	906	PSIG			

INJECTION SYSTEM STATUS		
SYSTEM	PUMP STATUS	FLOW (GPM)
HPCI		4190
RCIC		409
CS A		0
CS B		0
LPCI A		0
LPCI B		0
CRD		0
FW		0
C-BSTR		



Proposed Answer: B

Explanation: Since Reactor water level went <177” and Drywell pressure went >2.7 psig during the transient, PCIS Group II was required to have isolated. The SPDS displays show Group II did not isolate (red back lighting on GRP 2). Since Reactor water level stayed above 59.5” and the Reactor Mode Switch was immediately placed in SHUTDOWN on the scram, no PCIS Group I isolation was required. The SPD display shows this since the GRP 1 indication is light gray (not red or green).

A. Incorrect – Group II should have isolated but did not. Plausible because Reactor water level stayed above 59.5”.
C. Incorrect – Group I responded properly (by not isolating). Plausible because Reactor water level went lower than normal for a full power scram and approached 59.5”. Also plausible because Reactor pressure went <825 psig, which would have caused a Group I isolation is the Mode Switch had been left in RUN. Group II should have isolated but did not. Plausible because Reactor water level stayed above 59.5”.
D. Incorrect – Group I responded properly (by not isolating). Plausible because Reactor water level went lower than normal for a full power scram and approached 59.5”. Also plausible because Reactor pressure went <825 psig, which would have caused a Group I isolation is the Mode Switch had been left in RUN.

Technical Reference(s): AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.07

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 #	239002 2.4.18
	Importance Rating	3.3

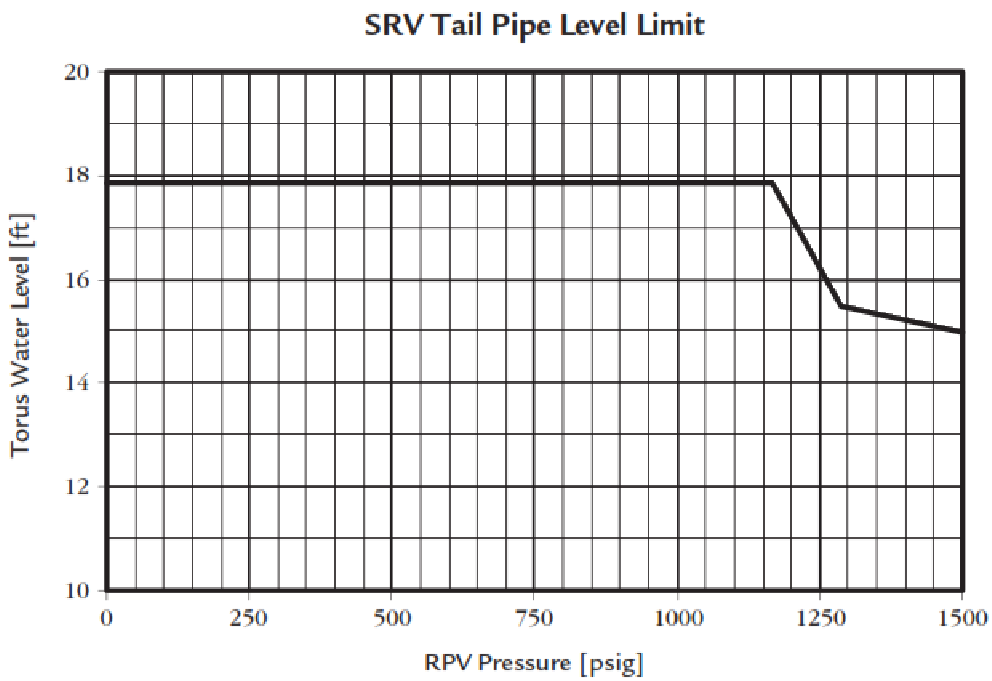
SRVs

Knowledge of the specific bases for EOPs.

Proposed Question: #21

The plant has experienced a transient with following:

- Multiple control rods failed to insert following a Reactor scram.
- Reactor pressure is 1200 psig.
- Torus water level is 17.7'.



Which one of the following describes the status of the SRV Tail Pipe Level Limit and the associated basis for the SRV Tail Pipe Level Limit at this Reactor pressure?

The SRV Tail Pipe Level Limit has ___(1)___. The basis for the SRV Tail Pipe Level Limit ***at this Reactor pressure*** is to prevent damage to the ___(2)___.

	(1)	(2)
A.	been exceeded	Containment in the event of a LOCA
B.	been exceeded	SRV tail pipes upon SRV opening
C.	NOT been exceeded	Containment in the event of a LOCA
D.	NOT been exceeded	SRV tail pipes upon SRV opening

Proposed Answer: B

Explanation: The SRV Tail Pipe Level Limit (SRVTPLL) is limited to 17.85 feet between 0 and 1165 psig. The limiting component in this range of Reactor pressures is the containment due to the reduced air space to cushion the pressure surge should a LOCA occur. The SRVTPLL lowers at Reactor pressures above 1165 psig. The combination of 1200 psig and Torus water level of 17.7’ exceeds the limit. The limiting component in this range of Reactor pressures is the SRV tailpipe due to high clearing loads upon SRV actuation.

A. Incorrect – The Containment is the limiting component of the SRVTPLL at Reactor pressures below 1165 psig.
C. Incorrect – The SRVTPLL has been exceeded. Plausible because the operating point is close to the limit. Also plausible because gray shading has been removed from graph. The Containment is the limiting component of the SRVTPLL at Reactor pressures below 1165 psig.
D. Incorrect – The SRVTPLL has been exceeded. The SRVTPLL has been exceeded. Plausible because the operating point is close to the limit. Also plausible because gray shading has been removed from graph.

Technical Reference(s): EOP-11, MIT-301.11B

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11B 1.01

Question Source: Bank – 9/12 NRC #26

Question History: 9/12 NRC #26

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 #	259002 2.1.7
	Importance Rating	4.4

Reactor Water Level Control

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

Proposed Question: #22

A plant cooldown is in progress with the following:

- Reactor pressure is 200 psig and slowly lowering.
- Narrow range Reactor water level instruments are indicating 200 inches and stable.
- Wide range Reactor water level instruments are indicating 210 inches and stable.
- The deviation between narrow and wide range Reactor water level instruments has been getting larger as the cooldown has progressed.

Which one of the following explains which instruments are more accurate under these plant conditions and the reason why?

The more accurate instruments under these plant conditions are the…

A. wide range instruments due to density compensation.

B. narrow range instruments due to density compensation.

C. wide range instruments due to differences in calibration conditions.

D. narrow range instruments due to differences in calibration conditions.

Proposed Answer: B

Explanation: The wide and narrow range Reactor water level instruments are calibrated at the same conditions (1000 psig, 546ºF, 135ºF DW temperature). However, only the narrow range Reactor water level instruments are density compensated. This density compensation makes the narrow range instruments more accurate as Reactor water temperature lowers. The wide range instruments tend to indicate higher than actual Reactor water level as Reactor water temperature lowers.

A. Incorrect – The narrow range instruments are more accurate under these conditions due to density compensation. Plausible if the candidate mixes up which instruments are density compensated.
C. Incorrect – The wide and narrow range Reactor water level instruments are calibrated at the same conditions. Plausible because not all instruments are calibrated at the same conditions and this does affect indications. The narrow range instruments are more accurate under these conditions due to density compensation. Plausible if the candidate mixes up which instruments are density compensated.
D. Incorrect – The wide and narrow range Reactor water level instruments are calibrated at the same conditions. Plausible because not all instruments are calibrated at the same conditions and this does affect indications.

Technical Reference(s): EOP-11, SDLP-02B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02B 1.10.g

Question Source: Bank – 3/14 NRC #37

Question History: 3/14 NRC #37

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K4.01
	Importance Rating	2.8

Instrument Air

Knowledge of INSTRUMENT AIR SYSTEM design feature(s) and or interlocks which provide for the following: Manual/automatic transfers of control

Proposed Question: #23

The plant is operating at 50% power during a startup with the following:

- An operator reports a major Instrument Air (IA) leak in the plant.
- IA header pressure is 106 psig and lowering.

Which one of the following describes the response of 39FCV-110, Service Air Header Auto Isolation Valve?

39FCV-110 will begin to close when header pressure reaches __(1)__.

If header pressure is later recovered to the normal band, 39FCV-110 __(2)__.

	(1)	(2)
A.	95 psig	automatically re-opens
B.	95 psig	requires manual action to re-open
C.	85 psig	automatically re-opens
D.	85 psig	requires manual action to re-open

Proposed Answer: B

Explanation: Lowering IA header pressure results in the following automatic actions:

- 107 psig - first standby IA compressor starts
- 104 psig - second standby IA compressor starts
- 95 psig - 39FCV-110 closes
- 85 psig - 39AOV-111 closes

Therefore, 39FCV-110 closes when header pressure reaches 95 psig. Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed.

A. Incorrect – Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed. Plausible because the valve responds automatically on lowering pressure.
C. Incorrect – 39FCV-110 closes when header pressure reaches 95 psig. Plausible because 39AOV-111 (Breathing Air) closes at 85 psig. Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed. Plausible because the valve responds automatically on lowering pressure.
D. Incorrect – 39FCV-110 closes when header pressure reaches 95 psig. Plausible because 39AOV-111 (Breathing Air) closes at 85 psig.

Technical Reference(s): AOP-12, OP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.c.1

Question Source: Bank - 9/14 NRC #49

Question History: 9/14 NRC #49

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 #	262001 K6.02
	Importance Rating	3.6

AC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: Off-site power

Proposed Question: #24

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

- MOD-10017, NORTH-SOUTH 115KV BUS DISC SW, is open due to a request from National Grid.
- The rest of the electrical distribution system is in a normal alignment for 100% power.
- Then, Line 3, FITZPATRICK – LIGHTHOUSE HILL, de-energizes.

Which one of the following describes the impact of the Line 3 loss on the electrical distribution system?

A. Bus 10300 de-energizes.

B. Bus 10400 de-energizes.

C. Bus 10300 remains energized, but loses its alternate power source.

D. Bus 10400 remains energized, but loses its alternate power source.

Proposed Answer: D

Explanation: With MOD-10017 open, the primary windings of T2 receive power from Line 3 alone. Therefore on a loss of Line 3, T2 will de-energize. This will cause a loss of the reserve supply to Buses 10200 and 10400. However, at 100% power, these buses are powered from house service transformer T4. Therefore Bus 10400 will remain energized.

A. Incorrect – With MOD-10017 open, Bus 10300 is affected by off-site power Line 4, not Line 3. Plausible that Line 3 would affect Bus 10300 vice Bus 10400.
B. Incorrect – A normal alignment at 100% power has Bus 10400 powered from house service transformer T4, not T2. Plausible because under other conditions T2 would be the normal supply to Bus 10400 and it would de-energize.
C. Incorrect – With MOD-10017 open, Bus 10300 is affected by off-site power Line 4, not Line 3. Plausible that Line 3 would affect Bus 10300 vice Bus 10400.

Technical Reference(s): SDLP-71O Figure 1, OP-44, OP-46A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D 1.09.a

Question Source: Bank – 3/14 NRC #1

Question History: 3/14 NRC #1

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 #	263000 A1.01
	Importance Rating	2.5

DC Electrical Distribution

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

Proposed Question: #25

The plant was operating at 10% power during a startup with the following:

- Lines 3 and 4 de-energized.
- The Reactor scrammed.
- All Emergency Diesel Generators failed to start.

Which one of the following describes the requirement to perform DC load shedding and the reason for the load shedding, in accordance with AOP-49, Station Blackout?

The first load shedding attachment must be completed within a maximum of…

A. 30 minutes to ensure meeting the coping time requirement of 4 hours.

B. 30 minutes to ensure meeting the coping time requirement of 8 hours.

C. 1 hour to ensure meeting the coping time requirement of 4 hours.

D. 1 hour to ensure meeting the coping time requirement of 8 hours.

Proposed Answer: A

Explanation: AOP-49 requires some of the load shedding actions in attachment 3 to be completed within 30 minutes of the start of the station blackout. This load shedding is required to meet the required station coping time of 4 hours.

<u>Note:</u> The question meets the K/A because battery discharge rate is monitored/controlled during a station blackout through proper timing of DC load shedding actions. The basis for load shed timing also allows predicting the discharge rate.

B. Incorrect – The coping time requirement is 4 hours, not 8. Plausible because surveillance testing tests battery capacity based on the 8 hour rating.
C. Incorrect – The load shedding time requirement is 30 minutes, not 1 hour. Plausible because some of the load shedding actions in AOP-49 attachment 3 have a 1 hour time requirement.
D. Incorrect – The load shedding time requirement is 30 minutes, not 1 hour. Plausible because some of the load shedding actions in AOP-49 attachment 3 have a 1 hour time requirement. The coping time requirement is 4 hours, not 8. Plausible because surveillance testing tests battery capacity based on the 8 hour rating.

Technical Reference(s): AOP-49, UFSAR section 8.11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.14.f

Question Source: Bank - 3/14 NRC #46

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 #	2
	Group #	1
	K/A #	215003 K2.01
	Importance Rating	2.5

IRM

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

Proposed Question: #26

Which one of the following describes the power supply to the IRM detectors?

A. 24VDC Battery Bus

B. 125VDC Battery Bus

C. 120VAC UPS Bus

D. 120VAC Instrument Bus

Proposed Answer: A

Explanation: IRM detectors are power from the 24 VDC Battery Buses.

B. Incorrect – IRM detectors are power from the 24 VDC Battery Buses. Plausible because this is another possible DC source.
C. Incorrect – IRM detectors are power from the 24 VDC Battery Buses. Plausible because this is the power supply to the IRM recorders and detector drive control circuits.
D. Incorrect – IRM detectors are power from the 24 VDC Battery Buses. Plausible because this is the power supply to the detector drives.

Technical Reference(s): OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B

Question Source: New

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 #	201001 K1.06
	Importance Rating	2.8

Control Rod Drive Hydraulic System

Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD DRIVE HYDRAULIC SYSTEM and the following: Component cooling water systems: Plant-Specific

Proposed Question: #27

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

- A complete loss of RBCLC pumps occurs.
- Annunciator 09-6-2-32, RBC HDR PRESS LO-LO, alarms.
- The Reactor is manually scrammed.

Which one of the following describes the status of cooling water to the CRD pumps?

Backup cooling water to the CRD pumps…

A. is automatically supplied from Normal Service Water.

B. is automatically supplied from Emergency Service Water.

C. must be manually aligned from Normal Service Water.

D. must be manually aligned from Emergency Service Water.

Proposed Answer: D

Explanation: RBCLC supplies the normal source of cooling water to the CRD pumps. On loss of all RBCLC pumps, backup cooling water must be manually aligned to the CRD pumps from the Emergency Service Water system.

A. Incorrect – Backup cooling water must be manually aligned to the CRD pumps from the Emergency Service Water system. Plausible because Normal Service Water supplies many loads in the Reactor Building and with a complete loss of normal cooling water pressure, ESW does automatically start and supply certain loads.
B. Incorrect – Backup cooling water must be manually aligned to the CRD pumps from the Emergency Service Water system. Plausible because with a complete loss of normal cooling water pressure, ESW does automatically start and supply certain loads.
C. Incorrect – Backup cooling water must be manually aligned to the CRD pumps from the Emergency Service Water system. Plausible because Normal Service Water supplies many loads in the Reactor Building.

Technical Reference(s): AOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.10.d

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 #	202002 K6.04
	Importance Rating	3.5

Recirculation Flow Control

Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: Feedwater flow inputs: BWR-3,4,5,6.

Proposed Question: #28

The plant is operating at 100% power when the Feedwater flow signal to Recirc pump A drifts low to 0.

Which one of the following describes the resulting operation of Recirc pump A?

Recirc pump A…

A. trips.

B. runs back to 30% speed.

C. runs back to 44% speed.

D. remains at the initial speed with the scoop tube locked.

Proposed Answer: B

Explanation: Feedwater flow less than 20% initiates Speed Limiter #1. This runs back Recirc pump A speed to 30%.

A. Incorrect – Recirc pump A remains running. Plausible because this condition does initiate automatic response and other conditions do trip the Recirc pump.
C. Incorrect – Recirc pump A runs back, but to 30% speed, not 44%. Plausible because there is a 44% speed run back and it is initiated by other adverse Feedwater system conditions.
D. Incorrect – Recirc pump A runs back. Plausible because other signal failures do result in automatic scoop tube lock.

Technical Reference(s): SDLP-02I

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.10.c

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 #	290003 K3.03
	Importance Rating	2.9

Control Room HVAC

Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: Control room temperature

Proposed Question: #29

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

- 70AHU-3A, Control Room Ventilation air handling unit (AHU) A, is in service.
- 70AHU-3B, Control Room Ventilation AHU B, is in standby.

Then, 70AHU-3A trips on motor thermal overload.

Which one of the following describes the response of Control Room air temperature and 70AHU-3B?

Control Room air temperature…

A. remains stable due to immediate auto-start of 70AHU-3B.

B. rises. 70AHU-3B will auto-start when Control Room air temperature reaches 90^{o}F.

C. rises. 70AHU-3B will auto-start when Control Room air temperature reaches 98^{o}F.

D. rises. 70AHU-3B remains in standby unless manually started, regardless of Control Room air temperature.

Proposed Answer: A

Explanation: The standby Control Room Ventilation AHU immediately auto-starts upon trip or low air flow of the companion AHU. With Control Room Ventilation AHU immediately starting, Control Room air temperature remains stable.

B. Incorrect – Control Room air temperature remains stable due to immediate auto-start of 70AHU-3B. Plausible because there is an auto-start on high temperature and 90ºF is used in TS 3.7.4 for entry into a shutdown Required Action.
C. Incorrect – Control Room air temperature remains stable due to immediate auto-start of 70AHU-3B. Plausible because there is an auto-start on high temperature of 98ºF.
D. Incorrect – Control Room air temperature remains stable due to immediate auto-start of 70AHU-3B. Plausible because this would be correct if there was no auto-start feature.

Technical Reference(s): HV-5A-04

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.05.b.6

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 #	241000 K4.03
	Importance Rating	3.0

Reactor/Turbine Pressure Regulator

Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine speed control

Proposed Question: #30

A plant startup is in progress with the following:

- Main Turbine startup is in progress.
- Main Turbine speed is 1700 rpm and rising.

Then, both the primary and backup speed signals to the Main Turbine speed control unit are lost.

Which one of the following describes the Main Turbine response?

The Main Turbine…

A. immediately trips.

B. does NOT trip, but slowly decelerates due to Turbine Control Valve closure.

C. continues to accelerate until it trips on overspeed at approximately 1980 rpm.

D. continues to accelerate until it trips on backup overspeed at approximately 2007 rpm.

Proposed Answer: A

Explanation: The Main Turbine speed control unit gets an input from two of the three speed devices (primary and backup). Given loss of both signals, the Master Trip Relay energizes to trip the Main Turbine immediately.

B. Incorrect – The Main Turbine immediately trips. Plausible that no immediate trip would occur and loss of speed signal would result in speed control unit closing down on TCVs as a fail-safe action.
C. Incorrect – The Main Turbine immediately trips. Plausible that no immediate trip would occur, speed would rise due to low indicated speed, and overspeed trip device would still operate at 1980 rpm.
D. Incorrect – The Main Turbine immediately trips. Plausible that no immediate trip would occur, speed would rise due to low indicated speed, and overspeed trip device would not work due to loss of speed signals, and backup overspeed would still operate at 2007 rpm based on the third speed device.

Technical Reference(s): ARP 09-5-2-18

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94C 1.14.c

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 #	234000 K5.01
	Importance Rating	2.9

Fuel Handling Equipment

Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT: Crane/hoist operation

Proposed Question: #31

A refueling outage is in progress with the following:

- The Reactor Mode Switch is in REFUEL.
- A fuel bundle is latched on the Refuel Bridge main hoist in center of the Spent Fuel Pool.
- The Refuel Bridge main hoist is in the normal-up position.
- One control rod is withdrawn to position 04.

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

This fuel movement would first be interrupted by an interlock…

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

B. after the Refuel Bridge begins moving towards the core, but before it gets above 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 fuel movement would first be interrupted by the interlock that stops Refuel Bridge motion when a control rod is withdrawn, the main hoist is loaded with a bundle, and the Refuel Bridge is nearing a location over the core.

A. Incorrect – Motion towards the core is allowed until the Refuel Bridge nears a position over the core. Plausible because motion towards the core will eventually be interrupted, just not until the Refuel Bridge nears the core.
C. Incorrect – The fuel movement would be interrupted earlier than this (before getting over the core). Plausible because if the earlier interlock failed, another interlock would prevent lowering the main hoist.
D. Incorrect – The fuel movement would be interrupted earlier than this (before getting over the core). Plausible because if the earlier interlock failed, another interlock would prevent lowering the main hoist. Also plausible that this interlock would not take effect until the fuel bundle was near the top guide because this is when reactivity would be inserted.

Technical Reference(s): OP-66A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08B 1.05.b.3

Question Source: Bank – SSES LOC28 NRC #31

Question History: SSES LOC28 NRC #31

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 #	290001 K6.01
	Importance Rating	3.5

Secondary Containment

Knowledge of the effect that a loss or malfunction of the following will have on the SECONDARY CONTAINMENT: Reactor building ventilation: Plant-Specific

Proposed Question: #32

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

- Secondary containment differential pressure is -0.25" WG.
- Refuel Floor Exhaust fan A is running.
- Refuel Floor Exhaust fan B is in standby.

Then, the discharge damper on Refuel Floor Exhaust fan A fails closed.

Which one of the following describes the status of Secondary Containment differential pressure five (5) minutes later?

Secondary containment differential pressure…

A. remains negative with Standby Gas Treatment running due to auto-start.

B. remains negative with Refuel Floor Exhaust fan B running due to auto-start.

C. remains negative with NO Refuel Floor Exhaust fan or Standby Gas Treatment fan running.

D. becomes approximately 0" WG and remains there until operator action is taken to start Refuel Floor Exhaust fan B or a Standby Gas Treatment fan.

Proposed Answer: B

Explanation: When the discharge dampers leaves the full open position, the associated Refuel Floor Exhaust fan trips. Trip of this fan causes the other Refuel Floor Exhaust fan to automatically start. This ensures Secondary Containment differential pressure is maintained negative and prevents an auto-start of Standby Gas Treatment

A. Incorrect – Standby Gas Treatment does not auto-start on this failure. Plausible because Standby Gas Treatment is a backup to Refuel Floor Exhaust fans and does auto-start on other signals.
C. Incorrect – The other Refuel Floor Exhaust fan auto-starts. Plausible that no auto-start would occur on this failure and that the running Below Refuel Floor Exhaust fan would be able to maintain negative building differential pressure.
D. Incorrect – The other Refuel Floor Exhaust fan auto-starts and negative building differential pressure is maintained. Plausible that no auto-start would occur on this failure and that the still running Supply fan would cause building differential pressure to go to approximately 0" WG.

Technical Reference(s): SDLP-66A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-66A 1.05.c.4

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 #	2
	K/A #	259001 A1.06
	Importance Rating	2.7

Reactor Feedwater

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: Feedwater heater level

Proposed Question: #33

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

- The Feedwater Heater (FWH) 3A level controller, 35LIC-113A1, has been observed operating erratically.
- An operator takes manual control of 35LIC-113A1 and throttles the associated drain valves further closed.

Which one of the following describes the ***initial*** effect of this operator action?

Level in FWH…

A. 2A rises.

B. 2A lowers.

C. 4A rises.

D. 4A lowers.

Proposed Answer: B

Explanation: FWH 3A drains to FWH 2A. Closing the drain valve slows the flow from FWH 3A to FWH 2A, which raises FWH 3A level and initially lowers FWH 2A level.

A. Incorrect – FWH 2A level lowers, not rises. Plausible because FWH 3A level rises. Also plausible if candidate reverses the flow path of extraction steam drainage.
C. Incorrect – FWH 4A level is initially unchanged. Plausible if the direction of Feedwater Heater drainage is reversed. Also plausible because FWH 4A level would eventually change if FWH 3A level rose high enough.
D. Incorrect – FWH 4A level is initially unchanged. Plausible if the direction of Feedwater Heater drainage is reversed. Also plausible because FWH 4A level would eventually change if FWH 3A level rose high enough.

Technical Reference(s): FM-35A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-35 1.05.a.1

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 #	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

The plant 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 3.8 gpm in the last hour.
- Drywell pressure is 2.03 psig and slowly rising.
- Drywell temperature is 140°F and slowly rising.

With one of the following describes:

(1) the current ability of the Drywell Floor Drain sumps to pump to Radwaste, and

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

	(1) Currently, Drywell Floor Drain sumps…	(2) Leakage Exceeds Any Technical Specification Limit?
A.	can pump to Radwaste.	No
B.	can pump to Radwaste.	Yes
C.	CANNOT pump to Radwaste.	No
D.	CANNOT pump to Radwaste.	Yes

Proposed Answer: B

Explanation: Drywell Floor Drain sumps can currently pump to Radwaste because Drywell pressure remain below 2.7 psig. Plausible because input has exceeded a limit. Also plausible because this would be correct if Drywell pressure were higher. Technical Specification 3.4.4 requires ≤ a 2 gpm rising is unidentified leakage in a 24 hour period. Since unidentified leakage has risen by 3.5 gpm in a single hour, the Technical Specification limit has been exceeded.

A. Incorrect – A Technical Specification limit is exceeded. Plausible because total unidentified leakage remains below the Technical Specification limit of 5 gpm.
C. Incorrect – Drywell Floor Drain sumps can currently pump to Radwaste. Plausible because input has exceeded a limit. Also plausible because this would be correct if Drywell pressure were higher. A Technical Specification limit is exceeded. Plausible because total unidentified leakage remains below the Technical Specification limit of 5 gpm.
D. Incorrect – Drywell Floor Drain sumps can currently pump to Radwaste. Plausible because input has exceeded a limit. Also plausible because this would be correct if Drywell pressure were higher.

Technical Reference(s): AOP-15, OP-50, Technical Specification

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-20B 1.05.a.1.a

Question Source: Bank – SSES LOC27 NRC #34

Question History: SSES LOC27 NRC #34

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 #	215002 A3.03
	Importance Rating	3.1

RBM

Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Alarm and indicating lights: BWR-3,4,5

Proposed Question: #35

The plant is operating at 50% power with the following:

- A hardware problem has occurred on Rod Block Monitor (RBM) A.
- Annunciator 09-5-2-45, RBM UPSCALE OR INOP, alarms.
- The INOP white light is illuminated for RBM A on Panel 09-14.

Which one of the following describes the status of RBM A and control rod blocks?

RBM A is…

A. auto-bypassed. NO control rod blocks are enforced.

B. NOT auto-bypassed. NO control rod blocks are enforced.

C. NOT auto-bypassed. A control rod withdraw block is enforced, only.

D. NOT auto-bypassed. Control rod withdraw and insert blocks are enforced.

Proposed Answer: C

Explanation: The RBM is in service since power is above 30% (NOT auto-bypassed). The INOP trip has been received due to the hardware failure. This trip initiates a rod withdraw block, but not an insert block.

A. Incorrect – RBM A is not auto-bypassed. Plausible because this would be correct if power were <30%.
B. Incorrect – A withdraw block is received. Plausible because this would be correct if the hardware failure was not causing the INOP trip.
D. Incorrect – No insert block is received. Plausible because this would be correct for a RWM INOP trip during a startup.

Technical Reference(s): OP-16, ARP 09-5-2-45

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.05.c.4.e

Question Source: Bank – SSES LOC26R NRC #30

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 #	2
	K/A #	256000 A4.01
	Importance Rating	3.3

Reactor Condensate

Ability to manually operate and/or monitor in the control room: Hotwell condensate / condensate booster pumps

Proposed Question: #36

The plant is operating at 50% power with the following:

- Condensate pumps B and C are running.
- Condensate Booster pumps B and C are running.
- Condensate Booster pump A is tagged out of service for repair.

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

- The Reactor scrams.
- The 10400 Bus de-energizes due to a sustained electrical fault.
- Reactor water level is 40” and slowly lowering.
- Reactor pressure is 800 psig and slowly lowering.

Which one of the following describes the ability to inject to the Reactor with Condensate/Feedwater?

Injection to the Reactor with Condensate/Feedwater…

A. is currently available and a Feedwater pump is running.

B. is currently available, but NO Feedwater pump is running.

C. is currently NOT available, but will first become available when Reactor pressure lowers to approximately 700 psig.

D. is currently NOT available, but will first become available when Reactor pressure lowers to approximately 255 psig.

Proposed Answer: D

Explanation: On the Reactor scram, the 10700 Bus de-energizes, which causes a loss of Condensate pump C and Condensate Booster pump C. The loss of the 10400 Bus results in the loss of Condensate pump B and Condensate Booster pump B. Additionally, the MSIVs are closed due to Reactor water level <59.5”, which results in the Feedwater pumps being unavailable due to lack of steam. With Condensate Booster pump A tagged out, this leaves only Condensate pump A available. This pump will first be able to inject to the Reactor when Reactor pressure lowers to approximately 255 psig.

A. Incorrect – No Feedwater pumps are available because the MSIVs are closed due to Reactor water level <59.5”. Plausible because this would be correct if a Condensate Booster pump was available and Reactor water level were >59.5”.
B. Incorrect – Injection is not currently possible because Reactor pressure is above the shut-off head of the one running Condensate pump. Plausible because this would be correct if a Condensate Booster pump was available and Reactor pressure was <700 psig.
C. Incorrect – Injection will first be possible at approximately 255 psig. Plausible because this would be correct if a Condensate Booster pump was available.

Technical Reference(s): OP-46A, OP-3, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.15

Question Source: Modified Bank - 3/12 NRC #63

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:

Level	RO/SRO
Tier #	2
Group #	2
K/A #	256000 K6.04
Importance Rating	2.8/2.8

Reactor Condensate: Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CONDENSATE SYSTEM: A.C. power

Proposed Question: 63

The Plant is operating at 65% with the following Condensate lineup:

- Condensate Pumps 'B' & 'C' are running
- Condensate Booster Pumps 'B' & 'C' running.
- Condensate Booster Pump 'A' Auxiliary Oil Pump is tagged out of service for repair.

The following event occurs:

- The Rx scrams.
- 4160VAC Bus 10400 de-energizes.

Regarding Condensate and Condensate Booster pump status, which of the following is correct?

A.	Only Condensate Pump 'A' auto starts
B.	No Condensate or Condensate Booster Pumps are running
C.	Both Condensate Pump 'A' & Condensate Booster Pump 'A' auto start
D.	Only Condensate Pump 'C' & Condensate Booster Pump 'C' remain running

Proposed Answer: A, Only Condensate Pump 'A' auto starts

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 2.4.4
	Importance Rating	4.5

Main and Reheat Steam

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

Proposed Question: #37

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

- Annunciator 09-5-1-16, PCIS SYS A STM TUN TEMP HI, alarms.
- Annunciator 09-5-1-17, PCIS SYS B STM TUN TEMP HI, is NOT yet in alarm.
- Operators in the field report that steam is coming from the Main Steam tunnel.

Which one of the following describes the current need to enter AOP-40, Main Steam Line Break, and EOP-5, Secondary Containment Control?

	AOP-40 Entry	EOP-5 Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: B

Explanation: ARP 09-5-1-16 requires entering AOP-40 based on occurrence of the alarm plus visual indication of a steam leak (operator report). EOP-5 entry is not required because the Main Steam tunnel opens to the Turbine Building, not the Reactor Building.

A. Incorrect – EOP-5 entry is not required because the Main Steam tunnel opens to the Turbine Building, not the Reactor Building. Plausible because some plants have Main Steam tunnels connected to the Reactor Building such that EOP-5 entry would be required.
C. Incorrect – AOP-40 entry is required. Plausible because only one annunciator is currently in alarm, such that MSIVs have not yet automatically closed, and indications of a severe Main Steam Line break are not given (high Main Steam flow, lowering Reactor pressure, etc.). EOP-5 entry is not required because the Main Steam tunnel opens to the Turbine Building, not the Reactor Building. Plausible because some plants have Main Steam tunnels connected to the Reactor Building such that EOP-5 entry would be required.
D. Incorrect – AOP-40 entry is required. Plausible because only one annunciator is currently in alarm, such that MSIVs have not yet automatically closed, and indications of a severe Main Steam Line break are not given (high Main Steam flow, lowering Reactor pressure, etc.).

Technical Reference(s): ARP 09-5-1-16, AOP-40, EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.01

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 #	2
	Group #	2
	K/A #	216000 K3.24
	Importance Rating	3.9

Nuclear Boiler Instrumentation

Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER INSTRUMENTATION will have on following: Vessel level monitoring

Proposed Question: #38

The plant was operating at 100% power when a significant event resulted in the following:

- Reactor pressure is 225 psig and slowly lowering.
- Drywell temperature is 415ºF and slowly rising.
- Narrow range Reactor water level instruments indicate 175" and stable.
- Wide range Reactor water level instruments indicate 32" and stable.
- Fuel zone Reactor water level indicators and recorder are indicating erratically.
- Refuel zone Reactor water level instrument indicates 210" and stable.
- EPIC is NOT operating properly.
- Portions of EOP-11, EOP & SAOG Graphs, are provided on the following page.

Which one of the following identifies which Reactor water level instrument(s), if any, is(are) indicating a valid reading?

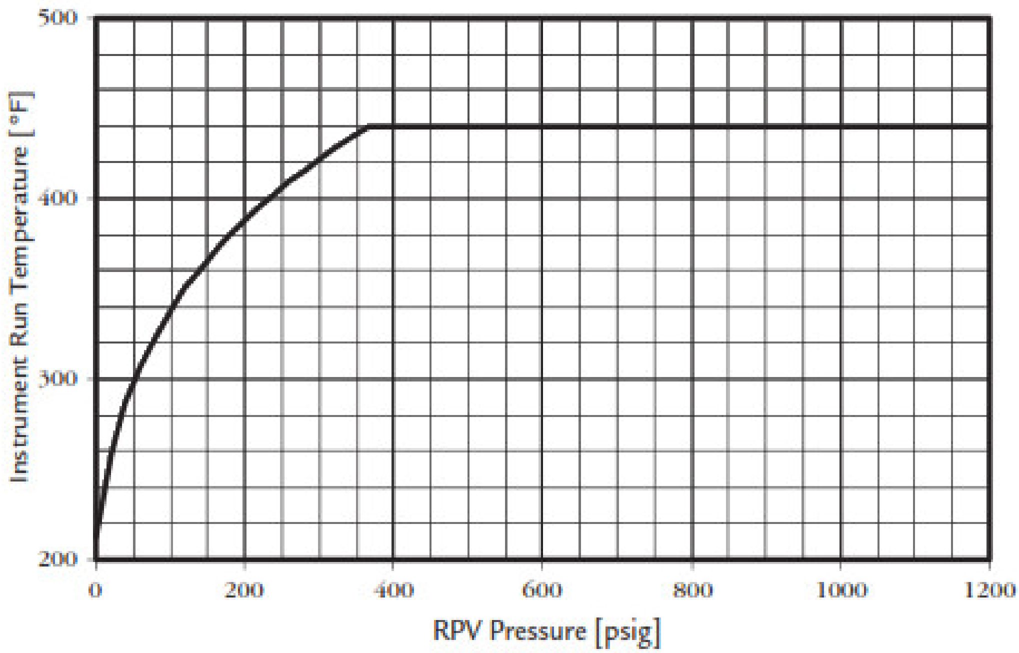
A. None

B. Wide Range

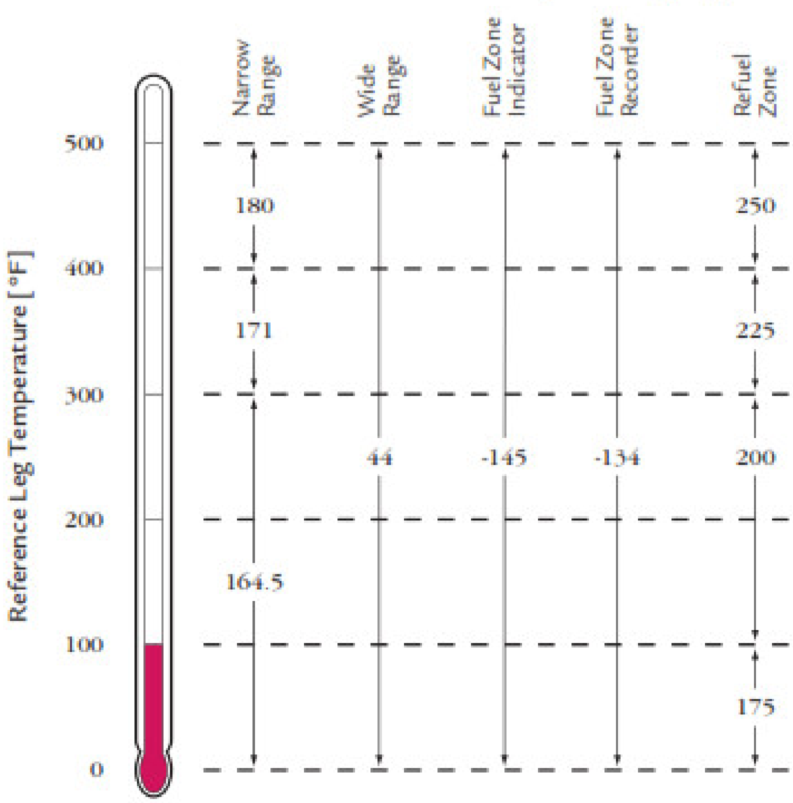
C. Refuel Zone

D. Narrow Range

RPV Saturation Temperature



Minimum Usable Indicating Levels [in.]



Proposed Answer: A

Explanation: The given combination of Reactor pressure and Drywell temperature are on the "bad" side of the RPV Saturation Temperature curve. This means Reactor water level instruments may be unreliable due to boiling, but are not invalid for use until the effects of boiling are observed. Since Wide range, Narrow range, and Refuel zone instruments are indicating stable level, no evidence of boiling is present. With Drywell temperature above 400°F, Wide range, Narrow range, and Refuel zone indicators are below their Minimum Usable Indicating Levels of 44", 180", and 250", respectively. This makes all of these indicators invalid for use.

B. Incorrect – Wide range is indicating below the minimum usable level of 44". Plausible because no indication of boiling in the instrument runs is present and Wide range is above the bottom of scale (14.5").
C. Incorrect – Refuel zone is indicating below the minimum usable level of 250". Plausible because no indication of boiling in the instrument runs is present and Refuel Zone is above the bottom of scale (164.5"). Also plausible because Refuel Zone would be valid if Drywell temperature were <400°F.
D. Incorrect – Narrow range is indicating below the minimum usable level of 180". Plausible because no indication of boiling in the instrument runs is present and Narrow range is above the bottom of scale (164.5"). Also plausible because Narrow range would be valid if Drywell temperature were <300°F.

Technical Reference(s): EOP-11 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.01

Question Source: Modified Bank - 3/12 NRC #68

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:

Level	RO/SRO
Tier #	3
Group #	G
K/A #	2.1.45
Importance Rating	4.3/4.3

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Question: 68

The Plant was operating at full power when a significant event occurred.

The following conditions are present:

- Drywell temperature: 405F
- RPV pressure: 225 psig
- Narrow Range level instruments: 165 inches and steady
- Wide Range level instruments: 32 inches and lowering
- Fuel Zone level indicator: Erratic, upscale and downscale
- Fuel Zone level recorder: De-energized
- Refuel Zone instrument: 300 inches and rising slowly
- EPIC: "locked-up", not updating

Which of the one of the following choices correctly lists which RPV level instrument(s) (if any) is\are indicating a valid reading?

Valid RPV level instrument(s)

A.	Narrow Range
B.	Refuel Zone
C.	Wide Range
D.	None

Proposed Answer: B, Refuel Zone

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK1.03
	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: Under-excitation

Proposed Question: #39

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

- Current time is 1100 on a Tuesday.
- Grid disturbances have resulted in changes to Main Generator operation.
- The Main Generator voltage regulator has been placed in Manual.
- Current Main Generator parameters are as follows:
- Main Generator real load is 873 MWe.
- Main Generator reactive load is 175 MVAR to bus (out).
- Main Generator terminal voltage is 24.2 kV.
- Main Generator hydrogen gas pressure is 60 psig.
- The applicable Main Generator limit curves from OP-11A, Main Generator, Transformers, and Isolated Bus Phase Cooling, are provided on the following two (2) pages.

Which one of the following describes the status of Main Generator parameters in accordance with OP-11A?

Main Generator…

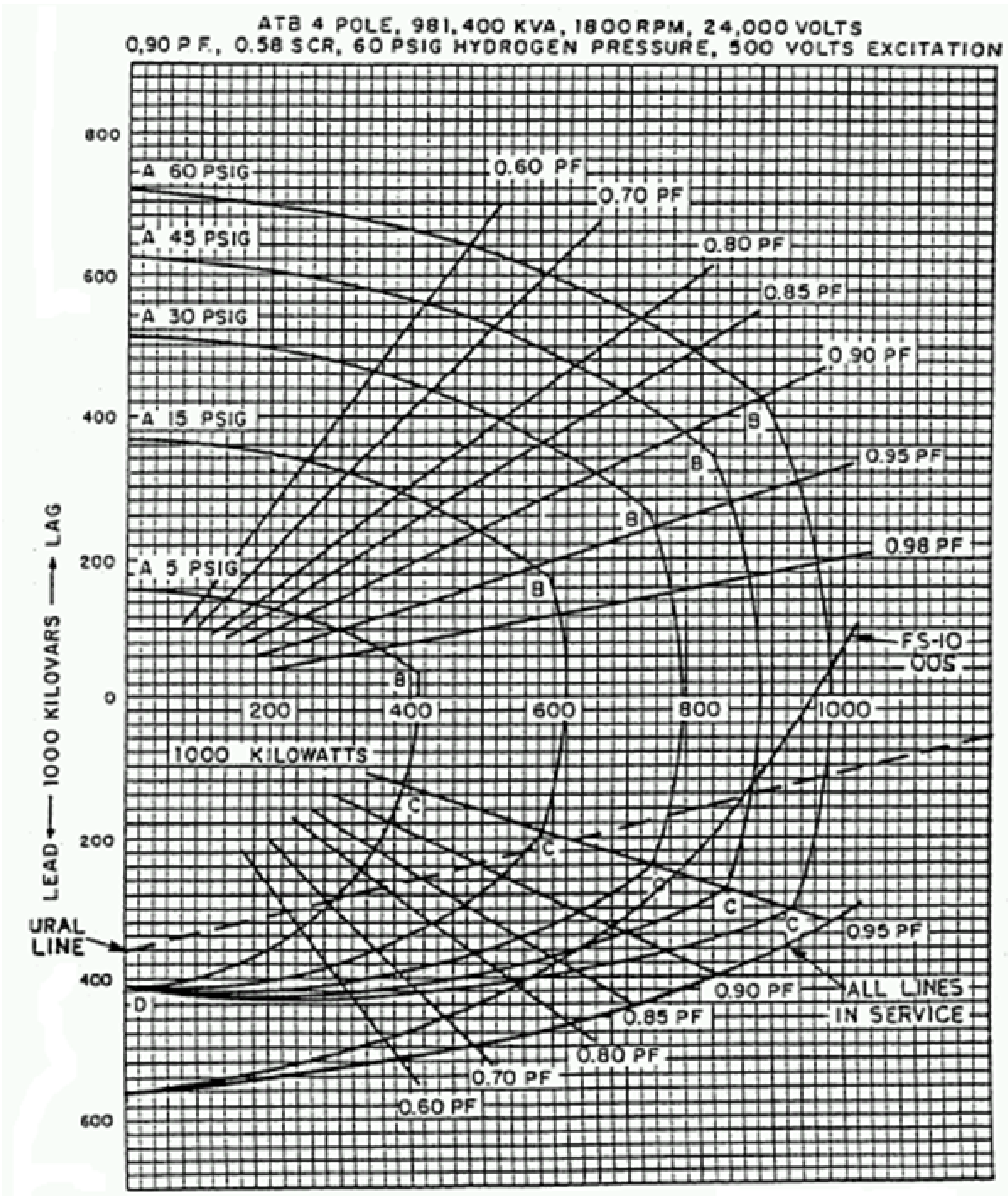
A. parameters are acceptable.

B. reactive load is too high for the given real load.

C. reactive load is too low for the given terminal voltage.

D. reactive load is too high for the given terminal voltage.

ESTIMATE CAPABILITY CURVES

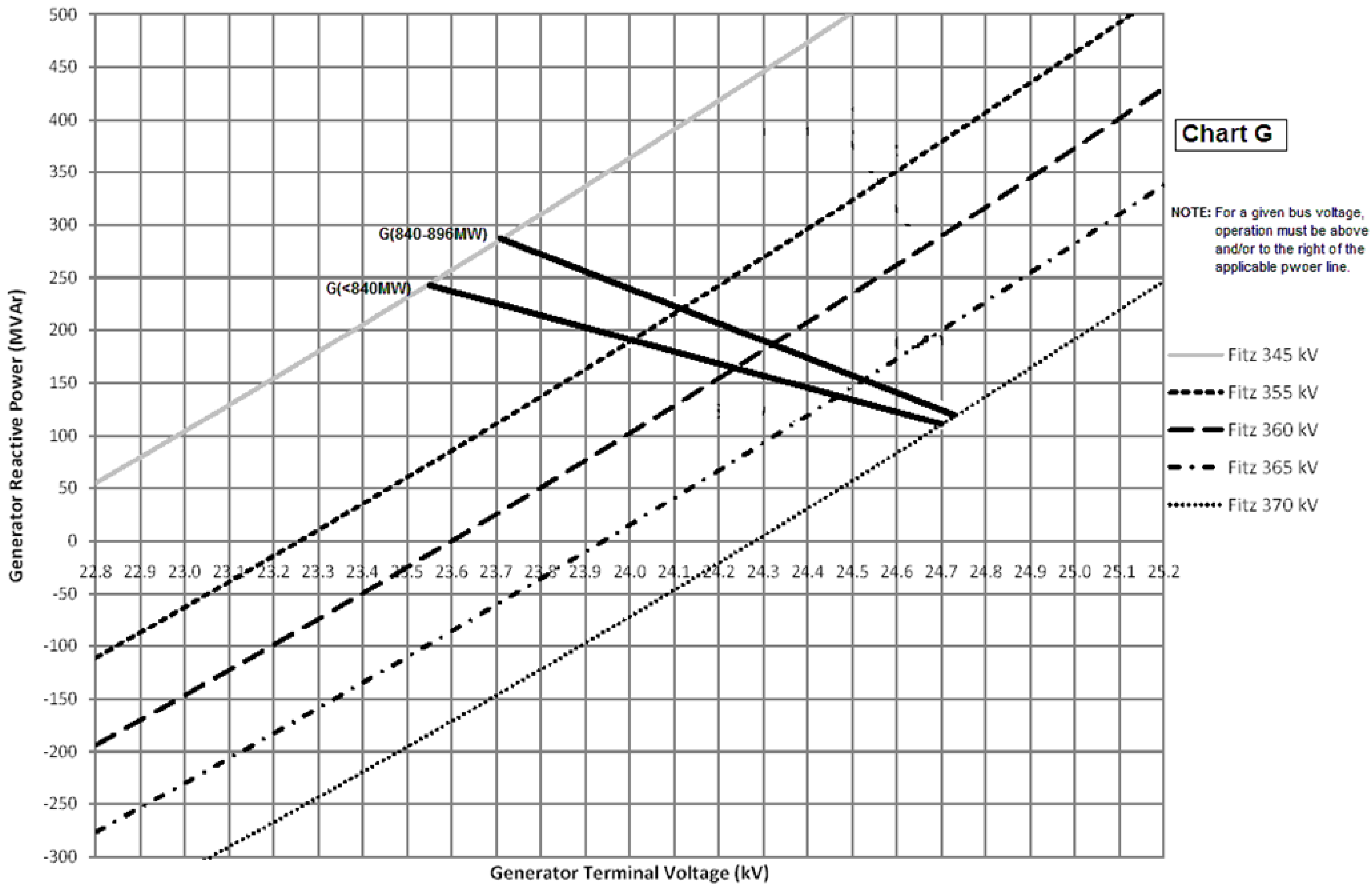


ATTACHMENT 9

CHART G - MINIMUM VOLTAGE/MVAR OPERATING LIMITS

ON-PEAK HOURS

VOLTAGE REGULATOR IN MANUAL



Proposed Answer: C

Explanation: OP-11A Section E.1 and associated attachments provide limitations on Main Generator parameters. The given values for real load, reactive load, and hydrogen gas pressure are within the limitations of attachment 2. However, the given values for real load, reactive load, and generator terminal voltage are outside of the required range for attachment 9. The voltage regulator must be adjusted to raise MVARs above the power line for 840-896 MWe operation.

A. Incorrect – Parameters are not within the limits of OP-11A attachment 9. Plausible because parameters are within the limits of OP-11A attachment 2.
B. Incorrect – Reactive load and real load are within the limitations of OP-11A attachment 2. Plausible because this would be correct if machine gas pressure was 45 psig or if reactive load were higher.
D. Incorrect – Reactive load is too low for the given terminal voltage, not too high. Plausible if the required operating point on the graph is misinterpreted.

Technical Reference(s): OP-11A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94D 1.13.a

Question Source: Bank – 9/12 NRC #48

Question History: 9/12 NRC #48

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 EK1.06
	Importance Rating	4.0

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Cooldown effects on reactor power

Proposed Question: #40

The plant has experienced a failure to scram with the following:

- Multiple control rods remain stuck full out.
- Reactor water level is being controlled between -19” and 110” using Feedwater.
- Reactor pressure is 920 psig and stable on Turbine Bypass Valves.
- All APRMs indicate downscale.
- All IRMs indicate mid-scale on range 4.
- Boron is being injected using Standby Liquid Control (SLC) pump A.
- Initial SLC tank level was 90%.
- Current SLC tank level is 50%.

Which one of the following describes the ability to commence a Reactor cooldown, in accordance with EOP-3, Failure to Scram?

A Reactor cooldown is currently…

A. allowed based on Reactor power indications.

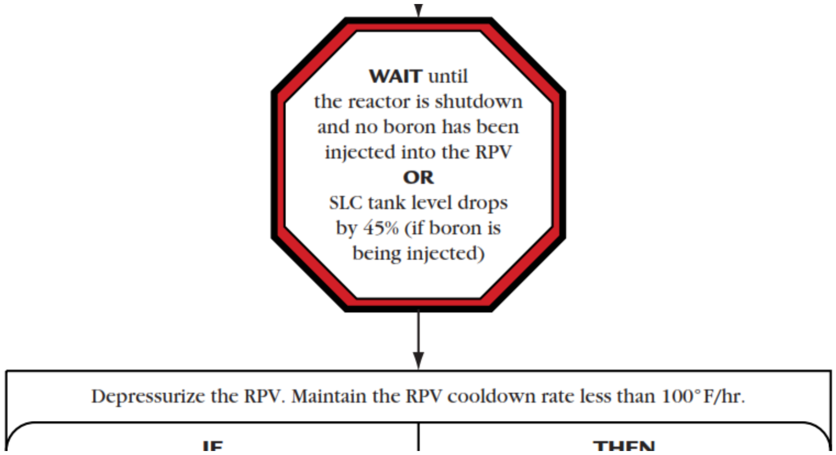
B. allowed based on the amount of boron injected.

C. NOT allowed because Reactor power is too high.

D. NOT allowed because of the amount of boron injected.

Proposed Answer: D

Explanation: EOP-3 contains the following step:



Since boron is being injected, cooldown cannot be commenced until SLC tank level drops by 45%. SLC tank level has only dropped by 40%, so cooldown is not yet allowed.

A. Incorrect – SLC tank level has only dropped by 40%, so cooldown is not yet allowed. Plausible because this would be correct if no boron had been injected.
B. Incorrect – SLC tank level has only dropped by 40%, so cooldown is not yet allowed. Plausible because this would be correct if SLC tank level were 5% lower.
C. Incorrect – SLC tank level, not power level, is the reason a cooldown is not allowed currently. Plausible because this would be the correct answer if no boron were injected and IRMs were indicating a few ranges higher.

Technical Reference(s): EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11d 1.07

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 #	1
	K/A #	295028 EK1.02
	Importance Rating	2.9

High Drywell Temperature

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification

Proposed Question: #41

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 water hammer concerns in cooling water piping inside the containment.

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

C. 347°F. This requirement is based on water hammer concerns in cooling water piping inside the containment.

D. 347°F. This requirement is based on applicable component qualification or structural design temperature limits.

Proposed Answer: D

Explanation: EOP-4 contains the following step:

IF	THEN
Drywell temperature <u>cannot</u> be restored and maintained below 347°F	**EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.**

The EOP bases state that this temperature is based on “applicable component qualification or structural design limits”.

A. Incorrect – The requirement is 347ºF, not 260ºF. Plausible because 260ºF is a limit in EOP-4 for isolating the cooling water supply to Drywell coolers, RWR pump and motor coolers, and Drywell equipment sump coolers. The temperature is based on component qualification or structural design limits. Plausible because this is the basis for action taken in EOP-4 for Drywell temperature of 260ºF.
B. Incorrect – The requirement is 347ºF, not 260ºF. Plausible because 260ºF is a limit in EOP-4 for isolating the cooling water supply to Drywell coolers, RWR pump and motor coolers, and Drywell equipment sump coolers.
C. Incorrect – The temperature is based on component qualification or structural design limits. Plausible because this is the basis for action taken in EOP-4 for Drywell temperature of 260ºF.

Technical Reference(s): EOP-4, MIT-301.11e

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 1.07

Question Source: Bank – 3/14 NRC #45

Question History: 3/14 NRC #45

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 #	295006 AK2.03
	Importance Rating	3.7

SCRAM

Knowledge of the interrelations between SCRAM and the following: CRD hydraulic system

Proposed Question: #42

The plant is operating at 100% power with the in-service Control Rod Drive (CRD) Flow Control Valve in automatic when a Reactor scram occurs.

Which one of the following describes the CRD total system flow change and the CRD drive water differential pressure change?

	CRD Total System Flow	CRD Drive Water Differential Pressure
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: B

Explanation: During normal operation, the CRD Flow Control Valve automatically maintains CRD total system flow at approximately 60 gpm. During a scram, CRD total system flow rises significantly due to flow through all 137 scram inlet valves. This flow is sensed through CRD system flow element FE-203 and transmitted to the Flow Control Valve, which closes in an attempt to limit total system flow below setpoint. Even with the Flow Control Valve closed, the CRD total system flow indication remains higher than normal while the scram condition exists. Drive water differential pressure lowers because the closed FCV cuts off the drive water header from the discharge of the CRD pumps.

A. Incorrect – Drive water differential pressure lowers. Plausible if candidate believes drive water header taps off upstream of the FCV, as the charging water header does.
C. Incorrect – Total system flow rises due to flow to all 137 scram inlet valves. Plausible because flow to the portions of the system after the FCV lower significantly. Drive water differential pressure lowers. Plausible if candidate believes drive water header taps off upstream of the FCV, as the charging water header does.
D. Incorrect – Total system flow rises due to flow to all 137 scram inlet valves. Plausible because flow to the portions of the system after the FCV lower significantly.

Technical Reference(s): OP-25, FM-27A, SDLP-03C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.05.c.3

Question Source: Modified Bank – 16-1 NRC #31

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 # 2
K/A # 201001 K5.02
Importance Rating 2.6

CRD Hydraulic

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM: Flow indication

Proposed Question: #31

The plant is operating at 100% power with the in-service Control Rod Drive (CRD) Flow Control Valve in automatic when a Reactor scram occurs.

Which one of the following describes the CRD Flow Control Valve response and the CRD total system flow change?

	CRD Flow Control Valve	CRD Total System Flow
A.	Closes	Rises
B.	Closes	Lowers
C.	Opens	Rises
D.	Opens	Lowers

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK2.15
	Importance Rating	3.8

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Containment spray logic: Plant-Specific

Proposed Question: #43

The plant has experienced a loss of coolant accident with the following:

Time (mm:ss)	Condition(s)
00:00	• Reactor water level is 120" and slowly lowering. • Reactor pressure is 800 psig and slowly lowering. • Drywell pressure is 10 psig and slowly rising. • Torus spray is initiated on RHR loop A.
03:00	• Reactor water level is 80" and slowly lowering. • Reactor pressure is 725 psig and slowly lowering. • Drywell pressure is 18 psig and slowly rising. • Drywell spray is initiated on RHR loop A.
05:00	• Reactor water level is 59.5" and slowly lowering. • Reactor pressure is 630 psig and slowly lowering. • Drywell pressure is 8 psig and slowly lowering.
07:30	• Reactor water level is 0" and slowly lowering. • Reactor pressure is 550 psig and slowly lowering. • Drywell pressure is 5 psig and slowly lowering.
09:00	• Reactor water level is -10" and slowly lowering. • Reactor pressure is 425 psig and rapidly lowering. • Drywell pressure is 2.7 psig and slowly lowering.

Which one of the following describes the status of Drywell spray during this transient?

Drywell spray…

A. remains in service continually after time 03:00.

B. isolates at time 05:00.

C. isolates at time 07:30.

D. isolates at time 09:00.

Proposed Answer: C

Explanation: When Drywell spray is initiated, Reactor water level is above 10”, therefore the DW & TORUS SPRAY VLV OVERRIDE OF FUEL ZONE LVL 10A S18A(B) keylock switch is NOT placed in MANUAL OVERRD. Therefore, when Reactor water level reaches 0” at time 07:30, the Drywell spray valves (10MOV-31A and 26A) automatically close and isolate Drywell spray.

A. Incorrect – Drywell spray isolates at time 07:30. Plausible because Reactor water level would not have isolated Drywell spray if the appropriate override switch was manipulated. Also plausible because Drywell pressure stays above 0 psig.
B. Incorrect – Drywell spray isolates at time 07:30. Plausible because this is the time when Reactor water level reaches the LPCI initiation setpoint of 59.5”.
D. Incorrect – Drywell spray isolates earlier at time 07:30. Plausible because Drywell spray would also isolate when Drywell pressure lowered to 2.7 psig if it were still in service.

Technical Reference(s): OP-13B, SDLP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.05.a.3.b

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 #	1
	Group #	1
	K/A #	295038 EK2.06
	Importance Rating	3.4

High Off-site Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Process liquid radiation monitoring system

Proposed Question: #44

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

- Torus Cooling is in service.
- Annunciator 09-3-2-30, Liquid Process Rad Monitor Hi-Hi, alarms.
- The Service Water Effluent radiation monitor is in alarm high.

Given the following possible leaks:

(1) Tube leak in RBCLC heat exchanger
(2) Tube leak in TBCLC heat exchanger
(3) Tube leak in RHR heat exchanger

Which one of the following identifies which of these given leaks is a potential cause of this alarm?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: B

Explanation: The Service Water Effluent radiation monitor is located such that it will detect a high release from an RBCLC heat exchanger tube leak or an RHR heat exchanger tube leak, but not a TBCLC heat exchanger tube leak.

A. Incorrect – The Service Water Effluent radiation monitor is located such that it will detect a high release from an RHR heat exchanger tube leak. Plausible because Service Water does not supply the RHR heat exchanger. The Service Water Effluent radiation monitor is located such that it will not detect a high release from a TBCLC heat exchanger tube leak. Plausible Service Water supplies the TBCLC heat exchangers and TBCLC does cool components that could be contaminated (Condensate/Feedwater/Offgas).
C. Incorrect – The Service Water Effluent radiation monitor is located such that it will detect a high release from an RBCLC heat exchanger tube leak. Plausible because RBCLC also has a separate rad monitor. The Service Water Effluent radiation monitor is located such that it will not detect a high release from a TBCLC heat exchanger tube leak. Plausible Service Water supplies the TBCLC heat exchangers and TBCLC does cool components that could be contaminated (Condensate/Feedwater/Offgas).
D. Incorrect – The Service Water Effluent radiation monitor is located such that it will not detect a high release from a TBCLC heat exchanger tube leak. Plausible Service Water supplies the TBCLC heat exchangers and TBCLC does cool components that could be contaminated (Condensate/ Feedwater).

Technical Reference(s): ARP 09-3-2-30, FM-46A, FM-36A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-46A 1.11.c.4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK3.01
	Importance Rating	3.8

Suppression Pool High Water Temperature

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Emergency/normal depressurization

Proposed Question: #45

Which one of the following describes (1) when Emergency RPV Depressurization is required based on high Torus temperature and (2) the basis for this requirement, in accordance with EOP-4, Primary Containment Control?

A. (1) Before Torus temperature reaches the Boron Injection Initiation Temperature (BIIT)
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

B. (1) Before Torus temperature reaches the Boron Injection Initiation Temperature (BIIT)
(2) Avoid failure of the Containment or equipment inside the Containment

C. (1) When Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL)
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

D. (1) When Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL)
(2) Avoid failure of the Containment or equipment inside the Containment

Proposed Answer: D

Explanation: The EOP-4 Torus temperature leg requires Emergency RPV Depressurization if Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL). The basis is to not raise Torus water temperature or pressure above limits before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent, such that failure of the containment and equipment inside the containment is avoided.

A. Incorrect – Entering EOP-2, not Emergency RPV Depressurization, is required before Torus temperature reaches BIIT. ECCS pump NPSH is lowered as Torus temperature rises, however it is not the specific basis for the Emergency RPV Depressurization requirement.
B. Incorrect – Entering EOP-2, not Emergency RPV Depressurization, is required before Torus temperature reaches BIIT.
C. Incorrect – ECCS pump NPSH is lowered as Torus temperature rises, however it is not the specific basis for the Emergency RPV Depressurization requirement.

Technical Reference(s): EOP-4, MIT-301.11e

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Bank – 9/14 NRC #56

Question History: 9/14 NRC #56

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 #	295001 AK3.05
	Importance Rating	3.2

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements: Plant-Specific

Proposed Question: #46

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

- A lockout condition develops on the 10300 bus.
- Reactor Recirculation (RWR) pump B trips on overcurrent.

Which one of the following describes (1) the required operator action and (2) the reason for that action, in accordance with the Abnormal Operating Procedures?

	(1) Required Operator Action	(2) Reason
A.	Establish Single Loop Operation (SLO)	Stop the idle RWR pump B impeller rotation
B.	Establish Single Loop Operation (SLO)	Activate SLO rod block and scram setpoints
C.	Insert a manual scram	Preclude a Cold Water Accident reactivity addition
D.	Insert a manual scram	No forced circulation through the Reactor core

Proposed Answer: D

Explanation: A lockout condition on the 10300 bus results in that bus de-energizing and remains de-energized until repaired. When the 10300 bus de-energizes, two RWR MG Set ‘A’ Lube Oil pumps 02-184P-2A1 and 2A3 trip. This results in a trip of the ‘A’ RWR pump on low Lube Oil pressure (see AOP-16, Loss 10300 Bus). The second bullet in the stem states that RWR pump B trips on overcurrent. This trip results in no RWR pumps running. AOP-8 (Loss or Reduction of Reactor Coolant Flow) is entered. Immediate Actions and Override statements in AOP-8 direct a manual scram be inserted if no RWR pumps are running with the Mode Switch in RUN.

A. Incorrect – SLO is not established because both RWR pumps are tripped. Plausible because the electrical loss is not directly the bus that powers RWR pump A. Also plausible because part of responding to trip of RWR pump B would be to close discharge valve to prevent flow through the idle loop.
B. Incorrect – SLO is not established because both RWR pumps are tripped. Plausible because the electrical loss is not directly the bus that powers RWR pump A. Also plausible because part of responding to trip of RWR pump B would be to adjust rod block and scram setpoints.
C. Incorrect – The reason for the scram requirement is lack of forced circulation. Plausible because restart of a tripped RWR pump can cause a cold water addition, and actions in the AOPs and OPs address this concern.

Technical Reference(s): AOP-8, AOP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H: 1.10, 1.14, 1.15

Question Source: Bank – 9/12 NRC #1

Question History: 9/12 NRC #1

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 #	295030 EK3.02
	Importance Rating	3.5

Low Suppression Pool Water Level

Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: HPCI operation: Plant-Specific

Proposed Question: #47

Which one of the following states the reason why EOP-4, Primary Containment Control, directs HPCI to be tripped if Torus water level CANNOT be maintained greater than 10.75 feet?

HPCI is tripped to prevent…

A. over-pressurizing the Torus.

B. localized boiling in the Torus.

C. vortex damage to the HPCI pump.

D. exceeding the Heat Capacity Temperature Limit (HCTL).

Proposed Answer: A

Explanation: HPCI steam discharge pipe becomes uncovered < 10.75 feet. EOP-4 bases states that operation of the HPCI System with its exhaust discharge device not submerged will directly pressurize the Torus. HPCI operation is therefore secured as required to preclude the occurrence of this condition.

B. Incorrect – The reason is to prevent over-pressurizing the Torus. Plausible because this is the reason for sequentially opening different SRVs in EOP-2.
C. Incorrect – The reason is to prevent over-pressurizing the Torus. Plausible because vortexing / loss of NPSH is a concern for HPCI on lowering Torus water level.
D. Incorrect – The reason is to prevent over-pressurizing the Torus. Plausible because HPCI operation adds heat to the Torus, which when combined with lowering Torus water level, challenges the HCTL.

Technical Reference(s): EOP-4, OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.05

Question Source: Bank – 9/12 NRC #15

Question History: 9/12 NRC #15

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 AA1.03
	Importance Rating	3.0

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air compressor power supplies

Proposed Question: #48

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

- The Reactor spuriously scrams.
- 4160VAC Buses 10100 and 10200 fail to fast transfer.
- 4160VAC Buses 10300 and 10400 fail to fast or slow transfer.
- Attempts to manually transfer these buses have NOT been successful.

Which one of the following identifies the number of Instrument Air compressors, if any, that are available?

A. 0

B. 1

C. 2

D. 3

Proposed Answer: A

Explanation: Instrument Air compressor A is powered from L23 (10300). Instrument Air compressor B is powered from L24 (10400). Instrument Air Compressor c is powered from L33 (10300). When the Reactor scrams, the Main Generator trips on reverse power and Transformer T4 de-energizes (normal supply at 100% power to the given buses). With failure of these buses to transfer to T2 and T3, they are de-energized and no Instrument Air compressors are available.

B. Incorrect – No Instrument Air compressors are available. Plausible because this would be correct if just 10400 was energized or if one of the three compressors was powered from 10500 or 10600.
C. Incorrect – No Instrument Air compressors are available. Plausible because this would be correct if just 10300 was energized or if two of the three compressors were powered from 10500 or 10600.
D. Incorrect – No Instrument Air compressors are available. Plausible because this would be correct if the compressors were all powered from 10500 or 10600.

Technical Reference(s): OP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.03, 1.05

Question Source: Modified Bank – 3/12 NRC #50

Question History: 3/12 NRC #50

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

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

Instrument Air: Knowledge of electrical power supplied to the following: Instrument air compressor.

Proposed Question: 50

The Plant is operating at 100% power with MOD 10017 (NORTH-SOUTH 115KV BUS DISC SW) tagged **open** due to Maintenance, when the following transient occurs:

- An NPO reports lightning strike in the 115KV switchyard
- Annunciators received:
  - 09-8-6-15 NMP-FITZ 115KV LINE 4 BKR 10012 TRIP
  - 09-5-1-2 MSIVs NOT FULL OPEN TRIP
- The SNO reports all MSIVs have both red and green lights illuminated
  - **No** Operator actions are taken

Ten (10) minutes after the transient begins, how many instrument air compressors, if any, would be available?

A.	0
B.	1
C.	2
D.	3

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA1.03
	Importance Rating	3.3

Partial or Complete Loss of CCW

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Affected systems so as to isolate damaged portions

Proposed Question: #49

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

- RBCLC leakage is suspected.
- Given the following RBCLC loads:

(1) Drywell Equipment Sump Cooler
(2) RWCU Pump Coolers
(3) Recirc Pump and Motor Coolers

Which one of the following lists which of these loads may be isolated from the RBCLC system using valves controlled from the Control Room?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: B

Explanation: Both the Drywell Sump Cooler and the Recirc Pump and Motor Coolers have RBCLC isolation valves that can be operated from Control Room panel 09-75. The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room.

A. Incorrect – The Recirc Pump and Motor Coolers have RBCLC isolation valves that can be operated from Control Room panel 09-75. Plausible because the RWCU Pump Coolers do NOT have similar valves. The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room. Plausible because other systems do have isolation valves that can be operated from the Control Room and RWCU itself has isolation valves that can be operated from the Control Room.
C. Incorrect – The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room. Plausible because other systems do have isolation valves that can be operated from the Control Room and RWCU itself has isolation valves that can be operated from the Control Room.
D. Incorrect – The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room. Plausible because other systems do have isolation valves that can be operated from the Control Room and RWCU itself has isolation valves that can be operated from the Control Room.

Technical Reference(s): OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.11.b

Question Source: Bank – 9/14 NRC #2

Question History: 9/14 NRC #2

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 #	600000 AA1.05
	Importance Rating	3.0

Plant Fire On-site

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Plant and control room ventilation systems

Proposed Question: #50

The plant has experienced a steam leak in the Reactor Building with the following:

- Standby Gas Treatment (SBGT) train A is in service.
- Annunciator 09-75-1-16, SGT SYS A ACT CHAR TEMP HI, alarms.
- The amber SBGT A WTR SPRAY SYS HI TEMP light is on at panel FPP.
- An operator in the field has confirmed that there is an active fire in the SBGT train A charcoal filter.

Which one of the following describes the response of SBGT fan A and SBGT train A fire protection water spray?

	SBGT Fan A	SBGT Train A Fire Protection Water Spray
A.	Automatically trips	Automatically initiates
B.	Automatically trips	Must be manually initiated
C.	Must be manually shutdown	Automatically initiates
D.	Must be manually shutdown	Must be manually initiated

Proposed Answer: D

Explanation: With the amber SBGT A WTR SPRAY SYS HI TEMP light on at panel FPP, OP-20 section G.3 is required to be performed. SBGT fan A remains running and must be manually shutdown. The fire protection water spray does not automatically initiate but is required to be manually initiated.

A. Incorrect – SBGT fan A remains running and must be manually shutdown. Plausible because a fire has been detected and further operation of the fan will exacerbate the condition by providing air flow through the area of the fire. The fire protection water spray does not automatically initiate but is required to be manually initiated. Plausible because a fire has been detected and many fire protection systems automatically initiate.
B. Incorrect – SBGT fan A remains running and must be manually shutdown. Plausible because a fire has been detected and further operation of the fan will exacerbate the condition by providing air flow through the area of the fire.
C. Incorrect – The fire protection water spray does not automatically initiate but is required to be manually initiated. Plausible because a fire has been detected and many fire protection systems automatically initiate.

Technical Reference(s): ARP 09-75-1-16, OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B 1.14.d

Question Source: New

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 #	295005 AA2.01
	Importance Rating	2.6

Main Turbine Generator Trip

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Turbine speed

Proposed Question: #51

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

- High vibrations develop on the Main Turbine.
- The Reactor is scrammed.
- The Main Turbine is manually tripped.
- The CRS directs continuing the shutdown of the Main Turbine in accordance with OP-9, Main Turbine.

Which one of the following describes the required operation of the Turning Gear, in accordance with OP-9?

A. Manually engage it when Turbine speed is less than 2 rpm.

B. Manually engage it when Turbine speed is approximately 100 rpm.

C. Verify it automatically engages when Turbine speed is less than 2 rpm.

D. Verify it automatically engages when Turbine speed is approximately 100 rpm.

Proposed Answer: C

Explanation: OP-9 requires verifying the Turbine Gear automatically engages when Turbine speed is less than 2 rpm.

A. Incorrect – The Turning Gear automatically engages. Plausible because during a startup the Turbine is placed on the Turning Gear manually and during a Turbine shutdown other components are manually started (Bearing Lift pumps).
B. Incorrect – The Turning Gear automatically engages. Plausible because during a startup the Turbine is placed on the Turning Gear manually and during a Turbine shutdown other components are manually started (Bearing Lift pumps).
D. Incorrect – The Turning Gear automatically engages at approximately 2 rpm, not 100 rpm. Plausible because during a Turbine startup, the Turning Gear is verified disengaged as the Turbine rolls to 100 rpm.

Technical Reference(s): OP-9

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.13.c

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 #	1
	Group #	1
	K/A #	295004 AA2.04
	Importance Rating	3.2

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: System lineups

Proposed Question: #52

The plant is operating at 25% power with the following:

- 4160V Buses have been transferred to Normal Station Service and Bus 10700 has been energized.
- RWR pump B is running at 60% speed.

Then, 125 VDC Bus B de-energizes due to a sustained electrical fault.

- AOP-46, Loss of DC Power System B, is being executed.
- The Reactor is manually scrammed due to rising Reactor water level.
- Reactor water level reaches a low of 150” following the scram.
- Reactor water level is maintained below 222.5” during the entire transient.

Which one of the following describes the status of RWR pump B three (3) minutes later?

A. NOT running

B. Running at 30% speed

C. Running at 44% speed

D. Running at 60% speed

Proposed Answer: A

Explanation: Buses 10200, 10400, and 10600 de-energize due to loss of power from T-4 on the Main Generator trip and failure to transfer because of loss of DC control power. With Bus 10200 de-energized, RWR pump B is not running.

B. Incorrect – RWR pump B is not running. Plausible because some conditions cause an RWR pump to run back to 30% speed (FW low flow) and a run back would normally occur on this scram due to Reactor water level <196.5” and only one FW pump in service at 25% power.
C. Incorrect – RWR pump B is not running. Plausible because a run back would normally occur on this scram due to Reactor water level <196.5” and only one FW pump in service at 25% power.
D. Incorrect – RWR pump B is not running. Plausible because the loss of DC Bus B causes RWR pump B speed control to lock up, such that a runback following a scram will not occur.

Technical Reference(s): AOP-46

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.09.a.12

Question Source: New

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 #	295003 AA2.02
	Importance Rating	4.2

Partial or Complete Loss of AC Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Reactor power, pressure, and level

Proposed Question: #53

The plant has experienced a Station Blackout and a failure of the UPS inverter to transfer to DC.

Which one of the following identifies indications that can be used to determine Reactor power and water level under these conditions?

	Reactor Power	Reactor Water Level
A.	IRM Downscale lights on Panel 09-5	Fuel Zone level indicator 02-3LI-92 on Panel 09-4
B.	IRM Downscale lights on Panel 09-5	Wide Range level recorder 06LR-97 on Panel 09-5
C.	IRM / APRM recorder on Panel 09-5	Fuel Zone level indicator 02-3LI-92 on Panel 09-4
D.	IRM / APRM recorder on Panel 09-5	Wide Range level recorder 06LR-97 on Panel 09-5

Proposed Answer: A

Explanation: With the loss of all AC power and failure of the UPS to transfer to DC, IRM and APRM recorders are unavailable, but IRM Downscale lights are still functional. Also, Wide Range level recorder is unavailable, but Fuel Zone level indicator 02-3LI-92 on Panel 09-4 is available.

B. Incorrect – Wide Range level recorder is unavailable. Plausible because other level indications remain available. Also plausible that Fuel Zone level indicator would be powered from UPS.
C. Incorrect – IRM and APRM recorders are unavailable. Plausible because other power indications remain available. Also plausible that IRM/APRM recorders would be powered from UPS.
D. Incorrect – IRM and APRM recorders are unavailable. Plausible because other power indications remain available. Also plausible that IRM/APRM recorders would be powered from UPS. Wide Range level recorder is unavailable. Plausible because other level indications remain available. Also plausible that Fuel Zone level indicator would be powered from UPS.

Technical Reference(s): AOP-21, AOP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.14

Question Source: Bank – 9/12 NRC #2

Question History: 9/12 NRC #2

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 #	295016 2.1.23
	Importance Rating	4.3

Control Room Abandonment

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

Proposed Question: #54

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

- Both RPS buses are powered from the normal supply.
- Then, a Control Room Evacuation becomes necessary.
- The Reactor fails to scram from the Control Room.

Given the following possible actions:

(1) Place RPS MG Set GENERATOR OUTPUT breakers in OFF.
(2) Open breakers to RPS on MCC-252 and MCC-262.
(3) Manually isolate and vent the scram air header.

Which one of the following identifies which of these actions is required to be performed, in accordance with AOP-43, Plant Shutdown From Outside the Control Room?

A. (1) only

B. (2) only

C. (1) and (3) only

D. (2) and (3) only

Proposed Answer: C

Explanation: Since both RPS buses are powered from the normal supply (MG set), the ATC is required to place RPS MG Set GENERATOR OUTPUT breakers in OFF in the West and East Electric Bays per AOP-43 Attachment 1. The scram air header is also manually isolated and vented by the SNO2 per AOP-43 Attachment 2.

A. Incorrect – Manually isolating and venting the scram air header is required. Plausible that this would not be performed because alternate electrical methods are used.
B. Incorrect – Opening breakers to RPS on MCC-252 and MCC-262 is not required to be performed. Plausible because this would be required if RPS was on alternate power. Manually isolating and venting the scram air header is required. Plausible that this would not be performed because alternate electrical methods are used.
D. Incorrect – Opening breakers to RPS on MCC-252 and MCC-262 is not required to be performed. Plausible because this would be required if RPS was on alternate power.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.05, 1.11

Question Source: Modified Bank – 9/12 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 # 1
Group # 1
K/A # 295016 AA1.01
Importance Rating 3.8

Control Room Abandonment

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPS

Proposed Question: #6

The Plant was operating at 78% power with the following RPS electrical power alignment:

- RPS 'A': powered from its Normal Power Supply
- RPS 'B': powered from its Alternate Power Supply

The following takes place:

- A significant fire in the Relay Room occurs.
- Smoke from the fire requires the Control Room to be abandoned.
- AOP-43 (Plant Shutdown From Outside The Control Room) is entered.

All actions by the ATC (At the Controls) Operator to scram the Reactor from the Control Room fail.

- No Control Rods move.
- RPS Group 'A' and 'B' white lights are ON.

Which one of the following states how the Reactor is to be shutdown?

(1) **<u>RPS 'A'</u>**
(2) **<u>RPS 'B'</u>**

A. **(1)** Place RPS MG Set 'A' GENERATOR OUTPUT breaker in OFF
(2) Place RPS MG Set 'B' GENERATOR OUTPUT breaker in OFF

B. **(1)** Place RPS MG Set 'A' GENERATOR OUTPUT breaker in OFF
(2) Open breaker 71MCC-262-OB2 71-05-6B REACTOR PROTECTION BUS B DISTRIBUTION PANEL.

C. **(1)** Open breaker 71MCC-252-OC1 71-05-6A REACTOR PROTECTION BUS A DISTRIBUTION PANEL.
(2) Place RPS MG Set 'B' GENERATOR OUTPUT breaker in OFF

D. **(1)** Open breaker 71MCC-252-OC1 71-05-6A REACTOR PROTECTION BUS A DISTRIBUTION PANEL.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 2.2.37
	Importance Rating	3.6

Loss of Shutdown Cooling

Ability to determine operability and / or availability of safety related equipment.

Proposed Question: #55

The plant is shutdown during a refueling outage with the following:

- New fuel is being loaded into the Reactor core.
- RHR pump A is running for Shutdown Cooling.
- All other RHR pumps are available and in standby.
- Then, a leak develops from a through-wall crack between 10MOV-18, RHR Shutdown Cooling Outboard Isolation Valve, and 10MOV-17, RHR Shutdown Cooling Inboard Isolation Valve.
- The Shift Manager has directed securing RHR pump A and isolating the leak by closing 10MOV-18 and 10MOV-17.

Which one of the following describes the impact of this evolution on Shutdown Cooling?

A. Shutdown Cooling will be secured temporarily during RHR pump A shutdown, but can be re-established with any of the other three RHR pumps once the leak is isolated.

B. Shutdown Cooling will be secured temporarily during RHR pump A shutdown, but can be re-established with either RHR pump B or D once the leak is isolated. RHR pump C will be unavailable for Shutdown Cooling.

C. The RHR Shutdown Cooling mode will be unavailable as long as 10MOV-18 and 10MOV-17 are closed. Alternate decay heat removal using RHR in the Fuel Pool Cooling Assist mode is still available.

D. The RHR Shutdown Cooling mode will be unavailable as long as 10MOV-18 and 10MOV-17 are closed. Alternate decay heat removal using RHR in the Fuel Pool Cooling Assist mode is also unavailable.

Proposed Answer: C

Explanation: 10MOV-18 and 10MOV-17 are required to be open to align any RHR pump for Shutdown Cooling. Therefore, with these valves closed, the RHR Shutdown Cooling mode will be unavailable. However, RHR may still be placed in the Fuel Pool Cooling Assist mode, which requires 10MOV-18 and 10MOV-17 to be closed. With the SFP gates removed, as evidenced by loading of new fuel into the Reactor core, this provides alternate decay heat removal.

A. Incorrect – 10MOV-18 and 10MOV-17 are required to be open to align any RHR pump for Shutdown Cooling. Plausible because a different leak location and isolation strategy would make this correct (e.g. closing 10MOV-15A).
B. Incorrect – 10MOV-18 and 10MOV-17 are required to be open to align any RHR pump for Shutdown Cooling. Plausible that the other division of RHR would have separate suction path for SDC to provide redundancy.
D. Incorrect – The Fuel Pool Cooling Assist mode is available for decay heat removal with 10MOV-18 and 10MOV-17 closed. Plausible because normal use of RHR for SDC is unavailable.

Technical Reference(s): OP-13D, AOP-30, OP-13F, FM-20

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.06.e and 1.06.h

Question Source: Bank – 3/14 NRC #43

Question History: 3/14 NRC #43

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

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: #56

A refueling outage is in progress with the following:

- The plant is in Mode 5.
- One control rod is stuck at position 02.
- All other control rods are fully inserted.
- It is desired to begin irradiated fuel movements from the Spent Fuel Pool into the Reactor core.

Which one of the following describes the ability to commence these irradiated fuel movements, in accordance with Technical Specifications?

These irradiated fuel movements into the Reactor core are…

A. NOT allowed.

B. allowed without any additional requirements because NO control rod is beyond position 02.

C. allowed without any additional requirements because only one control rod is NOT fully inserted.

D. allowed if additional independent verifiers are stationed on the Refuel Bridge and at Control Room panel 09-5.

Proposed Answer: A

Explanation: Technical Specification 3.9.3 requires all control rods to be fully inserted when loading fuel assemblies into the core. With one control rod not fully inserted, it is required to immediately suspend loading fuel assemblies into the core.

B. Incorrect – With one control rod not fully inserted, it is required to immediately suspend loading fuel assemblies into the core. Plausible because calculations for EOP-3 determine that the Reactor will stay shutdown under all conditions without boron because no rod is withdrawn beyond position 02.
C. Incorrect – With one control rod not fully inserted, it is required to immediately suspend loading fuel assemblies into the core. Plausible because SDM requirements are based on any control rod being withdrawn to any position.
D. Incorrect – With one control rod not fully inserted, it is required to immediately suspend loading fuel assemblies into the core. Plausible because the rod is not very far withdrawn and Technical Specification 3.3.2.1 contains a similar requirement for starting up without the RWM.

Technical Reference(s): Technical Specification 3.9.3

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08B 1.16

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 #	295025 EA1.02
	Importance Rating	3.8

High Reactor Pressure

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Reactor/turbine pressure regulating system

Proposed Question: #57

The plant has experienced a failure to scram from 100% power with the following:

- Reactor power now indicates 20% on APRMs.
- Reactor water level reached a low of 75” during the initial transient.
- Reactor water level is now 140” and slowly rising with Feedwater injecting.
- Main Condenser vacuum is 21” Hgv and stable.
- EOP-3, Failure to Scram, is being executed.
- An Operator has SRVs open.
- Reactor pressure is now 890 psig and slowly lowering.

Which one of the following describes the status of Turbine Bypass Valves?

Turbine Bypass Valves are currently…

A. available for pressure control and open.

B. available for pressure control and closed.

C. unavailable for pressure control because MSIVs are closed and CANNOT be re-opened.

D. unavailable for pressure control because MSIVs are closed, but can be made available by re-opening MSIVs.

Proposed Answer: B

Explanation: MSIVs are currently open because Reactor water level stayed above 59.5”. Combined with Main Condenser vacuum >8” Hgv, this makes Turbine Bypass Valves available for pressure control. Turbine Bypass Valves are currently closed because Reactor pressure has been reduced below 970 psig with SRVs.

A. Incorrect – Turbine Bypass Valves are currently closed because Reactor pressure has been reduced below 970 psig with SRVs. Plausible because if Reactor pressure was >970 psig, they would be open.
C. Incorrect – MSIVs are currently open. Plausible because this would be correct if Reactor water level went less than 59.5”, or if vacuum was further degraded, or if there was evidence of a steam leak.
D. Incorrect – MSIVs are currently open. Plausible because this would be correct if Reactor water level went less than 59.5”, because EOP-3 would allow re-opened MSIVs, bypassing interlocks if necessary.

Technical Reference(s): EOP-3, AOP-15, ODSO-49

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94C

Question Source: New

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 #	295031 EA2.02
	Importance Rating	4.0

Reactor Low Water Level

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor power

Proposed Question: #58

A failure to scram has occurred with the following:

- Reactor power is 20%.
- Torus water temperature is 115°F.
- Drywell pressure is 3.5 psig.
- All MSIVs are closed.
- Reactor pressure is 900 psig and rising with two SRVs open.
- Injection has been terminated and prevented from Condensate and Feedwater, HPCI, RHR, and Core Spray.
- Reactor water level has been lowered to +100" in accordance with EOP-3, Failure to Scram.

Which one of the following conditions would allow the operator to resume injection from Condensate and Feedwater, in accordance with EOP-3?

A. Reactor power drops to 2%.

B. Reactor water level drops to +19".

C. Drywell pressure drops to 2.2 psig.

D. Reactor pressure control needs only one SRV open to maintain pressure.

Proposed Answer: A

Explanation: EOP-3 contains the following guidance on lowering Reactor water level:

IF	THEN
Reactor power is above 2.5% or cannot be determined **AND** RPV water level is above 110 in.	Lower RPV water level to below 110 in. by terminating and preventing all injection except SLC, RCIC and CRD (EP-5).

IF	THEN
Reactor power is above 2.5% or cannot be determined **AND** RPV water level is above 0 in., **AND** Torus temperature is above the Boron Injection Initiation Temperature **AND** An SRV is open or drywell pressure is above 2.7 psig	Irrespective of reactor power or RPV water level oscillations, lower RPV water level by terminating and preventing all injection except SLC, RCIC and CRD (EP-5) until either: ➢ Reactor power drops below 2.5%, **OR** ➢ RPV water level drops to 0 in., **OR** ➢ All SRVs remain closed and drywell pressure remains below 2.7 psig.

Of the given conditions, the only one that would allow re-injection is Reactor power lowering to 2% (<2.5%).

B. Incorrect – Reactor water level would need to drop to 0" before re-injection would be allowed. Plausible because this would be correct if a lower water level were given.
C. Incorrect – Drywell pressure would need to drop AND SRVs would have to be closed before re-injection would be allowed. Plausible because Drywell pressure <2.7 psig is part of the conditional step for re-injection.
D. Incorrect – All SRVs would have to be closed AND Drywell pressure would need to drop before re-injection would be allowed. Plausible because SRVs are part of the conditional step for re-injection and this would also prove Reactor power has lowered further.

Technical Reference(s): EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.07

Question Source: Bank – 2010 NRC #16

Question History: 2010 NRC #16

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 #	295013 AK1.03
	Importance Rating	3.0

High Suppression Pool Temperature

Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Localized heating

Proposed Question: #59

A steam leak in the Main Steam tunnel has resulted in the following:

- Operators have scrammed the Reactor.
- The CRS has directed Reactor pressure controlled 800-1000 psig using SRVs.

Which one of the following describes how SRVs should be operated, in accordance with the Emergency Operating Procedures?

Cycle…

A. through all SRVs sequentially, as needed, to verify they are all seated properly.

B. the same SRV repeatedly, as needed, to prevent thermal cycling of multiple SRV tailpipes.

C. through all SRVs sequentially, as needed, to distribute the heat load uniformly around the Torus.

D. the same SRV repeatedly, as needed, to maintain Torus heating as close as possible to the Torus Cooling injection point.

Proposed Answer: C

Explanation: SRV discharges are designed to be equally spaced at points around the Torus. EOP-2 contains direction on the use of SRVs for Reactor pressure control based on this design to ensure more equal heat load distribution around the Torus by opening SRVs in a preferred sequence, rather than just opening the same SRV repeatedly.

A. Incorrect – The reason for operating SRVs sequentially is to distribute the heat load uniformly around the torus and equalize the number of actuations among the SRVs, NOT to ensure they are all seated. Plausible because cycling a leaking SRV is the method used to attempt to reseat the valve.
B. Incorrect – EOP-2 contains a note that directs opening SRVs per a pre-defined sequence, vice opening the same SRV multiple times. Plausible because opening a new SRV will cause a larger thermal transient since its tailpipe is colder.
D. Incorrect – EOP-2 contains a note that directs opening SRVs per a pre-defined sequence, vice opening the same SRV multiple times. Plausible because one loop of Torus Cooling does discharge to one point in the Torus, so this area will be cooled more directly.

Technical Reference(s): EOP-2, MIT-301.11c

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.07

Question Source: Bank – NMP1 2017 NRC #7

Question History: NMP1 2017 NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295015 AK2.05
	Importance Rating	2.6

Incomplete SCRAM

Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Rod worth minimizer: Plant-Specific

Proposed Question: #60

The plant has experienced a Reactor scram. The following conditions now exist:

- Multiple control rods remain fully withdrawn.
- The Reactor Mode Switch is in SHUTDOWN.
- All APRMs indicate 10%.
- All IRMs indicate upscale on range 9.
- EP-3, Backup Control Rod Insertion, is in progress.
- It is desired to insert control rods using RMCS.
- The Rod Worth Minimizer (RWM) is currently NOT bypassed.

Which one of the following describes the need to bypass the RWM?

The RWM…

A. MUST be bypassed because IRM upscale rod blocks are being enforced.

B. does NOT have to be bypassed because Reactor power is in the transition zone.

C. MUST be bypassed because the RWM is invoking an insert error control rod block.

D. does NOT have to be bypassed because the Emergency Rod In switch will insert control rods regardless of RWM operation.

Proposed Answer: C

Explanation: The RWM ensures the reactor operator adheres to a predetermined sequence of control rod withdrawals or insertions when the reactor is operating at low power levels. Given multiple control rods remain full out, insert blocks will be active and prevent the Operator from inserting control rods. Therefore, the RWM must be bypassed to allow rod insertion. Transition zone is between 16% and 35% core thermal power. While in the transition zone, all normal RWM rod block actions are removed. Displays are still active, with exception of withdraw error windows which will not be displayed in the transition zones. Low Power Setpoint: If steam flow goes < 16%, the RWM is returned to operation.

A. Incorrect – The RWM must be bypassed, but not because of IRM rod blocks. Plausible because IRMs are causing withdrawal blocks.
B. Incorrect – The RWM must be bypassed. Plausible because APRMS are indicating fairly high, but below the transition zone (16%).
D. Incorrect – The RWM must be bypassed. Plausible because the Emergency Rod In switch may be used and does bypass some features (RMCS timer).

Technical Reference(s): OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.05, 1.08, 1.15

Question Source: Bank - 9/12 NRC #23

Question History: 9/12 NRC #23

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295007 AK3.02
	Importance Rating	3.7

High Reactor Pressure

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

Proposed Question: #61

A scram has occurred with the following:

- MSIVs are closed.
- HPCI automatically started and is the only available injection source.
- The HPCI flow controller has been placed in MAN.
- HPCI injection has been throttled to 200 gpm to the Reactor.
- Reactor water level is 180 inches and stable.
- Reactor pressure is 900 psig and slowly rising.

Which one of the following describes the response of HPCI flow rate if Reactor pressure rises to 1100 psig?

HPCI flow rate will…

A. lower because HPCI turbine speed is controlled at a constant value.

B. lower because the design discharge pressure range of the HPCI pump is exceeded.

C. remain approximately constant because the control system will throttle the governor based on a flow feedback signal.

D. remain approximately constant because rising Reactor steam supply pressure balances rising pump discharge pressure without the need for governor adjustment.

Proposed Answer: A

Explanation: The HPCI pump is designed to supply 4250 gpm over a Reactor pressure range from 150 to 1195 psig. With the controller in MAN, the HPCI governor valve will be automatically adjusted to maintain constant speed. With rising Reactor pressure, HPCI flow will lower with the turbine maintained at the same speed.

B. Incorrect – HPCI is rated for 4250 gpm up to 1195 psig. Plausible because 1100 psig is high pressure and above the Reactor scram setpoint.
C. Incorrect – Flow will lower. Plausible because this would be correct with the controller in AUTO.
D. Incorrect – Flow will lower. Plausible because with no governor response, rising steam inlet pressure would tend to raise flow, while rising discharge pressure would tend to lower flow.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.a.22

Question Source: Modified Bank – 9/14 NRC #46

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 # 295025 EK3.03
Importance Rating 3.8

High Reactor Pressure

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

Proposed Question: #46

A scram has occurred with the following:

- MSIVs are closed.
- HPCI automatically started and is the only available injection source.
- HPCI injection has been throttled to 200 gpm to the Reactor with the controller in AUTO.
- Reactor water level is 180 inches and stable.
- Reactor pressure is 900 psig and slowly rising.

Which one of the following describes the response of HPCI flow rate if Reactor pressure rises to 1100 psig?

HPCI flow rate will…

A. lower because HPCI turbine speed is controlled at a constant value.

B. lower because the design discharge pressure range of the HPCI pump is exceeded.

C. remain approximately constant because the control system will throttle the governor based on a flow feedback signal.

D. remain approximately constant because rising Reactor steam supply pressure balances rising pump discharge pressure without the need for governor adjustment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AA1.03
	Importance Rating	2.7

Loss of CRD Pumps

Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: Recirculation system: Plant-Specific

Proposed Question: #62

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

- CRD pump B is tagged out of service for maintenance.
- Then, CRD pump A trips.

Which one of the following Reactor Recirculation pump components / parameters will require more frequent monitoring?

A. Pump seals

B. Motor current

C. Pump vibration

D. Motor temperatures

Proposed Answer: A

Explanation: The CRD system provides clean, cool water (called mini-purge) to the RWR pump seals. This mini-purge flushes or maintains the seals clean and cool. A loss of this mini-purge will cause the seals to over-heat and degrade.

B. Incorrect – RWR pump seals lose cooling. Plausible if candidate believes there are additional motor current limitations with mini-purge unavailable due to lowered cooling capacity.
C. Incorrect – RWR pump seals lose cooling. Plausible if candidate believes mini-purge flow is required to maintain balanced pump rotation based on hydraulic considerations or temperature considerations.
D. Incorrect – RWR pump seals lose cooling. Plausible because RWR pump motors require cooling, but it is from RBCLC.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.09

Question Source: Bank – 9/12 NRC #25

Question History: 9/12 NRC #25

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 #	295020 AA2.04
	Importance Rating	3.9

Inadvertent Containment Isolation

Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor pressure

Proposed Question: #63

The plant is operating at 100% power when MSIVs spuriously close and the Reactor fails to scram.

Which one of the following describes the Reactor pressure at which the first SRVs will begin to open and the Reactor pressure at which the last SRVs will begin to open?

The first SRVs will begin to open at…

A. 1080 psig and the last SRVs will begin to open at 1145 psig.

B. 1080 psig and the last SRVs will begin to open at 1153 psig.

C. 1135 psig and the last SRVs will begin to open at 1145 psig.

D. 1135 psig and the last SRVs will begin to open at 1153 psig.

Proposed Answer: C

Explanation: SRV electric lift setpoints are as follows:

Valve	Setpoint
02RV-71 K, L	1135 psig
02RV-71 D, E	1140 psig
02RV-71 A, B, C, F, G, H, J	1145 psig

Therefore the first SRVs begin to lift at 1135 psig and the last SRVs begin to lift at 1145 psig.

A. Incorrect – The first SRVs begin to open at 1135 psig. Plausible because 1080 psig is associated with the Reactor scram setpoint.
B. Incorrect – The first SRVs begin to open at 1135 psig. Plausible because 1080 psig is associated with the Reactor scram setpoint. The last SRVs being to open at 1145 psig. Plausible because 1153 psig is associated with ATWS RPT/ARI.
D. Incorrect – The last SRVs being to open at 1145 psig. Plausible because 1153 psig is associated with ATWS RPT/ARI.

Technical Reference(s): SDLP-02D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05.a.1

Question Source: Bank – 3/14 NRC #62

Question History: 3/14 NRC #62

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295012 2.4.1
	Importance Rating	4.6

High Drywell Temperature

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: #64

The plant has experienced a loss of coolant accident with the following:

- Drywell pressure is 5 psig and rising slowly.
- Drywell temperature is 140^{o}F and rising slowly.
- Torus pressure is 3 psig and rising slowly.
- Torus temperature is 87^{o}F and rising slowly.
- Torus water level is 14.1’ and rising slowly.

Which one of the following identifies the total number of entry conditions currently met or exceeded for EOP-4, Primary Containment Control?

A. One

B. Two

C. Three

D. Four

Proposed Answer: C

Explanation: The EOP-4 entry conditions are:

ENTRY CONDITIONS				
Primary containment hydrogen above 3%	**Drywell temperature** above 135°F	**Drywell pressure** above 2.7 psig	**Torus temperature** above 95°F	**Torus water level** below 13.88 ft **OR** above 14.0 ft

With the given conditions, a total of three entry conditions have been exceeded (Drywell temperature, Drywell pressure, and Torus water level.

A. Incorrect – A total of three entry conditions have been exceeded. Plausible because Drywell temperature and Torus water level are both just above their respective entry conditions.
B. Incorrect – A total of three entry conditions have been exceeded. Plausible because Plausible because Drywell temperature and Torus water level are both just above their respective entry conditions.
D. Incorrect – A total of three entry conditions have been exceeded. Plausible because Torus temperature is elevated but not yet above the entry condition. Also plausible because Torus pressure is above 2.7 psig, but not an entry condition.

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e EO-4.02

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 #	295014 AK2.06
	Importance Rating	3.4

Inadvertent Reactivity Addition

Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: Moderator temperature

Proposed Question: #65

The plant is operating at 40% power with the following:

- Recirculation pump A is operating.
- Preparations are underway to start Recirculation pump B, in accordance with OP-27, Recirculation System.

Which one of the following identifies the maximum differential temperature between RWR loop A and RWR loop B, in accordance with OP-27?

A. 50ºF

B. 70ºF

C. 90ºF

D. 145ºF

Proposed Answer: A

Explanation: The maximum allowable difference in temperature between the coolant in Recirculation loops A and B is 50ºF for start of Recirculation pump B.

Note: The question meets the K/A by testing a limitation placed on moderator temperature (idle vs. running Recirculation loop temperature) that is required to prevent an inadvertent reactivity addition (excess cold water addition beyond the plant's analysis).

B. Incorrect – The maximum difference is 50ºF. Plausible because 70ºF is a limit placed on RBC temperature to Recirculation pump B prior to start.
C. Incorrect – The maximum difference is 50ºF. Plausible because 90ºF is a limit placed on Recirculation MG set B lube oil temperature prior to start.
D. Incorrect – The maximum difference is 50ºF. Plausible because 145ºF is a limit placed on the temperature difference between the bottom head and bulk RPV coolant temperature prior to start.

Technical Reference(s): OP-27, ST-26K

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.13.d

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 #	2.1.1
	Importance Rating	3.8

Knowledge of conduct of operations requirements.

Proposed Question: #66

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

- You are a licensed Reactor Operator.
- You have been on vacation for the previous 14 days.
- Today was your first shift back on watch.
- You worked the normal day shift from 0600 to 1800.
- You then were required to stay a total of four (4) hours past end of shift due to an on-coming Operator calling in sick.
- You were originally scheduled to cover day shift tomorrow from 0600 to 1800.

Which one of the following identifies the earliest time you can return to work tomorrow without receiving a waiver, in accordance with LS-AA-119, Fatigue Management and Work Hour Limits?

A. 0600

B. 0700

C. 0800

D. 1000

Proposed Answer: C

Explanation: LS-AA-119 Section 3.4 requires a 10 hour break between the previous work period, or an 8 hour break between the previous work period when a break of less than 10 hours was necessary to accommodate a crew's scheduled transition between work schedules or shifts. Since this situation is not related to a crew's scheduled shift transition, the individual needs a minimum of a 10 hour break before returning back to work. The individual worked 4 hours past 1800, so they left work at 2200. 10 hours later is 0800.

A. Incorrect – The earliest allowed return is 0800. Plausible because this would be correct if this was part of a planned shift rotation.
B. Incorrect – The earliest allowed return is 0800. Plausible because this is in the range of acceptable times for various situations.
D. Incorrect – The earliest allowed return is 0800. Plausible because this is 12 hours later, which is the originally scheduled amount of time off between shifts.

Technical Reference(s): LS-AA-119

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2017 NRC #67

Question History: NMP1 2017 NRC #67

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 #	2.1.30
	Importance Rating	4.4

Ability to locate and operate components, including local controls.

Proposed Question: #67

The plant was operating at 100% with the following:

- A fire developed in the Control Room and required evacuation.
- AOP-43, Plant Shutdown From Outside the Control Room, is being performed.
- The Shift Manager has directed starting RHRSW pump B and RHR pump D.

Which one of the following describes the panel location(s) where these actions are performed?

A. Both actions are performed at 25RSP (Reactor Building 300' North).

B. RHRSW pump B is started at 25RSP (Reactor Building 300' North) and RHR pump D is started at 25ASP-2 (East Crescent Stairway).

C. RHRSW pump B is started at 25ASP-2 (East Crescent Stairway) and RHR pump D is started at 25RSP (Reactor Building 300' North).

D. Both actions are performed at 25ASP-2 (East Crescent Stairway).

Proposed Answer: A

Explanation: 25RSP, located on Reactor Building 300' North, contains the controls to both start RHRSW pump B and RHR pump D. 25ASP-2, located at the East Crescent Stairway, contains multiple RHR valve controls, but no controls for pumps.

B. Incorrect – Both controls are at 25RSP. Plausible because 25ASP-2, located at the East Crescent Stairway, contains multiple RHR valve controls, but no controls for pumps.
C. Incorrect – Both controls are at 25RSP. Plausible because 25ASP-2, located at the East Crescent Stairway, contains multiple RHR valve controls, but no controls for pumps.
D. Incorrect – Both controls are at 25RSP. Plausible because 25ASP-2, located at the East Crescent Stairway, contains multiple RHR valve controls, but no controls for pumps.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.11.b.3

Question Source: Bank – 3/14 NRC #42

Question History: 3/14 NRC #42

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.38
	Importance Rating	3.6

Knowledge of conditions and limitations in the facility license.

Proposed Question: #68

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

- A 3D Monicore case has just been run.
- The most limiting thermal limits are currently:
  - MFLCPR: 0.97
  - MFLPD: 0.89
  - 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, is NOT met.

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

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

Proposed Answer: B

Explanation: MFLPD, 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 all ratios that are required to be ≤1.
C. Incorrect – Since MFLCPR is less than 1, the associated MCPR thermal limit is SAT. Plausible if the candidate confuses which ratio corresponds to which thermal limit.
D. Incorrect – Since MFLPD is less than 1, the associated LHGR thermal limit is SAT. Plausible if the candidate confuses which ratio corresponds to which thermal limit.

Technical Reference(s): TS 3.2.1, COLR

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-09B 1.14

Question Source: Bank - 2017LOR0032

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.21
	Importance Rating	2.9

Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: #69

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

- A piece of safety-related equipment required by Technical Specifications has been out-of-service for two days for maintenance overhaul.
- All maintenance is now complete.
- All clearances have been removed.
- The equipment has been restored to a normal standby lineup per the associated Operating Procedure.
- Post-maintenance testing of the equipment is about to start.

Which one of the following defines the current status of the equipment, in accordance with OP-AA-108-115, Operability Determinations?

The equipment is…

A. inoperable, but available.

B. non-functional, but available.

C. both inoperable and unavailable.

D. both non-functional and unavailable.

Proposed Answer: A

Explanation: OP-AA-108-115, Operability Determinations, contains definitions for various system status designations, such as operable, available, and functional. Since the equipment has been physically restored to a normal standby lineup, it is available for use. However, operability is not restored until the equipment is tested (PMT) to ensure it meets all required design functions. Additionally, the term functional does not apply because this is equipment required by Technical Specifications, and functional is used to describe non-Technical Specifications required equipment.

B. Incorrect – The term functional does not apply because this is equipment required by Technical Specifications. Plausible because this would be correct If this equipment was not required by Technical Specifications.
C. Incorrect – The equipment is considered available. Plausible because this would be correct if the equipment had not yet been restored to a normal standby lineup.
D. Incorrect – The equipment is considered available. Plausible because this would be correct if the equipment had not yet been restored to a normal standby lineup and If this equipment was not required by Technical Specifications.

Technical Reference(s): OP-AA-108-115

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP2 2014 NRC #74

Question History: NMP2 2014 NRC #74

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 #	2.3.13
	Importance Rating	3.4

Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: #70

The plant is operating at 20% power with the following:

- Personnel are preparing to enter the Drywell to investigate a problem.
- Drywell oxygen concentration is 20.1%.

Which one of the following describes the acceptability of the current Reactor power level and Drywell oxygen concentration for personnel entry into the Drywell, in accordance with AP-12.02, Drywell Entries During Primary Containment?

	Reactor Power	Drywell Oxygen Concentration
A.	Acceptable	Acceptable
B.	Acceptable	NOT acceptable
C.	NOT acceptable	Acceptable
D.	NOT acceptable	NOT acceptable

Proposed Answer: C

Explanation: AP-12.02 requires Reactor power to be less than 15% and Drywell oxygen concentration to be between 19.5% and 23.5% to allow personnel entry to the Drywell when Primary Containment is required. Reactor power is currently unacceptable because it is above 15%. Drywell oxygen concentration is acceptable because it is within the required range.

A. Incorrect – Reactor power is currently unacceptable because it is above 15%. Plausible because Reactor power is significantly below full power.
B. Incorrect – Reactor power is currently unacceptable because it is above 15%. Plausible because Reactor power is significantly below full power. Drywell oxygen concentration is acceptable because it is within the required range of 19.5-23.5%. Plausible because concentration is low in range.
D. Incorrect – Drywell oxygen concentration is acceptable because it is within the required range of 19.5-23.5%. Plausible because concentration is low in range.

Technical Reference(s): AP-12.02

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP EO-45.04 and EO-45.05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.7
	Importance Rating	3.5

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

Proposed Question: #71

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

- An Operator is supporting a job under the Radiation Work Permit with the following limits:
  - Dose limit: 20 mRem
  - Dose rate limit: 80 mRem
- The Operator's electronic dosimeter is currently reading the following:
  - Dose: 17 mRem
  - Dose rate: 72 mRem/hr

Which one of the following describes the Operator's compliance with the RWP?

A. The Operator is within the RWP limits and may continue working on this job without contacting Radiation Protection.

B. The Operator must leave the area and contact Radiation Protection due to both dose and dose rate.

C. The Operator must leave the area and contact Radiation Protection due to dose rate. Dose is currently acceptable.

D. The Operator must leave the area and contact Radiation Protection due to dose. Dose rate is currently acceptable.

Proposed Answer: D

Explanation: The RWP has a dose limit of 20 mRem and a dose rate limit of 80 mRem/hr. The Operator's current dose of 17 mRem is below the 20 mRem dose limit, but above the backoff limit (80% of 20 mRem = 16 mRem) required by RP-AA-1008. This requires the Operator to leave the area and contact RP. The Operator's current dose rate of 72 mRem/hr is below the 80 mRem/hr dose rate limit, and there is no 80% backoff limit imposed on dose rate.

A. Incorrect – The Operator's current dose of 17 mRem is below the dose limit, but above the backoff dose of 16 mRem. This requires the Operator to leave the area and contact RP.
B. Incorrect – The Operator's current dose rate of 72 mRem/hr is below the 80 mRem/hr dose rate limit. Plausible because dose rate is >80% of the limit, which is the threshold that requires exiting the area for dose, but not dose rate.
C. Incorrect – The Operator's current dose rate of 72 mRem/hr is below the 80 mRem/hr dose rate limit. Plausible because dose rate is >80% of the limit, which is the threshold that requires exiting the area for dose, but not dose rate.

Technical Reference(s): RP-AA-1008

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2015 NRC #75

Question History: NMP1 2015 NRC #75

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 #	2.4.26
	Importance Rating	3.1

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

Proposed Question: #72

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

- The Electric Fire Pump unexpectedly starts.
- AOP-51, Unexpected Fire Pump Start, is entered.
- Investigation reveals a significant fire main leak has developed in the North Cable Tunnel.

Which one of the following describes the required operator action and the maximum allowed time for completing this action, in accordance with AOP-51?

A. Close a manual valve to isolate the leak within a maximum of 5 minutes.

B. Shutdown all Fire pumps to stop the leak within a maximum of 5 minutes.

C. Close a manual valve to isolate the leak within a maximum of 20 minutes.

D. Shutdown all Fire pumps to stop the leak within a maximum of 20 minutes.

Proposed Answer: C

Explanation: The immediate actions of AOP-51 require closing manual valve 76FPS-295 within a maximum of 20 minutes.

A. Incorrect – The maximum allowed time is 20 minutes. Plausible because AOP-51 immediate actions have a 5 minute requirement for commencing action to investigate for leakage.
B. Incorrect – The required action is the close manual valve 76FPS-295. Plausible because shutting down all Fire pumps would also stop the leak and a Fire pump has unexpectedly started. Also plausible that manual valve isolation would not be possible due to leak in a tunnel/corridor vs. a discrete room. The maximum allowed time is 20 minutes. Plausible because AOP-51 immediate actions have a 5 minute requirement for commencing action to investigate for leakage.
D. Incorrect – The required action is the close manual valve 76FPS-295. Plausible because shutting down all Fire pumps would also stop the leak and a Fire pump has unexpectedly started. Also plausible that manual valve isolation would not be possible due to leak in a tunnel/corridor vs. a discrete room.

Technical Reference(s): AOP-51

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP

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 #	2.4.14
	Importance Rating	3.8

Knowledge of general guidelines for EOP usage.

Proposed Question: #73

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

Time (minutes)	Event
0	• The Reactor is manually scrammed. • Reactor water level reaches a low of 135" before beginning to recover. • EOP-2, RPV Control, is entered.
1	• Both Feedwater pumps trips. • Reactor water level reaches a high of 165" before beginning to lower.
3	• HPCI is manually started after failing to automatically start. • Reactor water level reaches a low of 58" before beginning to recover.
5	• A steam leak has developed in the Drywell. • Drywell pressure is 3 psig and slowly rising.

Which one of the following describes the required ***re-entry*** into EOP-2, if any?

A. NO re-entry is required.

B. EOP-2 must be re-entered once, at time 3 minutes, only.

C. EOP-2 must be re-entered once, at time 5 minutes, only.

D. EOP-2 must be re-entered twice, at both times 3 and 5 minutes.

Proposed Answer: C

Explanation: When an EOP entry condition is exceeded, the associated EOP is entered and all legs are entered concurrently. When an additional EOP entry condition is exceeded, the entire EOP is re-entered and all legs are re-entered. After the initial EOP entry at time 0 minutes, the Reactor water level entry condition never clears (level never rises above 177”). Therefore, when level lowers again (including hitting the ECCS setpoint at 59.5”), no re-entry to EOP-2 is required based on this parameter. Re-entry to EOP-2 is required at time 5 minutes when Drywell pressure has exceeded 2.7 psig.

A. Incorrect – Re-entry to EOP-2 is required at time 5 minutes when Drywell pressure has exceeded 2.7 psig. Plausible because EOP-2 is never exited because Reactor water level doesn’t rise above 177”.
B. Incorrect – Re-entry is NOT required at 3 minutes. Plausible because Reactor water level has lowered below another important threshold. Re-entry to EOP-2 is required at time 5 minutes when Drywell pressure has exceeded 2.7 psig. Plausible because EOP-2 is never exited because Reactor water level doesn’t rise above 177”.
D. Incorrect – Re-entry is NOT required at 3 minutes. Plausible because Reactor water level has lowered below another important threshold. Also plausible because this would be correct if Reactor water level rose above 177” prior to time 3 minutes.

Technical Reference(s): EOP-2, EP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11B

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 #	3
	Group #	
	K/A #	2.4.3
	Importance Rating	3.7

Ability to identify post-accident instrumentation.

Proposed Question: #74

A Control Room evacuation has been performed.

Given the following plant parameters:

(1) Drywell pressure
(2) SRV tailpipe temperature
(3) Torus water temperature

Which one of the following identifies which of these parameters can be monitored on installed indicators at the Remote Shutdown Panel?

A. (2) only

B. (3) only

C. (2) and (3), only

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

Proposed Answer: B

Explanation: The Remote Shutdown Panel has an installed indicator for Torus water temperature, but not for Drywell pressure or SRV tailpipe temperature.

A. Incorrect – The Remote Shutdown Panel has an installed indication of Torus water temperature. Plausible because Remote Shutdown Panel indication is limited. The Remote Shutdown Panel does not have installed indication of SRV tailpipe temperature. Plausible because SRVs are opened from outside the Control Room during a Control Room evacuation.
C. Incorrect – The Remote Shutdown Panel does not have installed indication of SRV tailpipe temperature. Plausible because SRVs are opened from outside the Control Room during a Control Room evacuation.
D. Incorrect – The Remote Shutdown Panel does not have installed indication for Drywell pressure. Plausible because this is a requires post-accident instrument in the Control Room per Technical Specifications. The Remote Shutdown Panel does not have installed indication of SRV tailpipe temperature. Plausible because SRVs are opened from outside the Control Room during a Control Room evacuation.

Technical Reference(s): ST-43I

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – SSES LOC27 NRC #73

Question History: SSES LOC27 NRC #73

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 #	2.2.13
	Importance Rating	4.1

Knowledge of tagging and clearance procedures.

Proposed Question: #75

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

- An inspection must be performed on a 120 VAC electrical panel.
- The Electricians performing the inspection need to be able to open the panel's disconnect switch for personnel protection during some portions of the inspection.
- The Electricians also need to be able to close the panel's disconnect switch for periodic verifications.
- It is desired for the Tagout to be continuously hung during the activity, so repeated tag clearing and re-hanging is NOT required.

Which one of the following describes the tagging arrangement needed to support the requested maintenance activity, in accordance with EN-OP-102, Protective and Caution Tagging?

Tag the electrical panel's disconnect switch with…

A. a Danger Tag, only.

B. a Caution Tag, only.

C. a Test and Maintenance Tag with a Lockout Device, only.

D. both a Danger Tag and a Test and Maintenance Tag, simultaneously.

Proposed Answer: C

Explanation: A Test and Maintenance Tag has equal authority to a Danger Tag to provide personnel protection, but also allows the tagged component to be manipulated without clearing the tag, as required by this maintenance activity. During periods when personnel protection is required, the Lockout Device must be applied.

A. Incorrect – A Danger Tag would not allow re-positioning the disconnect switch without clearing the tag. Plausible because the Danger Tag does provide the personnel protection required.
B. Incorrect – A Caution Tag does not provide the required personnel protection. Plausible because the Caution Tag would allow repositioning the disconnect switch without clearing the tag.
D. Incorrect – A Danger Tag and Test and Maintenance Tag are not allowed to be placed on the same component at the same time. Plausible because the Danger Tag does provide the personnel protection required and the Test and Maintenance Tag is used to allow manipulation of a tagged component.

Technical Reference(s): EN-OP-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – PB 2017 NRC #68

Question History: PB 2017 NRC #68

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295028 EA2.05
	Importance Rating	3.8

High Drywell Temperature

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/suppression chamber pressure: Plant-Specific

Proposed Question: #76

A loss of coolant accident has resulted in the following:

- Reactor water level is 100” and slowly rising.
- Reactor pressure is 700 psig and slowly lowering.
- Drywell pressure is 6 psig and slowly rising.
- Drywell temperature is 220^{o}F and slowly rising.
- Torus pressure is 5 psig and slowly rising.
- Torus water level is 13.5 feet and stable.
- No actions have yet been directed from EOP-4, Primary Containment Control.

<u>**Note:**</u> The Drywell Spray Initiation Limit and Pressure Suppression Pressure curves are provided on the following page.

Which one of the following identifies the ***current*** procedural direction regarding Torus and Drywell spray, in accordance with EOP-4?

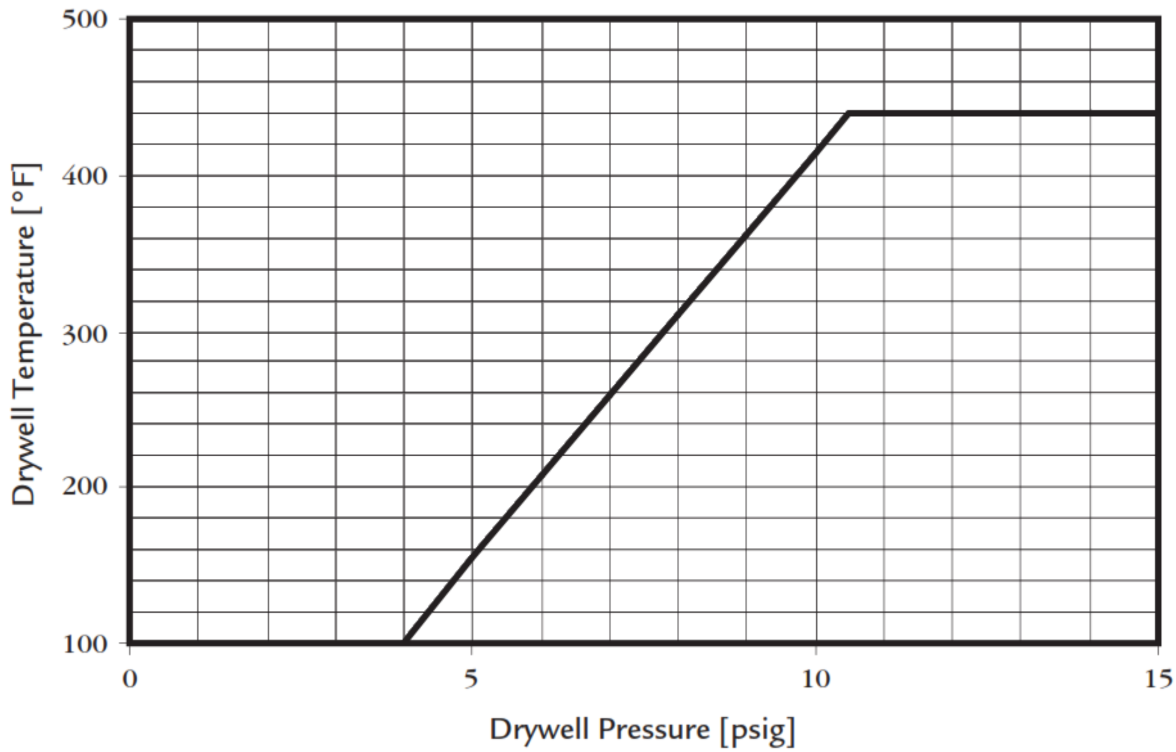
A. NEITHER Torus spray NOR Drywell spray are allowed.

B. Torus spray is allowed, but Drywell spray is NOT.

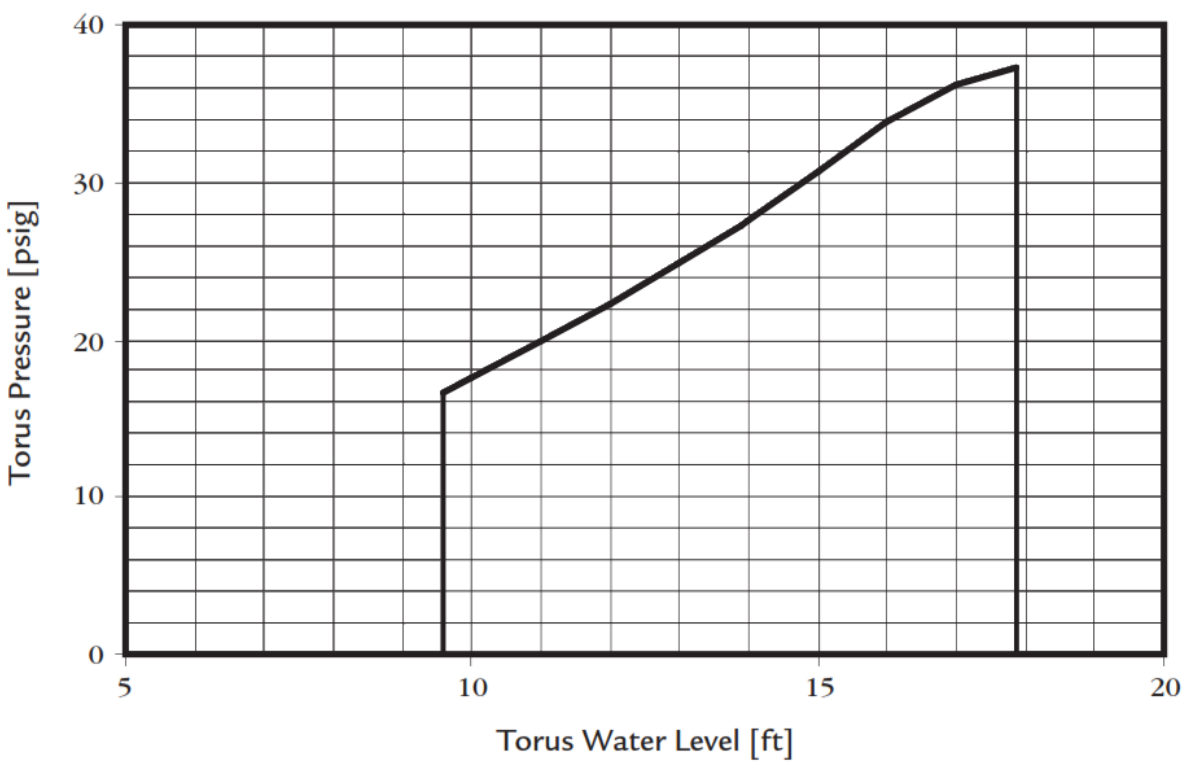
C. Torus spray is NOT allowed, but Drywell spray is.

D. Both Torus spray and Drywell spray are allowed.

Drywell Spray Initiation Limit



Pressure Suppression Pressure



Proposed Answer: B

Explanation: With Torus pressure >2.7 psig, Torus spray is allowed. The given combination of Drywell temperature and Drywell pressure are above the Drywell Spray Initiation Limit, therefore Drywell spray is NOT allowed.

Note: The question meets SRO level guidelines because it requires assessment of plant conditions (multiple Containment parameters) and then selection of a procedure subsection (Torus spray, Drywell spray) with which to proceed. Additionally, the question cannot be answered solely based on systems knowledge, immediate actions, entry conditions, or RO-level knowledge of overall strategy.

A. Incorrect – Torus spray is allowed. Plausible because Torus pressure is still below the 15 psig deadline for initiating Torus spray and the Drywell Spray Initiation Limit curve is exceeded (just does not also apply to Torus spray).
C. Incorrect – Torus spray is allowed. Plausible because Torus pressure is still below the 15 psig deadline for initiating Torus spray and the Drywell Spray Initiation Limit curve is exceeded (just does not also apply to Torus spray). Drywell spray is not allowed. Plausible because at higher Drywell pressures, Drywell spray would be allowed with the given Drywell temperature.
D. Incorrect – Drywell spray is not allowed. Plausible because at higher Drywell pressures, Drywell spray would be allowed with the given Drywell temperature.

Technical Reference(s): EOP-4, EOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

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 #	295019 AA2.01
	Importance Rating	3.6

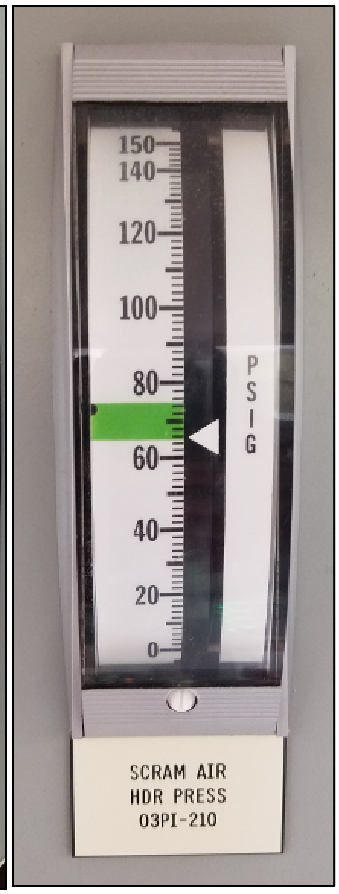
Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure

Proposed Question: #77

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

- An air leak develops in the Reactor Building.
- Operators have located the leak and are attempting to isolate it.
- Air pressure indications on Control Room Panels 09-5 and 09-6 are as shown in the following pictures, and have been slowly lowering since the leak developed:



Which one of the following describes the need for a manual Reactor scram, in accordance with AOP-12, Loss of Instrument Air, and/or Alarm Response Procedures?

A manual Reactor scram is…

A. required now due to the Scram Air Header pressure indication.

B. required now due to the Instrument Air Header pressure indication.

C. NOT required now, but will be **IF** annunciator 09-5-2-3, ROD DRIFT, alarms with one control rod drifting.

D. NOT required now, but will be **IF** annunciator 09-5-1-54, SCRAM AIR HDR PRESS HI OR LO, alarms low.

Proposed Answer: C

Explanation: No procedure (ARP or AOP) currently requires a manual Reactor scram based on the given pressures. AOP-12 requires a manual Reactor scram if any control rod drifts.

A. Incorrect – No requirement exists to scram based on the current Scram Air Header pressure. Plausible because Scram Air Header pressure is low and about to alarm, and many other plants scram on Scram Air Header pressure alarming low.
B. Incorrect – No requirement exists to scram based on the current Instrument Air Header pressure. Plausible because Instrument Air Header pressure is very low (below all automatic action setpoints) and many plants have a requirement to scram at a certain pressure.
D. Incorrect – No requirement exists to scram based on the Scram Air Header pressure alarm. Plausible because many other plants scram on Scram Air Header pressure alarming low, which is a precursor to drifting control rods.

Technical Reference(s): ARP 09-5-1-54, ARP 09-5-2-3, AOP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

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 #	295005 AA2.05
	Importance Rating	3.9

Main Turbine Generator Trip

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor power

Proposed Question: #78

The plant is operating at 20% power during a startup with the following:

- Ten minutes ago, the Main Generator was synchronized to the grid and loaded to 60 MWe per OP-11A, Main Generator, Transformers, and Isolated Bus Phase Cooling.
- No subsequent steps in the startup have been performed yet.
- Then, a Main Turbine trip occurs.
- The Reactor does NOT scram.
- Reactor power begins to slowly rise.

Which one of the following describes whether this is the expected Reactor response to the Main Turbine trip and a required action to be directed, in accordance with the AOPs and EOPs?

This is…

A. the expected Reactor response. Enter AOP-2, Main Turbine Trip Without Scram, and direct a manual Reactor scram.

B. the expected Reactor response. Enter AOP-2, Main Turbine Trip Without Scram, and direct a Reactor power reduction. A manual Reactor scram is NOT required.

C. NOT the expected Reactor response. Enter AOP-32, Unplanned Power Change, to determine the cause of the power rise. A manual Reactor scram is NOT required.

D. NOT the expected Reactor response. Enter EOP-2, RPV Control, and direct a manual Reactor scram. If the Reactor still does NOT scram, transition to EOP-3, Failure to Scram.

Proposed Answer: B

Explanation: This is the proper Reactor response. The Reactor does NOT scram on the Main Turbine trip because Reactor power is below 29%. Reactor power rises due to the loss of Feedwater heating on the Main Turbine trip. AOP-2 entry is required. The subsequent actions of AOP-2 require directing a Reactor power reduction due to the expected rise in power due to loss of Feedwater heating.

A. Incorrect – While AOP-2 does require responding to the Reactor power rise that results from the loss of Feedwater heating, it requires a power reduction, but NOT a Reactor scram. A Reactor scram would become required if appropriate action was not taken and Reactor power reached 29%.
C. Incorrect – This is the expected response. Plausible because the reason for the Reactor power rise is not obvious and under other condition, a Reactor scram would be expected. AOP-32 entry is required, although the cause of the power rise is already known to be loss of Feedwater heating.
D. Incorrect – This is the expected response. Plausible because the reason for the Reactor power rise is not obvious and under other condition, a Reactor scram would be expected. Also plausible because EOP-2 entry would only be required if the Reactor was supposed to scram.

Technical Reference(s): AOP-2, AOP-32, OP-65, EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.02

Question Source: Bank – 3/14 NRC #78

Question History: 3/14 NRC #78

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 #	295038 2.4.41
	Importance Rating	4.6

High Off-site Release Rate

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #79

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

- Significant fuel damage occurs.
- The Reactor is scrammed.
- The MSIVs fail to isolate the Main Steam Lines.
- The following radiation monitors have indicated upscale for the past 30 minutes:
  - Turbine Building Operating Floor (HP End) area radiation monitor.
  - Turbine Building Operating Floor (LP End) area radiation monitor.
  - Stack ventilation low range radiation monitor.
- Stack ventilation high range radiation monitors have indicated 1050 mR/hr and stable for the past 30 minutes.
- Field survey teams have completed multiple surveys off-site.
- The surveys found the highest closed window dose rate of 80 mR/hr at the intersection of State Route 104 and County Route 29.
- The Offsite Dose Assessment team expects this release rate to continue for the next hour.

Which one of the following identifies the highest Emergency Action Level that is met or exceeded, in accordance with IAP-2, Classification of Emergency Conditions?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: B

Explanation: The given conditions meet or exceed multiple EALs, including the following:

- Unusual Event AU1.1 due to Stack low range radiation monitors upscale.
- Unusual Event AU2.2 due to Turbine Building ARMs upscale.
- Alert AA1.1 due to Stack high range radiation monitors >116 mR/hr.

No Site Area Emergency or General Emergency EALs are met or exceeded, although AS1.1, AS1.3, AG1.1 and AG1.3 are possible if radiation conditions degrade further. Therefore, the highest EAL that is met or exceeded is an Alert.

A. Incorrect – Multiple Unusual Event EALs are exceeded, however the highest EAL that is met or exceeded is an Alert.
C. Incorrect – The highest EAL that is met or exceeded is an Alert. Plausible because AS1.1 and AS1.3 are close to being exceeded.
D. Incorrect – The highest EAL that is met or exceeded is an Alert. Plausible because the Stack high range reading is above the 1000 mR/hr threshold that would result in declaration of a General Emergency if it were an offsite radiation level.

Technical Reference(s): IAP-2

Proposed references to be provided to applicants during examination: Hot EAL Matrix (Table F-1 row D removed, Notes removed)

Note: The provided reference must have Table F-1 row D removed to prevent assistance with question 96. The provided reference must have Notes removed to prevent assistance with question 94.

Learning Objective: LP-AOP 1.12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.4.21
	Importance Rating	4.6

Partial or Complete Loss of AC Power

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

Proposed Question: #80

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

- 115 KV System voltage is 114.5 KV and slowly lowering.
- All load tap changers are operating in Auto.
- AOP-72, 115 KV Grid Loss, Instability, or Degradation, is being executed.
- Power Control reports that the Low Voltage Post Contingency Alarm voltage is 111.5 KV.

Which one of the following describes the parameter used by AOP-72 to determine operability of the 115 KV Lines and current operability of the 115 KV Lines based on this parameter, in accordance with AOP-72?

	Parameter Used By AOP-72 To Determine Operability of 115 KV Lines	Current Operability of 115 KV Lines Based On This Parameter
A.	Actual system voltage	Operable
B.	Actual system voltage	Inoperable
C.	Post contingency voltage	Operable
D.	Post contingency voltage	Inoperable

Proposed Answer: C

Explanation: AOP-72 and Technical Specifications base 115 KV Line operability on the status of the post contingency voltage. With load tap changers in Auto and post contingency voltage greater than 109.3 KV, the 115 KV Lines are operable.

A. Incorrect – AOP-72 bases operability on post contingency voltage. Plausible because actual system voltage is what is affecting current plant operation.
B. Incorrect – AOP-72 bases operability on post contingency voltage. Plausible because actual system voltage is what is affecting current plant operation. The 115 KV Line are currently operable. Plausible because actual system voltage and post contingency voltage is below nominal/normal.
D. Incorrect – The 115 KV Line are currently operable. Plausible because actual system voltage and post contingency voltage is below nominal/normal.

Technical Reference(s): AOP-72, Technical Specification 3.8.1 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D 1.16

Question Source: Modified Bank – 9/14 NRC #80

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 #	700000 2.2.37
	Importance Rating	4.6

Generator Voltage and Electric Grid Disturbances

Ability to determine operability and/or availability of safety related equipment.

Proposed Question: #80

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

- 115 KV System voltage is 114.5 KV and slowly lowering.
- All load tap changers are operating in Auto.
- AOP-72, 115 KV Grid Loss, Instability, or Degradation, is being executed.
- Power Control reports that the Low Voltage Post Contingency Alarm voltage is 105 KV.

Which one of the following describes the parameter used by AOP-72 to determine operability of the 115 KV Lines and current operability of the 115 KV Lines based on this parameter, in accordance with AOP-72?

	Parameter Used By AOP-72 To Determine Operability of 115 KV Lines	Current Operability of 115 KV Lines Based On This Parameter
A.	Actual system voltage	Operable
B.	Actual system voltage	Inoperable
C.	Post contingency voltage	Operable
D.	Post contingency voltage	Inoperable

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 2.2.36
	Importance Rating	4.2

Partial or Complete Loss of DC Power

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: #81

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

- Temporary battery charger 71BC-9 is placed in service on Battery Bus B.
- Then, 125 VDC Station Battery Charger 71BC-1B is removed from service for maintenance.
- The associated Battery Bus voltage is 131 VDC and stable.

Which one of the following identifies the need for Condition entry in Technical Specification 3.8.4, DC Sources - Operating, and Technical Specification 3.8.7, Distribution Systems - Operating?

	Technical Specification 3.8.4, DC Sources - Operating	Technical Specification 3.8.7, Distribution Systems - Operating
A.	Condition entry required.	Condition entry required.
B.	Condition entry required.	Condition entry NOT required.
C.	Condition entry NOT required.	Condition entry required.
D.	Condition entry NOT required.	Condition entry NOT required.

Proposed Answer: B

Explanation: TS 3.8.4 requires both the battery terminal voltage ≥127.8 VDC and the 125 VDC battery charger in service and capable of supplying ≥270 amps. While the temporary battery charger may maintain bus voltage, it does not satisfy the requirements of TS 3.8.4. Therefore, TS 3.8.4 Condition A must be entered. TS 3.8.7 remains satisfied because the associated bus is still energized to the appropriate voltage and capable of supplying the required loads.

A. Incorrect – TS 3.8.7 condition entry is not required. Plausible because the current capacity of Battery Bus B is much lower with 71BC-9 in service (150 amps) and entry would be required if voltage lowered as a result.
C. Incorrect – TS 3.8.4 condition entry is required. Plausible because the temporary battery charger is placed in service and is maintaining bus voltage. TS 3.8.7 condition entry is not required. Plausible because the current capacity of Battery Bus B is much lower with 71BC-9 in service (150 amps) and entry would be required if voltage lowered as a result.
D. Incorrect – TS 3.8.4 condition entry is required. Plausible because the temporary battery charger is placed in service and is maintaining bus voltage.

Technical Reference(s): OP-43A, Technical Specifications 3.8.4 and 3.8.7

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B EO-1.16

Question Source: Modified Bank – JAF 16-1 NRC #77

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 # 295004 AA2.02
Importance Rating 3.9

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: Extent of partial or complete loss of D.C. power

Proposed Question: #77

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

- 125 VDC Battery Charger A trips.
- The associated Battery Bus voltage is 121 VDC and lowering slowly.

Which one of the following identifies the need for Condition entry in Technical Specification 3.8.4, DC Sources - Operating, and Technical Specification 3.8.7, Distribution Systems - Operating?

	Technical Specification 3.8.4, DC Sources - Operating		Technical Specification 3.8.7, Distribution Systems - Operating
A.	Condition entry required.		Condition entry required.
B.	Condition entry required.		Condition entry NOT required.
C.	Condition entry NOT required.		Condition entry required.
D.	Condition entry NOT required.		Condition entry NOT required.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295031 2.4.6
	Importance Rating	4.7

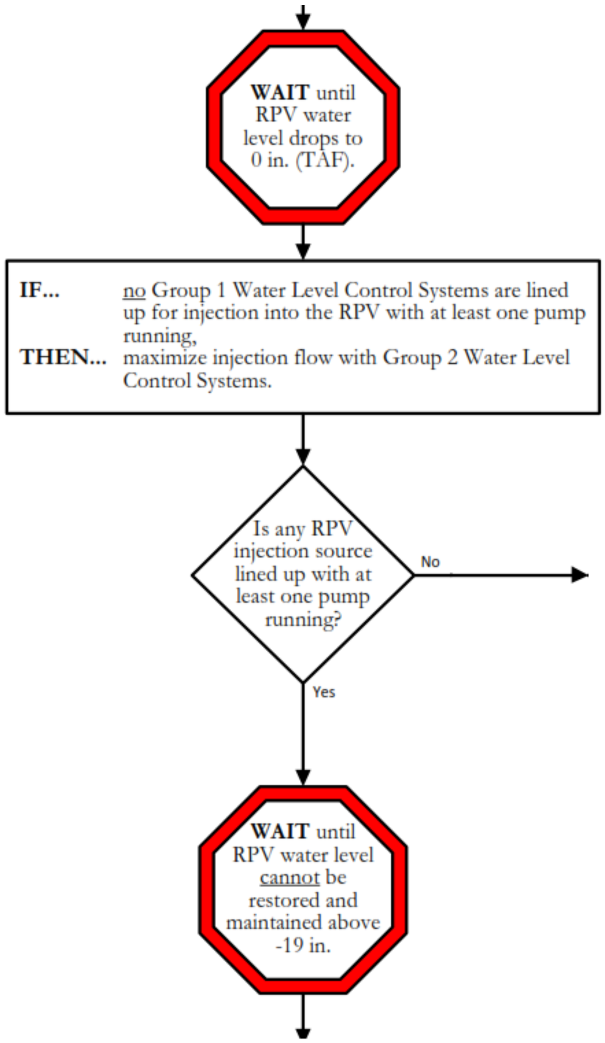
Reactor Low Water Level

Knowledge of EOP mitigation strategies.

Proposed Question: #82

The plant was operating at 100% power when a transient resulted in the following:

- Reactor water level is -10” and slowly lowering.
- Reactor pressure is 850 psig and slowly lowering with an SRV open.
- NO Reactor injection sources are available.
- EOP-2, RPV Control, has been executed up to the decision diamond shown below in the Alternate Level Control leg:



Which one of the following describes how Reactor pressure is now required to be controlled, in accordance with the Emergency Operating Procedures?

A. Stabilize Reactor pressure around the current value.

B. Lower Reactor pressure. Do NOT exceed a cooldown rate of 100ºF/hr.

C. Rapidly lower Reactor pressure. The cooldown rate is allowed to exceed 100ºF/hr.

D. Close the SRV and allow Reactor pressure to rise until SRVs automatically actuate.

Proposed Answer: A

Explanation: With no injection sources and Reactor water level -10” and lowering, EOP-2 requires transitioning from the Alternate Level Control leg to the Steam Cooling leg. This leg requires stabilizing Reactor pressure.

B. Incorrect – Reactor pressure must be stabilized. Plausible because this is the correct strategy in EOP-2 if a low pressure injection source is available and Reactor water level is higher.
C. Incorrect – Reactor pressure must be stabilized. Plausible because this is the correct strategy in EOP-2 if any injection source is available and Reactor water level is -10” and lowering.
D. Incorrect – Reactor pressure must be stabilized. Plausible because closing the SRV will be required, but allowing Reactor pressure to rise to ~1100 psig would violate the requirement to stabilize pressure. Also plausible because this is the strategy that would minimize inventory losses.

Technical Reference(s): EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c EO-1.05

Question Source: Bank - NMP1 2015 NRC #82

Question History: NMP1 2015 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 #	2
	K/A #	295015 AA2.02
	Importance Rating	4.2

Incomplete SCRAM

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: Control rod position

Proposed Question: #83

The plant is operating at 50% power with the following:

- Feedwater pump A trips.
- An Operator places the Reactor Mode Switch in SHUTDOWN.
- All control rods fully insert except:
    - Control rod 18-15 is at position 02.
    - Control rod 18-11 is at position 10.
    - Control rod 22-15 is at position 04.
- All other control rods fully insert.
- APRMs indicate downscale.
- Reactor water level reaches a low of 130" and then is restored to 200".
- Control rods are inserted as follows:

Time (minutes)	Event
3	Control rod 18-15 is inserted from position 02 to 00.
6	Control rod 18-11 is inserted from position 10 to 00.
9	Control rod 22-15 is inserted from position 04 to 00.

Which one of the following describes the proper Emergency Operating Procedure (EOP) execution for this transient?

Enter EOP-2, RPV Control, and…

A. remain in EOP-2. Entry into EOP-3, Failure to Scram, is NOT required.

B. then exit to EOP-3, Failure to Scram. EOP-3 may ***first*** be exited at time 3 minutes.

C. then exit to EOP-3, Failure to Scram. EOP-3 may ***first*** be exited at time 6 minutes.

D. then exit to EOP-3, Failure to Scram. EOP-3 may ***first*** be exited at time 9 minutes.

Proposed Answer: C

Explanation: With Reactor water level below 177”, EOP-2 must be entered. At the beginning of EOP-2, there is a series of diagnostic steps that determine if the Reactor will remain shutdown under all conditions without boron. These steps determine if the operator remains in EOP-2 or transitions to EOP-3. One criteria that allows the operator to determine that the Reactor will remain shutdown under all conditions without boron is if all control rods inserted to or beyond position 02 (defined as the Maximum Subcritical Banked Withdrawal Position at JAF). With two control rods beyond 02, EOP-3 entry is required. The first time EOP-3 may be exited is when only one control rod remains withdrawn past 02 at time 6 minutes. This is based on a second criteria that allows determining the Reactor will remain shutdown under all conditions without boron if all control rods are fully inserted other than one control rod, which may be at any position.

A. Incorrect – With two control rods beyond 02, EOP-3 entry is required. Plausible because this would be the correct answer if all three control rods were stuck at position 02. Also plausible because APRMs are downscale.
B. Incorrect – The first time EOP-3 may be exited is when only one control rod remains withdrawn past 02 at time 6 minutes. Plausible because this would be correct if the order insertion of control rods 18-15 and 18-11 were reversed.
D. Incorrect – The first time EOP-3 may be exited is when only one control rod remains withdrawn past 02 at time 6 minutes. Plausible because this is the first time all control rods are in.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.03

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 #	295034 2.2.25
	Importance Rating	4.2

Secondary Containment Ventilation High Radiation

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: #84

The plant is shutdown for an outage with the following:

- The Reactor Mode Switch is in REFUEL.
- Reactor coolant temperature is 110ºF.
- The Reactor has been shutdown for 80 hours.
- Irradiated fuel is being removed from the Reactor vessel and placed in the Spent Fuel Pool.
- There are NO operations with the potential to drain the Reactor vessel (OPDRVs) in progress.

Which one of the following describes the requirement for operability of the Secondary Containment Isolation on Reactor Building Exhaust High Radiation, in accordance with Technical Specifications?

The Secondary Containment Isolation on Reactor Building Exhaust High Radiation is currently…

A. required to be operable, based on mitigating a refueling accident.

B. required to be operable, based on mitigating a loss of coolant accident.

C. NOT required to be operable, based on the operating mode alone.

D. NOT required to be operable, based on the operating mode combined with the time after shutdown.

Proposed Answer: A

Explanation: Technical Specification Table 3.3.6.2-1 function 2 requires Secondary Containment isolation based on Reactor Building Exhaust High Radiation to be operable in Modes 1, 2, and 3. Notes (a) and (b) also require operability during OPDRVs and during movement of recently irradiated fuel assemblies in Secondary Containment. Technical Specification Bases define recently irradiated fuel assemblies as those that have been in a critical Reactor within 96 hours. Since the time after shutdown is <96 hours, operability is required. The basis for operability in this condition is to mitigate refueling accidents.

B. Incorrect – The basis for operability in this condition is to mitigate refueling accidents. Plausible because operability in Modes 1, 2, and 3 is based on mitigating a loss of coolant accident.
C. Incorrect – Operability is currently required. Plausible because no OPDRVs are in progress, the plant is in Mode 5, and time after shutdown is significant.
D. Incorrect – Operability is currently required. Plausible because no OPDRVs are in progress, the plant is in Mode 5, and time after shutdown is significant.

Technical Reference(s): Technical Specification 3.3.6.2 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.17

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 #	295032 2.4.2
	Importance Rating	4.6

High Secondary Containment Area Temperature

Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: #85

The plant is operating at 50% power with the following:

- The Reactor is manually scrammed due to indications of fuel failure.
- RCIC is manually started.
- RCIC develops a steam leak.
- RCIC automatically isolates based on high RCIC equipment area temperature.
- The Reactor Building isolates on a valid Reactor Building Ventilation high radiation signal.
- Reactor Building Ventilation radiation has been verified to be 3 times (3x) the Reactor Building isolation setpoint.

Given the following EOPs:

(1) EOP-5, Secondary Containment Control
(2) EOP-6, Radioactivity Release Control

Note: A portion of EOP-5, Secondary Containment Control, is provided on the following page.

Based on the given conditions, which one of the following identifies which of these EOPs **must** have entry conditions met, if any?

A. (1) only

B. (2) only

C. Both (1) and (2)

D. Neither (1) nor (2)

REACTOR BUILDING AREA TEMPERATURES

AREA	INSTRUMENT	MAXIMUM NORMAL	MAXIMUM SAFE
Reactor Building 369 ft elevation 66RTD-106 66RTD-108	66TI-106, Panel 09-75 66TI-108, Panel 09-75	104°F	112°F
Outside 'A' LPCI Battery Enclosure 66RTD-115	EPIC Only	104°F	113°F
Below Refuel Floor Exhaust 66RTD-105	66TI-105, Panel 09-75	104°F	113°F
Outside 'B' LPCI Battery Enclosure 66RTD-116	EPIC Only	104°F	113°F
SLC Pump Area 66RTD-114	EPIC Only ❶	104°F	133°F
Fuel Pool Cooling Pump Room 66RTD-113	EPIC Only	104°F	133°F
Reactor Building 300 ft elevation northeast 66RTD-112	EPIC Only ❶	104°F	158°F
RWCU Heat Exchanger Room 12TE-117E 12TE-117F	Panel 09-21 Panel 09-21	115°F	203°F
'B' RWCU Pump Room 12TE-117C 12TE-117D	Panel 09-21 Panel 09-21	135°F	225°F
'A' RWCU Pump Room 12TE-117A 12TE-117B	Panel 09-21 Panel 09-21	125°F	225°F
Reactor Building 300 ft elevation southwest 66RTD-111	EPIC Only ❶	104°F	173°F
'B' RHR Heat Exchanger Room 23RTD-02A 23RTD-02B	23-203A, Panel 09-95 23-203B, Panel 09-96	130°F	242°F
Reactor Building 272 ft elevation southeast 23RTD-02C 23RTD-02D	23-204A, Panel 09-95 23-204B, Panel 09-96	104°F	153°F
HPCI Drywell Entrance 13RTD-102C 13RTD-102D	13-202C, Panel 09-95 13-202D, Panel 09-96	120°F	251°F
RCIC Drywell Entrance 13RTD-102A 13RTD-107B	13-202A, Panel 09-95 13-207B, Panel 09-96	120°F	218°F
Reactor Building 272 ft elevation soutwest 23RTD-01C 23RTD-01D	23-202A, Panel 09-95 23-202B, Panel 09-96	104°F	196°F
'A' RHR Heat Exchanger Room 23RTD-01A 23RTD-01B	23-201A, Panel 09-95 23-201B, Panel 09-96	130°F	242°F
Torus Room - South HPCI Steamline 13RTD-107C 13RTD-107D	13-207C, Panel 09-95 13-207D, Panel 09-96	120°F	280°F
Torus Room - Southwest RCIC Steamline 13RTD-107A 13RTD-102B	13-207A, Panel 09-95 13-202B, Panel 09-96	120°F	280°F
East Crescent 66RTD-109B	66TI-109B, Panel 09-75	104°F	137°F
HPCI Room 23RTD-94A 23RTD-94B 23RTD-117A 23RTD-117B	23-294A, Panel 09-95 23-294B, Panel 09-96 23-217A, Panel 09-95 23-217B, Panel 09-96	104°F	137°F
RCIC Room 13RTD-89A 13RTD-89B	13-289A, Panel 09-95 13-289B, Panel 09-96	104°F	137°F
West Crescent 13RTD-76A 13RTD-76B	13-276A, Panel 09-95 13-276B, Panel 09-96	104°F	137°F

Proposed Answer: A

Explanation: RCIC isolation occurs on a high area temperature of 133°F. This is above the EOP-5 entry condition of 104°F. Reactor Building isolation occurs on Reactor Building Ventilation Exhaust radiation level of $1x10^4$ cpm, therefore Reactor Building Ventilation Exhaust radiation is $3x10^4$ cpm (3x isolation setpoint). This exceeds the EOP-5 entry condition of 10^3 cpm. This also exceeds the value of $2x10^4$ cpm in EAL Table A-1 related to declaration of an Unusual Event, but does not meet the criteria for an Alert ($9.9x10^5$ cpm). Therefore, EOP-6 entry is not required based on the given conditions.

B. Incorrect – EOP-5 entry is required because RCIC area temperature is >104°F and Reactor Building Ventilation Exhaust radiation is $>10^3$ cpm. Plausible because Reactor Building Ventilation Exhaust radiation is only $3x10^4$ cpm, which is well below upscale and well below the level requiring declaration of an Alert. EOP-6 entry is not required because Reactor Building Ventilation Exhaust radiation is $<9.9x10^5$ cpm. Plausible because the UE value is exceeded (therefore elevated offsite release did occur) and EOP-5 entry is required.
C. Incorrect – EOP-6 entry is not required because Reactor Building Ventilation Exhaust radiation is $<9.9x10^5$ cpm. Plausible because the UE value is exceeded (therefore elevated offsite release did occur) and EOP-5 entry is required.
D. Incorrect – EOP-5 entry is required because RCIC area temperature is >104°F and Reactor Building Ventilation Exhaust radiation is $>10^3$ cpm. Plausible because Reactor Building Ventilation Exhaust radiation is only $3x10^4$ cpm, which is well below upscale and well below the level requiring declaration of an Alert.

Technical Reference(s): ARP 09-3-3-2, ARP 09-3-2-40, EOP-5, EOP-6, Hot EAL Matrix

Proposed references to be provided to applicants during examination: Hot EAL Matrix (Table F-1 row D removed; Notes removed)

<u>Note:</u> The provided reference must have Table F-1 row D removed to prevent assistance with question 96. The provided reference must have Notes removed to prevent assistance with question 94.

Learning Objective: MIT-301.11G 6.05

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 #	259002 A2.03
	Importance Rating	3.7

Reactor Water Level Control

Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of reactor water level input

Proposed Question: #86

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

- Feedwater level control is operating in three-element control.
- 06LT-52A, the Narrow Range Reactor water level transmitter that is currently selected to Feedwater Level Control, fails downscale.
- AOP-41, Feedwater Malfunction, is entered.
- Feedwater level control is placed in MAN and Reactor water level is stabilized in the normal band.

Which one of the following describes:

- the initial direction of the actual Reactor water level change due to the level transmitter failure, and
- the Technical Specification impact of the level transmitter failure?

(1) Actual Reactor water level initially…

(2) Technical Specification condition entry is…

A. (1) rises.
(2) required.

B. (1) rises.
(2) NOT required.

C. (1) lowers.
(2) required.

D. (1) lowers.
(2) NOT required.

Proposed Answer: A

Explanation: With 06LT-52A failing downscale, the Feedwater level control system senses that Reactor water level lowering. Therefore, the Feedwater level control system raises actual Feedwater flow, causing actual Reactor water level to rise. This instrument and two others input to the Feedwater and Main Turbine High Water Level Trip circuitry (06LT-52A(B)(C)). Technical Specification 3.3.2.2 requires all three of these instruments to be operable. With 06LT-52A failed downscale, it is incapable of causing a high level trip, and therefore must be declared inoperable. This requires entry into Technical Specification 3.3.2.2 Condition A.

B. Incorrect – Part (2): Technical Specification 3.3.2.2 Condition A must be entered. Plausible because there are still 2 operable instruments, such that trip capability is maintained. Also plausible because this would be correct for the opposite failure.
C. Incorrect – Part (1): Actual Reactor water level initially rises. Plausible because indicated level rises and because this would be correct for the opposite failures.
D. Incorrect – Part (1): Actual Reactor water level initially rises. Plausible because indicated level rises and because this would be correct for the opposite failures. Part (2): Technical Specification 3.3.2.2 Condition A must be entered. Plausible because there are still 2 operable instruments, such that trip capability is maintained. Also plausible because this would be correct for the opposite failure.

Technical Reference(s): OP-2A, OP-27A, FM-47A, Technical Specification 3.3.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-06 1.17.a.6

Question Source: Modified Bank – 9/14 NRC #86

Question History:

Question Cognitive Level: Comprehension or Analysis

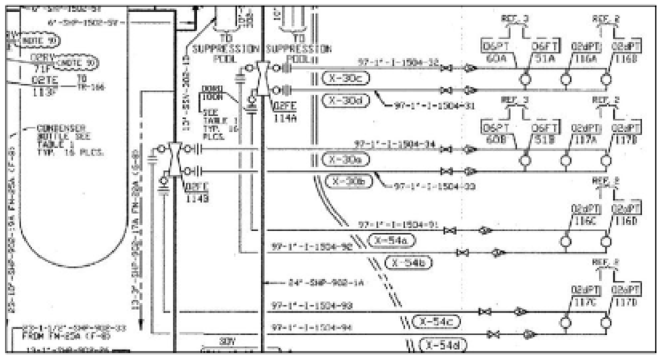
10 CFR Part 55 Content: 55.43(2)

Comments:

Proposed Question: #86

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

- Feedwater level control is operating in three-element control.
- 06FT-51A, the Main Steam Line A flow transmitter that inputs to Feedwater Level Control, fails downscale (shown in a portion of FM-29A below).
- AOP-41, Feedwater Malfunction, is entered.
- Feedwater level control is placed in MAN and Reactor water level is stabilized in the normal band.



Which one of the following describes:

(1) the initial direction of the Reactor water level change due to the steam flow transmitter failure, and

(2) the Technical Specification impact of the steam flow transmitter failure?

(1) Reactor water level…
(2) Technical Specification condition entry is…

A. (1) rises.
(2) required.

B. (1) rises.
(2) NOT required.

C. (1) lowers.
(2) required.

D. (1) lowers.
(2) NOT required.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	218000 A2.05
	Importance Rating	3.6

ADS

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. or D.C. power to ADS valves

Proposed Question: #87

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is 57" and stable.
- Reactor pressure is 650 psig and lowering very slowly.
- Drywell pressure is 10 psig and rising very slowly.
- All RHR and Core Spray pumps are running.
- ADS has NOT been overridden.
- The following annunciators came into alarm 10 seconds ago:
    - 09-4-1-17, ADS AUX RELAY ENERGIZED
    - 09-4-1-28, ADS TIMERS ACTUATED

Then, 125 VDC Bus A de-energizes due to a sustained electrical fault.

Which one of the following describes:

(1) the response of the ADS valves if the current conditions continue for 2 more minutes, and

(2) the required control of ADS under these conditions, in accordance with EOP-2, RPV Control?

	(1) If current conditions continue, ADS valves will…	(2) EOP-2 requires…
A.	open	overriding ADS.
B.	open	allowing ADS valves to automatically open.
C.	remain closed	overriding ADS.
D.	remain closed	manually opening ADS valves now.

Proposed Answer: A

Explanation: A valid ADS initiation signal exists due to Reactor water level <59.5". 125 VDC Bus A normally supplies power to the SRV ADS solenoids and ADS logic sub-channels A and C. The SRV ADS solenoids receive automatic backup power from 125 VDC Bus B. Logic sub-channels A and C are inoperable due to the loss of 125 VDC Bus A. However, logic sub-channels B and D are still operable and capable of initiating ADS without sub-channels A and C. Therefore, ADS valves will open despite the two impacts from loss of 125 VDC Bus A. EOP-2 requires overriding ADS in this situation.

B. Incorrect – EOP-2 requires overriding ADS in this situation. Plausible because Reactor water level is low, Reactor pressure is above the discharge head of ECCS pumps, and ECCS pumps are available.
C. Incorrect – ADS valves will open. Plausible because 125 VDC Bus A normally supplies power to the SRV ADS solenoids and ADS logic sub-channels A and C.
D. Incorrect – ADS valves will open. Plausible because 125 VDC Bus A normally supplies power to the SRV ADS solenoids and ADS logic sub-channels A and C. EOP-2 requires overriding ADS in this situation. ADS valves are only manually opened if Reactor water level continues to lower to 0". Plausible because Reactor water level is low, Reactor pressure is above the discharge head of ECCS pumps, and ECCS pumps are available.

Technical Reference(s): AOP-45, EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J

Question Source: Modified Bank - 16-1 NRC #87

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 #	218000 A2.06
	Importance Rating	4.3

ADS

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS initiation signals present

Proposed Question: #87

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is 59" and lowering slowly.
- Reactor pressure is 650 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- All RHR pumps have failed to start.
- Core Spray pump A has failed to start.
- Core Spray pump B is operating properly.
- ADS has NOT been overridden.
- The following annunciators came into alarm 10 seconds ago:
    - 09-4-1-17, ADS AUX RELAY ENERGIZED
    - 09-4-1-28, ADS TIMERS ACTUATED

Which one of the following describes:

(1) the response of the ADS valves if the current conditions continue until the ADS timers time out, and

(2) the required control of ADS under these conditions, in accordance with EOP-2, RPV Control?

	(1) If current conditions continue, ADS valves will…	(2) EOP-2 requires…
A.	open	overriding ADS.
B.	open	allowing ADS valves to open.
C.	remain closed	overriding ADS.
D.	remain closed	manually opening ADS valves.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	400000 2.1.25
	Importance Rating	4.2

Component Cooling Water

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: #88

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

- Thunderstorms with high winds are in progress.
- Wind speeds on site have been averaging 40 mph with gusts up to 75 mph.
- Computer point EPIC-A-1094, SCRNWL WL, alarms.
- Lake water level is 245'.
- Heavy wave action has resulted in an accumulation of debris in the intake.
- ESW intake bay water level is currently 237.3' and slowly lowering.
- ESW intake bay water level has dropped 5' in the past 10 minutes.

Which one of the following describes if an Emergency Action Level (EAL) is met or exceeded and the required action in accordance with AOP-56, Intake Water Level Trouble?

	Has an EAL been met or exceeded?	Required Action
A.	Yes	Perform a power reduction. A scram is NOT required.
B.	Yes	Scram the Reactor.
C.	No	Perform a power reduction. A scram is NOT required.
D.	No	Scram the Reactor.

Proposed Answer: D

Explanation: Since wind speeds are <90 mph and ESW intake bay water level is >237’, no EAL has been met or exceeded. Since intake level is <237.5’, AOP-56 requires scramming the Reactor.

A. Incorrect – Since wind speeds are <90 mph and ESW intake bay water level is >237’, no EAL has been met or exceeded. Plausible because if wind speeds rise higher or ESW intake bay water level lowers more, an EAL will be met or exceeded. A Reactor scram is required. Plausible because if intake water level was above 237.5’, then a power reduction would be required without a scram.
B. Incorrect – Since wind speeds are <90 mph and ESW intake bay water level is >237’, no EAL has been met or exceeded. Plausible because if wind speeds rise higher or ESW intake bay water level lowers more, an EAL will be met or exceeded.
C. Incorrect – A Reactor scram is required. Plausible because if intake water level was above 237.5’, then a power reduction would be required without a scram.

Technical Reference(s): IAP-2, AOP-56

Proposed references to be provided to applicants during examination: Hot EAL Matrix (Table F-1 row D removed; Notes removed)

Note: The provided reference must have Table F-1 row D removed to prevent assistance with question 96. The provided reference must have Notes removed to prevent assistance with question 94.

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 #	262001 2.2.37
	Importance Rating	4.6

AC Electrical Distribution

Ability to determine operability and/or availability of safety related equipment.

Proposed Question: #89

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

Time and Date	Condition
1300 on 5/1	Emergency Diesel Generator (EDG) A is declared inoperable.
1600 on 5/1	The Nine Mile Point Unit 1 Control Room reports that they have experienced a loss of 115KV Line 1, South Oswego to NMP1.
1700 on 5/1	EDG B is declared inoperable.

Which one of the following identifies the **latest** time that the plant must be in Mode 3 if NONE of these issues are corrected, in accordance with Technical Specifications?

A. 0400 on 5/2

B. 0600 on 5/2

C. 0700 on 5/2

D. 1600 on 5/2

Proposed Answer: B

Explanation: The loss of EDG A results in entry to TS 3.8.1 Condition B, which allows up to 14 days to restore EDG operability. The report from NMP1 results in the need to declare Line 4 inoperable because Line 1 connects Line 4 to the rest of the offsite 115 KV power grid. This results in entry into TS 3.8.1 Condition D (one EDG and one Line inoperable), which allows 12 hours to restore operability before entering Condition F, which gives another 12 hours before needing to be in Mode 3. The loss of EDG B results in entry to TS 3.8.1 Condition G, which requires entering LCO 3.0.3 immediately, and then being in Mode 3 within 13 hours. This is the limiting Condition, and results in a need to be in Mode 3 by 0600 on 5/2 (1700 on 5/1 + 13 hours).

A. Incorrect – The plant must be in Mode 3 by 0600 on 5/2 at the latest. Plausible because this is the time if the 12 hours from Condition D is used without the additional 13 hours allowed by LCO 3.0.3 (1600 on 5/1 + 12 hours).
C. Incorrect – The plant must be in Mode 3 by 0600 on 5/2 at the latest. Plausible because this is the time if Condition E is applied but Condition G is missed (1700 on 5/1 + 2 hours +12 hours).
D. Incorrect – The plant must be in Mode 3 by 0600 on 5/2 at the latest. Plausible because this is the time if Condition D is applied but Conditions E and G are missed (1600 on 5/1 + 12 hours + 12 hours).

Technical Reference(s): OP-44, Technical Specification 3.8.1 and 3.0.3

Proposed references to be provided to applicants during examination: Technical Specification 3.8.1

Learning Objective: SDLP-71D 1.18

Question Source: Modified Bank - 2010 NRC #77

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

QUESTION 77.

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

1300 on 5/03/10: 'A' Emergency Diesel Generator is declared INOPERABLE
1600 on 5/03/10: Line 4 offsite AC circuit is declared INOPERABLE
1700 on 5/03/10: 'B' Emergency Diesel Generator is declared INOPERABLE

When will the plant be required to enter MODE 3, in accordance with Technical Specifications?

A.	0500 on 5/04/10
B.	0600 on 5/04/10
C.	0700 on 5/04/10
D.	1600 on 5/04/10

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	203000 2.2.40
	Importance Rating	4.7

RHR/LPCI: Injection Mode

Ability to apply technical specifications for a system.

Proposed Question: #90

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

Time and Date	Event
0800 on 5/1	HPCI is taken out of service for corrective maintenance.
0800 on 5/4	Annunciator 09-3-1-3, RHR A VLV OVERLOAD OR PWR LOSS, alarms. Investigation reveals the breaker for 10MOV-25A, LPCI INBD INJ VLV, has spuriously tripped and CANNOT be reclosed.

Which one of the following identifies the <u>**latest**</u> time by which the plant must be in Mode 3 if NEITHER of these issues are corrected, in accordance with Technical Specifications?

A. 2100 on 5/4

B. 2000 on 5/7

C. 2000 on 5/11

D. 2000 on 5/15

Proposed Answer: B

Explanation: 10MOV-25A is normally closed and must open for either RHR pumps A or C to inject. Technical Specifications define this combination of pumps as a single subsystem. Therefore, with 10MOV-25A closed with no power, one subsystem of RHR is inoperable. Combined with HPCI being inoperable, this requires entry into Technical Specification 3.5.1 Condition D, which requires restoring either HPCI or RHR to operable within a maximum of 72 hours. If this does not occur, Condition G must be entered, which requires being in Mode 3 within a maximum of 12 hours. This results in the latest time by which the plant must be in Mode 3 being 2000 on 5/7 (0800 on 5/4 + 72 hours + 12 hours).

A. Incorrect – The latest time by which the plant must be in Mode 3 is 2000 on 5/7. Plausible because this is the answer if the 10MOV-25A failure resulted in two inoperable subsystems (0800 on 5/4 +13 hours).
C. Incorrect – The latest time by which the plant must be in Mode 3 is 2000 on 5/7. Plausible because this is the answer if HPCI were not inoperable (0800 on 5/4 + 7 days + 12 hours).
D. Incorrect – The latest time by which the plant must be in Mode 3 is 2000 on 5/7. Plausible because this is the answer if the initial HPCI inoperability were limiting (0800 on 5/1 + 14 days + 12 hours).

Technical Reference(s): Technical Specifications 3.5.1 and 3.0.3, FM-20A

Proposed references to be provided to applicants during examination: Technical Specification 3.5.1

Learning Objective: SDLP-10 1.17

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 #	201006 A2.01
	Importance Rating	2.8

RWM

Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply loss: P-Spec(Not-BWR6)

Proposed Question: #91

A plant startup is in progress with the following:

- Only the first eight (8) control rods have been withdrawn.
- Power is lost to the Rod Worth Minimizer (RWM).
- The RWM keylock switch is placed in BYPASS.
- The last startup performed with an inoperable RWM was on January 15, 2017.
- Today is May 1, 2018.

Which one of the following describes the ability to continue with the startup, in accordance with Technical Specifications?

The startup…

A. may continue with NO additional administrative requirements.

B. CANNOT continue until the RWM is restored to operable status due to the number of rods withdrawn.

C. CANNOT continue until the RWM is restored to operable status due to the date of the last startup with an inoperable RWM.

D. may continue if a second licensed operator or qualified individual is stationed to verify compliance with the approved rod withdrawal sequence.

Proposed Answer: D

Explanation: Loss of power to the RWM results in the RWM being inoperable. Technical Specification 3.3.2.1 Condition C provides the requirements for startups with an inoperable RWM. First, one of two conditions must be verified – either ≥12 rods are withdrawn (not met in this case) or that a startup has not been performed with an inoperable RWM in the current calendar year (met in this case). Since one of these requirements is met, the startup may continue, but only if a second licensed operator or qualified individual is stationed to verify rod movements are in accordance with the approved sequence.

<u>Note:</u> The question focuses on only the second half of the K/A since the first half is obvious (power supply loss results in inoperability) and cannot be readily tested at the SRO level. This is in accordance with NUREG 1021 ES-401 D.2.a.

A. Incorrect – The startup may continue, but only if a second licensed operator or qualified individual is stationed to verify rod movements are in accordance with the approved sequence. Plausible if the candidate believes this is not required since a startup hasn't been performed with an inoperable RWM in the current calendar year (confuses logical connectors).
B. Incorrect – The startup may continue. Plausible because less than 12 rods have been withdrawn.
C. Incorrect – The startup may continue. Plausible because a startup with an inoperable RWM has been performed relatively recently (within the length of a typical fuel cycle).

Technical Reference(s): Technical Specification 3.3.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.18

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 #	272000 2.4.45
	Importance Rating	4.3

Radiation Monitoring System

Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: #92

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

- Annunciator 09-4-0-01, CONT HI RANGE RAD MON A FAIL OR PWR LOSS, alarms.
- Investigation reveals that Containment high range radiation monitor A has failed downscale.

Which one of the following describes the need for Technical Specification condition entry, if any?

A. Technical Specification condition entry is required.

B. NO Technical Specification condition entry is required because this radiation monitor is only required by the TRM.

C. NO Technical Specification condition entry is required because this radiation monitor is only required by the ODCM.

D. NO Technical Specification condition entry is required because this radiation monitor is only required by the Emergency Plan.

Proposed Answer: A

Explanation: Both Technical Specifications 3.3.3.1 and 3.3.6.1 are impacted by Containment high range radiation monitor A. Technical Specification 3.3.6.1 condition entry is not required because this radiation monitor still provides its protective function when failed downscale. Technical Specification 3.3.3.1 condition entry is required because both Containment high range radiation monitors A and B are required to be operable per Table 3.3.3.1-1.

B. Incorrect – Technical Specification condition entry is required. Plausible because some PAM instrumentation requirements are in the TRM and not Technical Specifications.
C. Incorrect – Technical Specification condition entry is required. Plausible because many radiation monitors are covered by the ODCM and not Technical Specifications.
D. Incorrect – Technical Specification condition entry is required. Plausible because this monitor is required by the Emergency Plan for EAL and fuel damage assessment

Technical Reference(s): ARP 09-4-0-01, Technical Specifications 3.3.3.1 and 3.3.6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.16

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	286000 A2.03
	Importance Rating	3.0

Fire Protection System

Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. distribution failure: Plant-Specific

Proposed Question: #93

The plant is operating at 100% when the following occurs:

- L44 de-energizes due to a ground fault condition.

Which one of the following describes the Fire system pump that loses power and the need for Technical Requirements Manual (TRM) 3.7.H, High Pressure Water Fire Protection System, condition entry?

	Power is lost to…	TRM 3.7.H Condition Entry
A.	76P-2, Motor Driven Fire Pump	Required
B.	76P-2, Motor Driven Fire Pump	NOT required
C.	76P-3, Make Up Pump	Required
D.	76P-3, Make Up Pump	NOT required

Proposed Answer: A

Explanation: L44 supplies power to 76P-2, Motor Driven Fire Pump. 76P-3, Make Up Pump, maintains power from L34. TRM 3.7.H requires the Electric Motor Driven Fire Pump to be restored to functional status within a maximum of 7 days.

B. Incorrect – TRM condition entry is required. Plausible because the Diesel Driven Fire Pump is still fully functional.
C. Incorrect – Power is lost to the Motor Driven Fire Pump, not the Make Up Pump. Plausible because this would be correct for a loss of L34.
D. Incorrect – Power is lost to the Motor Driven Fire Pump, not the Make Up Pump. Plausible because this would be correct for a loss of L34. TRM condition entry is required. Plausible because the Diesel Driven Fire Pump is still fully functional.

Technical Reference(s): OP-33 Attachment 2, TRM 3.7.H

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76 1.03

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.39
	Importance Rating	4.3

Knowledge of conservative decision making practices.

Proposed Question: #94

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

Time (hhmm)	Event
0000	• The Fire Protection Panel (FPP) alarms. • A fire has been detected in the Cable Spreading Room.
0005	• The Fire Brigade Leader reports that there is a significant fire in the Cable Spreading Room. • The Fire Brigade Leader expects the fire will take 30 minutes to extinguish.
0007	• The Shift Manager is analyzing the need to declare the following EAL: **HU2.1** | **1** | **2** | **3** | **4** | **5** | **DEF** | Fire **not** extinguished within 15 minutes of control room notification or receipt of a valid Control Room alarm in **any** Table H-1 area (Note 3)

Which one of the following identifies:

(1) when the 15 minute clock included in this EAL expires

and

(2) the required timing of the EAL declaration,

in accordance with IAP-2, Classification of Emergency Conditions?

	(1) The EAL's 15 minute clock expires at time…	(2) Declaration of the EAL…
A.	0015.	must wait until time 0015.
B.	0015.	may be made before time 0015.
C.	0020.	must wait until time 0020.
D.	0020.	may be made before time 0020.

Proposed Answer: B

Explanation: The 15 minute clock starts at time 0000 due to receipt of a fire alarm (that is not disproven within 15 minutes). EAL bases allow use of a Fire Brigade Leader report to ensure the alarm is valid, but the beginning of the clock is not delayed for this report. EAL Note 3 allows declaration of this EAL before the 15 minute clock expires.

Note: The provided reference for Questions 79, 85, and 88 has the Notes section removed to avoid assisting with this question.

A. Incorrect – The EAL may be declared before 0015 based on Note 3. Plausible because without Note 3 and using general EAL declaration guidance, the EAL would not be declared until the clock expired.
C. Incorrect – The clock expires at time 0015. Plausible because 0020 would be correct if the alarm was not considered valid until it was investigated and the clock was held until such time. The EAL may be declared before the clock expires based on Note 3. Plausible because without Note 3 and using general EAL declaration guidance, the EAL would not be declared until the clock expired.
D. Incorrect – The clock expires at time 0015. Plausible because 0020 would be correct if the alarm was not considered valid until it was investigated and the clock was held until such time.

Technical Reference(s): IAP-2

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: 55.43(1)

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: #95

A Temporary Change is being processed for an operating procedure per AD-AA-101, Processing of Procedures and T&RMs.

Which one of the following describes:

(1) the individual(s) in Operations that are allowed to approved the Temporary Change for initial use,

and

(2) when full review, approval, and authorization of the Temporary Change must be completed by,

in accordance with AD-AA-101?

	(1)	(2)
A.	Any qualified SRO	14 days
B.	Any qualified SRO	31 days
C.	Operations Director, only	14 days
D.	Operations Director, only	31 days

Proposed Answer: A

Explanation: AD-AA-101 allows any qualified SRO to perform the initial approval of a Temporary Change. The full review, approval, and authorization must be completed within 14 days.

B. Plausible – 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.
C. Plausible – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval.
D. Plausible – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval. 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.

Technical Reference(s): AD-AA-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2013 Cert #97

Question History:

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 #	2.3.14
	Importance Rating	3.8

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

Proposed Question: #96

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 IAP-2, Classification of Emergency Conditions, Table F-1, Fission Product Barrier Matrix?

A. Drywell radiation; 450 R/hr

B. Drywell radiation; 3000 R/hr

C. Offgas radiation; hi alarm setpoint

D. Offgas radiation; hi-hi alarm setpoint

Proposed Answer: B

Explanation: IAP-2 Table F-1, Fission Product Barrier Matrix, identifies Drywell radiation > 3000 R/hr as a Loss of the Fuel Clad Barrier.

A. Incorrect – 3000 R/hr is the threshold value, NOT 450 R/hr. Plausible because 450 R/hr is the hi-hi alarm setpoint for the Drywell rad monitors.
C. Incorrect – This indicates fuel clad degradation, but does not define loss of the barrier. Plausible because Offgas radiation monitoring is used to define Unusual Event SU5.1.
D. Incorrect – This indicates fuel clad degradation, but does not define loss of the barrier. Plausible because Offgas radiation monitoring is used to define Unusual Event SU5.1.

Technical Reference(s): IAP-2

Proposed references to be provided to applicants during examination: None

Note: To avoid making this question a direct lookup, any EAL matrix provided for other questions must have Table F-1 row D removed.

Learning Objective:

Question Source: Bank – 3/14 NRC #99

Question History: 3/14 NRC #99

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

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

The plant has experienced a loss of coolant accident with the following:

- All control rods are in.
- NO Reactor injection sources are available.
- Emergency RPV Depressurization has been performed due to low Reactor water level.
- Reactor water level is -45” on Fuel Zone indicators and slowly lowering.

Which one of the following describes the required procedure execution?

A. Remain in EOP-2 and perform the Alternate RPV Level Control leg.

B. Remain in EOP-2 and perform the Steam Cooling leg.

C. Exit EOP-2 and execute EOP-7, RPV Flooding.

D. Exit all EOPs and enter the SAOGs.

Proposed Answer: D

Explanation: With no Reactor injection sources available and Reactor water level <-19", Steam Cooling has already been performed. When level dropped <-31.5", Emergency RPV Depressurization, was required. Since level now cannot be restored and maintained >-19", the Alternate RPV Level Control leg now requires Primary Containment Flooding, which requires exiting all EOPs and entering SAOGs.

A. Incorrect – With Steam Cooling and Emergency RPV Depressurization already performed and Reactor water level unable to be restore/maintained >-19", all EOPs must be exited and SAOGs entered. Plausible because the Alternate Level Control Leg is performed earlier, before the Emergency RPV Depressurization, when it is determined that Reactor water level cannot be restored and maintained above 0".
B. Incorrect – With Steam Cooling and Emergency RPV Depressurization already performed and Reactor water level unable to be restore/maintained >-19", all EOPs must be exited and SAOGs entered.
C. Incorrect – With Steam Cooling and Emergency RPV Depressurization already performed and Reactor water level unable to be restore/maintained >-19", Primary Containment Flooding is required. This requires exiting all EOPs and entering SAOGs, not EOP-7.

Technical Reference(s): EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C 1.03

Question Source: Bank – JAF 2017 Biennial Requal 2017LOR0026

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 #	2.3.11
Importance Rating	4.3

Ability to control radiation releases.

Proposed Question: #98

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

- Waste Sample Tank A is being discharged to the lake.
- Then, 17RM-350, Radwaste Effluent Monitor, fails downscale.
- The discharge is temporarily suspended to allow investigation.

Which one of the following describes the ability to continue the discharge?

A. The discharge must be stopped until 17RM-350 is fixed.

B. The discharge may continue if the appropriate grab samples are collected and analyzed at least once per 12 hours.

C. The discharge may continue with no further actions since the minimum required number of operable radiation monitor channels is still met.

D. The discharge may continue if two independent samples are analyzed and two technically qualified individuals verify the discharge valve lineup.

Proposed Answer: D

Explanation: 17RM-350 is the only radiation monitor installed to monitor discharges from Radwaste to the canal. With this monitor failed downscale, the Minimum Channels Operable of ODCM Table 2.1-1 is NOT met. The associated Action (a) states, "With the number of operable channels less than the required minimum number, effluent releases may continue provided that prior to initiating a release:

1. Two independent samples are analyzed;
2. Two technically qualified members of the facility staff verify the discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway."

A. Incorrect – ODCM Table 2.1-1 Action (a) provides alternate requirements that will allow continued discharge. Plausible because there will be no operable continuous radiation monitoring during the discharge.
B. Incorrect – This is the requirement for Service water system effluent line with less than the required Minimum Channels Operable in ODCM Table 2.1-1. The canal discharge relies on a separate pathway and radiation monitor.
C. Incorrect – 17RM-350 is the only radiation monitor installed to monitor discharges from Radwaste to the canal. Plausible that there would be a redundant monitor that would satisfy ODCM requirements, as most such monitors have a redundant backup.

Technical Reference(s): ODCM 2.1, OP-49, FM-17D

Proposed references to be provided to applicants during examination: ODCM 2.1

Learning Objective: SDLP-20 1.18

Question Source: Bank – 3/14 NRC #81

Question History: 3/14 NRC #81

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	
K/A #	2.4.42
Importance Rating	3.8

Knowledge of emergency response facilities.

Proposed Question: #99

The plant has scrammed from 100% power with the following:

- The Shift Manager has declared a Site Area Emergency due to a primary system discharging into the Reactor Building.
- Two NPOs must be dispatched to the Reactor Building to attempt manual closure of a valve to stop the discharge.
- All emergency response facilities are fully operational.

Which one of the following describes the required method of dispatching the NPOs, in accordance with EAP-13, Damage Control?

The NPOs are dispatched…

A. directly from the Control Room.

B. from the Technical Support Center (TSC).

C. from the Operations Support Center (OSC).

D. from the Emergency Operations Facility (EOF).

Proposed Answer: C

Explanation: Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC.

A. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. Prior to full activation, the NPOs would be dispatched directly by the Control Room staff.
B. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. The TSC may be used to come up with strategies that NPOs implement, but is not directly involved with dispatching NPOs.
D. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. The EOF maintains overall oversight of the emergency response, but is not directly involved with dispatching NPOs.

Technical Reference(s): EAP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.2 EO1.48

Question Source: Bank – 9/14 NRC #97

Question History: 9/14 NRC #97

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 #	2.2.42
	Importance Rating	4.6

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: #100

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

- I&C has just discovered that the Standby Liquid Control (SLC) tank 11TK-1 temperature indicating controller and alarm temperature switches have a calibration error.
- Actual SLC tank 11TK-1 conditions are as follows:
  - Temperature is 60ºF.
  - Net volume is 3900 gallons.
  - Concentration is 12% by weight.
  - Enrichment is 35.9 atom % B-10.

Which one of the following describes the operability of the SLC system and the required action, in accordance with OP-17, Standby Liquid Control System, and/or Technical Specifications?

The SLC system is…

A. inoperable. Enter Technical Specification 3.1.7 Condition A, only.

B. inoperable. Enter Technical Specification 3.1.7 Conditions A and B.

C. operable. Raise actual SLC tank temperature to between 65ºF and 75ºF.

D. operable. Raise actual SLC tank temperature to between 80ºF and 100ºF.

Proposed Answer: B

Explanation: The given combination of temperature and concentration are below the curve of Technical Specification Figure 3.1.7-2. This is the unacceptable region of the Figure, which makes SLC tank 11TK-1 inoperable. Since the SLC system has only one boron tank, this makes both SLC subsystems inoperable. This requires entering both Technical Specification 3.1.7 Conditions A and B.

A. Incorrect – The given combination of temperature and concentration make SLC tank 11TK-1 inoperable. Since the SLC system has only one boron tank, this makes both SLC subsystems inoperable. This requires entering both Technical Specification 3.1.7 Conditions A and B.
C. Incorrect – The given combination of temperature and concentration are below the curve of Technical Specification Figure 3.1.7-2. This is the unacceptable region of the Figure, which makes SLC tank 11TK-1 inoperable. 65-75ºF would restore SLC tank temperature to within the limits of Technical Specification Figure 3.1.7-2, but is below the band given by OP-17 Section G.
D. Incorrect – The given combination of temperature and concentration are below the curve of Technical Specification Figure 3.1.7-2. This is the unacceptable region of the Figure, which makes SLC tank 11TK-1 inoperable. 80-100ºF is the band required by OP-17 Section G.

Technical Reference(s): Technical Specification 3.1.7

Proposed references to be provided to applicants during examination: Technical Specification 3.1.7 (with region labels on both figures removed)

Learning Objective: SDLP-11 1.18

Question Source: Bank – 16-1 NRC #100

Question History: 16-1 NRC #100

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: