Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	262001 K1.05
Importance Rating	3.0

AC Electrical Distribution

Knowledge of the physical connections and/or cause-effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following: Main turbine/generator

Proposed Question: #1

The plant is operating at 40% power with a normal electrical distribution lineup.

Then, a Main Generator lockout occurs due to generator ground overcurrent.

Which one of the following describes the status of the 4160V busses (10100, 10200, 10300, 10400, 10500, 10600, and 10700) before the Main Generator lockout and one (1) minute after the Main Generator lockout?

	Before Main Generator Lockout	One (1) Minute After Main Generator Lockout
A.	All 4160V busses powered from Main Generator.	All 4160V busses energized.
B.	All 4160V busses powered from Main Generator.	One or more 4160V busses de-energized.
C.	One or more 4160V busses NOT powered from Main Generator.	All 4160V busses energized.
D.	One or more 4160V busses NOT powered from Main Generator.	One or more 4160V busses de-energized.

Proposed Answer: B

Explanation: After the Main Generator is loaded to > 70 MWe during a startup (which occurs before 40% power), all 4160 V busses are transferred to or energized by the Main Generator through Transformer T4. After the Main Generator lockout, 4160 V busses 10100, 10200, 10300, 10400, 10500, and 10600 are automatically energized from the 115 KV offsite power lines. However, 4160 V bus 10700 de-energizes after the Main Generator lockout.

A. Incorrect – 4160 V bus 10700 de-energizes after the Main Generator lockout.
C. Incorrect – At 40% power and a normal electrical lineup, all 4160 V busses are initially receiving power from the Main Generator. 4160 V bus 10700 de-energizes after the Main Generator lockout.
D. Incorrect – At 40% power and a normal electrical lineup, all 4160 V busses are initially receiving power from the Main Generator.

Technical Reference(s): OP-65, OP-46A, SDLP 71o Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.06.b, 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

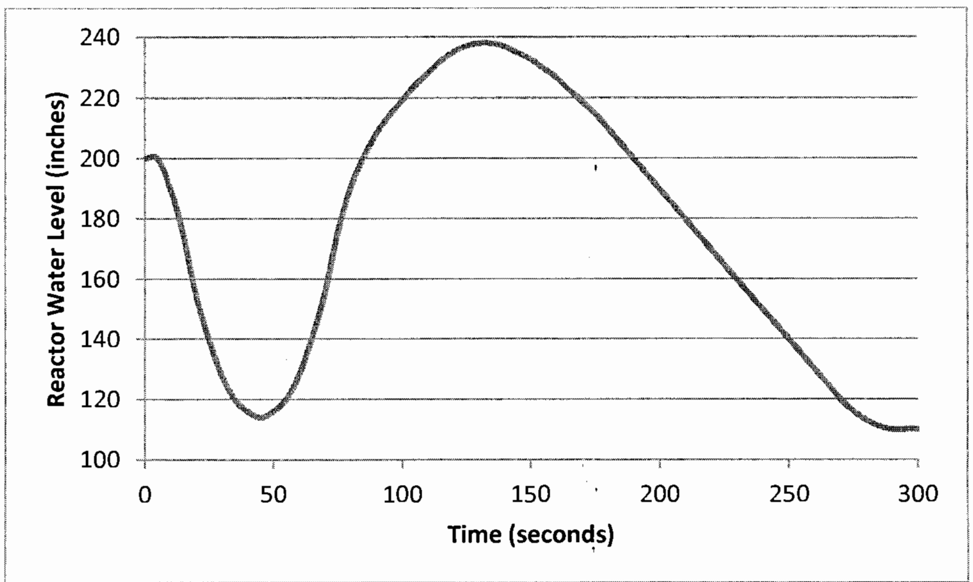
Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K1.02
	Importance Rating	3.5

RCIC

Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: Nuclear boiler system

Proposed Question: #2

The plant has experienced a Reactor scram with the following Reactor water level response:



Note: No operator actions have been taken with RCIC.

Which one of the following describes the response of RCIC during this transient?

RCIC...

A. remained in standby throughout the transient.

B. injected, then stopped injecting for the rest of the transient.

C. injected and then remained injecting for the rest of the transient.

D. injected, then stopped injecting, then automatically resumed injection.

Proposed Answer: D

Explanation: RCIC automatically started and injected when Reactor water level reached 126.5". The RCIC steam inlet isolation valve automatically closed when Reactor water level reached 222.5", stopping injection. When Reactor water level reached 126.5" again, RCIC automatically began injecting again.

A. Incorrect – While Reactor water level did not lower below the 59.5" setpoint that starts RHR and CS pumps, RCIC automatically starts at a higher setpoint of 126.5".
B. Incorrect – When Reactor water level reaches 126.5" after RCIC stopped injecting, RCIC automatically begins injecting again. Any other trip of RCIC would have prevented further injection without operator action.
C. Incorrect – RCIC automatically stopped injecting because Reactor water level reached 222.5". This causes the RCIC steam inlet isolation valve to close until Reactor water level lowers to 126.5" again.

Technical Reference(s): OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.05.c.3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K2.02
	Importance Rating	2.9

Component Cooling Water

Knowledge of electrical power supplies to the following: CCW valves

Proposed Question: #3

Which one of the following describes the power supply to the Drywell Cooling Assembly 'A' Supply Valve 15MOV-100A1?

A. L13

B. L14

C. MCC-152

D. MCC-252

Proposed Answer: C

Explanation: RBCLC is supplied to the 'A' Drywell Cooler thru Supply Valves 15MOV-100A1 (2, 3, and 4). Valve A1 is powered by 600VAC Motor Control Center (MCC) 152.

A. Incorrect - L13 is the power supply to RBCLC pumps 15P-2A and 2C.
B. Incorrect - L14 is the power supply to RBCLC pump 15P-2B.
D. Incorrect - MCC-252 is the power supply to the ESW supply header isolation valve (46MOV-101A) which could supply the Drywell Coolers.

Technical Reference(s): OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.04.b

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K2.01
	Importance Rating	3.1

Shutdown Cooling

Knowledge of electrical power supplies to the following: Pump motors

Proposed Question: #4

The Plant is operating in a Shutdown Cooling (SDC) lineup with the following:

- SDC Loop B is in service with RHR pump B running.
- RHR pump A has been started to place SDC Loop A in service.

Then, the 10500 bus de-energizes.

Which one of the following describes the resulting status of RHR pumps A and B?

A. Neither pump is running.

B. Only pump A is running.

C. Only pump B is running.

D. Both pumps are running.

Proposed Answer: A

Explanation: RHR loop A contains RHR pumps A and C, which are powered by Buses 10500 and 10600, respectively. RHR Loop B contains RHR pump B and D, which are power by Buses 10500 and 10600, respectively. A loss of the 10500 bus results in no RHR pumps running.

B. Incorrect - both pumps trip due to not having electrical power
C. Incorrect - both pumps trip due to not having electrical power
D. Incorrect - both pumps trip due to not having electrical power

Technical Reference(s): OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.03.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	262002 K3.02
Importance Rating	2.9

UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Recirculation pump speed: Plant-Specific

Proposed Question: #5

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

- A complete loss of the UPS occurs.
- UPS distribution bus 71ACUPS de-energizes.
- No operator action occurs.

Which one of the following describes the effect of this loss on the Reactor Recirculation system?

Both RWR MG Sets...

A. trip.

B. run back to the 44% limiter.

C. run back to minimum speed.

D. lock up at the current speed.

Proposed Answer: C

Explanation: Sustained loss of power from 71ACUPS causes RWR MG sets to automatically run-back to minimum speed.

A. Incorrect – Loss of UPS causes RWR MG sets to runback to minimum speed, but not trip.
B. Incorrect – Loss of UPS causes RWR MG sets to runback, but to minimum speed, not the 44% limiter.
D. Incorrect – Loss of UPS causes RWR MG sets to runback to minimum speed, not lock up. AOP-21 directs manual lock up of scoop tubes if time allows.

Technical Reference(s): AOP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K3.02
	Importance Rating	4.5

ADS

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

Proposed Question: #6

The plant was operating at 100% power when the following events occurred:

- An un-isolable Reactor Water Cleanup steam leak into the Reactor Building led to the need for an Emergency RPV Depressurization.
- Malfunctions have caused all seven (7) ADS SRVs to fail closed.
- MSIVs closed due to an invalid isolation signal that has now cleared.
- Main Condenser vacuum is 20 inches Hg.
- CW Pump A is in service.

Which one of the following describes the effect the malfunction of the ADS system has on the ability to rapidly depressurize the reactor, in accordance with EOP-2, RPV Control?

The Minimum Number of SRVs Required for Emergency Depressurization __(1)__ and Turbine Bypass Valves __(2)__ be used to rapidly depressurize the Reactor.

	(1)	(2)
A.	is available	should
B.	is available	should NOT
C.	is NOT available	should
D.	is NOT available	should NOT

Proposed Answer: C

Explanation: The plant has 11 SRVs and the Minimum Number of SRVs Required for Emergency Depressurization is 5. With 7 SRVs failed closed due to a malfunction of the ADS system, only 4 are available, which is less than the required 5. With less than 5 SRVs open, EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.

A. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 7 SRVs unavailable, only 4 remain available.
B. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 7 SRVs unavailable, only 4 remain available. EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.
D. Incorrect – EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C 4.04

Question Source: Bank – 9/12 NRC #42

Question History: 9/12 NRC #42

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 K4.05
	Importance Rating	3.4

SLC

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Dispersal of boron upon injection into the vessel

Proposed Question: #7

Which one of the following describes where the Standby Liquid Control (SLC) system injects into the Reactor vessel?

A. Into the 'A' Feedwater line.

B. Into the 'B' Feedwater line.

C. Into a dedicated sparger located below the core plate.

D. Into a dedicated sparger located above the core plate.

Proposed Answer: C

Explanation: The SLC system injects boron into the Reactor vessel through a dedicated sparger located below the core plate. This design ensures adequate boron dispersal and delivery as close as possible to the core region. This design relies on either Recirculation flow or natural circulation to carry the boron into the core region.

A. Incorrect – HPCI and RCIC inject into Feedwater lines B and A respectively, however SLC has its own dedicated sparger located below the Core Plate.
B. Incorrect – HPCI and RCIC inject into Feedwater lines B and A respectively, however SLC has its own dedicated sparger located below the Core Plate.
D. Incorrect – SLC has its own dedicated sparger located below the Core Plate, not above it.

Technical Reference(s): FM-21A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11 1.05.a.13

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 #	1
	K/A #	215003 K4.06
	Importance Rating	2.6

IRM

Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Alarm seal-in

Proposed Question: #8

A plant startup is in progress with the following:

- IRM A exceeds the upscale trip setpoint.
- Annunciator 09-5-2-42, IRM UPSCALE, alarms.
- The amber UPSCL ALARM light illuminates for IRM A on Panel 09-12.
- An operator restores IRM A to below the upscale trip setpoint by changing the range switch position.

Which one of the following describes the response of Annunciator 09-5-2-42 and the amber UPSCL ALARM light **AFTER** the Operator changes the range switch?

	Annunciator 09-5-2-42	Amber UPSCL ALARM Light on Panel 09-12
A.	Remains sealed-in	Remains sealed-in
B.	Remains sealed-in	Does NOT remain sealed-in
C.	Does NOT remain sealed-in	Remains sealed-in
D.	Does NOT remain sealed-in	Does NOT remain sealed-in

Proposed Answer: C

Explanation: The IRM trip unit produces two outputs – one that seals-in and one that automatically resets once the IRM returns to a normal value. The amber UPSCL ALARM light seals-in and does not reset until the operator manually resets the signal at Panel 09-12. Annunciator 09-5-2-42 does not seal-in.

A. Incorrect – Annunciator 09-5-2-42 does not seal-in.
B. Incorrect – Annunciator 09-5-2-42 does not seal-in, but the amber UPSCL ALARM light on Panel 09-12 does.
D. Incorrect – The amber UPSCL ALARM light on Panel 09-12 seals-in.

Technical Reference(s): SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.a.4.h

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 #	1
	K/A #	215005 K5.06
	Importance Rating	2.5

APRM / LPRM

Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Assignment of LPRMs to specific APRM channels

Proposed Question: #9

Which one of the following identifies:

(1) the minimum number of total LPRM inputs, and

(2) the minimum number of LPRM inputs per detector level,

required to maintain an APRM operable, in accordance with OP-16, Neutron Monitoring?

	(1)	(2)
A.	11	1
B.	11	2
C.	14	1
D.	14	2

Proposed Answer: B

Explanation: Each APRM receives input from multiple LPRMs from different levels in the core. For an APRM to remain operable, it must have at least 11 operable LPRM inputs, with at least 2 operable LPRMs from each detector level.

A. Incorrect – The minimum number of LPRMs from each detector level is 2, not just 1.
C. Incorrect – The minimum number of total LPRMs is 11, not 14. 14 is the number of total LPRMs that some APRMs are provided, while other APRMs have 17 LPRM inputs. The minimum number of LPRMs from each detector level is 2, not just 1.
D. Incorrect – The minimum number of total LPRMs is 11, not 14. 14 is the number of total LPRMs that some APRMs are provided, while other APRMs have 17 LPRM inputs.

Technical Reference(s): OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.07.a

Question Source: Bank – NMP2 2014 Audit #9

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K5.02
	Importance Rating	3.3

RPS

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements

Proposed Question: #10

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

- Turbine Stop Valve (TSV) 4 drifts 50% closed.
- Then, TSV 3 drifts 50% closed.

Considering only the Turbine Stop Valve Closure scram signal, which one of the following describes the resulting operation of the Reactor Protection System?

Based on the Turbine Stop Valve Closure scram signal, a full Reactor scram...

A. occurs while TSV 4 is drifting.

B. occurs while TSV 3 is drifting.

C. does NOT occur because only these two TSVs have drifted.

D. does NOT occur because these two TSVs have not tripped their scram position switches.

Proposed Answer: C

Explanation: Each Turbine Stop Valve (TSV) has two position switches that actuate when the valve is less than 90% open. These position switches input to the Reactor Protection System scram circuitry. The logic is arranged such that some combinations of 2 TSV (including TSV 3 & 4) cause a half scram, but 3 TSVs must close to cause an actual Reactor scram. In this case, both TSV 3 & 4 have drifted far enough to trip their scram position switches, which results in a half scram, but NOT an actual Reactor scram.

A. Incorrect – One TSV <90% open trips two scram position switches, but this is not enough to cause RPS logic to enforce a Reactor scram.
B. Incorrect – When TSV 3 drifts <90% open, a half scram is received, but not an actual Reactor scram.
D. Incorrect – These TSVs have both drifted far enough to trip their scram position switches (<90% open, not <90% closed).

Technical Reference(s): SDLP-05

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05

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 #	1
	K/A #	203000 K6.10
	Importance Rating	3.0

RHR/LPCI: Injection Mode

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Component cooling water systems

Proposed Question: #11

Which one of the following identifies the effect that a complete loss of the RBCLC system will have on RHR?

A. The normal source of cooling water to the RHR pump seals is lost.

B. The backup source of cooling water to the RHR pump seals is lost.

C. The normal source of cooling water to the RHR pump thrust bearing is lost.

D. The backup source of cooling water to the RHR pump thrust bearing is lost.

Proposed Answer: A

Explanation: Loss of RBCLC results is loss of the normal cooling water source to the RHR pump seals.

B. Incorrect – RBCLC is the normal source, not the backup source. ESW is the backup source.
C. Incorrect – The RHR pump thrust bearing is cooled by water from the discharge of the RHR pump, not RBCLC.
D. Incorrect – The RHR pump thrust bearing is cooled by water from the discharge of the RHR pump, not RBCLC.

Technical Reference(s): SDLP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10.g

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K6.05
	Importance Rating	2.6

SRM

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

Proposed Question: #12

A plant startup is in progress with the following:

- The Mode Switch is in START & HOT STBY.
- IRMs are on Range 1.
- Then, the SRM A trip unit fails upscale.

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

A. No rod block or scram signals are generated.

B. A rod block is generated, but no scram signals are generated.

C. A rod block and half scram are generated. A full scram signal is NOT generated.

D. A full scram signal is generated.

Proposed Answer: B

Explanation: Any SRM above 5 x 10^5 cps will cause a rod block. A half and\or a full scram would not occur. A full scram would only occur if the Shorting Links were removed.

A. Incorrect – A single SRM upscale trip causes a rod block.
C. Incorrect – A single SRM upscale trip causes a rod block, not a half or full scram.
D. Incorrect – A single SRM upscale trip causes a rod block, not a full scram.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.c.1.e

Question Source: Modified Bank – 3/12 NRC #39

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 #	259002 A1.05
	Importance Rating	2.9

Reactor Water Level Control

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

Proposed Question: #13

A plant startup is in progress with the following:

- Reactor power is approximately 2%.
- Feedwater pump A is in service and injecting through 34FCV-137, Feedwater Startup Valve.
- 34MOV-100A, RFP A DISCH, and 34MOV-100B, RFP B DISCH, are closed.
- The following picture shows the status of 06FIC-130, FDWTR STARTUP VLV (34FCV-137):

Which one of the following describes the response of 34FCV-137 and/or 06FIC-130 if the indicated control (inner knob) is operated in the clockwise direction indicated by the red arrow?

34FCV-137...

A. re-positions further open.

B. re-positions further closed.

C. remains in the same position. The top meter arrow on 06FIC-130 moves up.

D. remains in the same position. The top meter arrow on 06FIC-130 moves down.

Proposed Answer: A

Explanation: 06FIC-130 is currently in MAN, therefore the indicated inner knob will directly cause re-positioning of 34FCV-137. The arrow depicts the knob being move in the clockwise direction, which will cause 34FCV-137 to open further.

B. Incorrect – 34FCV-137 will re-position, but further open, not further closed.
C. Incorrect – 06FIC-130 is currently in MAN, therefore the indicated knob will directly cause re-positioning of 34FCV-137. If 06FIC-130 was in either AUTO or BAL, then 34FCV-137 would not re-position. The top meter arrow will move as Reactor water level responds to the change in 34FCV-137 position.
D. Incorrect – 06FIC-130 is currently in MAN, therefore the indicated knob will directly cause re-positioning of 34FCV-137. If 06FIC-130 was in either AUTO or BAL, then 34FCV-137 would not re-position. The top meter arrow will move as Reactor water level responds to the change in 34FCV-137 position.

Technical Reference(s): OP-2A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-06

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 #	209001 A1.02
	Importance Rating	3.2

LPCS

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

Proposed Question: #14

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

- Core Spray loop A is in a normal standby lineup.
- Preparations are underway to start Core Spray pump A per ST-3PA, Core Spray Loop A Quarterly Operability Test.

Which one of the following identifies the approximate expected Core Spray discharge pressure indications on Panel 09-3 indicator 14PI-48A (CORE SPRAY PMP A DISCH PRESS):

(1) while Core Spray loop A is in the normal standby lineup, and

(2) once Core Spray pump A is started?

	(1)	(2)
A.	0 psig	250 psig
B.	0 psig	450 psig
C.	50 psig	250 psig
D.	50 psig	450 psig

Proposed Answer: C

Explanation: With Core Spray loop A initially in a normal standby lineup, Core Spray Holding Pump A is operating, which causes 14PI-48A to indicate approximately 50 psig. Once Core Spray pump A is started, it will cause 14PI-48A to indicate approximately 250 psig.

A. Incorrect – With Core Spray loop A initially in a normal standby lineup, Core Spray Holding Pump A is operating, which causes 14PI-48A to indicate approximately 50 psig, not 0 psig as might be expected with Core Spray pump A secured.
B. Incorrect – With Core Spray loop A initially in a normal standby lineup, Core Spray Holding Pump A is operating, which causes 14PI-48A to indicate approximately 50 psig, not 0 psig as might be expected with Core Spray pump A secured. 450 psig is the Reactor pressure at which Core Spray injection valves will open, however the Core Spray pump only provides approximately 250 psig of discharge pressure.
D. Incorrect – 450 psig is the Reactor pressure at which Core Spray injection valves will open, however the Core Spray pump only provides approximately 250 psig of discharge pressure.

Technical Reference(s): ST-3PA, OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A2.03
	Importance Rating	2.9

SGTS

Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train temperature

Proposed Question: #15

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

- Standby Gas Treatment (SGT) train A is in service.
- SGT train B is secured.
- Annunciator 09-75-1-16, SGT SYS A ACT CHAR TEMP HI, alarms.
- SGT train A charcoal filter temperature is confirmed high, but is NOT yet high enough to require initiating fire protection water spray.

Which one of the following describes the required actions in response to this high temperature, in accordance with OP-20, Standby Gas Treatment System?

Start SGT train B and...

A. secure SGT train A. Then manually open 01-125MOV-100A, TRAIN A CLG VLV, to align cooling to the charcoal filter.

B. secure SGT train A. Then observe 01-125MOV-100A, TRAIN A CLG VLV, automatically opens to align cooling to the charcoal filter.

C. maintain SGT train A running. Then manually open 01-125MOV-100A, TRAIN A CLG VLV, to align cooling to the charcoal filter.

D. maintain SGT train A running. Observe 01-125MOV-100A, TRAIN A CLG VLV, automatically opened when the high temperature condition alarmed.

Proposed Answer: B

Explanation: ARP 09-75-1-16 directs use of OP-20 section G to provide cooling to the charcoal filter to prevent a fire. OP-20 section G.1 provides the directions for placing cooling in service. SGT train B is started. Then SGT train A is secured. When SGT train A is secured, 01-125MOV-100A automatically opens to align the required cooling flow path.

A. Incorrect – 01-125MOV-100A automatically opens to align the required cooling flow path.
C. Incorrect – SGT train A must be secured to align the required cooling flow path.
D. Incorrect – SGT train A must be secured to align the required cooling flow path.

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: Modified Bank – NMP1 2010 Audit #14

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 #	223002 A2.10
	Importance Rating	3.9

PCIS/Nuclear Steam Supply Shutoff

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of coolant accidents

Proposed Question: #16

A loss of coolant accident has resulted in the following:

- Reactor water level is 90 inches and lowering slowly.
- Drywell pressure is 12 psig and rising slowly.

Which one of the following describes the availability of the Containment Hydrogen and Oxygen monitors and the required action(s), in accordance with EOP-4, Primary Containment Control?

The Containment Hydrogen and Oxygen monitors are...

A. available and in service. Verify proper operation and monitor Drywell and Torus Hydrogen and Oxygen concentrations.

B. isolated. Restore the monitors to service per OP-37, Containment Atmosphere Dilution System. Overriding interlocks is NOT required.

C. isolated. Defeat the isolation per EP-2, Isolation/Interlock Overrides. Then, restore the monitors to service per OP-37, Containment Atmosphere Dilution System.

D. isolated and NOT available for restoration. Coordinate with Chemistry to manually sample the Containment.

Proposed Answer: C

Explanation: When Reactor water level lowers below 177 inches and/or Drywell pressure rises above 2.7 psig, a PCIS Group 2 isolation is received. This isolation remains in effect since Reactor water level is still below 177 inches and Drywell pressure is above 2.7 psig. The PCIS Group 2 isolation causes the Containment Hydrogen and Oxygen monitors to isolate. EOP-4 directs defeating the isolation per EP-2 and restoring these monitors to service per OP-37.

A. Incorrect – Containment Hydrogen and Oxygen monitors isolate on a PCIS Group 2 isolation (177 inches and/or 2.7 psig), not the PCIS Group 1 isolation (59.5 inches).
B. Incorrect – Since a PCIS Group 2 isolation signal still exists, the isolation must be defeated per EP-2 before the actions in OP-37 will work to restore the monitors.
D. Incorrect – Even though a PCIS Group 2 isolation signal still exists, EOP-4 provides direction to defeat the isolation and restore the monitors to service. Chemistry sampling would only be directed if the monitors were unavailable for some additional reason.

Technical Reference(s): AOP-15, EOP-4, EP-2, OP-37

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.14.c

Question Source: Bank – NMP1 2013 NRC #27

Question History: NMP1 2013 NRC #27

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 A3.08
	Importance Rating	3.6

SRVs

Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: Lights and alarms

Proposed Question: #17

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

- Safety Relief Valve (SRV) 'A' automatically opens and closes.
- Annunciator 09-4-1-16, SRV LEAKING, alarms.

Which one of the following identifies the parameter that causes Annunciator 09-4-1-16 to alarm and an alternate indication for the same parameter?

	Parameter	Alternate Indication for Same Parameter
A.	Acoustic monitor noise	White lights on Panel 09-4
B.	Acoustic monitor noise	Recorder on Panel 09-21
C.	Tail pipe temperature	White lights on Panel 09-4
D.	Tail pipe temperature	Recorder on Panel 09-21

Proposed Answer: D

Explanation: Annunciator 09-4-1-16, SRV LEAKING, is caused by high temperature as sensed by an SRV tail pipe thermocouple. A recorder on Control Room panel 09-21 also provides tail-pipe temperature indications.

A. Incorrect – Annunciator 09-4-1-16 is driven by high tail pipe temperatures, not acoustic monitor noise. Annunciator 09-4-2-06 is driven by acoustic monitor noise. The white lights by the SRV control switches on Panel 09-4 are based on acoustic monitor noise, not tail pipe temperature.
B. Incorrect – Annunciator 09-4-1-16 is driven by high tail pipe temperatures, not acoustic monitor noise. Annunciator 09-4-2-06 is driven by acoustic monitor noise.
C. Incorrect – The white lights by the SRV control switches on Panel 09-4 are based on acoustic monitor noise, not tail pipe temperature.

Technical Reference(s): ARP 09-4-1-16, SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.14.d

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 A3.03
	Importance Rating	3.4

EDGs

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Indicating lights, meters, and recorders

Proposed Question: #18

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

- Emergency Diesel Generator (EDG) C has just been started for post maintenance testing.
- EDG C is running at 900 rpm.

Then, the following occurs:

- 115KV Line 3 de-energizes.
- A small coolant leak develops in the Drywell.
- Operators manually scram the Reactor.
- Drywell pressure is 3 psig and rising slowly.
- One (1) minute later, the following indications are present for Emergency Diesel Generators (EDGs) A and C:

09-8 panel indications are on the following page:

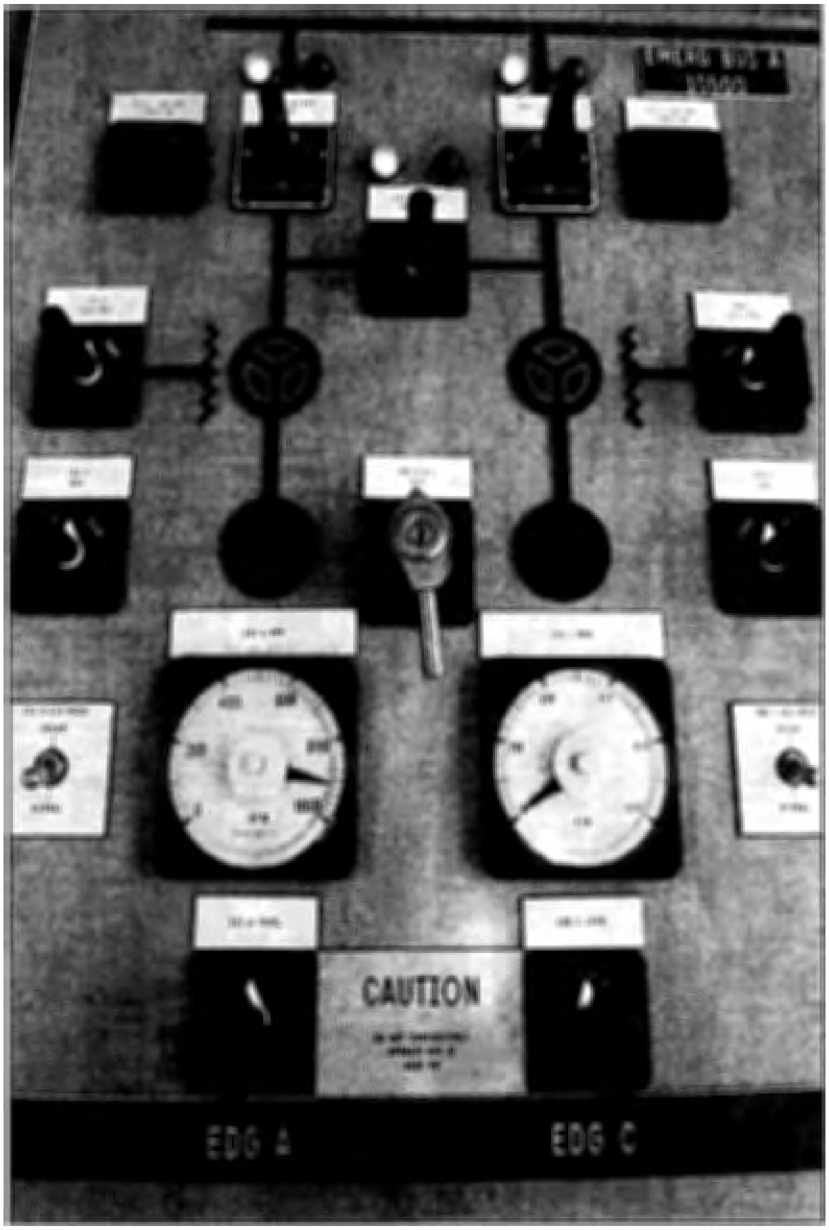
Which one of the following describes the status of EDGs A and C?

A. Both EDGs have responded properly.

B. EDG A has responded properly, but EDG C has NOT.

C. EDG C has responded properly, but EDG A has NOT.

D. Neither EDG has responded properly.



CAUTION
EDG A
EDG C

Proposed Answer: B

Explanation: Upon receipt of a LOCA signal (Drywell pressure ≥ 2.7 psig), both EDG A and C are designed to start and parallel together. With no undervoltage or degraded voltage present on Bus 10500 (because only Line 3 has been lost and Line 4 is sufficient to maintain Bus 10500 energized), neither EDG parallels with Bus 10500. The given indications show that EDG A is running (speed meter as expected) with its output breaker open (green light on, red light off), as required. The given indications show that EDG C has tripped (speed meter low).

A. Incorrect – EDG C is required to be running along with EDG A due to the LOCA signal, but is currently not running.
C. Incorrect – Even without an undervoltage or degraded voltage on Bus 10500, both EDGs A and C are required to be running due to the LOCA signal. EDG A is running, as required, but EDG C is NOT.
D. Incorrect – EDG A is running, as required. EDG C is also required to be running, but is not.

Technical Reference(s): OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.09.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A4.02
	Importance Rating	4.0

HPCI

Ability to manually operate and/or monitor in the control room: Flow controller: BWR-2,3,4

Proposed Question: #19

The plant has experienced a scram with the following conditions:

- Reactor water level is 200 inches and stable with RCIC injecting.
- HPCI has been started for pressure control per OP-15, High Pressure Coolant Injection.
- Reactor pressure is 975 psig and stable.
- HPCI speed is 3350 rpm and stable.
- HPCI flow is 3500 gpm and stable with 23FIC-108-1, HPCI FLOW CNTRL, in AUTO.
- 23MOV-21, TEST VLV TO CST, is throttled partially open.

Which one of the following describes the actions required to maximize the Reactor cooldown rate with HPCI, in accordance with OP-15?

A. Raise the 23FIC-108-1 setpoint and throttle 23MOV-21 further open.

B. Raise the 23FIC-108-1 setpoint and throttle 23MOV-21 further closed.

C. Lower the 23FIC-108-1 setpoint and throttle 23MOV-21 further open.

D. Lower the 23FIC-108-1 setpoint and throttle 23MOV-21 further closed.

Proposed Answer: B

Explanation: The Reactor cooldown rate is maximized by raising HPCI flow while also raising pump discharge pressure. This is accomplished by raising the setpoint on 23FIC-108-1 and throttling 23MOV-21 further closed.

A. Incorrect – Raising the flow controller setpoint is correct, however 23MOV-21 must be throttled further closed to raise pump discharge pressure.
C. Incorrect – Lowering the flow controller setpoint and throttling 23MOV-21 further open will lower the amount of work the HPCI turbine must perform, lowering steam draw to the HPCI turbine and Reactor cooldown rate.
D. Incorrect – 23MOV-21 does need to be throttled further closed to raise pump discharge pressure, however to maximize cooldown, the flow controller setpoint must also be raised.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.13.b

Question Source: Bank – SSES 1/15 NRC #19

Question History: SSES 1/15 NRC #19

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 A4.02
	Importance Rating	3.2

DC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Battery voltage indicator: Plant-Specific

Proposed Question: #20

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

- Both Station Batteries are on float charge.
- Station Battery A voltage is indicating 125 VDC.
- Station Battery B voltage is indicating 132 VDC.

Which one of the following describes the status of these voltages, in accordance with OP-43A, 125 VDC Power System?

A. Both voltages are within the required range.

B. Station Battery A voltage is within the required range, but Station Battery B voltage is too high.

C. Station Battery B voltage is within the required range, but Station Battery A voltage is too low.

D. Station Battery A voltage is too low and Station Battery B voltage is too high.

Proposed Answer: C

Explanation: OP-43A requires Station Battery float voltage to be 131-133 VDC. Station Battery B is within this required range, however Station Battery A voltage is too low.

A. Incorrect – While Station Battery A voltage is at the nominal voltage of the system, it is below the required float voltage of 131-133 VDC.
B. Incorrect – While Station Battery A voltage is at the nominal voltage of the system, it is below the required float voltage of 131-133 VDC. Station Battery B voltage is within the required 131-133 VDC range.
D. Incorrect – Station Battery B voltage is within the required 131-133 VDC range.

Technical Reference(s): OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.05.a.1

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 #	300000 2.4.31
	Importance Rating	4.2

Instrument Air

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #21

The plant is operating at 100% power when Annunciator 09-5-1-54, SCRAM AIR HDR PRESS HI OR LO, alarms.

Which one of the following describes:

(1) the location of the associated pressure indication(s) available to verify the alarm, and

(2) the status of the alarm if pressure indication is 75 psig, in accordance with ARP 09-5-1-54?

	Location of Associated Pressure Indication(s)	Status of Alarm if Pressure Indication is 75 Psig
A.	Local in Reactor Building only	Low Alarm
B.	Local in Reactor Building only	High Alarm
C.	Control Room Panel 09-5 and Local in Reactor Building	Low Alarm
D.	Control Room Panel 09-5 and Local in Reactor Building	High Alarm

Proposed Answer: D

Explanation: Pressure indication is provided for the Scram Air Header both locally in the Reactor Building and on Control Room Panel 09-5. The normal pressure is 70 psig. An indication of 75 psig means that the annunciator is caused by a High pressure condition. The Low alarm comes in ≤65 psig.

A. Incorrect – There is also a remote pressure indication located on Control Room Panel 09-5. Although much lower than Instrument Air pressure, Scram Air Header pressure of 75 psig is a High alarm, not a Low alarm.
B. Incorrect – There is also a remote pressure indication located on Control Room Panel 09-5.
C. Incorrect – Although much lower than Instrument Air pressure, Scram Air Header pressure of 75 psig is a High alarm, not a Low alarm.

Technical Reference(s): ARP 09-5-1-54, FM-27B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.11.a.6

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 #	1
	K/A #	263000 2.4.11
	Importance Rating	4.0

DC Electrical Distribution

Knowledge of abnormal condition procedures.

Proposed Question: #22

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

Which one of the following identifies the need to enter AOP-1, Reactor Scram, and AOP-16, Loss of 10300 Bus?

	AOP-1, Reactor Scram	AOP-16, Loss of 10300 Bus
A.	Entry required	Entry required
B.	Entry required	Entry NOT required
C.	Entry NOT required	Entry required
D.	Entry NOT required	Entry NOT required

Proposed Answer: A

Explanation: Loss of 125 VDC Battery Bus A while operating at 100% power causes a Main Turbine trip from loss of power to EHC trip logic, which results in a Reactor scram. Bus 10300 loses its original power source (Normal Station Service Transformer 71T-4) and the required control power to automatically transfer to reserve power. This causes Bus 10300 to de-energize. AOP-45, Loss of DC Power System A, must be entered. This procedure addresses many of the issues resulting from the loss of Battery Bus A, but does not replace the need for entering other AOPs as well. Therefore, both AOP-1, Reactor Scram, and AOP-16, Loss of 10300 Bus, must also be entered.

B. Incorrect – 10300 Bus de-energizes and AOP-16 must be entered.
C. Incorrect – The Reactor scrams and AOP-1 must be entered.
D. Incorrect – The Reactor scrams and AOP-1 must be entered. 10300 Bus de-energizes and AOP-16 must be entered.

Technical Reference(s): AOP-45, AOP-1, AOP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.02

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 #	2
	Group #	1
	K/A #	215004 A4.02
	Importance Rating	3.0

SRM

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

Proposed Question: #23

A plant startup is in progress with the following:

- The Reactor is critical.
- Control rod withdrawal is temporarily on hold.
- IRMs are indicating on Range 1.
- All Source Range Monitors (SRMs) are partially withdrawn.
- SRM recorders are indicating the following:
    - SRM A: 4×10^2 cps and stable
    - SRM B: 5×10^4 cps and stable
    - SRM C: 4×10^5 cps and stable
    - SRM D: 8×10^1 cps and stable

Which one of the following describes the action necessary to ensure these SRMs are indicating within the required range, in accordance with OP-65, Startup and Shutdown Procedure?

A. Insert SRM B and Insert SRM C

B. Withdrawal SRM C and Insert SRM D

C. Insert SRM A and Withdrawal SRM B

D. Withdrawal SRM A and Withdrawal SRM D

Proposed Answer: B

Explanation: With the Reactor critical and IRMs below range 3, OP-65 requires maintaining SRMs between 10^2 and 3×10^5 cps. SRM A and B are within this range, but SRM D is below the minimum value of this range and SRM C is above this range. To raise SRM D indication, it must be inserted further into the core. To lower SRM C indication, it must be withdrawn further out.

A. Incorrect – SRM B is in range and is not required to be inserted however, SRM C is high out of range and is required to be withdrawn.
C. Incorrect – SRM A and B are both in range and are not required to be inserted or withdrawn.
D. Incorrect – SRM A is in range and is not required to be withdrawn however, SRM D is low out of range and is required to be inserted.

Technical Reference(s): OP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.13.d

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 #	206000 K5.01
	Importance Rating	3.3

HPCI

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM: Turbine operation: BWR-2,3,4

Proposed Question: #24

The plant has scrammed with the following:

- HPCI is being utilized for Reactor water level and pressure control per EOP-2, RPV Control.
- Reactor water level is 200 inches and steady.
- Reactor pressure is 200 psig and lowering slowly.

Which one of the following identifies at which approximate Reactor pressure HPCI will FIRST become unusable?

HPCI will FIRST become unusable at a Reactor pressure of approximately...

A. 150 psig.

B. 60 psig.

C. 50 psig.

D. 10 psig.

Proposed Answer: B

Explanation: HPCI is rated to supply 4250 gpm across a range of Reactor pressures from 1195 psig to 150 psig. HPCI remains available below 150 psig until it isolates on low steam supply pressure. This isolation is required to occur between 61 psig and 90 psig by Technical Specifications (approximately 75 psig).

A. Incorrect – 150 psig is the bottom end of HPCI's rated pressure range, but HPCI will continue to be available below this pressure until it isolates on low steam supply pressure at a TS value of 61 psig minimum.

C. Incorrect – HPCI isolates at a TS value of 61 psig minimum, not 50 psig. 50 psig is the approximate value of when the SRVs close.

D. Incorrect – HPCI isolates at a TS value of 61 psig minimum, not 10 psig. 10 psig is the approximate setpoint for the high turbine exhaust pressure isolation.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.c.7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A2.11
	Importance Rating	4.0

RPS

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Main steamline isolation valve closure

Proposed Question: #25

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

- The Main Steam Isolation Valves (MSIVs) spuriously close.
- The Reactor scrams.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level is restored to approximately 200 inches using RCIC.
- Reactor pressure is being controlled 900-1000 psig using SRVs.
- Annunciator 09-5-1-1, SDIV A OR B HI LVL TRIP, is in alarm.

Given the following possible conditions:

(1) MSIVs are re-opened.
(2) SDIV HI LVL TRIP keylock switch is placed in BYPASS.

Which one of the following identifies which of these possible conditions, if any, MUST occur to be able to reset the reactor scram, in accordance with AOP-1, Reactor Scram?

A. Neither

B. (1) only

C. (2) only

D. (1) and (2)

Proposed Answer: C

Explanation: MSIV closure causes an automatic Reactor scram. However, when the Reactor Mode switch is taken out of RUN, this scram is bypassed. No further action to re-open the MSIVs is required to be able to reset the scram. The SDIV high level condition also causes a Reactor scram. This scram signal is NOT bypassed by the Reactor mode switch being taken out of RUN. The only way to lower SDIV level is to reset the scram. The only way to reset the scram with the SDIV high level condition present is to place the SDIV HI LVL TRIP keylock switch in BYPASS.

A. Incorrect – The SDIV HI LVL TRIP keylock switch must be placed in BYPASS, since the high SDIV level is still causing a scram signal and SDIV level will not lower until the scram is reset.
B. Incorrect – The MSIVs may remain closed since the Mode Switch being out of RUN bypasses the MSIV position scram signal. The SDIV HI LVL TRIP keylock switch must be placed in BYPASS, since the high SDIV level is still causing a scram signal and SDIV level will not lower until the scram is reset.
D. Incorrect – The MSIVs may remain closed since the Mode Switch being out of RUN bypasses the MSIV position scram signal.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.05.b.2

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 #	239002 K6.05
	Importance Rating	3.0

SRVs

Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: Discharge line vacuum breaker

Proposed Question: #26

A plant transient results in the following:

- SRV K opens due to high Reactor pressure and then closes.
- An SRV K vacuum breaker opens and sticks in the open position.

Which one of the following describes the consequence of the stuck open SRV vacuum breaker?

A. Steam will continue to flow from the Reactor to the Torus through the open SRV vacuum breaker.

B. If SRV K opens again, some of the steam passing through the SRV will be released directly to the Reactor Building.

C. If SRV K opens again, some of the steam passing through the SRV will be released directly into the Torus airspace.

D. If SRV K opens again, some of the steam passing through the SRV will be released directly into the Drywell airspace.

Proposed Answer: D

Explanation: After SRV operation, the vacuum breakers open to equalize pressure between the Drywell and tailpipes. Without vacuum breaker operation, condensation of steam in the tailpipe draws water from the Torus up into the tailpipe. Upon subsequent re-opening of the SRV, high forces would be experienced due to the clearing of the extra water from the tailpipe. With a stuck open vacuum breaker, subsequent SRV opening would admit steam directly to the Drywell airspace, resulting in rising Drywell temperature and pressure.

A. Incorrect – The closed SRV isolates the vacuum breaker from the Reactor. The vacuum breaker is connected between the SRV discharge piping and the Drywell air space, not the Reactor.
B. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Reactor Building.
C. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Torus.

Technical Reference(s): SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.09.f

Question Source: Bank – 9/12 NRC #44

Question History: 9/12 NRC #44

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 #	271000 K1.02
	Importance Rating	3.1

Off-gas

Knowledge of the physical connections and/or cause-effect relationships between OFFGAS SYSTEM and the following: Process radiation monitoring system

Proposed Question: #27

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

- Annunciator 09-3-2-17, OFF-GAS RAD MON DOWNSCALE OR INOP, alarms.
- Off-Gas Radiation Monitor 17RM-150B indicates downscale.
- Annunciator 09-3-2-27, OFF-GAS RAD MON HI, alarms.
- Annunciator 09-3-2-38, OFF-GAS RAD MON HI-HI, alarms.
- Off-Gas Radiation Monitor 17RM-150A indicates 1200 mR/hr and rising slowly.

Which one of the following describes the Off-Gas system response?

The Off-Gas system...

A. automatically isolated as soon as Annunciator 09-3-2-38 alarmed.

B. will automatically isolate approximately 15 minutes after Annunciator 09-3-2-38 alarmed.

C. will NOT automatically isolate due to 17RM-150B indicating downscale. Manual isolation of Off-Gas is required as soon as Annunciator 09-3-2-38 alarmed.

D. will NOT automatically isolate due to 17RM-150B indicating downscale. Manual isolation of Off-Gas is required approximately 15 minutes after Annunciator 09-3-2-38 alarmed.

Proposed Answer: B

Explanation: The Off-Gas isolation logic is setup such that timer 17-157 will initiate if either both 17RM-150A(B) trip on Hi-Hi, or one trips on Hi-Hi and the other is downscale/inop. Therefore, the downscale failure of 17RM-150B does not prevent automatic system isolation. Once the timer initiates, Off-Gas isolates approximately 15 minutes later.

A. Incorrect – The Off-Gas high radiation isolation logic includes a 15 minute time delay, therefore the system does not isolate immediately upon receipt of the Hi-Hi condition.
C. Incorrect – Downscale failure of 17RM-150B does not prevent automatic system isolation.
D. Incorrect – Downscale failure of 17RM-150B does not prevent automatic system isolation.

Technical Reference(s): AOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01A 1.05.c.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 #	2
	K/A #	202002 K1.03
	Importance Rating	3.7

Recirculation Flow Control

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

Proposed Question: #28

The plant is operating at 95% power with a down power in progress using Recirculation flow.

Given the following possible ***separate*** conditions:

(1) Upscale trip of a Recirculation flow unit.
(2) Low Recirculation MG Set lube oil pressure.
(3) High mismatch between Recirculation loop jet pump flows.

Which one of the following identifies which of these ***separate*** conditions would cause the core flow reduction to automatically stop?

A. (1) only

B. (2) only

C. (2) and (3) only

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

Proposed Answer: B

Explanation: Low Recirculation MG Set lube oil pressure causes an automatic scoop tube lock.

Upscale trip of a Recirculation flow unit could be caused by a drifting scoop tube, but does not cause an automatic scoop tube lock. Upscale trip of a Recirculation flow unit does cause an automatic control rod withdrawal block.

High mismatch between Recirculation loop jet pump flows could be caused by a drifting scoop tube, but does not cause an automatic scoop tube lock. High mismatch between Recirculation loop jet pump flows does affect Technical Specification compliance.

A. Incorrect – Low Recirculation MG Set lube oil pressure causes an automatic scoop tube lock. Upscale trip of a Recirculation flow unit could be caused by a drifting scoop tube, but does not cause an automatic scoop tube lock. Upscale trip of a Recirculation flow unit does cause an automatic control rod withdrawal block.
C. Incorrect – Low Recirculation MG Set lube oil pressure causes an automatic scoop tube lock. High mismatch between Recirculation loop jet pump flows could be caused by a drifting scoop tube, but does not cause an automatic scoop tube lock. High mismatch between Recirculation loop jet pump flows does affect Technical Specification compliance.
D. Incorrect – Neither upscale trip of a Recirculation flow unit nor high mismatch between Recirculation loop jet pump flows causes an automatic scoop tube lock.

Technical Reference(s): ARP 09-4-3-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.05.b.6

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 #	239003 K3.01
	Importance Rating	3.3

MSIV Leakage Control

Knowledge of the effect that a loss or malfunction of the MSIV LEAKAGE CONTROL SYSTEM will have on following: Radiation release to the environment: BWR-4,5,6 (P-Spec)

Proposed Question: #29

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

- All Main Steam Isolation Valves are closed.
- Both Main Steam Leakage Collection System (MSLCS) Train A and B **cannot** be placed in service due to multiple isolation valves failing closed.

Which one of the following describes the effect of the MSLCS Train A and B failure on radiation release?

A. There will be additional radiation release into the Drywell.

B. There will be additional radiation release into the Reactor Building.

C. There will be additional radiation release into the Turbine Building.

D. No additional radiation release will occur because the MSIVs are closed.

Proposed Answer: C

Explanation: Based on design of the MSLCS, AOP-39 only requires once train (A or B) to be placed in service in order to control the radiation release. Since the failure of both Train A and B due to isolation valves failing closed, additional radiation release will be admitted to the Turbine Bldg.

A. Incorrect – Only one train of MSLCS is required to control the radiation release. The MSLCS collects leak-by on the outboard MSIVs, not the inboard MSIVs, therefore the Drywell is not the area of concern.
B. Incorrect – Only one train of MSLCS is required to control the radiation release. Failure of all MSLCS would result in additional release to the Turbine Building, not the Reactor Building.
D. Incorrect – One train of MSLCS is required to control the radiation release. Failure of all MSLCS would still result in additional release to the Turbine Building due to outboard MSIV stem leak-by.

Technical Reference(s): AOP-39, OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29

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 #	241000 K4.06
	Importance Rating	3.6

Reactor/Turbine Pressure Regulator

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

Proposed Question: #30

A plant startup is in progress with the following:

- The Main Turbine is being started.
- Main Turbine speed is currently 800 rpm.

Which one of the following conditions will result in an automatic Main Turbine trip?

A. Exhaust Hood temperature of 230°F.

B. EHC pump discharge pressure of 680 psig.

C. Moisture Separator drain tank water level of 34".

D. Main Shaft Oil Pump (MSOP) discharge pressure of 75 psig.

Proposed Answer: B

Explanation: Low EHC oil pressure (<1100 psig), as sensed at the discharge of the EHC pumps, will cause an automatic Main Turbine trip.

A. Incorrect – High Main Turbine Exhaust Hood temperature (>225°F) originally caused a Main Turbine trip, but this trip has been removed by plant modification. A manual Main Turbine trip is still required if this temperature is exceeded.
C. Incorrect – High Moisture Separator drain tank water level does cause an automatic Main Turbine trip, but only if it exceeds 43". 34" is high, but below the trip setpoint.
D. Incorrect – Low MSOP discharge pressure (<105 psig) does normally provide an automatic Main Turbine trip, however this trip is bypassed when Main Turbine speed is <1300 rpm. This is to allow MSOP discharge pressure to build as the Main Turbine gains speed.

Technical Reference(s): OP-9

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.05.c.1

Question Source: Bank – LOI Bank #125779

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 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

Proposed Answer: A

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.

B. Incorrect – Total system flow rises due to flow to all 137 scram inlet valves.
C. Incorrect – The CRD Flow Control Valve closes in response to rising system flow.
D. Incorrect – The CRD Flow Control Valve closes in response to rising system flow. Total system flow rises due to flow to all 137 scram inlet valves.

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: Bank – 9/12 NRC #54

Question History: 9/12 NRC #54

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 #	223001 K6.09
Importance Rating	3.4

Primary Containment System and Auxiliaries

Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: Drywell vacuum relief system

Proposed Question: #32

Which one of the following describes the effect that a failed open Torus to Drywell vacuum breaker would have on Primary Containment response to a design-basis loss of coolant accident?

A. Both peak Torus and Drywell pressures would be lower than expected.

B. Both peak Torus and Drywell pressures would be greater than expected.

C. Peak Torus pressure would be higher than expected and peak Drywell pressure would be lower than expected.

D. Peak Torus pressure would be lower than expected and peak Drywell pressure would be higher than expected.

Proposed Answer: B

Explanation: With a failed open Torus to Drywell vacuum breaker, the Torus and Drywell airspaces are directly connected and the pressure-suppression feature of the Primary Containment is bypassed. In the event of a LOCA, this means steam will not be forced under the water in the Torus and condensed. Therefore, pressure will be higher in both the Torus and Drywell than expected for a design-basis LOCA.

A. Incorrect – This malfunction does effectively expand the size of the Drywell airspace, which would lead to lower pressures for very small LOCAs. However for a design-basis LOCA, the loss of pressure-suppression function would lead to higher peak pressures.
C. Incorrect – Peak Drywell pressure would also be higher during a design-basis LOCA.
D. Incorrect – Peak Torus pressure would also be higher during a design-basis LOCA.

Technical Reference(s): Technical Specification Bases 3.6.1.7 Action B.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.09.e

Question Source: Bank – 3/14 NRC #29

Question History: 3/14 NRC #29

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 #	256000 A1.04
	Importance Rating	2.9

Reactor Condensate

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

Proposed Question: #33

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

- Hotwell level is being maintained manually following maintenance on the automatic level controllers.
- Hotwell level is 83".
- Then, all Hotwell level controllers are returned to automatic with normal setpoints.

Which one of the following describes the resulting control of Hotwell level, as indicated on Panel 09-6 indicator 33LI-102, CNDSR LVL?

Hotwell level will...

A. rise due to makeup from the Condensate Storage Tanks.

B. rise due to makeup from the Demineralized Water System.

C. lower due to rejecting water to the Condensate Storage Tanks.

D. lower due to rejecting water to the Demineralized Water System.

Proposed Answer: A

Explanation: The Hotwell level control system utilizes three valves and three controllers to maintain Hotwell level under both normal and emergency conditions. During normal operation with all controllers in automatic at 100% power, Hotwell level will rise to approximately 87" due to makeup through the normal Hotwell makeup valve (33LCV-103), which injects water from the Condensate Storage Tanks to the Hotwell.

B. Incorrect – The emergency Hotwell makeup valve (33LCV-105) is capable of adding water to the Hotwell from the Demineralized Water System, but would only open if Hotwell level were lower than the initial 83".
C. Incorrect – Normal Hotwell level control is 87", which is higher than the initial 83", not lower.
D. Incorrect – Normal Hotwell level control is 87", which is higher than the initial 83", not lower.

Technical Reference(s): OP-3, SDLP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.05.a.1.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 #	201002 A2.04
	Importance Rating	3.2

RMCS

Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod block

Proposed Question: #34

A plant startup is in progress with the following:

- Reactor power is 18%.
- The Rod Worth Minimizer (RWM) is in service.
- Then, annunciator 09-5-2-1, RWM ROD BLOCK RPIS INOP, alarms due to failure of the RWM.

Which one of the following describes the impact of the inoperable RWM on control rod movement using Reactor Manual Control (RMCS)

AND

the need to station a second individual with no concurrent duties to independently verify continued control rod movements with the RWM bypassed, per OP-64, Rod Worth Minimizer?

	Impact on Control Rod Movement with RMCS	Per OP-64, stationing a second individual with no concurrent duties to verify continued control rod movements is...
A.	Withdrawal is blocked, only.	required.
B.	Withdrawal is blocked, only.	NOT required.
C.	Insertion and withdrawal are blocked.	required.
D.	Insertion and withdrawal are blocked.	NOT required.

Proposed Answer: D

Explanation: With the RWM originally in service and then becoming inoperable, both insert and withdraw rod blocks are generated. Reactor power is currently above the low power setpoint (>10% by Technical Specifications, actually set at 16%), therefore OP-64 allows bypassing the RWM and continuing control rod withdrawals without stationing a second individual with no concurrent duties to verify control rod movements.

A. Incorrect – While most rod blocks prevent withdrawal only, an inoperable RWM blocks both control rod insertion and withdrawal. Since power is above the LPSP, stationing a second individual is not required.
B. Incorrect – While most rod blocks prevent withdrawal only, an inoperable RWM blocks both control rod insertion and withdrawal.
C. Incorrect – Since power is above the LPSP, stationing a second individual is not required.

Technical Reference(s): OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.09.a

Question Source: New

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 #	233000 A3.02
	Importance Rating	2.6

Fuel Pool Cooling/Cleanup

Ability to monitor automatic operations of the FUEL POOL COOLING AND CLEAN-UP including: Pump trip(s)

Proposed Question: #35

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

- The Fuel Pool Cooling and Cleanup system is operating with the following:
  - 19P-1A, FUEL POOL CLN UP RECIRC PUMP A, is running.
  - Skimmer Surge Tank makeup is NOT in progress.

Then, a leak develops from the Fuel Pool Skimmer Surge tanks.

- Annunciator 09-3-1-9, FUEL POOL COOL & CLN UP TROUBLE, alarms.
- Skimmer Surge tank level indicates 1 foot and slowly lowering.

Which one of the following describes the automatic response of the Fuel Pool Cooling and Cleanup system?

Automatic addition of water to the Skimmer Surge tanks is…

A. in progress and 19P-1A is tripped.

B. in progress and 19P-1A is running.

C. NOT in progress and 19P-1A is tripped.

D. NOT in progress and 19P-1A is running.

Proposed Answer: C

Explanation: Annunciator 09-3-1-9 first alarms on low Skimmer Surge tank level of 3.6 feet. At 1.8 feet, the running Fuel Pool Recirc pump trips. However, makeup to the Skimmer Surge tanks does not automatically initiate. Manual action is required to add water to the tanks.

A. Incorrect – Makeup to the Skimmer Surge tanks does not automatically initiate. Manual action is required to add water to the tanks.
B. Incorrect – Makeup to the Skimmer Surge tanks does not automatically initiate. Manual action is required to add water to the tanks. 19P-1A tripped when Skimmer Surge tank level lowered to 1.8 feet.
D. Incorrect – 19P-1A tripped when Skimmer Surge tank level lowered to 1.8 feet.

Technical Reference(s): ARP-09-3-1-9, OP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-19 1.14.a

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 #	290003 A4.02
Importance Rating	2.8

Control Room HVAC

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

Proposed Question: #36

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

- AOP-39, Loss of Coolant, requires isolating Control Room and Relay Room Ventilation per OP-55B, Control Room Ventilation and Cooling.
- An operator places the Control Room Ventilation ISOL & PURGE CNTRL switch in the ISOL position.

Which one of the following describes the resulting status of the Control Room Ventilation fans?

	70AHU-3A(B), CLG SUPP	70FN-4A(B), EXH FN	70FN-6A(B), FRESH AIR SUPP
A.	One running	One running	One running
B.	One running	One running	Both stopped
C.	One running	Both stopped	One running
D.	Both stopped	Both stopped	One running

Proposed Answer: A

Explanation: Prior to placing the Control Room Ventilation ISOL & PURGE CNTRL switch in the ISOL position, the normal Control Room Ventilation has one 70AHU-3A(B), CLG SUPP, and one 70FN-4A(B), EXH FN, running. When the ISOL & PURGE CNTRL switch is placed in the ISOL position, multiple dampers reposition and one 70FN-6A(B), FRESH AIR SUPP, starts. 70AHU-3A(B), CLG SUPP, and 70FN-4A(B), EXH FN, remain running as in the initial lineup.

B. Incorrect – 70FN-6A(B) are normally both stopped, but one of these fans starts upon placing ISOL & PURGE CNTRL switch in the ISOL position.
C. Incorrect – While multiple dampers reposition to isolate Control Room Ventilation exhaust, 70FN-4A(B), EXH FN, remains running as in the initial normal lineup.
D. Incorrect – While multiple dampers reposition to isolate Control Room Ventilation exhaust, 70AHU-3A(B), CLG SUPP, and 70FN-4A(B), EXH FN, remain running as in the initial normal lineup.

Technical Reference(s): OP-55B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.06.b

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 #	2
	K/A #	204000 2.1.28
	Importance Rating	4.1

RWCU

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

Proposed Question: #37

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

- Boron is being injected with Standby Liquid Control pump A.
- Reactor water level has been maintained above 180 inches so far throughout the transient.

Given the following Reactor Water Cleanup valves:

(1) 12MOV-15, CLN UP INBD SUCT ISOL VALVE
(2) 12MOV-18, CLN UP OUTBD SUCT ISOL VALVE
(3) 12MOV-69, CLN UP RETURN ISOL VALVE

Which one of the following identifies which of these valves have received an isolation signal?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: C

Explanation: SLC initiation causes 12MOV-18 and 12MOV-69 to receive an isolation signal, but not 12MOV-15. 12MOV-15 would close if Reactor water level went <177 inches or Drywell pressure was >2.7 psig, but neither of these conditions are present.

A. Incorrect – 12MOV-15 does not receive an isolation signal on SLC initiation, and no low Reactor water level or high Drywell pressure condition is present. 12MOV-69 has received an isolation signal due to SLC initiation.
B. Incorrect – 12MOV-15 does not receive an isolation signal on SLC initiation, and no low Reactor water level or high Drywell pressure condition is present. 12MOV-18 has received an isolation signal due to SLC initiation.
D. Incorrect – 12MOV-15 does not receive an isolation signal on SLC initiation, and no low Reactor water level or high Drywell pressure condition is present.

Technical Reference(s): OP-17, AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12 1.05.c.1 & 4

Question Source: Modified Bank – 3/12 NRC #31

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 #	201003 A1.03
	Importance Rating	2.9

Control Rod and Drive Mechanism

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: CRD drive water flow

Proposed Question: #38

A plant startup is in progress with the following:

- An operator attempts to notch withdraw control rod 22-07 from position 08 to 10.
- Drive water flow initially indicates 4 gpm and the control rod moves into the core as expected for a control rod withdrawal.
- Then, drive water flow becomes 0 gpm and the control rod settles at the initial position.

Which one of the following solenoid operated directional control valve failures would cause the observed indications?

Directional control valve...

A. 120, WITHDRAW EXHAUST AND SETTLE VALVE, stuck open

B. 121, INSERT EXHAUST VALVE, stuck open

C. 122, WITHDRAW SUPPLY VALVE, stuck closed

D. 123, INSERT SUPPLY VALVE, stuck closed

Proposed Answer: C

Explanation: When the rod movement control switch is moved to the ROD OUT position, the RMCS timer opens the inlet drive water valve (123) and the exhaust valve (121) and the control rod moves into the core and off the collet fingers. RMCS then should open the withdraw valve (122) and exhaust valve (120). If the withdraw valve (122) does NOT open no pressure is applied to the collet fingers or the area above the drive piston the control rod will settle back onto the collet finger at its original position. This is further indicated by the 0.0 gpm drive water flow.

A. Incorrect – If 120 was stuck open, the control rod would not insert initially, as drive water flow through 123 would flow directly to the exhaust header through the open 120.
B. Incorrect – If 121 was stuck open, the control rod would not withdraw, but drive water flow would indicate high since drive water would flow directly through 122 to the exhaust header through the open 121.
D. Incorrect – If 123 was stuck closed, the control rod would not withdraw, but it also wouldn't initially insert or show 4 gpm drive water flow initially.

Technical Reference(s): SDLP-03F

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03F 1.10.e

Question Source: Bank – 2010 NRC #55

Question History: 2010 NRC #55

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK1.03
	Importance Rating	4.2

SCRAM Conditions 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: Boron effects on reactor power (SBLC)

Proposed Question: #39

A failure to scram has occurred with the following:

- Reactor pressure is being controlled 800-1000 psig on Turbine Bypass Valves.
- Standby Liquid Control (SLC) has been initiated and is injecting from an initial tank level of 84%.

Which one of the following identifies the amount of SLC injection required before Reactor depressurization is allowed, in accordance with EOP-3, Failure To Scram?

Reactor depressurization is **first** allowed when the...

A. Hot Shutdown Boron Weight is injected, as indicated by an SLC tank level of 58%.

B. Hot Shutdown Boron Weight is injected, as indicated by an SLC tank level of 39%.

C. Cold Shutdown Boron Weight is injected, as indicated by an SLC tank level of 58%.

D. Cold Shutdown Boron Weight is injected, as indicated by an SLC tank level of 39%.

Proposed Answer: D

Explanation: EOP-3 requires waiting until SLC tank level drops by 45% before initiating a Reactor depressurization. This is the amount of boron injection required to meet the Cold Shutdown Boron Weight. Achieving Cold Shutdown Boron Weight assures that Reactor power will be sufficiently suppressed even with the Reactor at a lower temperature. Therefore Reactor depressurization is first allowed when SLC tank level reaches 39% (84%-45%=39%).

A. Incorrect – At 58%, Hot Shutdown Boron Weight will have been injected (26% of tank level), which allows restoring Reactor water level, but not initiating a Reactor depressurization.
B. Incorrect – Hot Shutdown Boron Weight allows restoring Reactor water level, but not initiating a Reactor depressurization, and is first reached at a tank level of 58%.
C. Incorrect – 58% is the tank level for Hot Shutdown Boron Weight, not Cold Shutdown Boron Weight.

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 #17

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 #	295021 AK1.02
Importance Rating	3.3

Loss of Shutdown Cooling

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Thermal stratification

Proposed Question: #40

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

- RHR loop B is operating in the Shutdown Cooling (SDC) lineup.
- Reactor Water Recirculation (RWR) pump A is running and RWR pump B is secured.
- Reactor water level is 217 inches.

Then, 10300 and 10500 buses de-energize due to a sustained electrical fault.

Which one of the following describes the impact on reactor coolant temperature indication and the associated reason?

Reactor coolant temperature indication...

A. will remain valid due to sufficient SDC flow.

B. will remain valid due to sufficient RWR flow.

C. will remain valid due to sufficient natural circulation.

D. is NOT assured to remain valid under these conditions.

Proposed Answer: D

Explanation: For Reactor coolant temperature indications to remain valid, sufficient flow through the Reactor must be maintained to prevent thermal stratification. There are three recognized methods to assure valid Reactor coolant temperature indications: maintain flow with an RWR pump, maintain flow with an RHR pump in the SDC lineup, and/or maintain Reactor water level greater than 234.5 inches to provide adequate natural circulation. The loss of 10300 bus results in loss of RWR pump A. The loss of 10500 bus results in loss of RPS A, which also causes a loss of SDC. Since Reactor water level is below 234.5 inches, insufficient flow through the core exists to prevent thermal stratification and invalid Reactor coolant temperature indications.

A. Incorrect – SDC flow is lost due to loss of 10500 bus, which causes loss of RPS A and loss of SDC.
B. Incorrect – RWR flow is lost due to loss of 10300 bus.
C. Incorrect – Since Reactor water level is below 234.5 inches, insufficient natural circulation exists to assure accurate Reactor coolant temperature indications.

Technical Reference(s): AOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.09.e

Question Source: Bank – 3/12 NRC #29

Question History: 3/12 NRC #29

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 #	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

A plant shutdown is in progress with the following:

- A fire occurred in the Drywell following de-inerting.
- The Reactor was scrammed and EOP-2, RPV Control, was entered.
- Drywell temperature is 320°F and stable.
- Reactor pressure is 620 psig and stable.

Given the following Reactor water level instruments and current indications:

(1) Narrow Range: 165 inches
(2) Refuel Zone: 180 inches
(3) Wide Range: 160 inches

Note: A portion of EOP Figure 1 is provided on the following page.

Which one of the following identifies which of these reactor water level instruments can currently be used to determine actual reactor water level, in accordance with EOP-2, RPV Control?

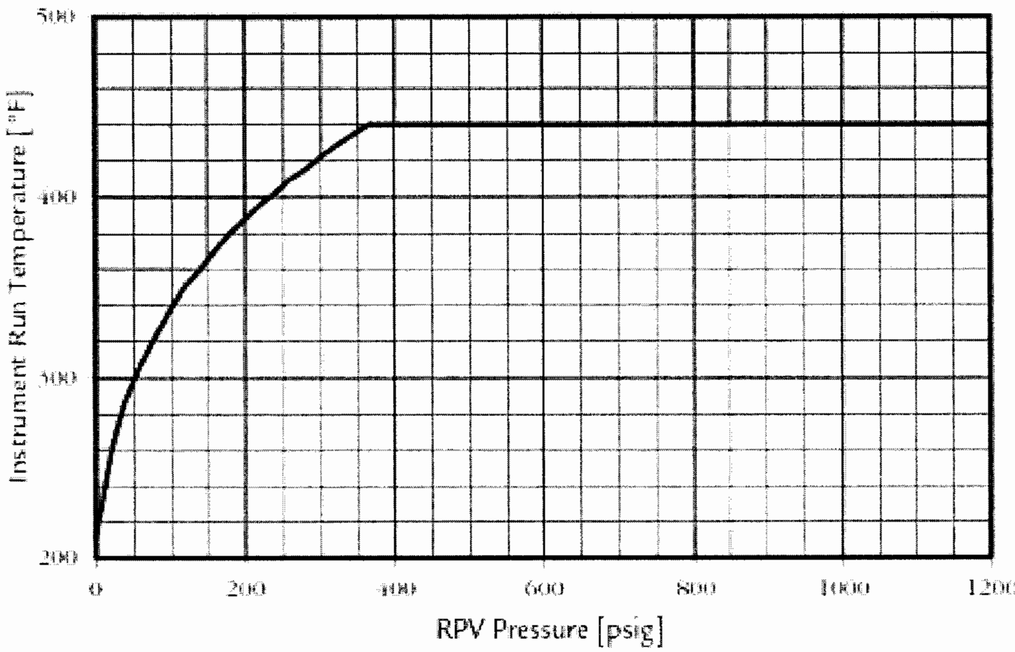
A. (1) only

B. (3) only

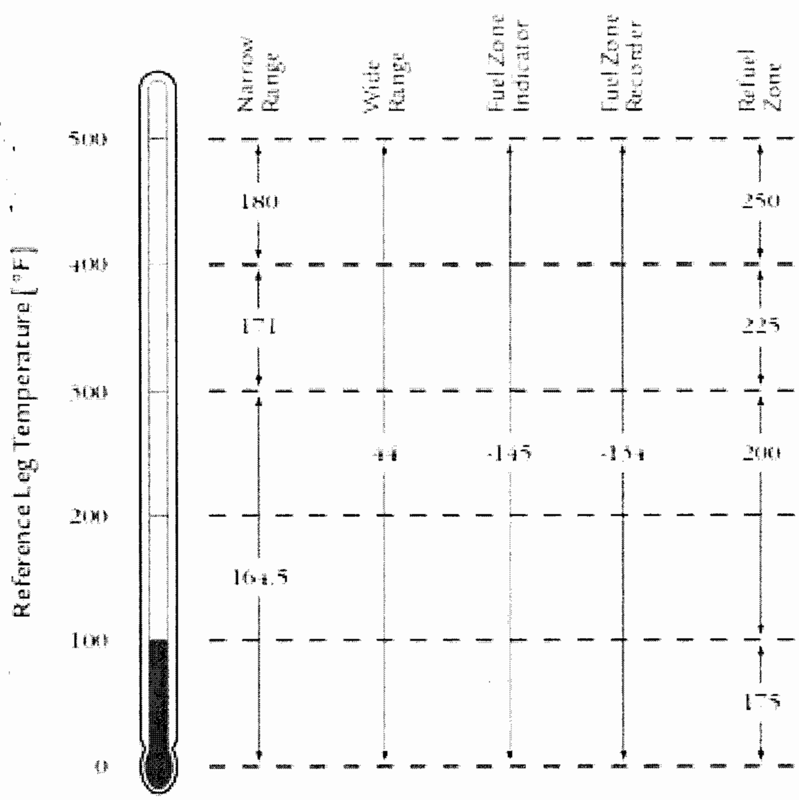
C. (1) and (2) only

D. (2) and (3) only

RPV Saturation Temperature



Minimum Usable Indicating Levels [in.]



Proposed Answer: B

Explanation: Elevated Drywell temperature affects the qualification of the Reactor water level instruments. Drywell temperature and Reactor pressure are on the "good" side of the RPV Saturation Temperature curve. However, both Narrow Range and Refuel Zone instruments are indicating below their Minimum Usable Indicating Levels of 171 inches and 225 inches, respectively. Therefore, of the given instruments, only Wide Range is currently available to determine actual Reactor water level.

A. Incorrect – Narrow Range is indicating below the Minimum Usable Indicated Level for the given Drywell temperature, and therefore cannot be used to determine current Reactor water level.
C. Incorrect – Narrow Range and Refuel Zone are indicating below the Minimum Usable Indicated Levels for the given Drywell temperature, and therefore cannot be used to determine current Reactor water level.
D. Incorrect – Refuel Zone is indicating below the Minimum Usable Indicated Level for the given Drywell temperature, and therefore cannot be used to determine current Reactor water level.

Technical Reference(s): EOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.01

Question Source: Bank – 9/12 NRC #14

Question History: 9/12 NRC #14

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EK2.01
	Importance Rating	4.1

High Reactor Pressure

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: RPS

Proposed Question: #42

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

- Annunciator 09-5-1-22, RPS HI RX PRESS TRIP, alarms.
- The following is reported:
  - Pressure transmitter 02-3PT-55A senses 1038 psig and steady.
  - Pressure transmitter 02-3PT-55B senses 1065 psig and steady.
  - Pressure transmitter 02-3PT-55C senses 1051 psig and steady.
  - Pressure transmitter 02-3PT-55D senses 1083 psig and steady.

Which one of the following identifies the required response of the RPS Auto Scram Annunciators, 09-5-1-03 and 09-5-1-04, based on these sensed pressures?

	09-5-1-03, RPS A AUTO SCRAM	09-5-1-04, RPS B AUTO SCRAM
A.	In alarm	In alarm
B.	In alarm	NOT in alarm
C.	NOT in alarm	In alarm
D.	NOT in alarm	NOT in alarm

Proposed Answer: C

Explanation: Annunciator 09-5-1-22 setpoint is 1062 psig (TS setpoint must be <1080 psig). Both detectors 02-3PT-55B and D are above the setpoint. These detectors input to RPS B. Only one of these two detectors must be above setpoint to cause a half scram and Annunciator 09-5-1-04 to alarm.

A. Incorrect – Pressure transmitter 02-3PT-55C inputs to RPS A and is at the 1051 psig setpoint for Annunciator 09-5-1-38, RX PRESS ALARM HI, but is below the 1062 psig setpoint to cause a half scram.
B. Incorrect – Pressure transmitter 02-3PT-55C inputs to RPS A and is at the 1051 psig setpoint for Annunciator 09-5-1-38, RX PRESS ALARM HI, but is below the 1062 psig setpoint to cause a half scram. Pressure transmitter 02-3PT-55D is above the 1062 psig setpoint to cause a half scram on RPS B.
D. Incorrect – Pressure transmitter 02-3PT-55D is above the 1062 psig setpoint to cause a half scram on RPS B.

Technical Reference(s): ARPs 09-5-1-22, 09-5-1-03, 9-5-1-04

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.11, 1.12

Question Source: Bank – 9/12 NRC #12

Question History: 9/12 NRC #12

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 #	295016 AK2.02
	Importance Rating	4.0

Control Room Abandonment

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific

Proposed Question: #43

The plant was operating at 100% power when the following occurred:

- The Control Room has been abandoned due to a fire.
- **No** Immediate Actions have been taken in the Control Room.
- AOP-43, Plant Shutdown From Outside the Control Room, is being performed and requires controlling the following valves:
    - 10MOV-25B, LPCI INBD INJ VLV
    - 10MOV-16B, MIN FLOW VLV

Which one of the following describes the panel location(s) from which these valves are controlled, in accordance with AOP-43?

	10MOV-25B, LPCI INBD INJ VLV	10MOV-16B, MIN FLOW VLV
A.	25RSP (Reactor Building 300' North)	25RSP (Reactor Building 300' North)
B.	25RSP (Reactor Building 300' North)	25ASP-2 (East Crescent Stairway)
C.	25ASP-3 (North Emergency Switchgear Room)	25RSP (Reactor Building 300' North)
D.	25ASP-3 (North Emergency Switchgear Room)	25ASP-2 (East Crescent Stairway)

Proposed Answer: B

Explanation: 25RSP contains the control and indications for 10MOV-25B, LPCI INBD INJ VLV, which must be opened during execution of AOP-43. 25ASP-2 contains the control and indications for 10MOV-16B, MIN FLOW VLV, which must be operated during execution of AOP-43.

A. Incorrect – 10MOV-16B, MIN FLOW VLV, is controlled from 25ASP-2, not 25RSP.
C. Incorrect – 10MOV-25B, LPCI INBD INJ VLV, is controlled from 25RSP, not 25ASP-3. 10MOV-16B, MIN FLOW VLV, is controlled from 25ASP-2, not 25RSP.
D. Incorrect – 10MOV-25B, LPCI INBD INJ VLV, is controlled from 25RSP, not 25ASP-3.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.11.b.9/10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AK2.04
	Importance Rating	3.4

Partial or Complete Loss of AC Power

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: A.C. electrical loads

Proposed Question: #44

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

- MOD 10017, NORTH-SOUTH 115KV BUS DISC SW, is tagged open for maintenance.
- Then, a Reactor scram occurs.
- Breaker 10012, NMP FITZ 115KV LINE 4 BKR, trips open and remains open.

Which one of the following describes the resulting number of Circulating Water pumps, if any, that are available five (5) minutes later?

A. 0

B. 1

C. 2

D. 3

Proposed Answer: B

Explanation: The three Circulating Water pumps are powered from separate buses – A from 10300, B from 10400, and C from 10700. The Reactor scram causes a Main Generator trip on reverse power, which causes loss of power to Bus 10700 and loss of Circulating Water pump C. The trip of Breaker 10012 with MOD 10017 open causes loss of power to Bus 10300 and loss of Circulating Water pump A. Circulating Water pump B remains available with power to Bus 10400 from Line 3.

A. Incorrect – The given electrical alignment maintains power available to Circulating Water pump B from Line 3.
C. Incorrect – Two Circulating Water pumps are lost due to different reasons. One is lost due to the scram and another is lost due to the trip of Breaker 10012.
D. Incorrect – Three Circulating Water pumps are initially available, but one is lost due to the scram and another is lost due to the trip of Breaker 10012.

Technical Reference(s): OP-4, SDLP-71o Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-36 1.10.a

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

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AK3.02
	Importance Rating	3.5

Partial or Complete Loss of Instrument Air

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Standby air compressor operation

Proposed Question: #45

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

- Air Compressor A is operating.
- Air Compressor B is in standby as the 1st standby compressor.
- Air Compressor C is in standby as the 2nd standby compressor.

Then, the following occur:

- Elevated levels of air leakage have been causing extended loading of Air Compressor A.
- The breaker for Air Compressor A trips due to motor overload.
- Air pressure is 100 psig and lowering slowly.

Which one of the following describes the response of Air Compressors B and C?

A. Both Air Compressors B and C start due to an Air Compressor A breaker position signal.

B. Air Compressor B starts due to an Air Compressor A breaker position signal. Air Compressor C starts due to a low air pressure signal.

C. Air Compressor B starts due to a low air pressure signal. Air Compressor C remains in standby.

D. Both Air Compressors B and C start due to a low air pressure signal.

Proposed Answer: D

Explanation: Following trip of the running Air Compressor, the standby Air Compressors will start if air pressure drops far enough. Neither the 1st nor 2nd standby Air Compressor will automatically start based on a breaker position signal from the tripped Air Compressor. The 1st standby Air Compressor starts when air pressure lowers to 107 psig. The 2nd standby Air Compressor starts if air pressure lowers to 104 psig.

A. Incorrect – Air Compressors B and C have both started, but due to a low air pressure signal, not a breaker position signal.
B. Incorrect – Air Compressor B has started, but due to a low air pressure signal, not a breaker position signal.
C. Incorrect – Air Compressor B has started due to a low air pressure signal (107 psig), but so has Air Compressor C (104 psig).

Technical Reference(s): OP-39, ARP 09-6-2-08

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.c.3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

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

High Drywell Pressure

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

Proposed Question: #46

Which one of the following describes the reason for the Reactor scram on Drywell high pressure, in accordance with the Final Safety Analysis Report (FSAR)?

A. Decrease the probability of exceeding Primary Containment design limits following a complete loss of Drywell cooling.

B. Prevent the loss of equipment inside the Drywell needed for accident mitigation following a complete loss of Drywell cooling.

C. Prevent the loss of equipment inside the Drywell needed for accident mitigation following a break in the Reactor Coolant Pressure Boundary.

D. Decrease the probability of fuel damage following a break in the Reactor Coolant Pressure Boundary.

Proposed Answer: D

Explanation: FSAR section 7.2.3.6.h describes the basis for the Reactor scram on Drywell high pressure:

> "High pressure inside the primary containment could indicate a break in the Reactor Coolant Pressure Boundary. It is prudent to scram the reactor in such a situation to minimize the possibility of fuel damage and to reduce the addition of energy from the core to the coolant."

A. Incorrect – The basis is for a break in the Reactor Coolant Pressure Boundary, not a loss of Drywell cooling. A loss of Drywell cooling may result in reaching the Drywell high pressure scram setpoint.
B. Incorrect – The basis is for a break in the Reactor Coolant Pressure Boundary, not a loss of Drywell cooling. A loss of Drywell cooling may result in reaching the Drywell high pressure scram setpoint.
C. Incorrect – The basis is for protection of the fuel cladding, not equipment needed for accident mitigation that is within the Drywell.

Technical Reference(s): FSAR Section 7.2.3.6.h

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.02

Question Source: Bank – SSES 2010 NRC #47

Question History: SSES 2010 NRC #47

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 #	295030 EK3.07
	Importance Rating	3.5

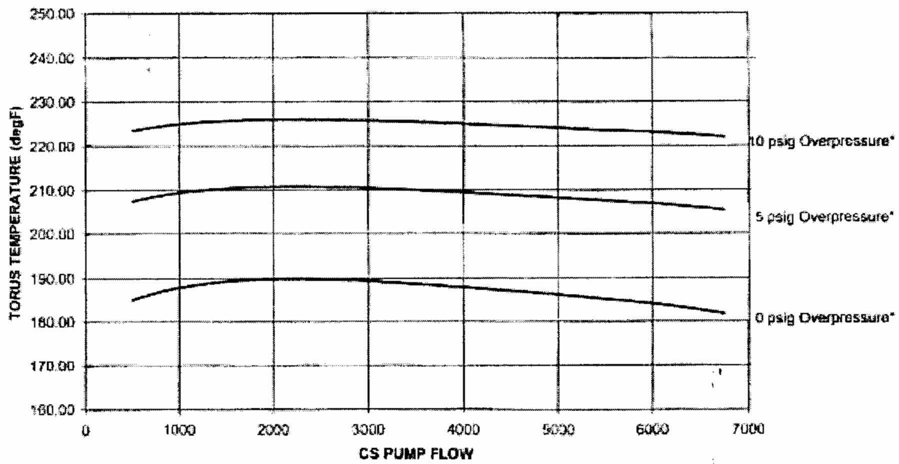
Low Suppression Pool Water Level

Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: NPSH considerations for ECCS pumps

Proposed Question: #47

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

- Reactor water level is +10 inches and slowly rising.
- Core Spray pump A is injecting 5000 gpm to the Reactor.
- Torus pressure is 4 psig and slowly rising.
- Torus water level is 10.5 feet and slowly lowering.
- Torus water temperature is 195°F and slowly rising.
- Given the following limit from OP-14, Core Spray System:



*Torus Overpressure = Torus Pressure + 0.4(Torus Water Level - 1.08)

Which one of the following describes the reason for this limit and the current status of Core Spray pump A operation with respect to this limit, in accordance with OP-14?

The reason for this limit is to ___(1)___.
Core Spray pump A is currently operating on the ___(2)___ side of the limit.

	(1)	(2)
A.	prevent air entrainment	satisfactory
B.	prevent air entrainment	unsatisfactory
C.	ensure adequate net positive suction head	satisfactory
D.	ensure adequate net positive suction head	unsatisfactory

Proposed Answer: C

Explanation: The given limit is the Core Spray Pump NPSH Limit. It is included in the OP-14 Posted Attachment along with the Vortex Limit. The purpose of the Core Spray Pump NPSH Limit is to ensure the Core Spray pump has adequate net positive suction head (NPSH) for the given combination of Torus pressure, Torus water level, Torus temperature, and Core Spray flow. The given Torus pressure and water level allow use of the 5 psig overpressure curve. The combination of Torus temperature and Core Spray flow are below the 5 psig overpressure curve, which means there Core Spray pump operation is satisfactory in relation to this limit.

A. Incorrect – The purpose of this limit is to ensure adequate NPSH. Preventing air entrainment is the purpose of the separate Vortex Limit.
B. Incorrect – The purpose of this limit is to ensure adequate NPSH. Preventing air entrainment is the purpose of the separate Vortex Limit. The given parameters place Core Spray pump A operation below the limit, which is the satisfactory side.
D. Incorrect – The given parameters place Core Spray pump A operation below the limit, which is the satisfactory side.

Technical Reference(s): OP-14, MIT-301.11b

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 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 #	1
	Group #	1
	K/A #	295031 EA1.12
	Importance Rating	3.9

Reactor Low Water Level

Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Feedwater

Proposed Question: #48

The plant has experienced a scram with the following:

- Multiple failures of HPCI and RCIC occurred.
- Reactor water level is now +50 inches and lowering slowly.
- Reactor pressure is 800 psig and lowering slowly.

Which one of the following describes the ability to inject to the reactor using **(1)** Feedwater pumps and/or **(2)** Condensate Booster pumps at the current reactor pressure?

	(1) Feedwater pumps	(2) Condensate Booster pumps
A.	can inject	can inject
B.	can inject	**cannot** inject
C.	**cannot** inject	can inject
D.	**cannot** inject	**cannot** inject

Proposed Answer: D

Explanation: With Reactor water level <+59.5 inches, the MSIVs are closed. This takes away the steam supply to the turbine-driven Feedwater pumps, making them unavailable for re-start and injection. Condensate Booster pumps are able to inject to the Reactor at a maximum pressure of 700 psig, therefore they are unable to inject at the current Reactor pressure of 800 psig.

A. Incorrect – Feedwater pumps are NOT available for re-start and injection because MSIVs are closed on low Reactor water level. Condensate Booster pumps do NOT provide enough discharge pressure to inject at the current Reactor pressure without a Feedwater pump also operating.
B. Incorrect – Feedwater pumps are NOT available for re-start and injection because MSIVs are closed on low Reactor water level.
C. Incorrect – Condensate Booster pumps do NOT provide enough discharge pressure to inject at the current Reactor pressure without a Feedwater pump also operating.

Technical Reference(s): AOP-15, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.05.a.2.c and 1.05.a.3.a

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 #	1
	Group #	1
	K/A #	600000 AA1.06
	Importance Rating	3.0

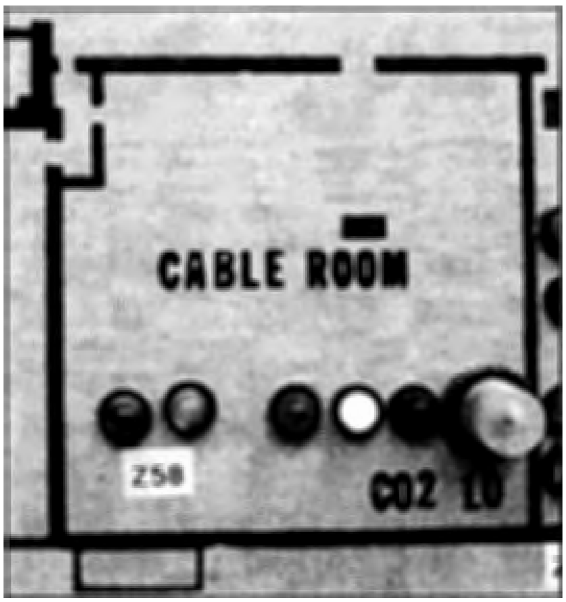
Plant Fire On-site

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: #49

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

- The Fire Protection Panel (FPP) alarms.
- The following indications are present for the Cable Spreading Room:



Which one of the following describes the status of the Fire Protection system for the Cable Spreading Room?

A. A failure has occurred with the zone control and/or detection circuit.

B. Heat has been detected in the area, but CO_2 has NOT been discharged.

C. Heat has been detected in the area and CO_2 is currently discharging.

D. Heat has been detected in the area, CO_2 has discharged, and CO_2 discharge has been manually locked-out.

Proposed Answer: B

Explanation: The green light is NOT illuminated, indicating there is no problem with the control/detection circuitry. The yellow light is illuminated, indicating heat has been detected in the area. The red light is extinguished, indicating CO_2 has NOT been discharged. The white light is extinguished, indicating the zone has NOT been manually locked-out.

A. Incorrect – The green light is NOT illuminated, indicating there is no problem with the control/detection circuitry.
C. Incorrect – The red light is extinguished, indicating CO_2 has NOT been discharged.
D. Incorrect – The red light is extinguished, indicating CO_2 has NOT been discharged. The white light is extinguished, indicating the zone has NOT been manually locked-out.

Technical Reference(s): SDLP-76

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76 1.11.a.11

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 #	1
	Group #	1
	K/A #	295001 AA1.05
	Importance Rating	3.3

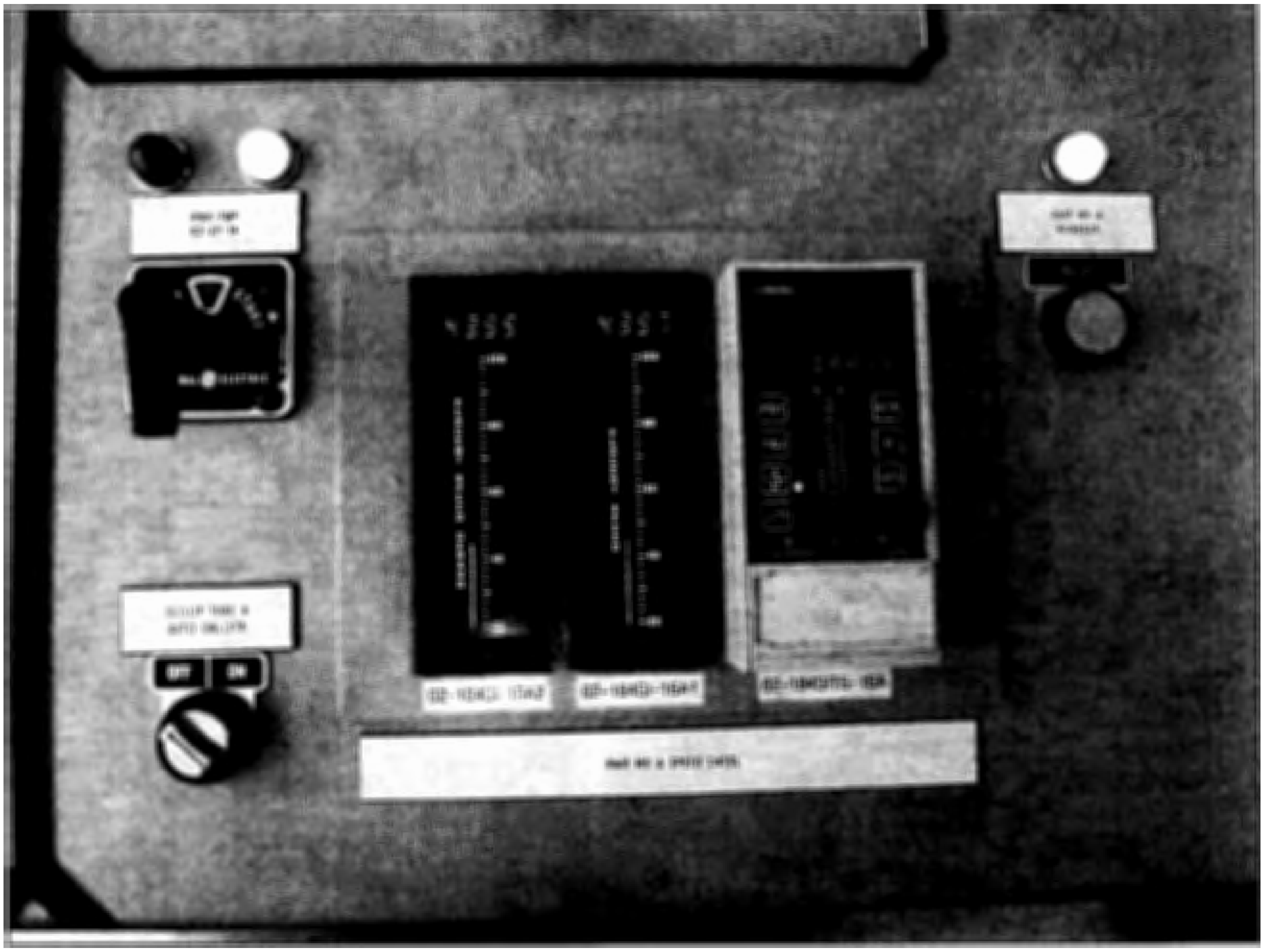
Partial or Complete Loss of Forced Core Flow Circulation

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Recirculation flow control system

Proposed Question: #50

The plant is operating at 65% power with the following:

- A loss of all Panel 09-5 annunciators occurs.
- Then, during monitoring of Panel 09-5, the following stable indications are observed for Reactor Water Recirculation (RWR) pump A:



Which one of the following describes the indicated condition of RWR pump A?

RWR pump A...

A. has tripped.

B. has received a 30% runback signal.

C. has received a 44% runback signal.

D. is in a normal condition for 65% power operation.

Proposed Answer: C

Explanation: The given indications show RWR pump A control switch red flagged with red light on and green light off. This indicates RWR pump A is running. The RWR pump A speed indicators show speed demand and actual speed to be ~44% with the RWR MG A RUNBACK red light on. This indicates a 44% runback has occurred. Normal indications for 65% power would have the RWR MG A RUNBACK red light off.

A. Incorrect – RWR pump A control switch is red flagged with red light on and green light off, indicating the pump has not tripped.
B. Incorrect – The RWR pump A speed indicators show speed demand and actual speed to be ~44%, indicating this is a 44% runback, not a 30% runback.
D. Incorrect – Normal indications for 65% power would have the RWR MG A RUNBACK red light off.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.05.b.3, 1.12.a

Question Source: Bank – 9/12 NRC #56

Question History: 9/12 NRC #56

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.01
	Importance Rating	3.3

Partial or Complete Loss of CCW

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Component temperatures

Proposed Question: #51

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

- A blockage has developed in the RBCLC supply line to the RWR pump 'A' bearing cooler in the Drywell.
- Annunciator 09-4-3-33, RWR PMP A OR B TMP HI, alarms.

Which one of the following describes the location of the corresponding temperature indication and the system response if this temperature continues to rise?

The corresponding temperature indication is located on Panel...

A. 09-4. RWR pump A will automatically trip if temperature continues to rise.

B. 09-4. RWR pump A will NOT automatically trip if temperature continues to rise.

C. 09-21. RWR pump A will automatically trip if temperature continues to rise.

D. 09-21. RWR pump A will NOT automatically trip if temperature continues to rise.

Proposed Answer: D

Explanation: While this alarm is received on Panel 09-4, the actual temperature indication is located on a recorder on Panel 09-21. RWR pump A does not automatically trip on high bearing temperature and would require manual shutdown if temperature exceeds 220°F.

A. Incorrect – While this alarm is received on Panel 09-4, the actual temperature indication is located on a recorder on Panel 09-21.
B. Incorrect – While this alarm is received on Panel 09-4, the actual temperature indication is located on a recorder on Panel 09-21. If RWR pumps A bearing temperature reaches 220°F, manual shutdown is required, but the pump will not automatically trip. Other RWR high temperature alarms, such as high MG fluid drive oil temperature, do result in automatic trip.
C. Incorrect – If RWR pumps A bearing temperature reaches 220°F, manual shutdown is required, but the pump will not automatically trip. Other RWR high temperature alarms, such as high MG fluid drive oil temperature, do result in automatic trip.

Technical Reference(s): ARP 09-4-3-33

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H

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 #	1
	Group #	1
	K/A #	295005 AA2.05
	Importance Rating	3.8

Main Turbine Generator Trip

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

Proposed Question: #52

The plant is operating at 22% power with the following:

- A Main Turbine trip occurs.
- Three (3) minutes later, Reactor power is 23% and rising slowly.

Which one of the following describes the response of Reactor power?

Reactor power has...

A. responded as expected.

B. NOT responded as expected because a Reactor scram should have occurred.

C. NOT responded as expected because it should be stable at the pre-transient value.

D. NOT responded as expected because it should be near the pre-transient value and lowering slowly.

Proposed Answer: A

Explanation: With Reactor power initially below 29%, a Reactor scram does not occur on a Main Turbine trip. Reactor power initially remains near the pre-transient value and will begin to rise due to the loss of Feedwater heating from the Main Turbine trip. The given conditions match this expected response.

B. Incorrect – Reactor power has responded as expected for a Main Turbine trip <29% power. If Reactor power was >29%, an automatic Reactor scram would have occurred.
C. Incorrect – Reactor power has responded as expected for a Main Turbine trip <29% power. Three minutes after the Main Turbine trip, the loss of Feedwater heating is causing Reactor power to rise.
D. Incorrect – Reactor power has responded as expected for a Main Turbine trip <29% power. This transient causes Feedwater temperature entering the Reactor to lower, not rise; therefore, Reactor power rises, not lowers.

Technical Reference(s): AOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.09.c

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 #	1
	Group #	1
	K/A #	295038 EA2.03
	Importance Rating	3.5

High Off-site Release Rate

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

Proposed Question: #53

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

- A steam leak has developed from the RCIC supply piping into the Reactor Building.
- Reactor Building ventilation exhaust radiation monitors indicate upscale.
- RCIC room area temperature is 150°F and slowly rising.
- West Crescent area radiation level is 300 mR/hr and slowly rising.
- Reactor Building differential pressure is +0.25" H_20 and stable.
- The Shift Manager has declared an Alert emergency condition due to high offsite release rate.

Note: Portions of EOP-5, Secondary Containment Control, are provided on the following pages.

Which one of the following identifies the total number of entry conditions met for EOP-5, Secondary Containment Control, and EOP-6, Radioactivity Release Control?

	EOP-5	EOP-6
A.	3	0
B.	3	1
C.	4	0
D.	4	1

REACTOR BUILDING AREA RADIATION LEVELS			
AREA	INSTRUMENT	MAXIMUM NORMAL	MAXIMUM SAFE
Spent Fuel Pool	18RIA-051-12	25 mr/hr	10^3 mr/hr
Reactor Building 344 ft elevation	18RIA-051-13	20 mr/hr	10^3 mr/hr
New Fuel Vault	18RIA-051-14	20 mr/hr	10^3 mr/hr
Cleanup Precoat Area	18RIA-051-15	80 mr/hr	10^3 mr/hr
RWCU Heat Exchanger Room	18RIA-051-16	50 mr/hr	10^3 mr/hr
Fuel Pool Pump Room	18RIA-051-17	300 mr/hr	10^3 mr/hr
Contaminated Equipment Storage	18RIA-051-18	50 mr/hr	10^3 mr/hr
RWCU Pump Area	18RIA-051-19	30 mr/hr	10^3 mr/hr
Rx Bldg Sample Area	18RIA-051-20	30 mr/hr	10^3 mr/hr
RBCLC Heat Exchanger Area	18RIA-051-21	5 mr/hr	10^3 mr/hr
Reactor Building Access 272 ft elevation	18RIA-051-23	40 mr/hr	10^3 mr/hr
TIP Cubicle	18RIA-051-24	125 mr/hr	10^3 mr/hr
East HCU Area	18RIA-051-25	30 mr/hr	10^3 mr/hr
West HCU Area	18RIA-051-26	35 mr/hr	10^3 mr/hr
East Crescent	18RIA-051-27	110 mr/hr	10^3 mr/hr
CRD Removal Hatch	18RIA-051-28	25 mr/hr	10^3 mr/hr
West Crescent	18RIA-051-29	100 mr/hr	10^3 mr/hr
Refuel Floor West	18RIA-052-30	10^3 mr/hr	$2x10^4$ mr/hr

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 southwest 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: D

Explanation: EOP-5 entry is required due to Reactor Building ventilation exhaust radiation (>$1x10^3$ cpm), RCIC room area temperature (>104°F), West Crescent area radiation level (>100 mR/hr), and Reactor Building differential pressure (at or above 0" H_20). EOP-6 entry is required due to offsite release rate above the Emergency Plan "Alert" level.

A. Incorrect – Four EOP-5 entry conditions are met, not just three. EOP-6 entry is required due to offsite release rate above the Emergency Plan "Alert" level.
B. Incorrect – Four EOP-5 entry conditions are met, not just three.
C. Incorrect – EOP-6 entry is required due to offsite release rate above the Emergency Plan "Alert" level.

Technical Reference(s): EOP-5, EOP-6, ARP-09-3-2-40, ARP-09-3-3-2(12)

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11F 1.04, MIT-301.11G 6.05

Question Source: Modified Bank – 3/14 NRC #49

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 #	295004 2.2.37
	Importance Rating	3.6

Partial or Complete Loss of DC Power

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

Proposed Question: #54

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

Which one of the following describes the resulting availability of HPCI and RCIC for injection to the Reactor?

A. Both HPCI and RCIC are available.

B. HPCI is available, but RCIC is NOT available.

C. RCIC is available, but HPCI is NOT available

D. Both HPCI and RCIC are NOT available.

Proposed Answer: C

Explanation: Loss of 125 VDC Battery Bus B results in HPCI being unavailable for injection to the Reactor due to loss of power to multiple MOVs. Power is also lost to part of the RCIC logic, however RCIC is still available for injection to the Reactor.

A. Incorrect – HPCI is unavailable for injection to the Reactor due to loss of power to multiple MOVs. If the fault were on 125 VDC Battery Bus A, HPCI would be available.
B. Incorrect – HPCI is unavailable for injection to the Reactor due to loss of power to multiple MOVs. Despite loss of power to some logic, RCIC remains available for injection to the Reactor. If the fault were on 125 VDC Battery Bus A, HPCI would be available and RCIC would NOT be available.
D. Incorrect – Despite loss of power to some logic, RCIC remains available for injection to the Reactor.

Technical Reference(s): AOP-46

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.14.d

Question Source: Modified Bank – 2008 NRC #45

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 2.4.35
	Importance Rating	3.8

SCRAM

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

Proposed Question: #55

A Reactor scram has occurred with the following:

- The CRS has directed Reactor pressure controlled 800-1000 psig using Turbine Bypass Valves (TBVs).
- An NPO has been dispatched to reduce the number of in-service Steam Jet Air Ejectors (SJAEs) to minimum in accordance with AOP-1, Reactor Scram, and OP-24C, Condenser Air Removal.
- One (1) Second Stage SJAE is in service

Which one of the following describes the number of **FIRST Stage** SJAEs that will remain in-service once this task is complete and the reason for changing the lineup, in accordance with AOP-1 and OP-24C?

	Number of First Stage SJAEs in-service	Reason for the SJAE lineup change
A.	1	Adjust reactor cooldown rate
B.	1	Adjust Main Condenser vacuum
C.	2	Adjust reactor cooldown rate
D.	2	Adjust Main Condenser vacuum

Proposed Answer: A

Explanation: AOP-1 directs securing SJAEs per OP-24C to lower steam loads to reduce Reactor cooldown rate. The lowest combination of in-service SJAEs allowed in OP-24C is 1 first stage and 1 second stage.

B. Incorrect – AOP-1 directs securing SJAEs to control Reactor cooldown rate, not adjust Main Condenser vacuum.
C. Incorrect – Only one first stage SJAE is left in service. A 2/1 lineup is a valid SJAE lineup allowed by OP-24C, but not the minimum.
D. Incorrect – Only one first stage SJAE is left in service. A 2/1 lineup is a valid SJAE lineup allowed by OP-24C, but not the minimum. AOP-1 directs securing SJAEs to control Reactor cooldown rate, not adjust Main Condenser vacuum.

Technical Reference(s): AOP-1, OP-24C

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

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 #	295023 2.4.45
Importance Rating	4.1

Refueling Accidents

Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: #56

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

- You are on-watch in the Control Room and assigned to be in contact with the Refuel Bridge during fuel movements.
- The Refuel Bridge has just picked up an irradiated fuel bundle in the Spent Fuel Pool and moved over the core.
- The following indications are present in the Control Room:
  - o The Reactor Mode Switch is in REFUEL.
  - o All control rods are fully inserted.
  - o Annunciator 09-5-2-2, ROD WITHDRAWAL BLOCK, is clear.
  - o SRMs are fully inserted and indicate:

SRM	Counts per second	Period
A	100	∞
B	110	∞
C	90	∞
D	115	∞

Which one of the following describes the implication of these indications?

A. These indications are appropriate for continued fuel movement.

B. You must direct the Refuel Bridge to stop fuel movement due to improper operation of the REFUELING INTERLOCKS rod block. SRM indications are appropriate for continued fuel movement.

C. You must direct the Refuel Bridge to stop fuel movement due to improper SRM indication. REFUELING INTERLOCKS rod block operation is appropriate for continued fuel movement.

D. You must direct the Refuel Bridge to stop fuel movement due to improper operation of the REFUELING INTERLOCKS rod block and improper SRM indications.

Proposed Answer: B

Explanation: With the Refuel Bridge hoist loaded with an irradiated fuel bundle and located over the core, the REFUELING INTERLOCKS rod block should cause Annunciator 09-5-2-2, ROD WITHDRAWAL BLOCK, to be in alarm. Fuel movement must be stopped until this malfunction is fixed.

A. Incorrect – With the Refuel Bridge hoist loaded with an irradiated fuel bundle and located over the core, the REFUELING INTERLOCKS rod block should cause Annunciator 09-5-2-2, ROD WITHDRAWAL BLOCK, to be in alarm. Fuel movement must be stopped until this malfunction is fixed.
C. Incorrect – SRMs are indicating appropriately for fuel movements (> 3 cps and stable) and with the Refuel Bridge hoist loaded with an irradiated fuel bundle and located over the core, the REFUELING INTERLOCKS rod block should cause Annunciator 09-5-2-2, ROD WITHDRAWAL BLOCK, to be in alarm. Fuel movement must be stopped until this malfunction is fixed.
D. Incorrect – SRMs are indicating appropriately for fuel movements (> 3 cps and stable).

Technical Reference(s): ST-20F, RAP-7.1.04B, RAP-7.1.04C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03F 1.05.b.1.j

Question Source: Bank – 3/14 NRC #56

Question History: 3/14 NRC #56

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA2.03
	Importance Rating	3.9

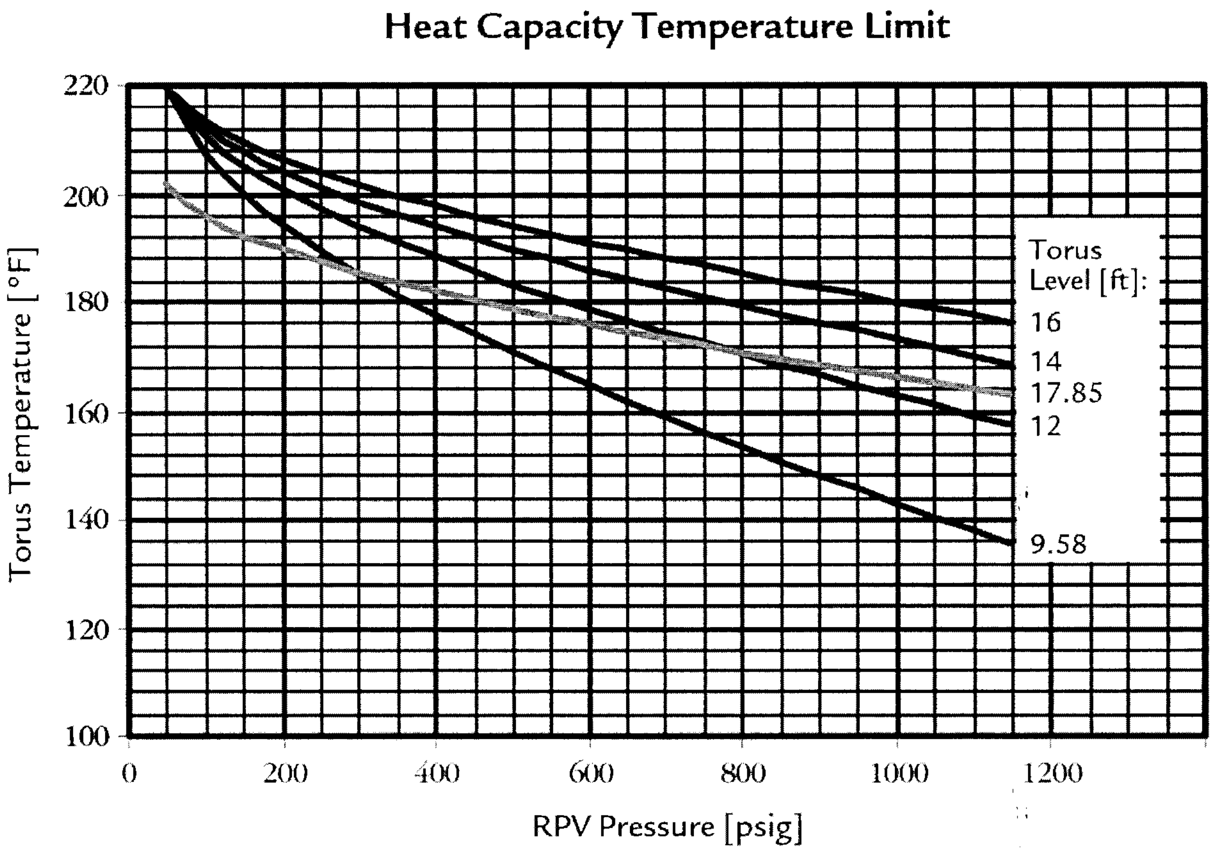
Suppression Pool High Water Temperature

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

Proposed Question: #57

The plant has experienced a transient with the following:

- Reactor pressure is 650 psig.
- Torus water level is 13'.
- Torus water temperature is 180°F.



Which one of the following identifies (1) the current status of the Heat Capacity Temperature Limit (HCTL) and (2) the required action if HCTL is exceeded, in accordance with EOP-4, Primary Containment Control?

	(1) HCTL is currently...	(2) Required Action if HCTL is Exceeded
A.	exceeded	Emergency RPV Depressurization
B.	exceeded	Venting the Primary Containment
C.	NOT exceeded	Emergency RPV Depressurization
D.	NOT exceeded	Venting the Primary Containment

Proposed Answer: A

Explanation: With Torus water level at 13', the next lowest Torus water level curve (12') must be used when evaluating HCTL. With Torus water temperature at 180°F and Reactor pressure at 650 psig, the 12' HCTL curve is being exceeded. EOP-4 requires performing an Emergency RPV Depressurization if HCTL is exceeded.

B. Incorrect – The required action for exceeding HCTL is an Emergency RPV Depressurization, not venting the Primary Containment. EOP-4 requires venting Primary Containment if PCPL is exceeded.
C. Incorrect – HCTL must be evaluated using the 12' curve, which is currently being exceeded. The 14' foot curve is not currently being exceeded, but is not the correct curve to use.
D. Incorrect – HCTL must be evaluated using the 12' curve, which is currently being exceeded. The 14' foot curve is not currently being exceeded, but is not the correct curve to use. The required action for exceeding HCTL is an Emergency RPV Depressurization, not venting the Primary Containment. EOP-4 requires venting Primary Containment if PCPL is exceeded.

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.07

Question Source: Modified Bank – NMP1 2010 NRC #39

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 #	700000 2.4.9
	Importance Rating	3.8

Generator Voltage and Electric Grid Disturbances

Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: #58

The plant is shutdown with the following:

- Shutdown Cooling (SDC) is in service with RHR Pump A running.
- Then, a grid disturbance causes 115KV Lines 3 and 4 to de-energize.
- Emergency Diesel Generators A and C fail to start.
- One minute later both Line 3 and 4 re-energize.
- Five minutes later Line 3 and 4 breakers 10012 and 10022 are still open.

Which one of the following describes the Abnormal Operating Procedure(s) (AOP), if any, to be executed and why?

A. **No** AOPs are required to be executed due to being in Mode 3.

B. **Only** AOP-30 (LOSS OF SHUTDOWN COOLING) is executed in order to re-establish SDC.

C. **Only** AOP-72 (115KV GRID LOSS, INSTABILITY OR DEGRADATION) is executed in order to re-energize the 4160 VAC busses.

D. **Both** AOP-30 and AOP-72 are executed in order to re-energize the 4160 VAC busses and re-establish SDC.

Proposed Answer: D

Explanation:

AOPs are applicable in **all** Modes of operation, whereas EOPs are NOT applicable in Mode 3.

When Lines 3 and 4 de-energize and EDGs A and C failing to start, the 10300 and 10500 buses de-energize. This results in a trip of RHR pump A, therefore SDC cooling is lost. AOP-30 will provide mitigating strategy guidance for re-establishing SDC. AOP-72 will provide mitigating strategy guidance to re-energize the 4160 VAC busses due to breakers 10012 and 10022 not automatically closing when the 115KV lines re-energize.

A. Incorrect – AOPs are still required to be entered in Mode 3. EOPs are not to be entered in Mode 3.
B. Incorrect – While AOP-30 will be entered to re-establish SDC; it is not the only AOP to be entered.
C. Incorrect – While AOP-72 will be entered to re-energize the 4160 VAC busses; it is not the only AOP to be entered.

Technical Reference(s): OP-13D, AOP-18, AOP-19, AOP-30, AOP-72

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10.a

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 #	1
	Group #	2
	K/A #	295034 EK1.01
	Importance Rating	3.8

Secondary Containment Ventilation High Radiation

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Personnel protection

Proposed Question: #59

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

- Fuel movement is in progress in the Spent Fuel Pool in preparation for an upcoming outage.
- A failure of the fuel grapple results in an irradiated fuel bundle being dropped.
- The following annunciators alarm in the Control Room:
  - 09-3-1-20, REFUEL AREA ARM RAD HI
  - 09-3-2-29, RX BLDG VENT RAD MON HI
- Operators perform all immediate **AND** subsequent actions of AOP-44, Dropped Fuel Assembly.

Which one of the following describes the resulting alignment of Reactor Building Ventilation and Control Room Ventilation, in accordance with AOP-44?

A. Both Reactor Building and Control Room Ventilation are isolated.

B. Reactor Building Ventilation is isolated, but Control Room Ventilation remains in a normal alignment.

C. Control Room Ventilation is isolated, but Reactor Building Ventilation remains in a normal alignment.

D. Both Reactor Building and Control Room Ventilation remain in a normal alignment.

Proposed Answer: A

Explanation: The given conditions present a dropped fuel assembly causing elevated area radiation levels and elevated Reactor Building ventilation radiation levels. Reactor Building ventilation radiation has not yet reached the threshold for automatic Reactor Building Ventilation isolation. However, AOP-44 requires isolation of Reactor Building Ventilation as part of the Immediate Actions. AOP-44 also requires isolation of Control Room Ventilation within 30 minutes as part of the Subsequent Actions. These actions both control the offsite release and provide protection for personnel in the Control Room.

B. Incorrect – Control Room Ventilation is also isolated as part of the Subsequent Actions of AOP-44.
C. Incorrect – Reactor Building Ventilation is isolated as part of the Immediate Actions of AOP-44.
D. Incorrect – Reactor Building Ventilation is isolated as part of the Immediate Actions of AOP-44. Control Room Ventilation is also isolated as part of the Subsequent Actions of AOP-44.

Technical Reference(s): AOP-44

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Modified Bank – 2010 NRC #18

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 #	295033 EK2.01
	Importance Rating	3.8

High Secondary Containment Area Radiation Levels

Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: Area radiation monitoring system

Proposed Question: #60

A Reactor scram has occurred with the following:

- A steam leak is in progress from the Scram Discharge Volume into the Reactor Building.
- All attempts to reset the scram and isolate the leak have been unsuccessful.
- Area radiation monitor 18RIA-051-25, East HCU Area, is indicating above the Maximum Safe limit.
- Area radiation monitor 18RIA-051-26, West HCU Area, is indicating above the Maximum Safe limit.
- All other area radiation monitors and area temperature instruments are indicating below their Maximum Safe limits and stable.

Which one of the following describes the required control of Reactor pressure, in accordance with EOP-2, RPV Control, and EOP-5, Secondary Containment Control?

A. Stabilize Reactor pressure below 1080 psig.

B. Depressurize the Reactor while maintaining the cooldown rate less than 100°F/hr.

C. Depressurize the Reactor with Turbine Bypass Valves irrespective of the cooldown rate.

D. Perform an Emergency RPV Depressurization.

Proposed Answer: D

Explanation: While the two given area radiation monitors are both on the same Reactor Building elevation and monitor the area around HCUs, they count as two separate areas for the purposes of evaluating the Emergency RPV Depressurization requirement in EOP-5. With an un-reset scram and a leak from the SDV, a primary system is discharging into the Reactor Building. This discharge is causing two separate areas to have area radiation levels above the Maximum Safe limit. Therefore, EOP-5 requires an Emergency RPV Depressurization.

A. Incorrect – EOP-5 requires an Emergency RPV Depressurization. This is the normal requirement immediately following a scram.
B. Incorrect – EOP-5 requires an Emergency RPV Depressurization. If only one area radiation level was above the Max Safe limit and no other area was approaching its Max Safe Limit, this would be the correct answer.
C. Incorrect – EOP-5 requires an Emergency RPV Depressurization. If only one area radiation level was above the Max Safe limit and another area was approaching its Max Safe Limit, this would be the correct answer.

Technical Reference(s): EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11F 1.07

Question Source: Bank – SSES LOC23 NRC #60

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 #	295002 AK3.06
	Importance Rating	2.9

Loss of Main Condenser Vacuum

Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM: Air ejector flow

Proposed Question: #61

The plant is operating at 100% power when an event results in the following symptoms:

- Main Condenser vacuum is 26.5" Hg and lowering slowly.
- Offgas flow is 150 scfm and lowering slowly.
- Main Stack radiation levels are lowering slowly.
- Annunciator 09-6-1-23, OFF GAS LINE FILTER DIFF PRESS HI, is in alarm.

Which one of the following identifies a possible reason for these symptoms?

A. Insufficient Low Conductivity Sump level

B. Loss of both Steam Packing Exhaust fans

C. Cracking in the Main Turbine exhaust boots

D. Combustion in the Steam Jet Air Ejector condenser

Proposed Answer: D

Explanation: AOP-5 identifies lowering Offgas flow rate, high Offgas filter differential pressure, lowering Stack radiation levels and lowering Main Condenser vacuum as symptoms of combustion in the Steam Jet Air Ejector after condenser.

A. Incorrect - A low level in the Low Conductivity Sump would allow air in-leakage into the Main Condenser via TB Equip Sump Vacuum Drag valve 20LCV-958 causing Main Condenser vacuum to lower and Offgas flow rate to rise.
B. Incorrect – Loss of both Steam Packing Exhaust fans could result in steam leakage into the Turbine Building, but not lowering Main Condenser vacuum and lowering Offgas flow.
C. Incorrect – Cracking in the Main Turbine exhaust boots could result in lowering Main Condenser vacuum, but would be accompanied by rising Offgas flow, not lowering Offgas flow.

Technical Reference(s): AOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.01

Question Source: Modified Bank – 2010 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	2
K/A #	295036 EA1.01
Importance Rating	3.2

Secondary Containment High Sump/Area Water Level

Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Secondary containment equipment and floor drain systems

Proposed Question: #62

Given the following conditions, which of the following combinations identifies the conditions that directly cause annunciator 09-4-1-39, RX BLDG EQUIP SUMP A LEAKAGE, to alarm?

1: Sump pump run time is too long
2: Time between pump downs is too short
3: Sump level exceeds the high level setpoint

A. 1 and 2, only

B. 1 and 3, only

C. 2 and 3, only

D. 1 and 2 and 3

Proposed Answer: A

Explanation: There are only two conditions that directly cause annunciator 09-4-1-39 to alarm. They are either the sump pump running for too long during a single pump down (>8 minutes) or too short of a timer period passing between successive pump downs (<55 minutes).

B. Incorrect – Sump level exceeding the high level setpoint causes annunciator 09-4-1-29, RX BLDG EQUIP SUMP A LVL HI, to alarm, but does not directly cause annunciator 09-4-1-39 to alarm. Too long of a sump pump runtime also directly causes annunciator 09-4-1-39 to alarm.
C. Incorrect – Sump level exceeding the high level setpoint causes annunciator 09-4-1-29, RX BLDG EQUIP SUMP A LVL HI, to alarm, but does not directly cause annunciator 09-4-1-39 to alarm. Too short of a time between pump downs also directly causes annunciator 09-4-1-39 to alarm.
D. Incorrect – Sump level exceeding the high level setpoint causes annunciator 09-4-1-29, RX BLDG EQUIP SUMP A LVL HI, to alarm, but does not directly cause annunciator 09-4-1-39 to alarm.

Technical Reference(s): ARPs 09-4-1-39 and 09-4-1-29

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-20

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 #	2
	K/A #	295015 AA2.01
	Importance Rating	4.1

Incomplete SCRAM

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

Proposed Question: #63

The plant was operating at 100% power when a spurious Group 1 Primary Containment Isolation signal resulted in the following:

- The Reactor Mode Switch is in SHUTDOWN.
- All control rods are NOT full in.
- APRMs are unavailable.
- Four SRVs are open.
- Reactor pressure is 1140 psig and steady.
- Reactor water level is 180 inches and lowering.
- RCIC and HPCI are NOT running.

Which one of the following ranges contains the approximate value of Reactor power?

A. 5-10%

B. 15-20%

C. 30-35%

D. 55-60%

Proposed Answer: C

Explanation: A Group 1 isolation signal causes the MSIVs to close. When the MSIVs are not full open, a Reactor Scram is generated. Each SRV will pass approximately 900,000 lbm/hr at 1145 psig. Therefore 4 SRVs = 3.6×10^6 lbm/hr. Total Steam Flow at 100% power is approximately 10.97×10^6 lbm/hr; therefore each SRV can pass approximately 8.2% Rx power. Four SRVs multiplied by 8.2% is ~32%. With HPCI and RCIC not running, there are no other steam loads since the MSIVs are shut.

A. Incorrect – Four SRVs open with Reactor pressure 1140 psig and stable indicates Reactor power is approximately 32%. 5-10% would be correct for a single SRV open.
B. Incorrect – Four SRVs open with Reactor pressure 1140 psig and stable indicates Reactor power is approximately 32%. 15-20% is plausible if SRV capacity were underestimated.
D. Incorrect – Four SRVs open with Reactor pressure 1140 psig and stable indicates Reactor power is approximately 32%. 55-60% power would be correct if four SRVs were open AND Turbine Bypass Valves were available and open (32% + 25% = 57%).

Technical Reference(s): FSAR Table 4.4-1 and Chapter 16.9.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05, SDLP-29 1.02 and 1.05

Question Source: Bank – 9/12 NRC #5

Question History: 9/12 NRC #5

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 #	295017 2.4.6
	Importance Rating	3.7

High Off-site Release Rate

Knowledge of EOP mitigation strategies.

Proposed Question: #64

A loss of coolant accident has resulted in the following:

- Hydrogen concentrations in the Primary Containment require venting.
- Torus water level is 17 feet and stable.

Which one of the following describes the path required to be used to vent the Containment and the reason why, in accordance with EOP-4a, Primary Containment Gas Control, and EP-6, Post Accident Containment Venting and Gas Control?

Vent from the...

A. Torus to minimize cycling of the Torus-to-Drywell vacuum breakers.

B. Drywell because Torus water level is too high for venting from the Torus.

C. Torus to better scrub fission products from the Containment atmosphere before release.

D. Drywell to more quickly reduce the risk of hydrogen ignition by electrical equipment operation.

Proposed Answer: C

Explanation: EOP-4a requires venting per EP-6. EP-6 requires venting from the Torus as long as Torus water level is below 29.5 feet. This is to scrub the Containment atmosphere through the Torus water volume prior to release to lower the radioactive release.

A. Incorrect – Torus venting is required, but the reason is to scrub the atmosphere. Lowering Torus pressure first does also prevent the need for vacuum breakers to cycle.
B. Incorrect – If Primary Containment water level were ≥29.5', then Drywell venting would be required. Since Torus water level is only 17', Torus venting is still allowed and preferred.
D. Incorrect – There is greater risk of hydrogen ignition in the Drywell due to presence of electrical equipment, however venting is required to be from the Torus to lower the radioactive release.

Technical Reference(s): EOP-4a, EP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Bank – NMP1 2015 Audit #61

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 #	295022 AA1.04
	Importance Rating	2.5

Loss of CRD Pumps

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

Proposed Question: #65

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

Time (minutes)	Event
0	CRD pump A trips.
5	CRD pump B is manually started.

Which one of the following describes the status of seal purge flow to the operating Reactor Water Cleanup pump during this transient?

Seal purge flow to the operating RWCU pump...

A. is lost at Time = 0 minutes and is automatically restored at Time = 5 minutes.

B. is lost at Time = 0 minutes and remains lost after Time = 5 minutes until it is manually aligned to CRD pump B.

C. is supplied from an alternate source at Time = 0 minutes and then automatically shifts back to CRD at Time = 5 minutes.

D. is supplied from an alternate source at Time = 0 minutes and remains aligned to this alternate source after Time = 5 minutes.

Proposed Answer: A

Explanation: CRD normally supplies seal purge flow to the RWCU system. When CRD pump A trips at Time = 0 minutes, all CRD flow is lost, including seal purge to RWCU. There is no automatic alternate seal purge supply to RWCU. When CRD pump B is manually started at Time = 5 minutes, it supplies seal purge flow to RWCU through the same lineup as existed with CRD pump A in service.

B. Incorrect – When CRD pump B is manually started at Time = 5 minutes, it supplies seal purge flow to RWCU through the same lineup as existed with CRD pump A in service. There is no need to manually realign seal purge flow to CRD pump B.
C. Incorrect – There is no automatic alternate seal purge supply to RWCU. Alternate seal purge to RWCU is only available manually through use of a temporary mod.
D. Incorrect – There is no automatic alternate seal purge supply to RWCU. Alternate seal purge to RWCU is only available manually through use of a temporary mod.

Technical Reference(s): AOP-69, FM-27A, FM-26A, FM-24A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12 1.10.h

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

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 the At-The-Controls (ATC) Operator.
- You are being relieved for lunch by another qualified Reactor Operator that is assigned to the FIN team.

Which one of the following describes the requirements that must be met for this watch station relief per EN-OP-115-03, Shift Turnover and Relief?

(1) Verbal turnover to on-coming Reactor Operator.
(2) Update brief informing the shift crew of the relief change.
(3) Permission from the Shift Manager (SM) or Control Room Supervisor (CRS).

A. (1) only.

B. (1) and (2) only.

C. (1) and (3) only.

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

Proposed Answer: D

Explanation: EN-OP-115-03 requires the following for a control room operator to be relieved during their shift:

- Permission granted by the SM or CRS as applicable.
- A verbal turnover conducted to a qualified individual as follows…
- An update brief performed informing the shift crew of the relief change.

A. Incorrect – An update brief and SM/CRS permission are also required.
B. Incorrect – SM/CRS permission is also required.
C. Incorrect – An update brief is also required.

Technical Reference(s): EN-OP-115-03

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: Bank – 9/12 NRC #67

Question History: 9/12 NRC #67

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 #	2.1.4
	Importance Rating	3.3

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: #67

A licensed Reactor Operator works the following schedule:

January 1, 2016	Off
January 2, 2016	Off
January 3, 2016	12-hour day shift as SNO
January 4, 2016	12-hour day shift as SNO
January 5, 2016	12-hour day shift as SNO
January 6, 2016	Off
January 7, 2016	Off
January 8, 2016	12-hour night shift as SNO
January 9, 2016	12-hour night shift as SNO

Then, the Reactor Operator is transferred to a non-watchstanding position to support a station project.

The Reactor Operator maintains requalification training current and remains medically qualified, but does NOT perform any watchstanding for a prolonged period of time.

Which one of the following identifies when the Reactor Operator's license will **first** go inactive if NO additional watchstanding is performed, in accordance with ODSO-30, Maintenance of NRC Licenses and STA Qualifications?

A. February 1, 2016

B. March 1, 2016

C. April 1, 2016

D. July 1, 2016

Proposed Answer: D

Explanation: The Reactor Operator has met the watchstanding requirements for the first quarter of 2016 (January 1 – March 31) by reaching the minimum of 5 12-hour shifts. Therefore, the Reactor Operator will remain active for the second quarter of 2016 (April 1 – June 30). The Reactor Operator will first go inactive on the first day of the third quarter of 2016 (July 1, 2016).

A. Incorrect – July 1, 2016 is the first day the Reactor Operator will go inactive based on watchstanding requirements. February 1, 2016 would be correct if the requirement was monthly, instead of quarterly, and the Reactor Operator had stood less than 5 12-hour watches in January 2016.
B. Incorrect – July 1, 2016 is the first day the Reactor Operator will go inactive based on watchstanding requirements. March 1, 2016 would be correct if the requirement was monthly, instead of quarterly.
C. Incorrect – July 1, 2016 is the first day the Reactor Operator will go inactive based on watchstanding requirements. April 1, 2016 would be correct if the Reactor Operator had stood less than 5 12-hour watches in the first quarter of 2016.

Technical Reference(s): ODSO-30

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 67.02

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 #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms.

Proposed Question: #68

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

- I&C is performing troubleshooting on an SDIV level instrument.
- Annunciator 09-5-1-44, SDIV A OR B NOT DRAINED, will be received intermittently for the next ten (10) minutes.
- The troubleshooting activity will NOT extend into the next shift.

Which one of the following describes requirements for flagging and logging for this annunciator per EN-OP-115-08, Annunciator Response?

	Flagging of the annunciator...	Logging of the annunciator in either the annunciator log or turnover sheet...
A.	may NOT be waived.	is required.
B.	may NOT be waived.	is NOT required.
C.	may be waived.	is required.
D.	may be waived.	is NOT required.

Proposed Answer: D

Explanation: EN-OP-115-08 Section 5.2[7] describes requirements for flagging of annunciators. For short duration alarms (less than a shift and is associated with a planned evolution), the CRS/SM may waive flagging requirements. Otherwise, for activities that cause an expected alarm and the annunciator is associated with equipment required in current mode, the following is required:

- Install an annunciator flag on the expected alarm.
- If projected activity duration will exceed remaining time of current shift, then update annunciator log or SRO/RO turnover sheet as appropriate with alarm status.

Since the activity is projected for less than a shift and does not extend into the next shift, flagging may be waived and logging is not required.

A. Incorrect – Since the activity is planned and is projected for less than a shift, waiver of flagging is allowable. Since the activity does not extend into the next shift, logging is not required.
B. Incorrect – Since the activity is planned and is projected for less than a shift, waiver of flagging is allowable.
C. Incorrect – Since the activity does not extend into the next shift, logging is not required.

Technical Reference(s): EN-OP-115-08

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: Modified Bank – 9/14 NRC #69

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.41
	Importance Rating	3.5

Ability to obtain and interpret station electrical and mechanical drawings.

Proposed Question: #69

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

- SRV A opens due to a spurious signal from the high pressure logic in the SRV Electric Lift System (SRVELS).

Note: A portion of electrical drawing 1.83-38 is provided on the following page. Red marks on the drawing identify the cause of the spurious signal. Additionally, a portion of electrical drawing ESK-11AAQ is provided.

Given the following possible methods to close SRV A:

(1) Placing SRV NORMAL-BYPASS switch for SRV A in BYPASS at Panel 09ECCS1-EP.
(2) Placing isolation switch (IS) for SRV A in LOCAL at Panel 25ASP-5.

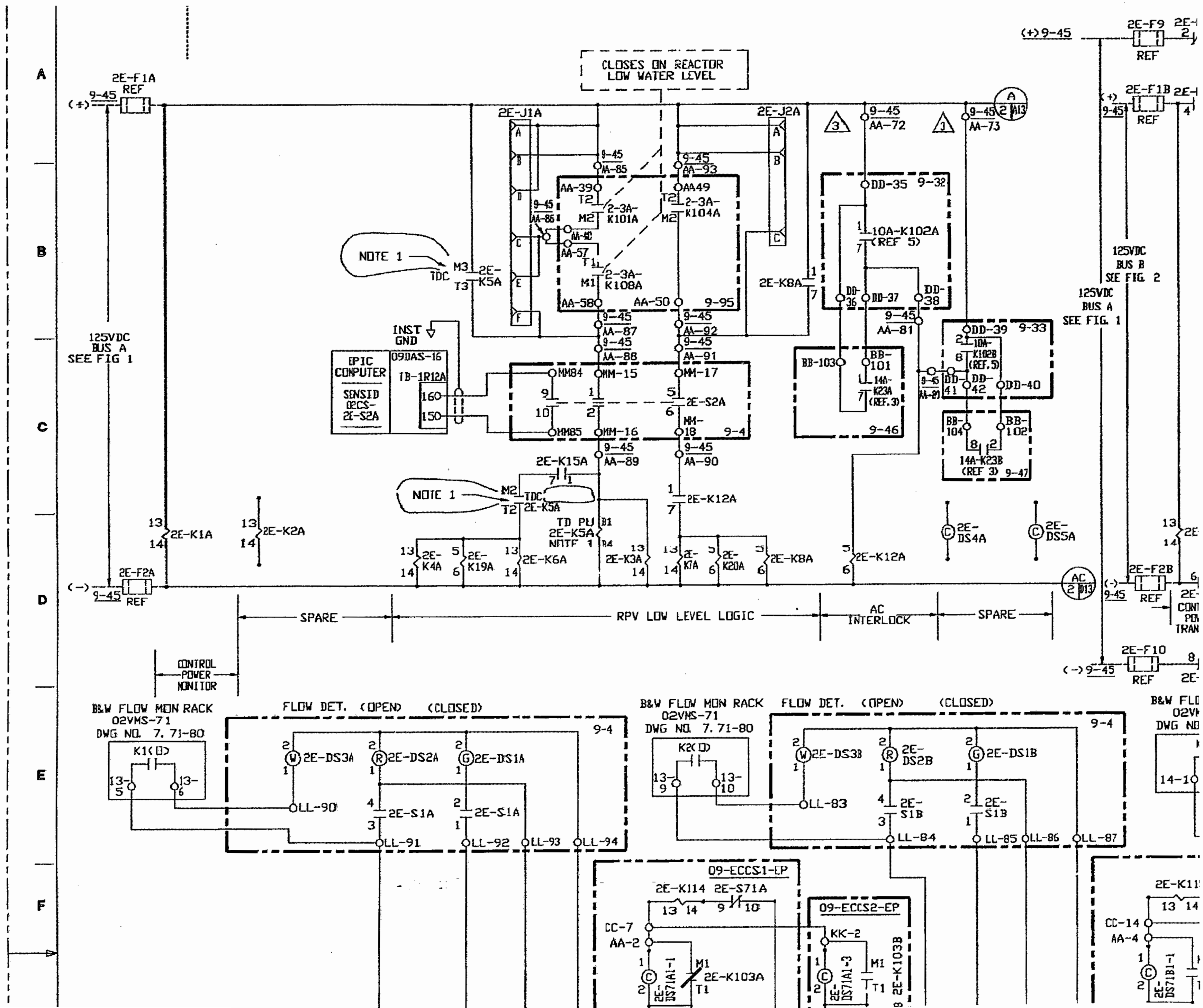
Which one of the following identifies which of these methods, if any, will close SRV A?

A. NEITHER method (1) NOR method (2) will close SRV A.

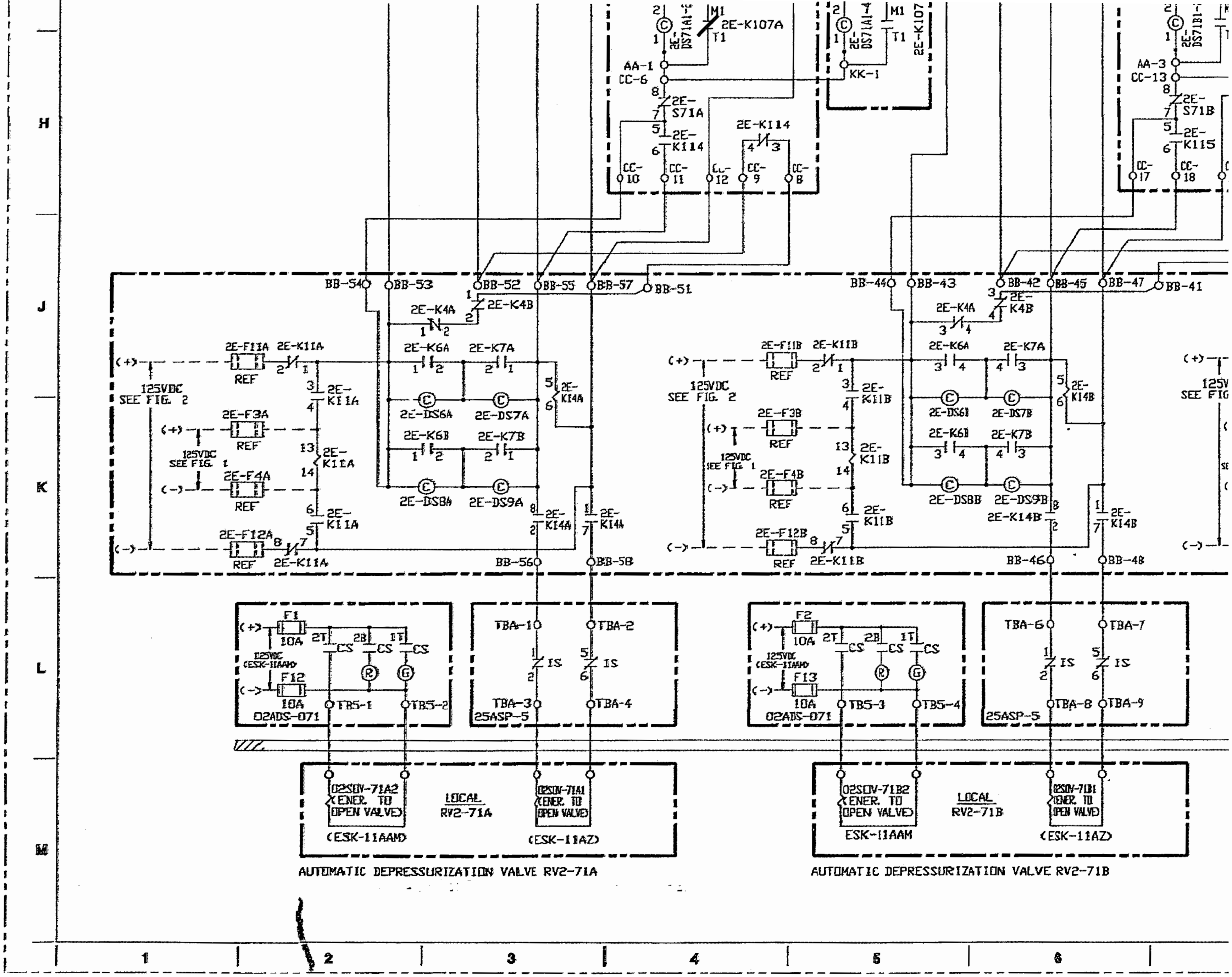
B. Method (1) will close SRV A, but method (2) will NOT.

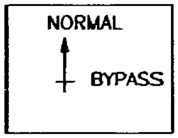
C. Method (2) will close SRV A, but method (1) will NOT.

D. Either method (1) or method (2) will close SRV A.



CLOSES ON REACTOR LOW WATER LEVEL
2E-F1A REF
9-45
125VDC BUS A SEE FIG 1
2E-J1A
2E-J2A
2-3A-K101A
2-3A-K104A
2-3A-K108A
NOTE 1
2E-K5A
2E-K8A
10A-K102A (REF 5)
INST GND
EPIC COMPUTER
SENSID 02CS-2E-S2A
09DAS-16
TB-1R12A
2E-S2A
2E-K15A
2E-K12A
TD PU 2E-K5A NOTE 1
2E-K1A
2E-K2A
2E-K4A
2E-K19A
2E-K6A
2E-K3A
2E-K7A
2E-K20A
2E-K8A
2E-K12A
14A-K23A (REF. 3)
14A-K23B (REF 3)
10A-K102B (REF. 5)
2E-DS4A
2E-DS5A
2E-F2A REF
2E-F9 REF
2E-F1B REF
2E-F2B REF
2E-F10 REF
125VDC BUS B SEE FIG 2
125VDC BUS A SEE FIG 1
SPARE
RPV LOW LEVEL LOGIC
AC INTERLOCK
SPARE
CONTROL POWER MONITOR
B&W FLOW MON RACK 02VMS-71 DWG NO. 7.71-80
FLOW DET. (OPEN) (CLOSED)
K1(D)
K2(D)
2E-DS3A
2E-DS2A
2E-DS1A
2E-S1A
2E-DS3B
2E-DS2B
2E-DS1B
2E-S1B
09-ECCS1-EP
2E-K114
2E-S71A
2E-K103A
09-ECCS2-EP
2E-K103B
2E-DS71A1-1
2E-DS71A1-3
2E-DS71B1-1


H
J
K
L
M
1
2
3
4
5
6
2E-K107A
AA-1
CC-6
2E-S71A
2E-K114
KK-1
2E-K107
AA-3
CC-13
2E-S71B
2E-K115
BB-54
BB-53
BB-52
BB-55
BB-57
BB-51
BB-44
BB-43
BB-42
BB-45
BB-47
BB-41
2E-K4A
2E-K4B
2E-F11A
2E-K11A
2E-K6A
2E-K7A
125VDC
SEE FIG. 2
REF
2E-DS6A
2E-DS7A
2E-K14A
2E-F3A
2E-K6B
2E-K7B
125VDC
SEE FIG. 1
2E-F4A
2E-DS8A
2E-DS9A
2E-F12A
BB-56
BB-58
2E-F11B
2E-K11B
2E-F3B
2E-F4B
2E-F12B
2E-DS6B
2E-DS7B
2E-DS8B
2E-DS9B
2E-K14B
BB-46
BB-48
F1
10A
CS
F12
02ADS-071
TB5-1
TB5-2
TBA-1
TBA-2
IS
TBA-3
TBA-4
25ASP-5
F2
F13
TB5-3
TB5-4
TBA-6
TBA-7
TBA-8
TBA-9
02SOV-71A2
(ENER. TO OPEN VALVE)
(ESK-11AAM)
LOCAL
RV2-71A
02SOV-71A1
(ESK-11AZ)
02SOV-71B2
ESK-11AAM
RV2-71B
02SOV-71B1
AUTOMATIC DEPRESSURIZATION VALVE RV2-71A
AUTOMATIC DEPRESSURIZATION VALVE RV2-71B



(F.V.)

SWITCH CONTACT DEVELOPMENT FOR
SRV ELECTRIC LIFT KEY OPERATED SELECTOR SWITCH

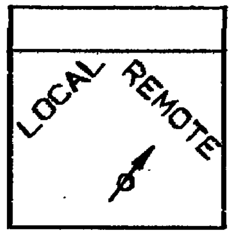
COMPONENT ID	CONTACTS	NORMAL	BYPASS	DWG. NO.
02E-S71A, 02E-S71B 02E-S71C, 02E-S71D	1 2 o—||—o		X	ESK-11AAS
	3 4 o—||—o	X		ESK-11AAS
	5 6 o—||—o		X	ESK-11AAS
	7 8 o—||—o	X		1.83-38
	9 10 o—||—o	X		1.83-38

SWITCH CONTACT DEVELOPMENT FOR
SRV ELECTRIC LIFT KEY OPERATED SELECTOR SWITCH

COMPONENT ID	CONTACTS	NORMAL	BYPASS	DWG. NO.
02E-S71E, 02E-S71F 02E-S71G, 02E-S71H 02E-S71J, 02E-S71K 02E-S71L	1 2 o—||—o		X	ESK-11AAS
	3 4 o—||—o	X		ESK-11AAS
	5 6 o—||—o		X	ESK-11AAS
	7 8 o—||—o	X		1.83-39
	9 10 o—||—o	X		1.83-39

DECK	CONTACTS HANDLE END	POSITION REMOTE	POSITION LOCAL
1	1o—||—||—o2	X	
	3o—||—||—o4		X
2	5o—||—||—o6	X	
	7o—||—||—o8		X
3	9o—||—||—o10	X	
	11o—||—||—o12		X
4	13o—||—||—o14	X	
	15o—||—||—o16		X

MAINTAINED CONTACTS-ELECTROSWITCH
TYPE 20K



PISTOL GRIP
HANDLE
"CG2"

DETAIL "CM"

Proposed Answer: D

Explanation: Since SRV A is open due to a spurious signal from the high pressure logic in the SRV Electric Lift System (SRVELS), the logic located near location F-4 on drawing 1.83-38 is at fault. The signal from this portion of the SRV logic can be isolated from the SRV opening solenoid (02SOV-71A1, located at drawing location M-3) by placing SRV NORMAL-BYPASS switch for SRV A in BYPASS at Panel 09ECCS1-EP (two contacts 2E-S71A, located near drawing location F-4). The signal may also be isolated from the SRV opening solenoid by placing isolation switch for SRV A in LOCAL at Panel 25ASP-5 (two contacts IS, located at drawing location L-3).

A. Incorrect – These methods will not solve all possible causes of SRV A spuriously opening, but they will both close SRV A given a spurious high pressure signal from the SRV Electric Lift System (SRVELS).
B. Incorrect – Method (2) will also close SRV A given a spurious high pressure signal from the SRV Electric Lift System (SRVELS).
C. Incorrect – Method (1) will also close SRV A given a spurious high pressure signal from the SRV Electric Lift System (SRVELS).

Technical Reference(s): AOP-36, 1.83-38

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05.c

Question Source: Modified Bank – SSES 2014 NRC #69

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.15
	Importance Rating	2.9

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

Proposed Question: #70

A plant startup is in progress with the following:

- A Condenser Air Removal Pump is in service to establish Main Condenser vacuum.
- Reactor pressure is 150 psig and slowly rising.
- One Turbine Bypass Valve is partially open.

Then, Annunciator 09-3-3-1, MAIN STM RAD MON HI, is received and followed shortly by Annunciator 09-5-1-32, MAIN STM LINE RADIATION HI-HI.

Main Steam Line radiation monitor indications are as shown **(see next page).**

Which one of the following describes the plant response to these indications?

A. No automatic plant response has occurred yet.

B. The Condenser Air Removal Pump has tripped.
The RWR loop sample valves remain open.

C. The Condenser Air Removal Pump remains in service.
The RWR loop sample valves have closed.

D. The Condenser Air Removal Pump has tripped.
The RWR loop sample valves have closed.

MAIN STM RAD MON A

MAIN STM RAD MON C

MAIN STM RAD MON B

MAIN STM RAD MON D

Proposed Answer: A

Explanation: Four radiation monitors (A, B, C, D) are provided on the Main Steam Lines. A and C input to trip system A. B and D input to trip system B. The trip logic requires at least one of the trip system A inputs and at least one of the trip system B inputs to be above the Hi-Hi setpoint (2871 mr/hr nominal) to cause the associated MSL rad monitor trips. These trips include:

- Recirc Loop Sample Valves 02-2AOV-39 and 02-2AOV-40 close
- Main Steam Line Drain Valve 29MOV-74 and 29MOV-77 close
- Condenser Air Removal Pumps 38P-2A and 38P-2B trip
- Condenser Air Removal Pump Suction and Discharge Isolation Valves 38AOV-111 and 38AOV-112 close

The given MSL rad monitor indications show 3 monitors (A, C, and D) above the Hi setpoint (1809 mr/hr nominal) but below the Hi-Hi setpoint (2871 mr/hr nominal), and 1 monitor (B) above the Hi-Hi setpoint. With only one monitor above the Hi-Hi setpoint, the trip logic is NOT satisfied, therefore the Condenser Air Removal Pump has NOT tripped. The RWR sample valves do not close on one high MSL rad level either.

B. Incorrect – With only one MSL rad monitor above the Hi-Hi setpoint, no automatic plant response has occurred yet.
C. Incorrect – With only one MSL rad monitor above the Hi-Hi setpoint, no automatic plant response has occurred yet.
D. Incorrect – With only one MSL rad monitor above the Hi-Hi setpoint, no automatic plant response has occurred yet.

Technical Reference(s): ARP 09-3-3-1, ARP-09-5-1-32, OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.05.a.3, 1.05.c.5

Question Source: Bank – 9/12 NRC #63

Question History: 9/12 NRC #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

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

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

Proposed Question: #71

The plant is operating at power with the following conditions:

- Hydrogen injection is in service per OP-89A (Hydrogen Addition System).
- Noble Metals injection is scheduled to commence next shift.

Which one of the following describes the expected plant response once Noble Metals are injected into the reactor?

A. Offgas radiation levels will lower.

B. Main Stack radiation levels will lower.

C. Main Steam Line radiation levels will rise.

D. Reactor Building ventilation radiation levels will rise.

Proposed Answer: C

Explanation: Per OP-65 G.5.3 Note 3: Introduction of noble metals with hydrogen injection increases the carryover of N-16 resulting in an increase Main Steam Line Radiation Monitor reading.

A. Incorrect – Offgas radiation levels would rise.
B. Incorrect – Main Stack radiation levels would rise.
D. Incorrect – RB ventilation radiation levels would be unaffected.

Technical Reference(s): OP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-89A, 1.09.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

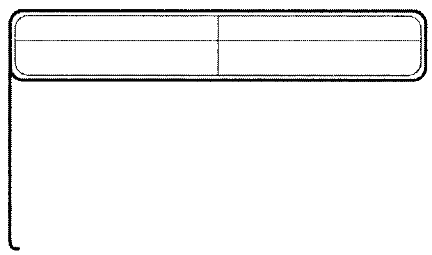
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.19
	Importance Rating	3.4

Knowledge of EOP layout, symbols, and icons.

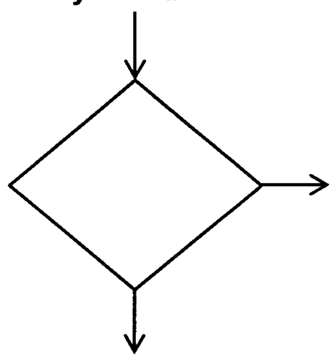
Proposed Question: #72

Which one of the following is indicated by the below two symbols in the Emergency Operating Procedures?

Symbol #1



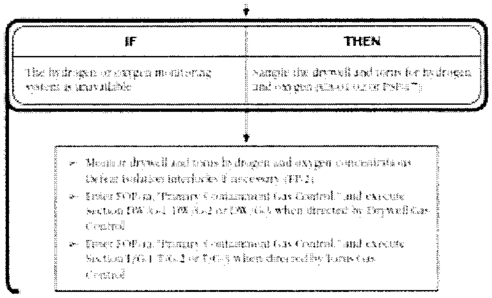
Symbol #2



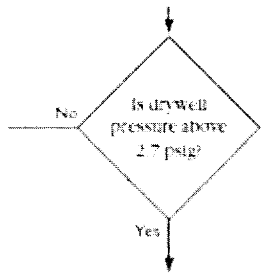
	Symbol #1	Symbol #2
A.	Before Step	Decision Point
B.	Before Step	Hold Point
C.	Override Step	Decision Point
D.	Override Step	Hold Point

Proposed Answer: C

Explanation: The given symbol #1 indicates an override step. This type of step remains in effect while subsequent steps are performed. Here is an example from EOP-4:



The given symbol #2 indicates a decision point. This type of step requires analysis of conditions prior to determining which leg is required to be executed. Here is an example from EOP-2:



A. Incorrect – Before steps are shown as arrow-shaped pentagons.
B. Incorrect – Before steps are shown as arrow-shaped pentagons. Hold points are shown as arrow-shaped pentagons.
D. Incorrect – Hold points are shown as arrow-shaped pentagons.

Technical Reference(s): EOP-4, 2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11A, 1.03

Question Source: Modified Bank – NMP2 2014 NRC #73

Question History: NMP2 2014 NRC #73

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 #	2.4.1
	Importance Rating	4.6

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: #73

Given the following parameters:

	Parameter	Value
1)	TB temperature	242°F
2)	RPV pressure	1070 psig
3)	DW water level	22 inches
4)	Torus water temperature	109°F
5)	DW pressure	14.7 psig
6)	DW H_2	5%
7)	RB ventilation radiation	500 cpm
8)	Control Rods	One rod at position 48, all others at 02

Which of the following lists the Emergency Operating Procedure(s) that are required to be entered?

A. EOP-2 and EOP-4 only

B. EOP-3 and EOP-4 only

C. EOP-5 and EOP-6 only

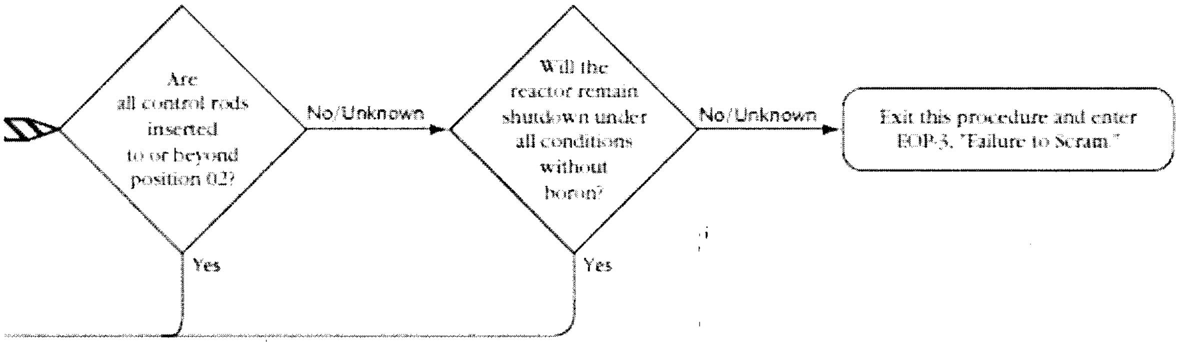
D. EOP-3 and EOP-7 only

Proposed Answer: A

EOP-2 RPV CONTROL

ENTRY CONDITIONS			
RPV pressure above 1080 psig	**RPV water level** below 177 in.	**Reactor power** above 2.5% or unknown when a scram is required	**Drywell pressure** above 2.7 psig

From EOP-2:



EOP-3 FAILURE TO SCRAM

START

EOP-4 PRIMARY CONTAINMENT CONTROL

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

EOP-5
SECONDARY CONTAINMENT CONTROL

ENTRY CONDITIONS					
Reactor Building Area Temperature above the Maximum Normal value in any area	**Reactor building floor drain sump level** above the 'high' setpoint	**Crescent area water level** above 0 in	**Reactor Building Area Radiation Level** above the Maximum Normal value in any area	**Reactor building ventilation exhaust radiation** above 10^3 cpm	**Reactor building differential pressure** at or above 0 in. of water

EOP-6
RADIOACTIVITY RELEASE CONTROL

ENTRY CONDITIONS
Offsite radioactivity release rate above the Emergency Plan "Alert" level

From EOP-2:

RPV water level cannot be determined	Exit this procedure and enter EOP-7, "RPV Flooding" at "Shutdown Flooding."

EOP-7 RPV FLOODING

The given parameters and values that meet or exceed EOP entry conditions are #4, #5 and #6. (Torus water temperature >95F, DW pressure >2.7 psig and DW H_2 >3%).

B. Incorrect – EOP-3 entry condition is not met via EOP-2 due to the reactor will remain shutdown under all conditions without boron with one rod at position 48 with all others at 02 (definition of shutdown margin).

C. Incorrect – EOP-5 entry condition is not met with a TB temperature of 242F. This is a value in EOP-5 for a RB temperature. EOP-6 entry is not met with a RB ventilation rad level of 500 cpm (this would be correct if Offsite rad levels were above the Alert level of EALs.

D. Incorrect - EOP-3 entry condition is not met with one rod at 48 and all others at 02. EOP-7 entry is not warranted with conditions provided however it could be mis-interpreted with a DW water level of 22 inches.

Technical Reference(s): EOP-2, EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.02, MIT-301.11e 4.02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

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

Proposed Question: #74

The plant is operating at 80% power with the following:

- HPCI surveillance testing is in progress in accordance with ST-4N, HPCI Quick-Start, In-service, and Transient Monitoring Test.
- Torus water temperature is 112°F and rising slowly.

Which one of the following describes the required action, if any, in accordance with Technical Specifications?

The surveillance test...

A. may continue. Continued Reactor operation is acceptable at the current power level.

B. must be immediately suspended. The Reactor mode switch must be placed in SHUTDOWN immediately.

C. must be immediately suspended. Continued Reactor operation is acceptable at the current power level as long as Torus water temperature does NOT exceed 120°F.

D. must be immediately suspended. Torus water temperature must be lowered below 110°F within one hour, or then the Reactor mode switch must be placed in SHUTDOWN.

Proposed Answer: B

Explanation: TS 3.6.2.1 allows Torus water temperature to go as high as 105°F during ST-4N (up from the normal limit of 95°F). Once Torus water temperature exceeds 105°F, TS 3.6.2.1 Condition C requires ST-4N to be immediately suspended. Once Torus water temperature exceeds 110°F, TS 3.6.2.1 Condition D requires immediately placing the Reactor mode switch in SHUTDOWN.

A. Incorrect – During performance of ST-4N, TS 3.6.2.1 does allow higher Torus water temperature than normal. Additionally, current Torus water temperature is below the highest limit of 120°F in TS 3.6.2.1. However, with Torus water temperature >110°F, TS 3.6.2.1 requires immediately suspending the surveillance test and scramming the Reactor.
C. Incorrect – During performance of ST-4N, TS 3.6.2.1 does allow higher Torus water temperature than normal, and 120°F is one of the limits on Torus water temperature. However, with Torus water temperature >110°F, TS 3.6.2.1 requires immediately scramming the Reactor.
D. Incorrect – During performance of ST-4N, TS 3.6.2.1 does allow higher Torus water temperature than normal, but does not allow continued Reactor operation for any period of time with Torus water temperature >110°F.

Technical Reference(s): 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: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.32
	Importance Rating	3.6

Knowledge of operator response to loss of all annunciators.

Proposed Question: #75

A plant shutdown is in progress for a refueling outage with the following:

- Reactor power is 80% and being lowered with Recirculation flow.
- Calibration of APRMs per ST-5D is about to begin.

Then, a loss of all Main Control Room annunciators occurs.

Which one of the following describes the required control of plant evolutions, in accordance with AOP-65, Loss of Control Room Annunciators?

A. The power reduction and APRM calibration may continue.

B. The power reduction and APRM calibration must be suspended until annunciators are restored.

C. The power reduction must be suspended until annunciators are restored, but the APRM calibration may continue.

D. The power reduction and APRM calibration must be suspended, but may continue once additional personnel are stationed to monitor parameters.

Proposed Answer: B

Explanation: AOP-65 requires suspending both plant transients (such as power reduction) and surveillance testing (such as ST-5D) until annunciators are restored.

A. Incorrect – AOP-65 requires suspending both plant transients (such as power ascension) and surveillance testing (such as ST-5D).
C. Incorrect – AOP-65 requires suspending both plant transients (such as power ascension) and surveillance testing (such as ST-5D).
D. Incorrect – AOP-65 requires suspending both plant transients (such as power ascension) and surveillance testing (such as ST-5D). Additional monitoring is also required, but no provision is provided to continue with transients and surveillance testing once the additional monitoring is in place.

Technical Reference(s): AOP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Bank – NMP1 2008 NRC #17

Question History: NMP1 2008 NRC #17

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 AA2.09
	Importance Rating	4.3

Generator Voltage and Electric Grid Disturbances

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operational status of emergency diesel generators

Proposed Question: #76

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

- There is strong thunderstorm activity in the area.
- Lines 3 and 4 have de-energized.
- AOP-72, 115 KV Grid Loss, Instability, or Degradation, has been entered.
- Plant shutdown is required by Technical Specifications.

Which one of the following describes the required actions, in accordance with AOP-72?

A. Insert a manual reactor scram per AOP-1 (Reactor Scram) from the current reactor power level **THEN** ensure the Emergency Diesels automatically supply power to the 10500 and 10600 busses.

B. Manually start and supply the 10500 and 10600 busses from the Emergency Diesel Generators per OP-22 (Diesel Generator Emergency Power), **THEN** insert a manual reactor scram per AOP-1 (Reactor Scram) from the current reactor power level.

C. Lower reactor power to 50-70 MWe per OP-65 (Startup and Shutdown Procedure), **THEN** insert a manual reactor scram per AOP-1 (Reactor Scram) and ensure the Emergency Diesels automatically supply power to the 10500 and 10600 busses.

D. Lower reactor power to 50-70 MWe per OP-65 (Startup and Shutdown Procedure), **THEN** manually start and supply the 10500 and 10600 busses from the Emergency Diesel Generators per OP-22 (Diesel Generator Emergency Power).

Proposed Answer: D

Explanation: The subsequent actions of AOP-72 coordinate control of various plant activities if a plant shutdown is required with Lines 3 and 4 unavailable. Reactor power is required to be lower to 50-70 MWe per OP-65. Operation of the EDGs is delayed until after lowering power to 50-70 MWe and just prior to the scram to minimize low-load operation of the EDGs. EDGs are manually started per OP-22, not allowed to auto-start after the scram.

A. Incorrect – Reactor power is required to be lower to 50-70 MWe before scramming. Plausible because a Reactor scram is eventually directed by AOP-72. EDGs are manually started, not allowed to auto-start after the scram. Plausible because EDG operation prior to the scram needs to be minimized and EDGs would auto-start on the scram.
B. Incorrect – Reactor power is required to be lower to 50-70 MWe before scramming. Plausible because a Reactor scram is eventually directed by AOP-72.
C. Incorrect – EDGs are manually started, not allowed to auto-start after the scram. Plausible because EDG operation prior to the scram needs to be minimized and EDGs would auto-start on the scram.

Technical Reference(s): AOP-72

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Modified Bank – 9/12 NRC #20

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	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.

Proposed Answer: A

Explanation: The current conditions have DC power system A energized from the Battery and bus voltage is slightly low, but still above the alarm setpoint of 120 VDC. The bases for Technical Specification 3.8.4 require both the Battery and Battery Charger to be operable in order to call the associated 125 VDC subsystem operable. Since the Battery Charger is not operable, Technical Specification 3.8.4 Condition A must be entered. Technical Specification 3.8.7 Surveillance Requirement 3.8.7.1 requires Battery Bus voltage to be greater than 128 VDC to be operable. Since Battery Bus voltage is less than this value, one 125 VDC power distribution subsystem is inoperable, which required entry into Technical Specification 3.8.7 Condition A.

B. Incorrect – TS 3.8.7 Condition entry is also required because actual voltage is less than 128 VDC.
C. Incorrect – TS 3.8.4 Condition entry is required because, even though the Battery is still supplying the bus, the Battery Charger is also required to be operable.
D. Incorrect – TS 3.8.4 Condition entry is required because, even though the Battery is still supplying the bus, the Battery Charger is also required to be operable. TS 3.8.7 Condition entry is also required because actual voltage is less than 128 VDC.

Technical Reference(s): Technical Specification 3.8.4 and 3.8.7 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.16

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 #	1
	K/A #	295019 AA2.02
	Importance Rating	3.7

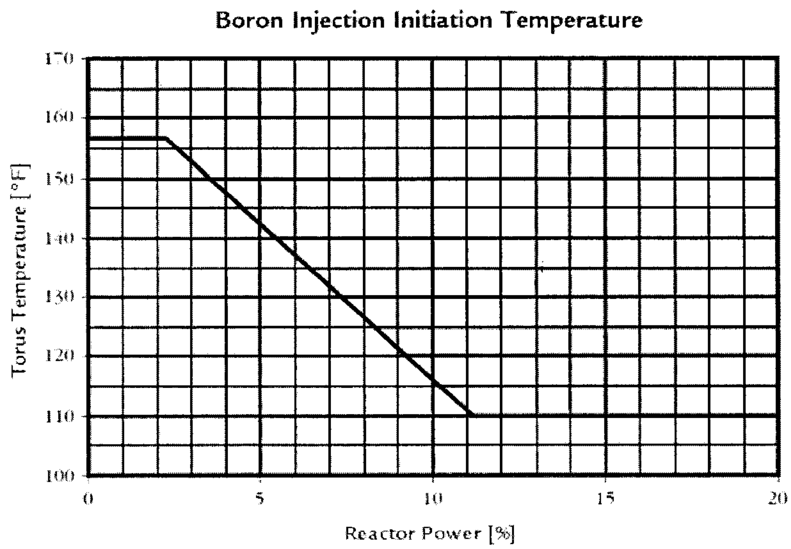
Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Proposed Question: #78

A failure to scram has occurred with the following:

- RCIC is out of service.
- A complete loss of Instrument Air pressure occurred 45 minutes ago.
- EOP-3, Failure to Scram, is being executed.
- Reactor water level is being intentionally lowered.
- Reactor water level is 90 inches and slowly lowering.
- Reactor pressure is 950 psig and slowly lowering with two SRVs open.
- Reactor power is 10% and slowly lowering.
- Torus temperature is 140°F and slowly rising.
- Drywell pressure is 1.5 psig and stable.



Which one of the following describes an appropriate Reactor water level control strategy, in accordance with EOP-3?

A. Re-commence injection with HPCI and maintain Reactor water level between -19 inches and 110 inches.

B. Re-commence injection with Feedwater and maintain Reactor water level between -19 inches and 110 inches.

C. Continue to lower Reactor water level until Reactor power drops below 2.5%, Reactor water level lowers to 0 inches, or all SRVs remain closed. Then re-commence injection with HPCI.

D. Continue to lower Reactor water level until Reactor power drops below 2.5%, Reactor water level lowers to 0 inches, or all SRVs remain closed. Then re-commence injection with Feedwater.

Proposed Answer: C

Explanation: EOP-3 contains the following steps:

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).
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.

The first step is satisfied with Reactor water level at 90 inches and would allow re-injection. However, since Reactor power is above 2.5%, Reactor water level is above 0 inches, Torus temperature is above the BIIT (~115°F at 10% power), and SRVs are open, the second step requires further lowering of Reactor water level. Once injection is allowed, Feedwater is unavailable due to the complete loss of instrument air, which causes outboard MSIVs to close and loss of motive steam to the Feedwater pumps. At 950 psig with RCIC out of service and Feedwater unavailable, only HPCI is available for injection. HPCI is unaffected by the loss of instrument air.

A. Incorrect – The given conditions require Reactor water level to be lowered further.
B. Incorrect – The given conditions require Reactor water level to be lowered further. Feedwater is unavailable due to the prolonged, complete loss of instrument air pressure.
D. Incorrect – Feedwater is unavailable due to the prolonged, complete loss of instrument air pressure.

Technical Reference(s): EOP-3, EOP-11, AOP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.07

Question Source: Bank – 3/14 NRC #77

Question History: 3/14 NRC #77

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 #	295001 2.1.23
	Importance Rating	4.4

Partial or Complete Loss of Forced Core Flow Circulation

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

Proposed Question: #79

The plant is operating at 34% power with the following:

- Reactor Water Recirculation (RWR) pump A tripped earlier in the shift.
- The cause of the trip has been determined and corrected.
- The plant is operating in the un-shaded region of the single-loop power to flow map.
- Preparations are underway for re-starting RWR pump A per OP-27, Recirculation System.
- RWR pump B speed is 45%.
- The temperature differential between the coolant in RWR loop A and the coolant in the RPV is 65°F.

Which one of the following describes the acceptability of these conditions for re-start of RWR pump A, in accordance with OP-27, Technical Specifications, and the Technical Requirements Manual?

A. Conditions are acceptable for the start of RWR pump A.

B. Reactor power is acceptable for the start of RWR pump A, but the temperature differential is NOT acceptable.

C. The temperature differential is acceptable for the start of RWR pump A, but Reactor power is NOT acceptable.

D. Both Reactor power and the temperature differential are NOT acceptable for the start of RWR pump A.

Proposed Answer: B

Explanation: OP-27, Technical Specifications, and the TRM place multiple limits on parameters for start of an RWR pump. Reactor power must be controlled as follows:

- All control rods are fully inserterd in core cells containing one or more fuel assemblies, OR
- Reactor is in Mode 1 and Reactor power is >10%.

The temperature differential between the coolant in the idle RWR loop and the Reactor must be ≤50°F. With the Reactor in Mode 1, Reactor power is acceptable since it is greater than 10%. However, the temperature differential is unacceptable since it is greater than 50°F.

A. Incorrect – Temperature differential is unacceptable.
C. Incorrect – Reactor power is acceptable, however temperature differential is unacceptable.
D. Incorrect – Reactor power is acceptable.

Technical Reference(s): OP-27, TS SR 3.4.9.5, PTLR

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: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b) (2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.1.20
	Importance Rating	4.6

Partial or Complete Loss of AC Power

Ability to interpret and execute procedure steps.

Proposed Question: #80

The plant has experienced a Station Blackout with the following:

- AOP-49, Station Blackout, has been entered.
- Reactor water level is 180 inches and lowering slowly with all injection secured.
- Both HPCI and RCIC are available for injection.
- Reactor pressure is 900 psig and lowering slowly with an SRV open.
- The current Reactor cooldown rate is 50°F/hr.

Which one of the following describes the required direction to be given regarding the preferred injection system and control of the Reactor cooldown rate, in accordance with AOP-49?

Direct use of...

A. RCIC as the preferred injection system and raising the cooldown rate.

B. RCIC as the preferred injection system and lowering the cooldown rate.

C. HPCI as the preferred injection system and raising the cooldown rate.

D. HPCI as the preferred injection system and lowering the cooldown rate.

Proposed Answer: B

Explanation: AOP-49 requires using RCIC preferentially over HPCI to minimize cycling of DC powered motor-operated valves. AOP-49 also requires an attempt to maintain the cooldown rate < 20°F/hr. The current cooldown rate of 50°F/hr with an SRV open must be lowered.

A. Incorrect – The current cooldown rate of 50°F/hr is above the desired rate of <20°F/hr and must be lowered, not raised. 50°F/hr is below the target cooldown rate during a normal plant cooldown per OP-13D.
C. Incorrect – RCIC, not HPCI, is the preferred system during a station blackout. The current cooldown rate of 50°F/hr is above the desired rate of <20°F/hr and must be lowered, not raised. 50°F/hr is below the target cooldown rate during a normal plant cooldown per OP-13D.
D. Incorrect – RCIC, not HPCI, is the preferred system during a station blackout.

Technical Reference(s): AOP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Modified Bank – 2010 NRC #81

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 #	295028 2.1.7
	Importance Rating	4.7

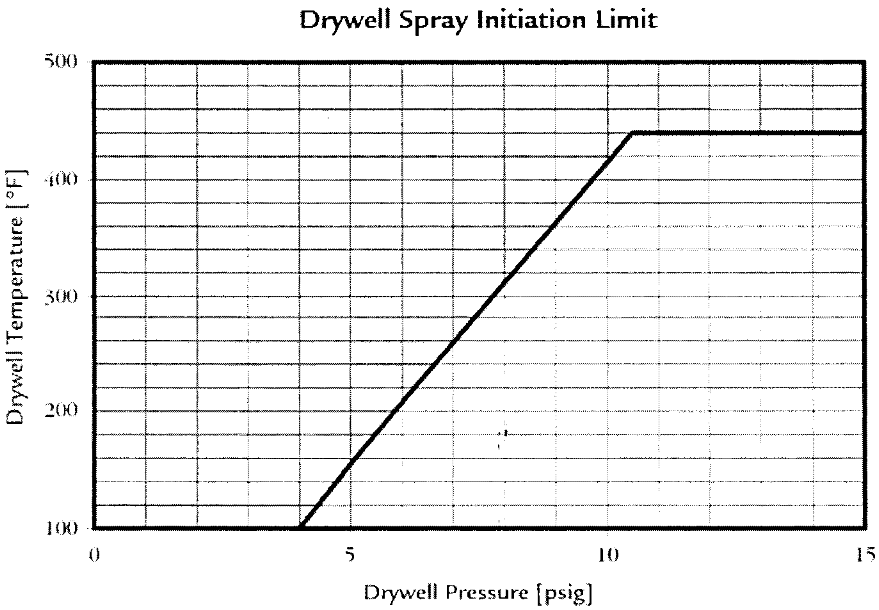
High Drywell Temperature

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

Proposed Question: #81

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

- Drywell temperature is 330°F and slowly rising.
- Drywell pressure is 8 psig and slowly rising.
- Torus water level is 14 feet and stable.
- Torus spray is in service.



Which one of the following describes the required action to be directed, in accordance with EOP-4, Primary Containment Control?

A. Spray the Drywell using RHR per OP-13B, Containment Control.

B. Perform an Emergency RPV Depressurization per EOP-2, RPV Control.

C. Spray the Drywell using RHR Service Water per EP-14, Alternate Containment Sprays.

D. Vent and purge the Containment per EP-6, Post Accident Containment Venting and Gas Control.

Proposed Answer: B

Explanation: With Drywell cooling unavailable due to high Drywell pressure and Drywell temperature greater than 309°F and rising, EOP-4 would normally require spraying the Drywell using RHR. However, the combination of Drywell pressure and temperature are on the BAD side of the Drywell Spray Initiation Limit. This prevents the use of both Drywell spray using RHR and alternate Drywell spray using RHR Service Water. With no other options available, Drywell temperature cannot be restored and maintained below 309°F, therefore Emergency RPV Depressurization is required.

A. Incorrect – The combination of Drywell pressure and temperature are on the BAD side of the Drywell Spray Initiation Limit. This prevents the use of Drywell spray, even though it would otherwise be required due to Drywell temperature exceeding 309°F.
C. Incorrect – The combination of Drywell pressure and temperature are on the BAD side of the Drywell Spray Initiation Limit. This prevents the use of Drywell spray, even though it would otherwise be required due to Drywell temperature exceeding 309°F. This applies to both Drywell spray using RHR pumps and alternate Drywell spray using RHR Service Water pumps.
D. Incorrect – While vent and purge would assist in lowering Drywell temperature, Containment venting is not authorized in EOP-4 for this purpose. EOP-4 only directs venting if Drywell pressure gets higher (approaching 60 psig) or if required to control hydrogen concentrations. Neither of these conditions are currently present.

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.03

Question Source: Bank – NMP2 2012 NRC #76

Question History: NMP2 2012 NRC #76

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 #	295037 2.4.18
	Importance Rating	4.0

SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the specific bases for EOPs.

Proposed Question: #82

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

- Reactor water level is being controlled -19 to 110 inches using Feedwater.
- Reactor pressure is 925 psig and stable on Turbine Bypass Valves.
- Reactor power is 5% and lowering slowly.
- Standby Liquid Control (SLC) is injecting.
- IRMs and SRMs are fully inserted.

Which one of the following identifies:

(1) the condition required to be met to allow restoring Reactor water level to a band of 177 to 222.5 inches while in EOP-3, Failure to Scram, and

(2) the basis for restoring Reactor water level back to this band, in accordance with EOP-3?

	(1) Condition Required to Allow Raising Reactor Water Level to Band of 177 to 222.5 Inches While in EOP-3	(2) Basis for Restoring Reactor Water Level
A.	SLC tank level must drop by 26%	Raise boron concentration in the core region
B.	SLC tank level must drop by 26%	Reduce Reactor vessel thermal stratification
C.	When all IRMs are less than Range 6	Raise boron concentration in the core region
D.	When all IRMs are less than Range 6	Reduce Reactor vessel thermal stratification

Proposed Answer: A

Explanation: Once Reactor water level has been lowered, EOP-3 requires waiting until SLC tank level drops by 26% before allowing Reactor water level to be restored to the normal band of 177 to 222.5 inches. This ensures the Hot Shutdown Boron Weight has been injected. Restoring Reactor water level is then desired to mix boron and raise the concentration of boron in the core region. Prior to raising Reactor water level, a large portion of the injected boron stays in the lower plenum of the Reactor, and must be mixed into the core region to insert negative reactivity.

Note: The question meets SRO level guidelines by requiring detailed knowledge of specific steps in EOP-3 and not just the general mitigating strategy. The candidate must know the exact criteria for when to raise Reactor water level and the reason for this action.

B. Incorrect – The reason for raising Reactor water level is to mix boron and raise the boron concentration in the core. Raising Reactor water level also does lessen any thermal stratification, but is not the specific basis in EOP-3. Reducing thermal stratification is the basis for raising Reactor water level in AOP-30, Loss of Shutdown Cooling.
C. Incorrect – The required condition is for SLC tank level to drop by 26%, not when IRMs are less than Range 6. The "Wait" block in the Reactor pressure leg of EOP-3 is conditional upon neutron monitoring reading sufficiently low before allowing Reactor cooldown.
D. Incorrect – The required condition is for SLC tank level to drop by 26%, not when IRMs are less than Range 6. The "Wait" block in the Reactor pressure leg of EOP-3 is conditional upon neutron monitoring reading sufficiently low before allowing Reactor cooldown. The reason for raising Reactor water level is to mix boron and raise the boron concentration in the core. Raising Reactor water level also does lessen any thermal stratification, but is not the specific basis in EOP-3. Reducing thermal stratification is the basis for raising Reactor water level in AOP-30, Loss of Shutdown Cooling.

Technical Reference(s): EOP-3, MIT-301.11D

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: Memory or Fundamental Knowledge

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 100% power with the following:

- All control rod position indication is lost.
- Then, a Main Turbine trip occurs.
- APRMs indicate downscale.
- Reactor water level lowers to 150 inches and then rises with Feedwater injecting.

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 until Reactor water level rises above 177".

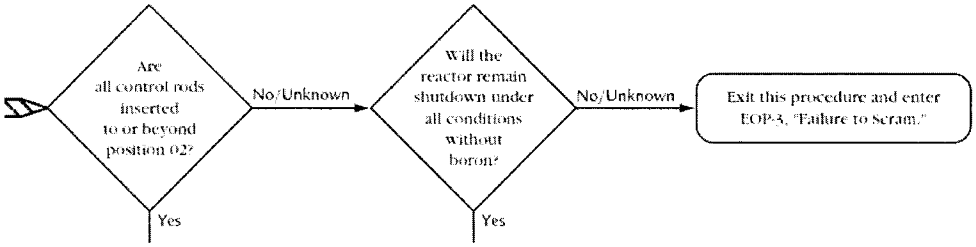
B. then exit to EOP-3, Failure to Scram. EOP-3 may NOT be exited until control rod positions are known.

C. then exit to EOP-3, Failure to Scram. EOP-3 may be exited once all IRMs are fully inserted and indicate less than range 6.

D. then exit to EOP-3, Failure to Scram. EOP-3 may be exited if Standby Liquid Control is used to inject the Cold Shutdown Boron Weight.

Proposed Answer: B

Explanation: EOP-2 entry is required because Reactor water level lowered below 177". With no control rod position information both of the following steps must be answered "No/Unknown":



Therefore EOP-2 must be exited and EOP-3 must be entered. EOP-3 is only exited once one of the following conditions is satisfied:

While executing this procedure:	
IF	**THEN**
All control rods are inserted to or beyond position 02 **OR** It has been determined the reactor will remain shutdown under all conditions without boron	1. **IF** .... boron is being injected to shut down the reactor, **THEN** ... terminate boron injection. 2. Exit this procedure and enter EOP-2, "RPV Control."

Neither of these two determinations can be made until control rod positions are known. Therefore EOP-3 cannot be exited until control rod positions are known.

A. Incorrect – With control rod position unknown, EOP-2 must be exited and EOP-3 must be entered. APRMs being downscale is not enough to avoid EOP-3 entry.
C. Incorrect – EOP-3 cannot be exited until control rod positions are known. IRMs less than range 6 only allows RPV cooldown while still in EOP-3.
D. Incorrect – EOP-3 cannot be exited until control rod positions are known. Injection of Cold Shutdown Boron Weight only allows RPV cooldown and restoration of normal water level while still in EOP-3.

Technical Reference(s): EOP-2, EOP-3, MIT-301.11C, MIT-301.11D

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C 1.03 and 1.07, MIT-301.11D 1.02 and 1.03

Question Source: Bank – 3/14 NRC #85

Question History: 3/14 NRC #85

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295034 2.4.41
	Importance Rating	4.6

Secondary Containment Ventilation High Radiation

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #84

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

- Fuel movement was occurring in the Spent Fuel Pool in preparation for an upcoming outage.
- An irradiated fuel bundle was dropped and damaged in the Spent Fuel Pool.
- The following radiation monitors have indicated upscale for over an hour:
    - o Refuel Floor Exhaust radiation monitors.
    - o Spent Fuel Pool area radiation monitor.
    - o Stack low range radiation monitors.
- Stack high range radiation monitors have indicated 900 mR/hr and stable for the past 30 minutes.
- Field survey teams have completed multiple surveys in the area.
- The surveys found the highest closed window dose rate of 150 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 two (2) hours.

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

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: C

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

- Unusual Event AU1.1 due to Stack and Refuel Floor Exhaust radiation monitors upscale.
- Unusual Event AU2.2 due to Spent Fuel Pool ARM upscale.
- Alert AA1.1 due to Refuel Floor Exhaust radiation monitors upscale.
- Alert AA2.1 due to damaged irradiated fuel causing high radiation.
- Site Area Emergency AS1.3 due to field survey indicating closed window dose rates > 100 mRem/hr and expected to continue for ≥ 1 hour at or beyond the site boundary.

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

A. Incorrect – Multiple Unusual Event EALs are exceeded, however the highest EAL that is met or exceeded is a Site Area Emergency.
B. Incorrect – Multiple Alert EALs are exceeded, however the highest EAL that is met or exceeded is a Site Area Emergency.
D. Incorrect – Multiple General Emergency EALs are approached, but not met or exceeded. The highest EAL that is met or exceeded is a Site Area Emergency.

Technical Reference(s): IAP-2

Proposed references to be provided to applicants during examination: Hot and Cold EAL Chart

Learning Objective: LP-AOP 1.12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 EA2.03
	Importance Rating	3.8

Secondary Containment High Sump/Area Water Level

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level

Proposed Question: #85

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

- A seismic event has occurred.
- The OBE EXCEEDED light is NOT lit at the Seismic Monitoring Panel.
- East Crescent area water level is 26" and rising slowly.
- West Crescent area water level is 20" and rising slowly.
- Torus water level is 13.7' and lowering very slowly.
- Attempts to isolate the leak have been unsuccessful.
- An operator has been directed to add water to the Torus, but has NOT yet completed this action.

Which one of the following describes the required control of the Reactor, in accordance with the Emergency Operating Procedures?

A. The Reactor may continue to operate at the current power level.

B. A Reactor shutdown is required, but a Reactor scram is NOT required.

C. A Reactor scram is required, but an emergency RPV depressurization is not required.

D. A Reactor scram and emergency RPV depressurization are required.

Proposed Answer: B

Explanation: The high area water levels require entry into EOP-5, Secondary Containment Control. Both crescent area water levels are above the Max Safe value of 18". Since the leak is from the Torus, it is a non-primary system discharge. Therefore, EOP-5 requires a Reactor shutdown. EOP-5 would only require a Reactor scram and emergency RPV depressurization if these water levels were caused by a primary system discharge. Torus water level below 13.88' also requires entry into EOP-4, Primary Containment Control. Since Torus water level remains well above 10.75' and is only very slowly lowering, EOP-4 does not require a Reactor scram or emergency RPV depressurization.

Note: The K/A requires determining and/or interpreting the ***cause of*** a high water level condition as it applies to a high sump or area water level in the Secondary Containment. The question satisfies this K/A by giving a high water level in multiple areas of the Reactor Building (Secondary Containment) and requiring the candidate to interpret the cause/source of the high water level (***crack in Torus***) to determine the required course of action (procedural path in EOP-5 based on ***non-primary*** system discharge)

A. Incorrect – EOP-5 requires a Reactor shutdown due to both crescent area water levels >18".
C. Incorrect – Since this is a non-primary system discharge, a Reactor scram is NOT required.
D. Incorrect – Since this is a non-primary system discharge, a Reactor scram and an emergency RPV depressurization is NOT required.

Technical Reference(s): EOP-5, EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11F 1.07

Question Source: Bank – Limerick 2008 NRC #85

Question History: Limerick 2008 NRC #85

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	217000 A2.07
	Importance Rating	3.1

RCIC

Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of lube oil

Proposed Question: #86

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

- RHR pump A is inoperable for corrective maintenance.
- HPCI operability has been verified by administrative means.
- RCIC is being operated for surveillance testing.
- A lube oil leak results in the following annunciators alarming:
  - 09-4-1-5, RCIC TURB CPLG END BRG TEMP HI
  - 09-4-1-15, RCIC TURB GOV END BRG TEMP HI
  - 09-4-1-25, RCIC TURB OIL PRESS LO
- The associated RCIC turbine bearing temperatures are 200°F and rising slowly.
- RCIC turbine oil pressure is 2 psig and stable.

Which one of the following describes:

(1) the required action, in accordance with the Alarm Response Procedures, and

(2) the most limiting Required Action in Technical Specification (TS) 3.5.3, RCIC System, that applies as a result of this event?

A. (1) Direct manually tripping RCIC.
(2) Be in Mode 3 in 12 hours.

B. (1) Direct manually tripping RCIC.
(2) Restore RCIC to operable status within 14 days.

C. (1) Direct verification of an automatic RCIC trip.
(2) Be in Mode 3 in 12 hours.

D. (1) Direct verification of an automatic RCIC trip.
(2) Restore RCIC to operable status within 14 days.

Proposed Answer: B

Explanation: RCIC bearing high temperatures and low oil pressure all result in control room alarms, but do not result in an automatic trip of RCIC. The associated alarm response procedures require directing a manual trip of RCIC. With RCIC inoperable, Technical Specification 3.5.3 Condition A must be entered. This requires verifying HPCI operable (already complete per stem conditions) and restoring RCIC to operable status within 14 days. Nothing in the question indicates that HPCI is inoperable, so a 12 hour shutdown to Mode 3 is NOT required. An inoperable RHR pump does NOT have the same impact in Technical Specification 3.5.3 as inoperable HPCI would. HPCI and RHR operability are intertwined in Technical Specifications. RCIC and HPCI operability are intertwined in Technical Specifications. However, RCIC and RHR operability are NOT intertwined in Technical Specifications.

A. Incorrect – An inoperable RHR pump does not have the same impact in Technical Specification 3.5.3 as inoperable HPCI would. If HPCI were inoperable, then Technical Specifications would require being in Mode 3 in 12 hours.
C. Incorrect – RCIC does not automatically trip due to low oil pressure or high bearing temperatures, but must be manually tripped. An inoperable RHR pump does not have the same impact in Technical Specification 3.5.3 as inoperable HPCI would. If HPCI were inoperable, then Technical Specifications would require being in Mode 3 in 12 hours.
D. Incorrect – RCIC does not automatically trip due to low oil pressure or high bearing temperatures, but must be manually tripped.

Technical Reference(s): ARPs 09-4-1-5, 09-4-1-15, 09-4-1-25, OP-19 Section B, Technical Specifications 3.5.1 and 3.5.3

Proposed references to be provided to applicants during examination: Technical Specifications 3.5.1 and 3.5.3

Learning Objective: SDLP-13 1.18.a

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 #	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:
    - o 09-4-1-17, ADS AUX RELAY ENERGIZED
    - o 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.

Proposed Answer: A

Explanation: A valid ADS initiation signal exists due to Reactor water level <59.5". When the ADS timer times out, ADS will open ADS valves since Core Spray pump B is operating properly. The ADS logic receives discharge pressure signals from all RHR and Core Spray pumps, however it only requires a single pump to be developing proper discharge pressure in order to open ADS valves. EOP-2 requires overriding ADS in this situation. ADS valves are only manually opened if Reactor water level continues to lower to 0".

B. Incorrect – EOP-2 requires overriding ADS in this situation. ADS valves are only manually opened if Reactor water level continues to lower to 0".
C. Incorrect – A valid ADS initiation signal exists due to Reactor water level <59.5". When the ADS timer times out, ADS will open ADS valves since Core Spray pump B is operating properly. The ADS logic receives discharge pressure signals from all RHR and Core Spray pumps, however it only requires a single pump to be developing proper discharge pressure in order to open ADS valves.
D. Incorrect – A valid ADS initiation signal exists due to Reactor water level <59.5". When the ADS timer times out, ADS will open ADS valves since Core Spray pump B is operating properly. The ADS logic receives discharge pressure signals from all RHR and Core Spray pumps, however it only requires a single pump to be developing proper discharge pressure in order to open ADS valves. EOP-2 requires overriding ADS in this situation. ADS valves are only manually opened if Reactor water level continues to lower to 0".

Technical Reference(s): EOP-2, OP-68 Section B

Proposed references to be provided to applicants during examination: None

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

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 #	215005 2.4.46
	Importance Rating	4.2

APRM / LPRM

Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: #88

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- The following annunciators are all ***clear***:
  - 09-5-1-41, NEUTRON MON SYS TRIP
  - 09-5-2-2, ROD WITHDRAWAL BLOCK
  - 09-5-2-34, APRM DOWNSCALE
  - 09-5-2-44, APRM UPSCALE
  - 09-5-2-54, APRM TRIP SYS A INOP OR UPSCALE TRIP
  - 09-5-2-55, APRM TRIP SYS B INOP OR UPSCALE TRIP
- APRMs indicate as follows:

APRM	Indication
A	10%
B	18%
C	8%
D	18%
E	16%
F	10%

Given the following requirements:

- Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation
- Technical Requirements Manual (TRM) 3.3.B, Control Rod Block Instrumentation

Which one of the following describes the need for Condition entry in TS 3.3.1.1 and/or TRM 3.3.B?

	TS 3.3.1.1 Condition Entry Required?	TRM 3.3.B Condition Entry Required?
A.	No	No
B.	Yes	No
C.	No	Yes
D.	Yes	Yes

Proposed Answer: D

Explanation: With the Reactor Mode Switch in STARTUP/HOT STANDBY, TS 3.3.1.1 requires 2 operable APRMs per trip system to initiate an upscale trip ≤15%. APRMs B, D, and E indicate >15%, but have failed to cause a scram, as evidenced by the clear annunciators. Therefore, these 3 APRMs must be inoperable for the high flux scram. RPS trip system A still has 2 potentially operable APRMs (A and C), but RPS trip system B only has 1 potentially operable APRM (F). Since RPS trip system B does not have the required number of operable channels, TS 3.3.1.1 Condition A must be entered.

With the Reactor Mode Switch in STARTUP/HOT STANDBY, TRM 3.3.B requires 4 operable APRMs to initiate an upscale rod block ≤12% and a downscale rod block ≥2.5%. APRMs B, D, and E indicate >12%, but have failed to cause a rod block, as evidenced by the clear annunciators. Therefore, these 3 APRMs must be inoperable for the high flux rod block. With only 3 potentially operable APRMs remaining (A, C, F), the requirement of TRM 3.3.B is not met and Condition A must be entered.

A. Incorrect – Both TS 3.3.1.1 and TRM 3.3.B require Condition entry.
B. Incorrect – Both TS 3.3.1.1 and TRM 3.3.B require Condition entry.
C. Incorrect – Both TS 3.3.1.1 and TRM 3.3.B require Condition entry.

Technical Reference(s): TS 3.3.1.1, TRM 3.3.B

Proposed references to be provided to applicants during examination: TS 3.3.1.1 and TRM 3.3.B (with allowable values removed)

Learning Objective: SDLP-07C 1.16

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 #	1
	K/A #	212000 2.2.44
	Importance Rating	4.4

RPS

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: #89

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

- Annunciator 09-5-2-58, ATTS RPS DIV A1 OR A2 GROSS FAIL OR TU INOP, alarms.
- Annunciator 09-5-2-60, ATTS RPS DIV B1 OR B2 GROSS FAIL OR TU INOP, alarms.
- Investigation reveals that the following transmitters have failed **upscale**:
    - 02-3LT-101A (Reactor Vessel Water Level – Low (Level 3))
    - 02-3LT-101D (Reactor Vessel Water Level – Low (Level 3))

Which one of the following describes the most limiting requirement based on these failures, if any, in accordance with Technical Specification 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

A. Restore trip capability in a maximum of 1 hour.

B. Place one channel or the associated trip system in the tripped condition in a maximum of 6 hours.

C. Place one channel or the associated trip system in the tripped condition in a maximum of 12 hours.

D. The minimum operability requirements of Technical Specification 3.3.1.1 remain satisfied.

Proposed Answer: B

Explanation: 02-3LT-101A inputs to the RPS trip system A low Reactor water level scram circuitry and 02-3LT-101D inputs to the RPS trip system B low Reactor water level scram circuitry. No other transmitters input to this function on RPS trip system B. The RPS logic is one out of two taken twice. With both of these transmitters failed upscale, RPS trip system A and B are still capable of tripping on low Reactor water level, however one function in each channel is inop which meets TS 3.3.1.1 Condition B and is limiting, which requires placing the channel in trip within 6 hours.

A. Incorrect – This would be the limiting requirement if these two transmitters input to the same side of RPS.
C. Incorrect – TS 3.3.1.1 Condition A also must be entered, which requires placing one channel or the associated trip system in the tripped condition within a maximum of 12 hours. However, TS 3.3.1.1 Condition C is more limiting.
D. Incorrect –1 of these level transmitters input to RPS trip system A and B, respectively, for the low Reactor water level scram function. Therefore, the minimum required channels per trip system is NOT satisfied.

Technical Reference(s): OP-27 Attachment 7, SDLP-02B, Technical Specification 3.3.1.1

Proposed references to be provided to applicants during examination: Technical Specifications 3.3.1.1 (with allowable values removed)

Learning Objective: SDLP-02B 1.18.a

Question Source: Modified Bank – 2010 NRC #87

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 #	1
	K/A #	400000 2.2.40
	Importance Rating	4.7

Component Cooling Water

Ability to apply technical specifications for a system.

Proposed Question: #90

The plant is operating at 100% power with a lake temperature of 86°F and rising slowly.

Considering the following Technical Specifications:

- 3.7.1, Residual Heat Removal Service Water (RHRSW) System
- 3.7.2, Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

Which one of the following identifies which of these Technical Specifications (TS), if any, currently require Condition entry?

A. NEITHER TS 3.7.1 NOR TS 3.7.2 requires Condition entry.

B. TS 3.7.1 requires Condition entry, but TS 3.7.2 does NOT.

C. TS 3.7.2 requires Condition entry, but TS 3.7.1 does NOT.

D. Both TS 3.7.1 and TS 3.7.2 require Condition entry.

Proposed Answer: C

Explanation: Lake temperature >85°F exceeds the Surveillance Requirement 3.7.2.2, therefore Condition entry is required for Technical Specification 3.7.2. RHRSW also ultimately requires an acceptable lake temperature for proper operation, however Technical Specification 3.7.1 bases specifically do NOT require Condition entry for Technical Specification 3.7.1 based on lake temperature (credit is taken for Condition entry in Technical Specification 3.7.2).

KA justification: From TS 3.7.2 Bases (ESW System and UHS); The ESW System is designed to provide cooling water for the removal of heat from equipment, such as the emergency diesel generators, electric bay coolers, crescent area coolers, cable tunnel/switchgear room coolers and control room and relay room air handling units, required for a safe reactor shutdown following a DBA or transient.

A. Incorrect – TS 3.7.2 requires Condition entry because lake temperature is >85°F.
B. Incorrect – TS 3.7.2 requires Condition entry because lake temperature is >85°F. While lake temperature impacts RHRSW performance, TS 3.7.1 bases exclude any lake temperature requirement.
D. Incorrect – While lake temperature impacts RHRSW performance, TS 3.7.1 bases exclude any lake temperature requirement and take credit for TS 3.7.2 Condition entry on lake temperature.

Technical Reference(s): Technical Specifications 3.7.1 and 3.7.2, and associated bases

Proposed references to be provided to applicants during examination: Technical Specifications 3.7.1 and 3.7.2, with SR 3.7.2.1 and 3.7.2.2 allowable values removed

Learning Objective: SDLP-46B 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 #	204000 A2.03
	Importance Rating	2.9

RWCU

Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Flow control valve failure

Proposed Question: #91

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

- The air supply to 12FCV-55, RWCU Blowdown Flow Control Valve, is lost due to a coupling failure.
- The resulting air leak has been isolated such that no other components are affected.

Then, a seismic event results in the following:

- Emergency RPV Depressurization is required due to low Torus water level.
- Multiple SRVs fail to open.
- A total of four (4) SRVs are able to be opened.
- Reactor pressure is 800 psig and lowering.
- Reactor water level is 205 inches and stable.

Which one of the following identifies:

(1) if use of Group 2 Pressure Control Systems is authorized, in accordance with EOP-2, RPV Control, and

(2) whether RWCU may be aligned as a Group 2 Pressure Control System?

	Is Use of Group 2 Pressure Control Systems Authorized?	Is RWCU Available for Use as a Group 2 Pressure Control System?
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: D

Explanation: The Emergency RPV Depressurization leg of EOP-2 requires attempting to open all 7 ADS valves. If fewer than all 7 can be opened, EOP-2 directs opening additional SRVs in an attempt to get a total of 7 open. EOP-2 then contains two decision points, such that if less than 5 SRVs are open and RPV pressure is greater than 50 psig above Torus pressure, then use of Group 2 Pressure Control Systems is required. RWCU is included on the Group 2 Pressure Control System list twice – for RWCU Recirc mode and for RWCU Blowdown mode. Loss of air to 12FCV-55 causes the valve to fail closed, which makes RWCU Blowdown mode unavailable. However, RWCU Recirc mode does not require opening 12FCV-55. Therefore, RWCU remains available for use as a Group 2 Pressure Control System in the Recirc mode.

A. Incorrect – Since Emergency RPV Depressurization is required, fewer than 5 SRVs can be opened, and Reactor pressure is still high, use of Group 2 Pressure Control Systems is authorized. Loss of air to 12FCV-55 prevents use of RWCU in Blowdown mode, however RWCU remains available for use as a Group 2 Pressure Control System in the Recirc mode.
B. Incorrect – Since Emergency RPV Depressurization is required, fewer than 5 SRVs can be opened, and Reactor pressure is still high, use of Group 2 Pressure Control Systems is authorized.
C. Incorrect – Loss of air to 12FCV-55 prevents use of RWCU in Blowdown mode, however RWCU remains available for use as a Group 2 Pressure Control System in the Recirc mode.

Technical Reference(s): EOP-2, SDLP-12, OP-28

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.07, SDLP-12 1.10.b

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 #	2
	K/A #	272000 2.1.31
	Importance Rating	4.3

Radiation Monitoring

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

Proposed Question: #92

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

- Annunciator 09-3-2-20, LIQUID PROCESS RAD MON DOWNSCALE OR INOP, alarms.
- The pictures on the following page show the status of the radiation monitors associated with this annunciator.
- Chemistry and Radiation Protection have verified that all actual plant radiation levels and release rates are normal.
- Radwaste discharge was NOT in progress prior to receipt of this annunciator and is NOT required.

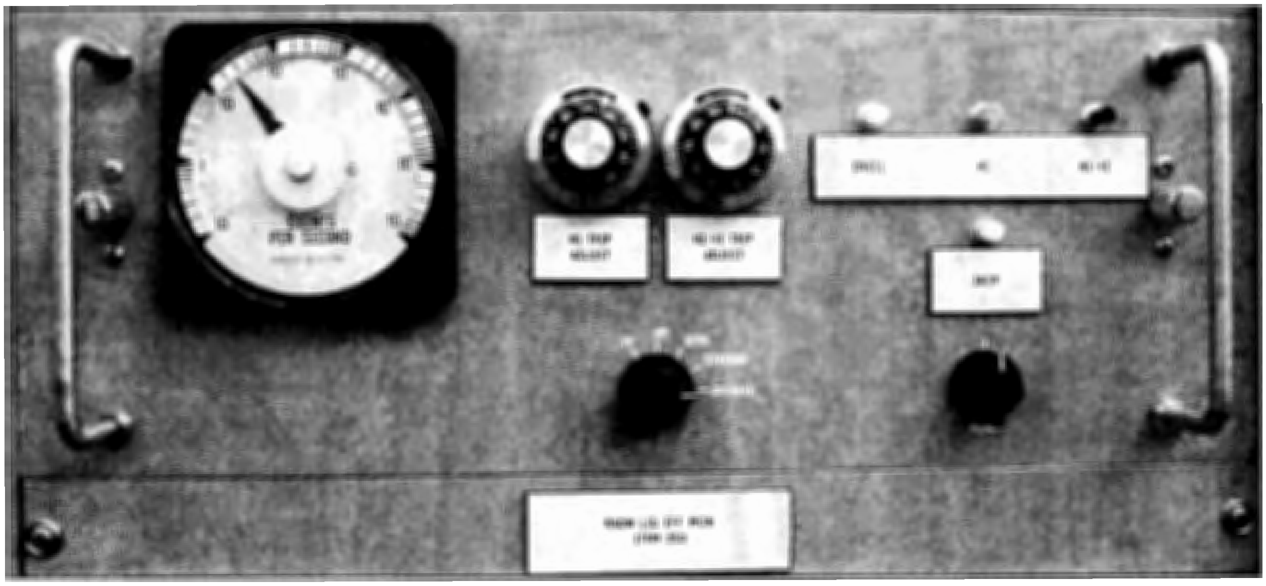
Which one of the following is required to allow discharge to occur or continue to occur from the affected system, in accordance with the Offsite Dose Calculation Manual (ODCM)?

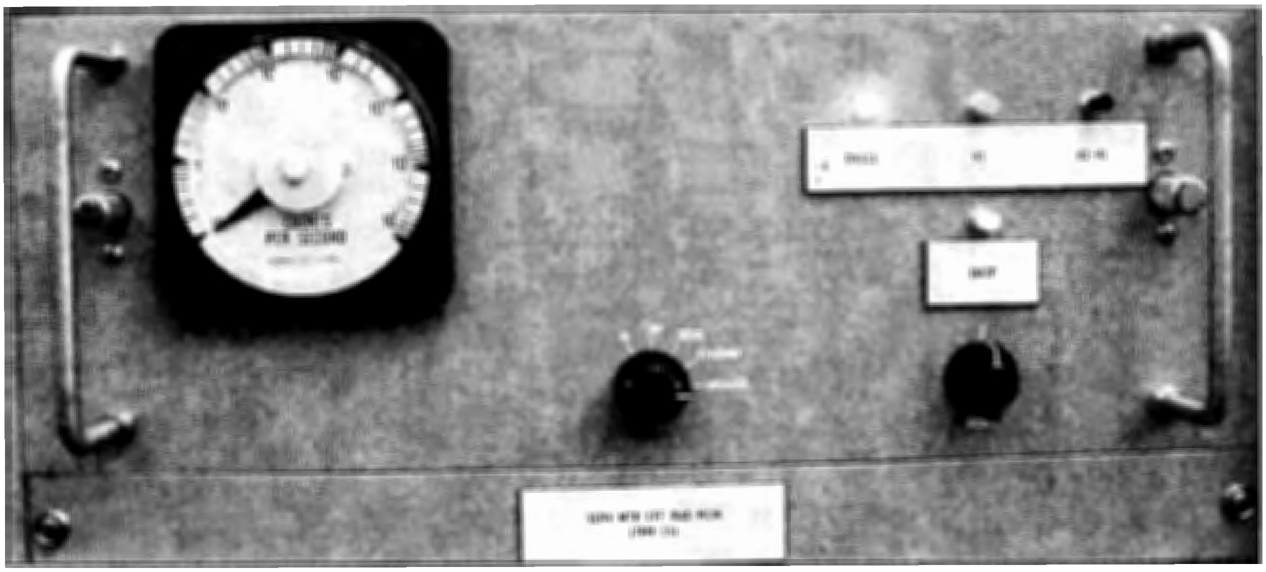
A. Verify the redundant radiation monitor is operable within 12 hours.

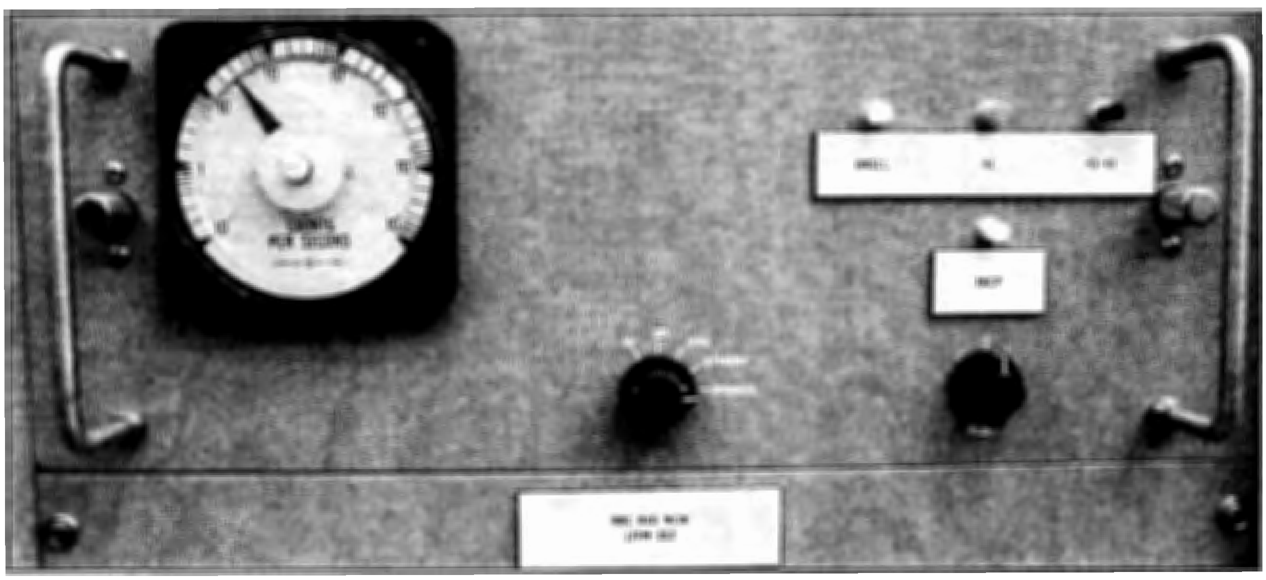
B. Collect and analyze effluent grab samples at least once each 12 hours.

C. Establish the ability for continuous monitoring of the effluent stream and monitor at least once each 15 minutes.

D. Collect and analyze two independent effluent grab samples and have two technically qualified individuals verify discharge line valving.







Proposed Answer: B

Explanation: The given indications show Service Water radiation monitor 17RE-332 failed downscale. ODCM Table 2.1-1 requires a Service Water effluent line radiation monitor to be operable. With the only installed Service Water effluent line radiation monitor inoperable, Action (b) applies and requires "With the number of operable channels less than the required minimum number, effluent releases in this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed..."

Note: The K/A requires testing the ability to locate control room switches/controls/indications related to Radiation Monitoring, and to determine their status relative to the desired plant lineup. The question meets the K/A by giving multiple Radiation Monitoring indications that are in the control room and requiring the candidate to determine which indications do and do not accurately reflect the desired plant conditions. From this determination, the candidate must determine the ODCM requirements for a failed Radiation Monitor.

A. Incorrect – ODCM Table 2.1-1 requires only one Service Water radiation monitor to be operable, so if there was a second one, the table would be met as long as it was operable. However, only one Service Water radiation monitor is installed.
C. Incorrect – The Service Water effluent line radiation monitor normally provides the ability to continuously monitor the effluent stream, but the ODCM does not require completely replacing that ability.
D. Incorrect – This is patterned after ODCM Table 2.1-1 Action (a), but this note applies to liquid radwaste radiation monitoring, not Service Water.

Technical Reference(s): ARP 09-3-2-20, ODCM section 2.1

Proposed references to be provided to applicants during examination: ODCM section 2.1

Learning Objective: SDLP-17 1.18

Question Source: Bank – NMP1 2015 NRC #96

Question History: NMP1 2015 NRC #96

Question Cognitive Level: Comprehension or Analysis

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

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

With regards to the High Pressure Water Fire,Protection System, which one of the following describes the required Technical Requirements Manual (TRM) action?

A. Immediately enter TRO 3.0.C.

B. Establish an hourly fire watch patrol.

C. Establish a backup fire suppression water system within 24 hours.

D. Restore the high pressure fire protection pump to functional status within 7 days.

Proposed Answer: D

Explanation: The Electric Motor Driven Fire Pump is powered from L44. Loss of L44 would make the Electric Motor Driven Fire Pump non-functional. TRM 3.7.H requires the Electric Motor Driven Fire Pump to be restored to functional status within a maximum of 7 days.

A. Incorrect – TRO 3.0.C is not required to be executed until all required actions of TRM 3.7.H (A.1, B.2, or B.3 and associated Completion Times are not met).
B. Incorrect – The required action is to restore the pump within 7 days. Establishing an hourly fire watch patrol is a required action in TRM 3.7.I for an inoperable water spray or sprinkler system, however the Diesel Driven Fire Pump maintains all water spray and sprinkler systems operable.
C. Incorrect - The Diesel Driven Fire Pump is credited as a backup fire suppression system pump iaw TRM 3.7.H Bases, thereby meeting the requirement of TRM 3.7.H B.2.

Technical Reference(s): OP-33 Attachment 2, TRM 3.7.H

Proposed references to be provided to applicants during examination: TRM 3.7.H

Learning Objective: SDLP-76 1.03

Question Source: Bank – NMP1 2015 Audit #93

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.35
	Importance Rating	3.9

Knowledge of the fuel-handling responsibilities of SROs.

Proposed Question: #94

The plant is refueling with the following:

- A fuel bundle is in transit from the Reactor core to the Spent Fuel Pool.
- The Reactor Analyst on the Refuel Bridge informs the Refuel SRO that the last fuel bundle placed in the Reactor core was in the wrong orientation.

Which one of the following describes the required actions, in accordance with OSP-66.001, Management of Refueling Activities?

	First...	Then...
A.	Place the current fuel bundle in its intended Spent Fuel Pool location.	Re-orientate the mis-orientated bundle and continue fuel moves.
B.	Place the current fuel bundle in its intended Spent Fuel Pool location.	Stop further fuel moves until GMPO permission is granted to continue.
C.	Return the current fuel bundle to its original location in the Reactor core.	Re-orientate the mis-orientated bundle and continue fuel moves.
D.	Return the current fuel bundle to its original location in the Reactor core.	Stop further fuel moves until GMPO permission is granted to continue.

Proposed Answer: B

Explanation: A mis-orientated fuel bundle is defined as a refuel error. Per OSP-66.001, should a refuel error occur, refueling activities shall be immediately stopped. Refueling activities shall not resume until the condition is resolved and GMPO has granted permission. OSP-66.001 states, "Suspension of core alterations shall not preclude completion of movement of a component to a safe position." With a mis-oriented fuel bundle in the Reactor, the safe position for the in-transit fuel bundle is in its intended location in the Spent Fuel Pool. OSP-66.001 only directs returning a component to its prior location if there is a reason it cannot be moved to the target location.

A. Incorrect – Refueling activities shall be immediately stopped and shall not resume until the condition is resolved and GMPO has granted permission.
C. Incorrect – OSP-66.001 states, "Suspension of core alterations shall not preclude completion of movement of a component to a safe position." With a mis-oriented fuel bundle in the Reactor, the safe position for the in-transit fuel bundle is in its intended location in the Spent Fuel Pool. OSP-66.001 only directs returning a component to its prior location if there is a reason it cannot be moved to the target location. Refueling activities shall be immediately stopped and shall not resume until the condition is resolved and GMPO has granted permission.
D. Incorrect – OSP-66.001 states, "Suspension of core alterations shall not preclude completion of movement of a component to a safe position." With a mis-oriented fuel bundle in the Reactor, the safe position for the in-transit fuel bundle is in its intended location in the Spent Fuel Pool. OSP-66.001 only directs returning a component to its prior location if there is a reason it cannot be moved to the target location.

Technical Reference(s): OSP-66.001

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 109.06

Question Source: Bank – 3/12 NRC #95

Question History: 3/12 NRC #95

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.22
	Importance Rating	4.7

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #95

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

- At time 0100 on March 1st, a common cause failure is discovered which makes multiple pieces of Technical Specification required equipment inoperable.
- The affected LCOs are applicable in Modes 1, 2, and 3.
- The Shift Manager immediately enters Technical Specification LCO 3.0.3.
- Corrective measures will take at least 3 days to restore compliance with Technical Specifications.

Which one of the following identifies the ***latest*** time by which Technical Specifications require the plant to be in Mode 4?

A. 1400 on March 1st

B. 2100 on March 1st

C. 1400 on March 2nd

D. 1000 on March 3rd

Proposed Answer: C

Explanation: LCO 3.0.3 requires initiating action within 1 hour to place the plant in Mode 2 within 7 hours, Mode 3 within 13 hours, and Mode 4 within 37 hours. Therefore, the plant must be in Mode 4 37 hours after 0100 on March 1, which is 1400 on March 2.

A. Incorrect – 1400 on March 1 is the latest time the plant must be in Mode 3, not Mode 4. The latest time the plant must be in Mode 4 is 1400 on March 2.
B. Incorrect – 2100 on March 1 is the latest time the plant would need to be in Mode 3, not Mode 4, if the 7 and 13 hour time requirements were added together. However, these time requirements run in parallel, not in series. The latest time the plant must be in Mode 4 is 1400 on March 2.
D. Incorrect – 1000 on March 3 is the latest time the plant would need to be in Mode 4 if the 7, 13, and 37 hour time requirements were added together. However, these time requirements run in parallel, not in series. The latest time the plant must be in Mode 4 is 1400 on March 2.

Technical Reference(s): Technical Specification LCO 3.0.3

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS01 1.01

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 #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	4.3

Ability to control radiation releases.

Proposed Question: #96

The plant was operating at 100% power when the following occurred:

- A Main Steam leak began discharging into the Turbine Building.
- The Reactor was scrammed.
- The MSIVs are stuck in mid-position.
- Turbine Building Ventilation has isolated due to exhaust duct high-high radiation.

Which one of the following describes the required direction to be given to control Turbine Building Ventilation per EOP-6, Radioactivity Release Control, and why?

Direct...

A. maintaining Turbine Building Ventilation isolated to contain the steam leak to the Turbine Building.

B. maintaining Turbine Building Ventilation isolated to limit total radioactivity release to the Site Boundary.

C. restoring Turbine Building Ventilation to service to limit unmonitored ground level radioactivity release.

D. restoring Turbine Building Ventilation to service to minimize steam damage to equipment in the Turbine Building.

Proposed Answer: C

Explanation: EOP-6 has a step that states, "IF Turbine Building Ventilation or Radwaste Building Ventilation is shutdown, or isolated due to high radiation, THEN Restart the ventilation system as required. Defeat isolation interlocks if necessary (EP-2)." The restart of Turbine Building Ventilation is required to direct any radioactive discharge to an elevated, monitored release point instead of a ground-level, unmonitored release point.

A. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not maintained isolated.
B. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not maintained isolated.
D. Incorrect – The reason for restarting Turbine Building Ventilation is to control radioactive release, not prevent equipment damage inside the Turbine Building.

Technical Reference(s): EOP-6, MIT-301.11G

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11G 6.04

Question Source: Modified Bank – 3/12 NRC #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.40
	Importance Rating	4.5

Knowledge of SRO responsibilities in emergency plan implementation.

Proposed Question: #97

The Plant has experienced an emergency radiological event with the following conditions:

Time	Condition
0800:	General Emergency (GE) declared.
0815:	Part 1 Notification sent with the following Protective Action Recommendations (ERPAs) circled: 1,2,3,4,5
0900:	GE still is still in effect **however**, environmental conditions **now** dictate ERPAs 3,4,5,6,7 warrant evacuation.

Which one of the following correctly lists the **latest time** a new Part 1 Notification is required to be transmitted by and the **ERPAs** that should be circled?

	Latest Time	ERPAs
A.	0915	3,4,5,6,7
B.	0915	1,2,3,4,5,6,7
C.	0930	3,4,5,6,7
D.	0930	1,2,3,4,5,6,7

Proposed Answer: B

Explanation: From EAP-1.1: Part 1 Notifications are required to be updated every 30 minutes unless: reclassification of EAL, initial PARs or a PAR changes occur. If PARs change, an updated Part 1 is required to be transmitted within 15 minutes. From EAP-4: Do not delete ERPAs from a PAR that previously recommended the ERPA for evacuation or sheltering unless an error was made and recognized prior to the County taking action to implement the PAR.

A. Incorrect – Do not delete ERPAs from a PAR that previously recommended the ERPA for evacuation or sheltering unless an error was made and recognized prior to the County taking action to implement the PAR.
C. Incorrect – Due to the PAR change, a new Part 1 is required within 15 minutes, not 30. Do not delete ERPAs from a PAR that previously recommended the ERPA for evacuation or sheltering unless an error was made and recognized prior to the County taking action to implement the PAR.
D. Incorrect – Due to the PAR change, a new Part 1 is required within 15 minutes, not 30.

Technical Reference(s): EAP-1.1, EAP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.2 EO-1.29

Question Source: Bank – 3/12 NRC #100

Question History: 3/12 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.8

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: #98

Which one of the following describes the approval requirements for personnel access to Very High Radiation Areas (VHRAs), in accordance with EN-RP-101, Access Control For Radiologically Controlled Areas?

Personnel access to VHRAs requires specific approval from the...

A. Shift Manager, only.

B. Radiation Protection Manager, only.

C. Shift Manager and the Radiation Protection Manager, only.

D. Shift Manager, Radiation Protection Manager and the Plant General Manager.

Proposed Answer: C

Explanation: EN-RP-101 5.6 [11] and Attachment 9.4 provide the required controls for personnel entry to a VHRA. Specific approval is required from both the Shift Manager and the Radiation Protection Manager.

A. Incorrect – Specific approval is also required from the Radiation Protection Manager for VHRA entry. Emergent entry to Steam Affected Areas (SAAs) is allowed with only approval from the Shift Manager per EN-RP-101.
B. Incorrect – Specific approval is also required from the Shift Manager for VHRA entry. Entry into certain Locked High Rad Areas (LHRAs) is allowed with only approval from the Radiation Protection Manager per EN-RP-101.
D. Incorrect – Approval is not required from the Plant General Manager. Plant General Manager approval is required for a Planned Special Exposure however.

Technical Reference(s): EN-RP-101

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 31.04.b

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 #	3
	Group #	
	K/A #	2.4.28
	Importance Rating	4.1

Knowledge of procedures relating to a security event (non-safeguards information).

Proposed Question: #99

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

- The NRC Operations Center calls with information about an airborne security threat.
- An airliner has been hi-jacked from Buffalo, NY.
- The airliner is approximately 20 minutes from the plant.
- A Probable Airborne Threat has been issued for James A. Fitzpatrick Nuclear Power Plant.
- This information has been validated.

Based on these current conditions, which one of the following describes the required control of the Reactor and the required Emergency Action Level (EAL) declaration, in accordance with AOP-70, Security Threat, and IAP-2, Classification of Emergency Conditions?

	Required Control of Reactor	Required EAL Declaration
A.	Insert a manual scram.	Alert
B.	Insert a manual scram.	Unusual Event
C.	Perform a rapid power reduction. A scram is NOT required.	Alert
D.	Perform a rapid power reduction. A scram is NOT required.	Unusual Event

Proposed Answer: C

Explanation: A Probable Airborne Threat (<30 minutes, >5 minutes) requires a rapid power reduction, but not a scram. A scram is only required if the time to threat arrival lowers to <5 minutes. Both EALs HU4.1 (Unusual Event) and HA4.1 (Alert) are met by the current threat, therefore the higher Alert declaration is required.

A. Incorrect – A scram is only required if the time to threat arrival lowers to <5 minutes.
B. Incorrect – A scram is only required if the time to threat arrival lowers to <5 minutes. Unusual Event EAL HU4.1 is met, however the higher Alert EAL HA4.1 is also met.
D. Incorrect – Unusual Event EAL HU4.1 is met, however the higher Alert EAL HA4.1 is also met.

Technical Reference(s): AOP-70, IAP-2

Proposed references to be provided to applicants during examination: Hot EAL Chart

Learning Objective: LP-AOP 1.03, 1.12

Question Source: Modified Bank – 3/12 NRC #99

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.42
	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 60°F.
  - 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 65°F and 75°F.

D. operable. Raise actual SLC tank temperature to between 80°F and 100°F.

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. 65-75°F 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. 80-100°F 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 – NMP1 2010 NRC #95

Question History: NMP1 2010 NRC #95

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: