Examination Outline Cross-Reference:	Level RO
	Tier # 1
	Group # 1
	K/A # 295001 AK2.07
	Importance Rating 3.5

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Core flow indication

Proposed Question: #1

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

Time (minutes)	Condition(s)
0	• RWR MG Set A Field Breaker trips. • AOP-8, Unexpected Change in Core Flow, is entered.
10	• A Reactor Operator reports that the required 5 minute wait period in AOP-8 has just commenced.

Which one of the following describes the status of flow in Recirculation loop A and indicated core flow at panel 09-5 at time = 11 minutes?

	Flow in Recirculation Loop A	Indicated Core Flow at Panel 09-5
A.	There is NO flow	Is accurate
B.	There is NO flow	Is NOT accurate
C.	Reverse flow is present	Is accurate
D.	Reverse flow is present	Is NOT accurate

Proposed Answer: A

Explanation: Given that the 5 minute wait period in AOP-8 has commenced, the discharge valve for Recirculation pump A has been fully closed and not yet throttled open. Therefore, there is no flow in Recirculation loop A. Indicated core flow is accurate under these conditions.

B. Incorrect – Indicated core flow is accurate under these conditions. Plausible because at this point in AOP-8, numerous required actions affecting the Recirculation system have not been taken, such as placing the TLO/SLO toggle switch in SLO.
C. Incorrect – There is currently no reverse flow in Recirculation loop A. Plausible because there was before the Reactor Operator closed the associated discharge valve and there will be again following the 5 minute wait period when the discharge valve is throttled back open.
D. Incorrect – There is currently no reverse flow in Recirculation loop A. Plausible because there was before the Reactor Operator closed the associated discharge valve and there will be again following the 5 minute wait period when the discharge valve is throttled back open. Indicated core flow is accurate under these conditions. Plausible because at this point in AOP-8, numerous required actions affecting the Recirculation system have not been taken, such as placing the TLO/SLO toggle switch in SLO.

Technical Reference(s): AOP-8, SDLP-02H

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.09.e

Question Source: Modified Bank - 3/12 NRC #1

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

AK 2.01 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: (CFR: 41.7 / 45.8)

Recirculation system

Proposed Question: 1

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

- Time = 00 min: 'A' RWR MG Set Field Breaker trips and AOP-8, Loss or Reduction of Reactor Coolant Flow, is entered.
- Time = 10 min: The Reactor Operator reports that the required 5 minute wait period for the idle RWR loop has commenced and that loop temperature is lowering 2°F\hr.

At Time = 11 min, the Operator who is monitoring total core flow would expect to see indicated flow to be __**(1)**__ actual flow because the reverse flow summer __**(2)**__.

	(1)	**(2)**
A.	the same as	senses no DP
B.	the same as	senses a negative DP and is subtracting the flow
C.	less than	senses a positive DP and is subtracting the flow
D.	greater than	senses no DP

Proposed Answer:

A.	the same as	senses no DP

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AA2.02
	Importance Rating	4.2

Partial or Complete Loss of AC Power

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

Proposed Question: #2

A startup is in progress with the following:

- The Reactor is at the point of adding heat (POAH).
- Reactor pressure is 0 psig.

Then, a Station Blackout occurs. The UPS inverter fails to transfer to DC.

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

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

Proposed Answer: A

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

Note: A psychometrically sound question was only able to be developed to address two of the three parameters mentioned in the sub-K/A. The plant conditions used in the question were selected such that only two of the three parameters would be of concern, based on this question construction limitation.

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.14

Question Source: Bank - 17-1 NRC #53

Question History: 17-1 NRC #53

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 #	295004 AA2.03
	Importance Rating	2.8

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: Battery voltage

Proposed Question: #3

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

- Battery Charger A is de-energized due to a malfunction.
- Station Battery A voltage is 130 VDC and slowly lowering.
- AOP-45, Loss of DC Power System A, has been entered.

Which one of the following identifies the threshold for Station Battery A voltage that requires mitigating action and the associated concern, in accordance with AOP-45 and OP-43A, 125 VDC Power System?

Mitigating action must be taken when Station Battery A voltage reaches the threshold of…

A. 125 VDC to prevent damage to loads due to high current.

B. 125 VDC to prevent damage to the battery due to voltage reversal.

C. 105 VDC to prevent damage to loads due to high current.

D. 105 VDC to prevent damage to the battery due to voltage reversal.

Proposed Answer: D

Explanation: The second override in AOP-45 requires mitigating action when Station Battery A voltage reaches 105 VDC. This is based on mitigating damage to the battery due to voltage reversal.

A. Incorrect – 105 VDC is the threshold. Plausible because 125 VDC is the nominal rating of the battery and much lower than normal operating voltage. The reason is damage to the battery due to voltage reversal. Plausible because low voltage will also cause loads to pull higher currents, which is not desirable.
B. Incorrect – 105 VDC is the threshold. Plausible because 125 VDC is the nominal rating of the battery and much lower than normal operating voltage.
C. Incorrect – The reason is damage to the battery due to voltage reversal. Plausible because low voltage will also cause loads to pull higher currents, which is not desirable.

Technical Reference(s): AOP-45, OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Bank - NMP1 2015 Cert #51

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 2.4.31
	Importance Rating	4.2

Main Turbine Generator Trip

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #4

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

- The following annunciators alarm:
  - 09-7-3-08, MAIN GEN STATOR WTR SYS TROUBLE
  - 94-HSC-7, CONDUCTIVITY ABOVE 0.45 MICROMHO'S STATOR WATER CONDUCTIVITY HIGH
  - 94-HSC-8, CONDUCTIVITY ABOVE 9.66 MICROMHO'S STATOR WATER CONDUCTIVITY HIGH-HIGH
- Operators have validated these alarms.

Which one of the following describes the plant response and/or required Operator response, in accordance with the Alarm Response Procedures?

A. An automatic Turbine trip occurs immediately.

B. An automatic Turbine trip occurs after a 70 second time delay.

C. NO automatic Turbine trip occurs and NO Reactor scram is required.

D. NO automatic Turbine trip occurs. A rapid power reduction and Reactor scram are required.

Proposed Answer: D

Explanation: The given annunciators indicate that Stator Water conductivity is high. With annunciator 94-HSC-8 in alarm, conductivity is above 9.66 mmho/cm. There is no automatic Turbine trip on this signal. This requires a rapid power reduction. Starting from 100% power, the rapid power reduction will end with power still above 25%. Therefore, a Reactor scram will also then be required, which will lead to tripping of the Main Turbine as part of the scram response.

A. Incorrect – There is no automatic Turbine trip on this signal. Plausible because other Stator Water Cooling alarms are associated with automatic Turbine trips and a manual Reactor scram (with ensuing Turbine trip) is required.
B. Incorrect – There is no automatic Turbine trip on this signal. Plausible because other Stator Water Cooling alarms are associated with automatic Turbine trips and a manual Reactor scram (with ensuing Turbine trip) is required. Also plausible because there is a 70 second time delay associated with Turbine trips coming from Stator Water Cooling signals.
C. Incorrect – A Reactor scram is required. Plausible because this would be correct if initial Reactor power were lower or if 94-HSC-8 was not in alarm.

Technical Reference(s): 09-7-3-8, 94-HSC-7, 94-HSC-8

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94E 1.15

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 #	295006 AK3.06
Importance Rating	3.2

Scram

Knowledge of the reasons for the following responses as they apply to SCRAM: Recirculation pump speed reduction: Plant-Specific

Proposed Question: #5

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

- A small steam leak develops in the Drywell.
- The Reactor is manually scrammed in accordance with AOP-1, Reactor Scram.
- Reactor water level reaches a low of 140” before recovering to the normal band and being stabilized at 200” using Feedwater.

Which one of the following describes the resulting Recirc pump speeds and a reason for this runback?

	Recirc Pump Speed	Reason for Runback
A.	30%	Maintain adequate Recirc pump NPSH
B.	30%	Prevent damage to Reactor vessel internals
C.	44%	Maintain adequate Recirc pump NPSH
D.	44%	Prevent damage to Reactor vessel internals

Proposed Answer: A

Explanation: Under scram conditions with Reactor water level stabilized at 200" using Feedwater, total Feedwater flow is less than 20% of rated. This initiates Recirc speed limiter #1, which causes a runback to 30% speed. One of the reasons for this runback is to ensure adequate NPSH to the Recirc pumps.

B. Incorrect – Prevent damage to Reactor vessel internals is not a reason for this runback. Plausible because lowering Recirc flow through the Reactor vessel does reduce stress to many Reactor vessel internal components and one other reason for the runback is preventing bearing damage to Recirc pumps.
C. Incorrect – Recirc pumps runback to 30%, not 44%. Plausible because there is a 44% runback that occurs under other conditions.
D. Incorrect – Recirc pumps runback to 30%, not 44%. Plausible because there is a 44% runback that occurs under other conditions. Prevent damage to Reactor vessel internals is not a reason for this runback. Plausible because lowering Recirc flow through the Reactor vessel does reduce stress to many Reactor vessel internal components and one other reason for the runback is preventing bearing damage to Recirc pumps.

Technical Reference(s): OP-27, SDLP-02I

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.05.b.2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK3.01
	Importance Rating	4.1

Control Room Abandonment

Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Reactor SCRAM

Proposed Question: #6

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

- Control Room Evacuation was ordered per AOP-43, Plant Shutdown From Outside the Control Room.
- The ATC Operator has completed all required Control Room actions.
- Operators have exited the Main Control Room and are traveling to the Remote Shutdown Panel.

Which one of the following describes the position of the Reactor Mode Switch and the reason for this position?

The Reactor Mode Switch is in…

A. SHUTDOWN to insert all control rods prior to leaving the Control Room.

B. SHUTDOWN to allow enabling controls at the Remote Shutdown Panel.

C. RUN to avoid initiating a plant transient prior to assuming control from the Remote Shutdown Panel.

D. RUN because the plant is scrammed using only the scram pushbuttons to provide a redundant MSIV closure signal.

Proposed Answer: D

Explanation: AOP-43 requires scramming the Reactor prior to leaving the Control Room, but leaving the Reactor Mode Switch in RUN to provide a redundant closure signal to the MSIVs.

A. Incorrect – The Reactor Mode switch remains in RUN. Plausible because the Reactor is scrammed, which normally results in placing the Reactor Mode Switch in SHUTDOWN.
B. Incorrect – The Reactor Mode switch remains in RUN. Plausible because the Reactor is scrammed, which normally results in placing the Reactor Mode Switch in SHUTDOWN. Also plausible because action is required to enable controls at the Remote Shutdown Panel.
C. Incorrect – The Reactor is scrammed, thereby initiating a plant transient, prior to assuming control from the Remote Shutdown Panel. Plausible because AOP-43 does have actions to initiate a scram from outside the Control Room and it is generally desired to not initiate plant transients when control is degraded.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Bank – NMP1 2009 Cert #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of CCW

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Proposed Question: #7

The plant is operating at 100% power when all Reactor Building Closed Loop Cooling (RBCLC) pumps become unavailable.

Which one of the following identifies a load that CANNOT have cooling restored from Emergency Service Water?

A. Drywell Ventilation Coolers

B. Drywell Equipment Drain Sump Cooler

C. Reactor Water Cleanup Pump Coolers

D. Control Rod Drive Hydraulic Pump Coolers

Proposed Answer: C

Explanation: Upon complete loss of all RBCLC pumps, ESW can be aligned to supply cooling water to many loads. Of the given loads, only Reactor Water Cleanup Pump Coolers CANNOT be supplied with cooling from ESW.

A. Incorrect – This load can be supplied from ESW. Plausible because it is normally supplied by RBCLC and not all RBCLC loads can be supplied from ESW.
B. Incorrect – This load can be supplied from ESW. Plausible because it is normally supplied by RBCLC and not all RBCLC loads can be supplied from ESW.
D. Incorrect – This load can be supplied from ESW. Plausible because it is normally supplied by RBCLC and not all RBCLC loads can be supplied from ESW.

Technical Reference(s): AOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12 1.10.a

Question Source: Bank – 2010 NRC #7

Question History: 2010 NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.01
	Importance Rating	3.5

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply

Proposed Question: #8

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: Bank – 16-1 NRC #45

Question History: 16-1 NRC #45

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 #	295021 AA1.02
	Importance Rating	3.5

Loss of Shutdown Cooling

Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling

Proposed Question: #9

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.

Note: The question meets the K/A by giving a loss of loss of Shutdown Cooling (due to loss of 10500 bus) and requiring the candidate to be able to monitor the RHR/shutdown cooling response (isolation of SDC due to RPS A loss) and resulting temperature indication impact.

A. Incorrect – SDC flow is lost due to loss of 10500 bus, which causes loss of RPS A and loss of SDC. Plausible because RHR loop B still has a pump with power available.
B. Incorrect – RWR flow is lost due to loss of 10300 bus. Plausible because this would be correct if the initial status of RWR pumps were swapped.
C. Incorrect – Since Reactor water level is below 234.5 inches, insufficient natural circulation exists to assure accurate Reactor coolant temperature indications. Plausible because Reactor water level is higher than the normal band.

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 - 16-1 NRC #40

Question History: 16-1 NRC #40

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK2.05
	Importance Rating	3.5

Refueling Accidents

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

Proposed Question: #10

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

- ISFSI cask loading activities in progress.
- 17RM-456A, Refuel Floor Exhaust Radiation Monitor A, spikes upscale and then fails downscale.

Then, an irradiated fuel bundle is dropped, resulting in the following:

- 17RM-456B, Refuel Floor Exhaust Radiation Monitor B, indicates $2x10^4$ cpm and stable.
- 17RM-452A, Below Refuel Floor Exhaust Radiation Monitor A, indicates 30 cpm and stable.
- 17RM-452B, Below Refuel Floor Exhaust Radiation Monitor B, reads 200 cpm and stable.

Which one of the following describes the response of the Reactor Building (RB) Ventilation system?

RB Ventilation train(s)…

A. A isolates, only.

B. B isolates, only.

C. A and B both isolate.

D. A and B both remain in service.

Proposed Answer: C

Explanation: 17RM-456A and B (Refuel Floor Exhaust Radiation Monitor) setpoints are:

- Downscale/inop – 10cpm
- High – $1x10^3$ cpm (EPIC only)
- High – $5x10^3$ cpm
- Hi-Hi – $1x10^4$ cpm

In the event of high-high radiation on either channel or downscale/inop on both channels, a RB Ventilation isolation signal will be generated. Therefore, with 17RM-456B >$1x10^4$ cpm, both RB Ventilation trains isolate.

A. Incorrect – Both trains isolate. Plausible that the A train would isolate on upscale/downscale failure and B train would not isolate until 17RM-456B indicated higher.
B. Incorrect – Both trains isolate. Plausible that the B train would isolate on 17RM-456B indication but A would not isolate with the downscale failure.
D. Incorrect – Both trains isolate. Plausible that isolation logic wouldn't be made up for either train with downscale failure and 17RM-456B at current indication.

Technical Reference(s): OP-51A, ARP 09-3-2-40, SDLP-66A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-66A 1.14

Question Source: Bank - 9/12 NRC #10

Question History: 9/12 NRC #10

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 #	295024 EK1.01
	Importance Rating	4.1

High Drywell Pressure

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific

Proposed Question: #11

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

- Reactor water level is 5” and rising slowly with Condensate pump B injecting.
- Core Spray pump A and HPCI have just been made available for injection.
- HPCI suction is aligned to the CSTs and CANNOT be swapped to the Torus.
- NO other injection sources are available.
- Drywell pressure is challenging the Primary Containment Pressure Limit (PCPL).

Which one of the following identifies the preferred injection source(s) under these conditions, in accordance with EOP-2, RPV Control?

A. HPCI

B. Core Spray pump A

C. Condensate pump B

D. HPCI and Condensate pump B

Proposed Answer: B

Explanation: EOP-2 contains the following guidance:

IF	THEN
Primary containment water level and pressure <u>cannot</u> be maintained below the Primary Containment Pressure Limit **AND** Adequate core cooling can be assured	Terminate injection into the RPV from sources external to the primary containment.

Condensate pump B injects from the Hotwell, which is external to the primary containment. With HPCI aligned to the CSTs and unable to be realigned to the Torus, it also injects from a source external to the primary containment. Core Spray pump A injects from the Torus, which is not external to the primary containment. Therefore, the preferred injection source, with PCPL being challenged, is Core Spray pump A.

A. Incorrect – Core Spray is the preferred injection source. Plausible because HPCI would be a preferred injection source if it could be realigned to take suction from the Torus. Also plausible because HPCI can inject at higher Reactor pressures, has finer control, and is from a relatively cleaner source. Also plausible because Core Spray is specifically not preferred in some situations (ATWS).
C. Incorrect – Core Spray is the preferred injection source. Plausible because Condensate is already injecting, has finer control, and is from a relatively cleaner source. Also plausible because Core Spray is specifically not preferred in some situations (ATWS).
D. Incorrect – Core Spray is the preferred injection source. Plausible because HPCI would be a preferred injection source if it could be realigned to take suction from the Torus. Also plausible because HPCI and Condensate have finer control and are from relatively cleaner sources. Also plausible because Core Spray is specifically not preferred in some situations (ATWS).

Technical Reference(s): EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.07

Question Source: Bank - SSES LOC28R NRC #39 (2017)

Question History: SSES LOC28R NRC #39 (2017)

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 #	295025 EA1.04
	Importance Rating	3.8

High Reactor Pressure

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

Proposed Question: #12

A scram has occurred with the following:

- MSIVs are closed.
- HPCI automatically started and is the only available injection source.
- HPCI injection has been throttled to 1000 gpm to the Reactor with the controller in AUTO.
- Reactor water level is 180 inches and slowly rising.
- Reactor pressure is 900 psig and slowly rising.

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

HPCI flow rate will…

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

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

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

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

Proposed Answer: C

Explanation: The HPCI pump is designed to supply 4250 gpm over a Reactor pressure range from 150 to 1195 psig. With rising Reactor pressure, HPCI flow will tend to lower. However, with the controller in AUTO, the HPCI governor valve will be automatically adjusted to maintain constant flow based on a flow feedback signal.

A. Incorrect – Flow will remain approximately constant. When in AUTO, the controller maintains constant flow, not speed.
B. Incorrect – Flow will remain approximately constant. HPCI is rated for 4250 gpm up to 1195 psig.
D. Incorrect – With rising Reactor pressure, HPCI flow will tend to lower based on the natural interplay of steam supply pressure vs. pump discharge pressure. Governor response is required to maintain flow approximately constant over this 200 psig change in Reactor pressure.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.a.22

Question Source: Bank - 14-2 NRC #46

Question History: 14-2 NRC #46

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

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

Suppression Pool High Water Temperature

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

Proposed Question: #13

Which one of the following describes

(1) when Emergency RPV Depressurization is required based on high Torus temperature

and

(2) the basis for this requirement,

in accordance with EOP-4, Primary Containment Control?

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

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

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

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

Proposed Answer: D

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Bank – 17-1 NRC #45

Question History: 17-1 NRC #45

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 2.2.42
	Importance Rating	3.9

High Drywell Temperature

Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Proposed Question: #14

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

- A dual seal failure occurred on Recirc pump A.
- Recirc pump A has been tripped and isolated.
- Drywell pressure is 1.85 psig and stable.
- Drywell average air temperature is 145ºF and stable.

Which one of the following describes the status of the Limiting Condition for Operation (LCO) for Technical Specifications 3.6.1.4, Drywell Pressure, and 3.6.1.5, Drywell Air Temperature?

	LCO 3.6.1.4, Drywell Pressure	LCO 3.6.1.5, Drywell Air Temperature
A.	NOT exceeded	NOT exceeded
B.	NOT exceeded	Exceeded
C.	Exceeded	NOT exceeded
D.	Exceeded	Exceeded

Proposed Answer: B

Explanation: Drywell pressure is below the LCO 3.6.1.4 limit of 1.95 psig. Drywell average air temperature is above the LCO 3.6.1.5 limit of 135°F.

A. Incorrect – Drywell average air temperature is above the LCO 3.6.1.5 limit of 135°F. Plausible because Drywell average air temperature is only slightly above the limit.
C. Incorrect – Drywell pressure is below the LCO 3.6.1.4 limit of 1.95 psig. Plausible because Drywell pressure is elevated and close to the limit. Drywell average air temperature is above the LCO 3.6.1.5 limit of 135°F. Plausible because Drywell average air temperature is only slightly above the limit.
D. Incorrect – Drywell pressure is below the LCO 3.6.1.4 limit of 1.95 psig. Plausible because Drywell pressure is elevated and close to the limit.

Technical Reference(s): Technical Specifications 3.6.1.4 and 3.6.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.16

Question Source: Modified Bank - SSES LOC27 NRC #65 (2017)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 1/30/20 – Changed to “Memory or Fundamental Knowledge” based on NRC comment.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295010 High Drywell Pressure / 5	Tier #	1	
2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits.	Group #	2	
	K/A #	295010	
	Importance Rating	4.0	

Proposed Question: **# 65**

Unit 1 is in Mode 3 with the following:

- A dual seal failure occurred on Reactor Recirculation pump (RRP) 1A.
- RRP 1A has been tripped and isolated.
- Drywell pressure is 2.2 psig, steady.
- Drywell average temperature is 145°F, steady.

Which one of the following describes the status of the Limiting Condition for Operation (LCO) for Technical Specifications 3.6.1.4, Containment Pressure, and 3.6.1.5, Drywell Air Temperature?

	LCO 3.6.1.4, Containment Pressure	LCO 3.6.1.5, Drywell Air Temperature
A.	NOT exceeded	NOT exceeded
B.	NOT exceeded	Exceeded
C.	Exceeded	NOT exceeded
D.	Exceeded	Exceeded

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK1.02
	Importance Rating	3.5

Low Suppression Pool Water Level

Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Pump NPSH

Proposed Question: #15

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

- Reactor water level is -10” and slowly lowering.
- Reactor pressure is 100 psig and slowly lowering.
- Torus pressure is 6 psig and slowly rising.
- Torus water level is 11’ and stable.
- RHR pumps A and C have just been made available for injection.
- NO other injection sources are available.

<u>**Note:**</u> The RHR Pump NPSH Limit is provided on the following page.

Which one of the following describes:

(1) the appropriate Torus Overpressure Line to use on the RHR Pump NPSH Limit

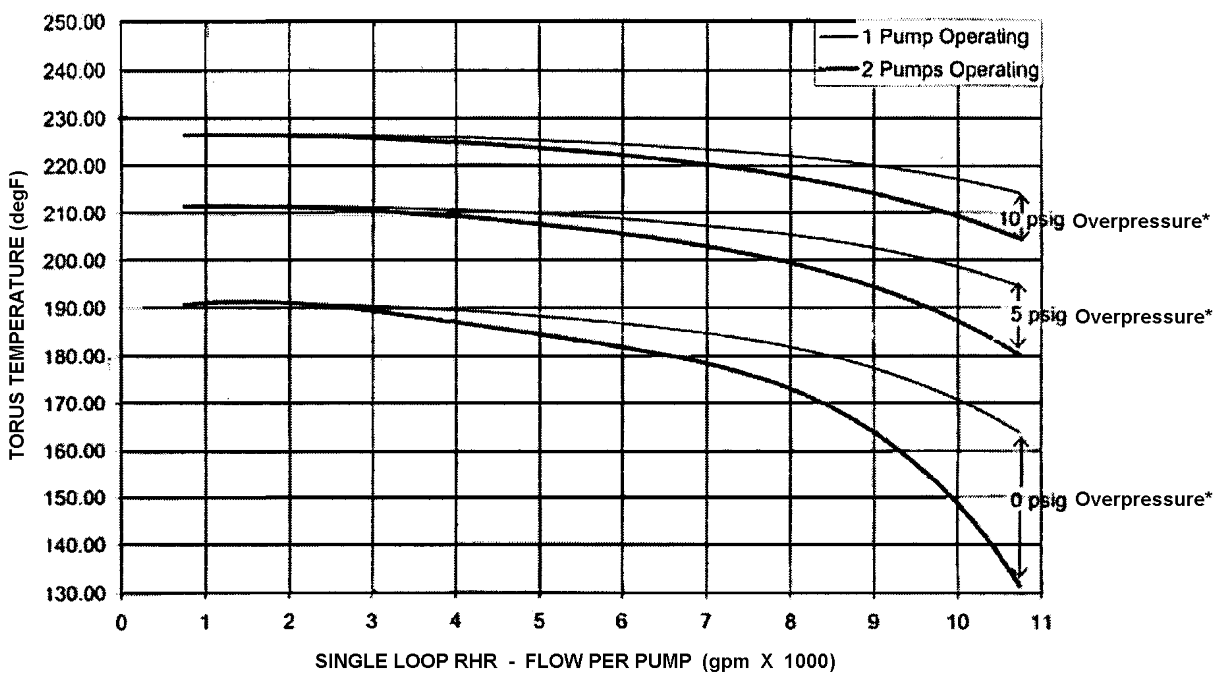
and

(2) whether or NOT RHR flow is allowed to temporarily exceed this limit,

in accordance with OP-13A, RHR – Low Pressure Coolant Injection?

	(1) Torus Overpressure Line to Use	(2) Is temporarily exceeding RHR Pump NPSH Limit allowed?
A.	5 psig	No
B.	5 psig	Yes
C.	10 psig	No
D.	10 psig	Yes

RHR PUMP NPSH LIMIT



*Torus Overpressure = Torus Pressure + 0.4 (Torus Water Level - 1.92) (See Note 1)

Proposed Answer: B

Explanation: The given conditions result in a Torus Overpressure calculation of 6 psig + 0.4*(11'-1.92) = 9.632. Note 1 on the associated Posted Attachment states "If the calculated value of Torus Overpressure falls between sets of curves, then use the lower set. (e.g.: Overpressure =2 psig, use the 0 psig curves)". Therefore, the 5 psig curve must be used. This limit is not an absolute limit, but rather a caution regarding the potential to cause pump damage. Therefore, the limit can be exceeded if necessary.

A. Incorrect – RHR flow is allowed to exceed the limit if necessary. Plausible because many, if not most, limits are not allowed to be exceeded.
C. Incorrect – The 5 psig curve must be used. Plausible because the calculated overpressure is 9.632, which is very close to 10. RHR flow is allowed to exceed the limit if necessary. Plausible because many, if not most, limits are not allowed to be exceeded.
D. Incorrect – The 5 psig curve must be used. Plausible because the calculated overpressure is 9.632, which is very close to 10.

Technical Reference(s): OP-13A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 2.4.34
	Importance Rating	4.2

Reactor Low Water Level

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: #16

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

- A fire in the Control Room requires execution of AOP-43, Plant Shutdown From Outside the Control Room.
- Operators have completed all required Control Room and field actions of AOP-43.
- Then, a small loss of coolant accident develops.
- Reactor water level is 150” and lowering.

Which one of the following describes the ability of HPCI and RCIC to automatically inject to the Reactor if Reactor water level continues to lower?

A. Both HPCI and RCIC will automatically inject.

B. HPCI will automatically inject, but RCIC will NOT.

C. HPCI will NOT automatically inject, but RCIC will.

D. NEITHER HPCI NOR RCIC will automatically inject.

Proposed Answer: D

Explanation: AOP-43 actions result in closing of the RCIC trip throttle valve, which prevents RCIC from automatically injecting to the Reactor given low water level. AOP-43 actions also result in taking the 23MOV-16 and -60 isolation switches to LOCAL, which prevents automatic initiation of HPCI.

A. Incorrect – HPCI will NOT automatically inject. Plausible because this would be correct if specific action was not taken in AOP-43 to prevent HPCI injection. RCIC will not automatically inject because operator action has been taken per AOP-43 to close the trip throttle valve. Plausible because this would be correct if specific action was not taken in AOP-43 to prevent RCIC injection.
B. Incorrect – HPCI will NOT automatically inject. Plausible because this would be correct if specific action was not taken in AOP-43 to prevent HPCI injection.
C. Incorrect – RCIC will not automatically inject because operator action has been taken per AOP-43 to close the trip throttle valve. Plausible because this would be correct if specific action was not taken in AOP-43 to prevent RCIC injection.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.08

Question Source: Bank - 17-2 NRC #21

Question History: 17-2 NRC #21

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK2.03
	Importance Rating	4.1

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: ARI/RPT/ATWS: Plant-Specific

Proposed Question: #17

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

- A malfunction occurs with the Main Turbine pressure regulator.
- Reactor pressure rises to a high value of 1170 psig and then lowers to 920 psig on Turbine Bypass Valves.

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

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

Proposed Answer: A

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

B. Incorrect – RWR pumps trip. Plausible because there was no low level condition, the pressure rise was mitigated by TBVs, and the pressure transient was short.
C. Incorrect – ARI solenoids are energized. Plausible because there was no low level condition, the pressure rise was mitigated by TBVs, and the pressure transient was short. Also plausible because normal scram solenoids are de-energize to function.
D. Incorrect – ARI solenoids are energized. Plausible because there was no low level condition, the pressure rise was mitigated by TBVs, and the pressure transient was short. Also plausible because normal scram solenoids are de-energize to function. RWR pumps trip. Plausible because there was no low level condition, the pressure rise was mitigated by TBVs, and the pressure transient was short.

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

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank - 14-1 NRC #50

Question History: 14-1 NRC #50

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK1.02
	Importance Rating	4.2

High Offsite Radioactivity Release Rate

Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Protection of the general public

Proposed Question: #18

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

- A Main Steam leak occurred in the Turbine Building.
- The Reactor was scrammed.
- The MSIVs are stuck in mid-position.
- The running Turbine Building Exhaust Fan has tripped.
- The standby Turbine Building Exhaust Fan has failed to auto-start.
- EOP-6, Radioactivity Release Control, has been entered.

Which one of the following describes the required control of Turbine Building Ventilation, in accordance with EOP-6, and the associated reason?

A. Restore a Turbine Building Exhaust Fan to service to limit unmonitored ground level radioactivity release.

B. Restore a Turbine Building Exhaust Fan to service to prevent equipment damage in the Turbine Building.

C. Ensure Turbine Building Ventilation is secured and isolated to contain the steam leak to the Turbine Building.

D. Ensure Turbine Building Ventilation is secured and isolated to limit total radioactivity release to the Site Boundary.

Proposed Answer: A

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.

B. Incorrect – The reason for restarting Turbine Building Ventilation is to control radioactive release, not prevent equipment damage inside the Turbine Building. Plausible because restoring ventilation will limit temperature rise in the Turbine Building and exhaust moisture, both of which will tend to prevent equipment damage.
C. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not isolated. Plausible because for a similar situation in the Reactor Building, Reactor Building ventilation is required to be isolated.
D. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not isolated. Plausible because for a similar situation in the Reactor Building, Reactor Building ventilation is required to be isolated.

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: Bank – 12-2 NRC #71

Question History: 12-2 NRC #71

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 #	600000 AA2.14
	Importance Rating	3.0

Plant Fire On Site

Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Equipment that will be affected by fire suppression activities in each zone

Proposed Question: #19

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

- A fire has developed in the Reactor Building.
- Zone 1A, West Crescent Area 227'6'', is in alarm.
- Automatic fire suppression has discharged in this zone.

Which one of the following identifies equipment in this area that may be affected and the automatic fire suppression agent in this area?

	Equipment in Area	Automatic Fire Suppression Agent
A.	RCIC	CO_2
B.	RCIC	Water
C.	HPCI	CO_2
D.	HPCI	Water

Proposed Answer: B

Explanation: RCIC is located in this area of the Reactor Building (Zone 1A, west crescent) and may be affected by the fire and fire suppression activities. The automatic fire suppression agent used in this area is water.

A. Incorrect – The automatic fire suppression agent used in this area is water. Plausible because CO_2 is used as an automatic fire suppression agent in many other plant areas.
C. Incorrect – HPCI is not located in this area of the Reactor Building. Plausible because this would be correct for a fire in the east portion of Reactor Building 227'. The automatic fire suppression agent used in this area is water. Plausible because CO_2 is used as an automatic fire suppression agent in many other plant areas.
D. Incorrect – HPCI is not located in this area of the Reactor Building. Plausible because this would be correct for a fire in the east portion of Reactor Building 227'.

Technical Reference(s): AOP-28, OP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.11.c.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 1/30/20 – Revised question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK2.02
	Importance Rating	3.1

Generator Voltage and Electric Grid Disturbances

Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Breakers, relays

Proposed Question: #20

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

Time (seconds)	Condition(s)
0	• A fault occurs on Line 3. • Breaker 10022, LHH-115KV LINE 3, opens.
5	• The fault on Line 3 clears. • Line 3 is energized with normal voltage.

Which one of the following identifies the status of Breaker 10022 and Disconnect 10017, NORTH-SOUTH 115 KV BUS DISC SW, at Time = 120 seconds?

	Breaker 10022	Disconnect 10017
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: D

Explanation: With the plant initially in a normal electrical lineup, Breaker 10022 is closed, Disconnect 10017 is closed, and the Breaker 10022 automatic reclosure feature is enabled. Since the fault clears in under 10 seconds and Line 3 has normal voltage, Breaker 10022 will automatically reclose at approximately Time = 10 seconds. Since Breaker 10022 is able to close at this time, Disconnect 10017 will remain closed throughout the transient.

A. Incorrect – Breaker 10022 will be closed. Plausible because this would be correct if the automatic reclosure feature was normally disabled, or if the fault did not clear for a longer period of time such that automatic reclosures attempted and then locked out. Disconnect 10017 will be closed. Plausible because this would be correct if the normal electrical lineup had this disconnect open, or in other fault responses.
B. Incorrect – Breaker 10022 will be closed. Plausible because this would be correct if the automatic reclosure feature was normally disabled, or if the fault did not clear for a longer period of time such that automatic reclosures attempted and then locked out.
C. Incorrect – Disconnect 10017 will be closed. Plausible because this would be correct if the normal electrical lineup had this disconnect open, or in other fault responses.

Technical Reference(s): SDLP-71D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D

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 #	295013 AK2.01
	Importance Rating	3.6

High Suppression Pool Water Temperature

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

Proposed Question: #21

The plant is operating at 90% power with the following:

- HPCI surveillance testing has just been completed in accordance with ST-4N, HPCI Quick-Start, In-service, and Transient Monitoring Test.
- RHR Loop A is operating in Torus Cooling.
- RHR pump A is operating.
- RHR pump C is secured.
- RHRSW flow has been established.
- Lake water temperature is 37°F.
- Torus temperature is 80°F and slowly lowering.

Which one of the following identifies the lowest allowable Torus temperature and whether RHR pump C is allowed to be started to raise the cooldown rate, in accordance with OP-13B, RHR – Containment Control?

The lowest allowable Torus temperature is…

A. 50°F. RHR pump C is allowed to be started.

B. 50°F. RHR pump C is NOT allowed to be started.

C. 72°F. RHR pump C is allowed to be started.

D. 72°F. RHR pump C is NOT allowed to be started.

Proposed Answer: C

Explanation: OP-13B states that the lowest allowable Torus temperature 72°F. RHR pump C is allowed to be started to raise the cooldown rate.

A. Incorrect – The lowest allowable Torus temperature 72°F. Plausible because 50°F is a temperature limit used in other plant procedures (Condenser waterbox temperature control, Offgas). RHR pump C is allowed to be started to raise the cooldown rate. Plausible that only one RHR pump would be allowed to be run in non-emergency conditions.
B. Incorrect – The lowest allowable Torus temperature 72°F. Plausible because 50°F is a temperature limit used in other plant procedures (Condenser waterbox temperature control, Offgas).
D. Incorrect – RHR pump C is allowed to be started to raise the cooldown rate. Plausible that only one RHR pump would be allowed to be run in non-emergency conditions.

Technical Reference(s): OP-13B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13

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 #	2
	K/A #	295020 AK1.02
	Importance Rating	3.5

Inadvertent Containment Isolation

Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION: Power / reactivity control

Proposed Question: #22

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

- The Reactor Mode Switch is in ***RUN***.
- An inadvertent isolation signal causes the following valves to close:
  - MSIV 29AOV-80A
  - MSIV 29AOV-80C
  - MSIV 29AOV-86B

Which one of the following describes Reactor power as indicated on the APRMs one (1) minute after this transient begins?

APRMs will indicate…

A. approximately 0%.

B. greater than 0% but less than 20%.

C. approximately 20%.

D. greater than 20%.

Proposed Answer: A

Explanation: On an isolation signal, MSIVs are required to close within 5 seconds. The MSIV closure scram logic is arranged so that with the Mode Switch in RUN, a scram will occur if an MSIV is closed in at least 3 of the 4 Main Steam Lines. The given valves are distributed between Main Steam Lines A, B, and C, therefore a Reactor scram will occur, even though 20% Reactor steam flow is well within the capacity of a single Main Steam Line. APRMs drop to 0% following the Reactor scram.

B. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. Plausible because Core Thermal Power will remain above 0% for an extended period of time due to decay heat.
C. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. Plausible because if only two Main Steam Lines isolated, the Reactor would NOT scram and power would stabilize back at approximately 20% within one minute. While a single Main Steam Line can handle 20% Reactor steam flow, the RPS logic will still enforce a scram in this situation.
D. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. Plausible because Reactor power will initially rise as MSIVs stroke closed and Reactor pressure rises, however the scram occurs well within one minute. While a single Main Steam Line can handle 20% Reactor steam flow, the RPS logic will still enforce a scram in this situation.

Technical Reference(s): OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29 1.05.b.1

Question Source: Bank – 14-2 NRC #63

Question History: 14-2 NRC #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EA1.03
	Importance Rating	2.9

High Suppression Pool Water Level

Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: RHR/LPCI

Proposed Question: #23

The plant is operating at 75% power with the following:

- An SRV inadvertently opened and was closed by Operators.
- Torus water level is now 14.1 feet and stable.

Which one of the following describes the method required to lower Torus water level, in accordance with EOP-4, Primary Containment Control?

A. Run an RHR pump to transfer Torus water to Radwaste

B. Run a Core Spray pump to transfer Torus water to Radwaste

C. Gravity drain through RHR piping to Reactor Building drains

D. Gravity drain through Core Spray piping to Reactor Building drains

Proposed Answer: A

Explanation: EOP-4 directs use of OP-13B to lower Torus water level. OP-13B sections G.1 and G.2 utilize a running RHR pump to transfer water Radwaste.

B. Incorrect – RHR, not Core Spray is used. Plausible because Core Spray also takes a suction on the Torus.
C. Incorrect – Torus water is actively pumped, not gravity drained. Plausible because the correct method does utilize the Reactor Building drain header as part of the flow path, and it would be physically possible to gravity drain to Reactor Building drains and then pump those drains using normal sump pumps.
D. Incorrect – Torus water is actively pumped using RHR, not gravity drained. Plausible because the correct method does utilize the Reactor Building drain header as part of the flow path, and it would be physically possible to gravity drain to Reactor Building drains and then pump those drains using normal sump pumps. Also plausible because Core Spray also takes a suction on the Torus.

Technical Reference(s): EOP-4, OP-13B

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 EK3.01
	Importance Rating	3.5

High Secondary Containment Area Temperature

Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Emergency/normal depressurization

Proposed Question: #24

Which one of the following describes a reason for Emergency RPV Depressurization due to high Secondary Containment temperature, in accordance with EOP-5, Secondary Containment Control?

Emergency RPV Depressurization is based on concerns for…

A. habitability of the Control Room

B. operability of the Primary Containment

C. loss of equipment located in the Reactor Building

D. loss of normal personnel access to the Reactor Building

Proposed Answer: C

Explanation: One of the reasons for Emergency RPV Depressurization due to high Secondary Containment temperatures is loss of equipment located in the Secondary Containment.

A. Incorrect – This is not a stated reason for Emergency RPV Depressurization due to high Secondary Containment temperatures. Plausible because an un-isolable primary system discharging into the Reactor Building does lead to temperature and radiation concerns that could impact the Control Room.
B. Incorrect – This is not a stated reason for Emergency RPV Depressurization due to high Secondary Containment temperatures. Plausible because high temperatures in the Secondary Containment do impact the Primary Containment structure, since it is located within the Secondary Containment.
D. Incorrect – This is not a stated reason for Emergency RPV Depressurization due to high Secondary Containment temperatures. Plausible because high temperatures in the Secondary Containment do impact the ability of personnel to enter the building.

Technical Reference(s): MIT-301.11f

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11f 1.07

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 1/30/20 – Reviewed wording of choice C based on NRC comment and found that it is in alignment with the wording in MIT-301.11f. Additionally, could not find wording such as "exceeding equipment environmental qualification" in the plant technical reference to support such a revision to choice C.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AA2.01
	Importance Rating	4.2

Low Reactor Water Level

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

Proposed Question: #25

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

- Reactor water level is -10” and slowly lowering.
- Reactor pressure is 150 psig and slowly lowering.
- CRD pump A is injecting to the Reactor.
- NO other Reactor injection sources are available.

Which one of the following describes the status of Adequate Core Cooling (ACC)?

ACC is…

A. NOT presently assured.

B. presently assured. ACC will be first lost if Reactor water level lowers below -19”.

C. presently assured. ACC will be first lost if Reactor water level lowers below -31.5”.

D. presently assured. ACC will be first lost if Reactor water level lowers below -44.5”.

Proposed Answer: B

Explanation: With Reactor water level below top of active fuel (0”), but above -19”, ACC is presently assured via “Steam Cooling with injection”. With no Core Spray pumps running and CRD injection, ACC will first be lost if Reactor water level reaches -19”.

A. Incorrect – With Reactor water level below top of active fuel (0”), but above -19”, ACC is presently assured via “Steam Cooling with injection”. Plausible because the preferred source of ACC, “Submergence”, is not currently in place (<0”).
C. Incorrect – With no Core Spray pumps running and CRD injection, ACC will first be lost if Reactor water level reaches -19”. Plausible because this would be correct if there was no injection to the Reactor.
D. Incorrect – With no Core Spray pumps running and CRD injection, ACC will first be lost if Reactor water level reaches -19”. Plausible because this would be correct if at least one Core Spray pump was running at or above 4725 gpm.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT 301.11C 2.04

Question Source: Modified Bank - 9/12 NRC #22

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 #	295009 AA2.01
Importance Rating	4.2

Low Reactor Water Level

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

Proposed Question: #22

The Plant has experienced a loss of coolant accident.

Core Spray Pump (CSP) 'A' is the only injecting source of water.

- RPV level: -10 inches and lowering
- CSP 'A' flow rate: 5075 gpm
- Safety Relief Valves: Seven open

Based on the above conditions, which one of the following describes the status of "Adequate Core Cooling" (ACC)?

ACC…

A. is NOT presently assured.

B. is presently assured. ACC will be first lost if RPV level reaches -19 inches.

C. is presently assured. ACC will be first lost if RPV level reaches -31.5 inches.

D. is presently assured. ACC will be first lost if RPV level reaches -44.5 inches.

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

Secondary Containment High Differential Pressure

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: #26

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 is 3×10^3 cpm and slowly rising.
- RCIC Drywell entrance area temperature is 140°F and slowly rising.
- Reactor Building differential pressure is 0" H_20 and stable.

Note: A portion of EOP-5, Secondary Containment Control, is provided on the following page

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

	EOP-5	EOP-6
A.	3	1
B.	3	0
C.	2	1
D.	2	0

SC-1	**REACTOR BUILDING AREA TEMPERATURES**						
AREA	**INSTRUMENT**	**MAXIMUM NORMAL**	**MAXIMUM SAFE**	**AREA**	**INSTRUMENT**	**MAXIMUM NORMAL**	**MAXIMUM SAFE**
Reactor Building 369 ft 66RTD-106 66RTD-108	Panel 09-75 66TI-106 66TI-108	**104 °F**	**112 °F**	Reactor Building 272 SE 23RTD-02C 23RTD-02D	23-204A, (09-95) 23-204B, (09-96)	**104 °F**	**153 °F**
Outside "A" LPCI Battery room 66RTD-115	EPIC only A-1559	**104 °F**	**113 °F**	HPCI Drywell Entrance 13RTD-102C 13RTD-102D	13-202C, (09-95) 13-202D, (09-96)	**120 °F**	**251 °F**
Below refuel floor exhaust 66RTD-105	Panel 09-75 66TI-105	**104 °F**	**113 °F**	RCIC Drywell Entrance 13RTD-102A 13RTD-107B	13-202A, (09-95) 13-207B, (09-96)	**120 °F**	**218 °F**
Outside "B" LPCI Battery room 66RTD-116	EPIC only A-1560	**104 °F**	**113 °F**	Reactor Building 272 SW 23RTD-01C 23RTD-01D	23-202A, (09-95) 23-202B, (09-96)	**104 °F**	**196 °F**
SLC Pump Area 66RTD-114	EPIC only ❶ A-1558	**104 °F**	**133 °F**	"A" RHR Heat Exchanger room 23RTD-01A 23RTD-01B	23-201A, (09-95) 23-201B, (09-96)	**130 °F**	**242 °F**
Fuel Pool Cooling pump room 66RTD-113	EPIC only A-1557	**104 °F**	**133 °F**	Torus Room – South HPCI Steamline 13RTD-107C 13RTD-107D	13-207C, (09-95) 13-207D, (09-96)	**120 °F**	**280 °F**
Reactor Building 300 ft NE 66RTD-112	EPIC only ❶ A-1556	**104 °F**	**158 °F**	Torus Room – South RCIC Steamline 13RTD-107A 13RTD-102B	13-207A, (09-95) 13-202B, (09-96)	**120 °F**	**280 °F**
RWCU Heat Exchanger room 12TE-117E 12TE-117F	Panel 09-21 2F-S2, Pos. 3 2F-S1, Pos. 3	**115 °F**	**203 °F**	East Crescent 66RTD-109B	Panel 09-75 66TI-109B	**104 °F**	**137 °F**
"B" RWCU pump room 12TE-117C 12TE-117D	Panel 09-21 2F-S2, Pos. 2 2F-S1, Pos. 2	**135 °F**	**225 °F**	HPCI Room 23RTD-94A 23RTD-94B 23RTD-117A 23RTD-117B	23-294A, (09-95) 23-294B, (09-96) 23-217A, (09-95) 23-217B, (09-96)	**104 °F**	**137 °F**
"B" RWCU pump room 12TE-117A 12TE-117B	Panel 09-21 2F-S2, Pos. 1 2F-S1, Pos. 1	**125 °F**	**225 °F**				
Reactor Building 300 ft SW 66RTD-111	EPIC only ❶ A-1555	**104 °F**	**173 °F**	RCIC Room 13RTD-89A 13RTD-89B	13-289A, (09-95) 13-289B, (09-96)	**104 °F**	**137 °F**
"B" RHR Heat Exchanger room 23RTD-02A 23RTD-02B	23-203A, (09-95) 23-203B, (09-96)	**130 °F**	**242 °F**	West Crescent 13RTD-76A 13RTD-76B	13-276A, (09-95) 13-276B, (09-96)	**104 °F**	**137 °F**

Proposed Answer: B

Explanation: EOP-5 entry is required due to Reactor Building ventilation exhaust radiation (>$1x10^3$ cpm), RCIC Drywell entrance area temperature (>120ºF), and Reactor Building differential pressure (at or above 0" $H_2$0). EOP-6 entry is not required, because although Reactor Building exhaust radiation is elevated, there is no indication provided that there is an off-site release rate above the alert level.

A. Incorrect – No EOP-6 entry condition is met by the given conditions. Plausible because there is elevated Reactor Building radiation and a loss of negative pressure, which will result in some elevated release.
C. Incorrect – Three EOP-5 entry conditions are met. Plausible if any of the three given parameters are judged to not be severe enough to require EOP entry. No EOP-6 entry condition is met by the given conditions. Plausible because there is elevated Reactor Building radiation and a loss of negative pressure, which will result in some elevated release.
D. Incorrect – Three EOP-5 entry conditions are met. Plausible if any of the three given parameters are judged to not be severe enough to require EOP entry.

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: Bank – 14-1 NRC #49

Question History: 14-1 NRC #49

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 #	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: #27

Given the following conditions:

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

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

A. 1 and 2, only

B. 1 and 3, only

C. 2 and 3, only

D. 1, 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 1.12

Question Source: Bank – 16-1 NRC #62

Question History: 16-1 NRC #62

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K1.17
	Importance Rating	4.0

RHR/LPCI: Injection Mode

Knowledge of the physical connections and/or cause/effect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following: Reactor pressure

Proposed Question: #28

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

- Reactor water level is 70” and slowly lowering.
- Reactor pressure is 445 psig and slowly lowering.
- Drywell pressure is 7.5 psig and slowly rising.

Which one of the following describes the status of RHR?

RHR…

A. is currently injecting to the Reactor.

B. remains in a standby lineup until Reactor water level reaches 59.5”.

C. pumps have automatically started, but will NOT inject until Reactor water level reaches 59.5”.

D. pumps have automatically started, but will NOT inject until Reactor pressure lowers further.

Proposed Answer: D

Explanation: Reactor water level has not yet reached the LPCI initiation setpoint of 59.5”. However, LPCI also initiates on high Drywell pressure (>2.7 psig). Therefore, RHR pumps have started. With Reactor pressure <450 psig, the LPCI injection valves have opened. RHR is not yet injecting to the Reactor because Reactor pressure is above the shutoff head of the pumps. Once Reactor pressure lowers below approximately 215 psig, RHR will begin to inject, regardless of current Reactor water level.

A. Incorrect – RHR is not injecting yet. Plausible because pumps are running and injection valves are open.
B. Incorrect – RHR pumps are running and injection valves are open. Plausible because Reactor water level is >59.5”.
C. Incorrect – RHR will inject when Reactor pressure lowers regardless of Reactor water level. Plausible because Reactor water level is >59.5”, which is another LPCI initiation signal.

Technical Reference(s): OP-13, ARP 09-3-1-27, Simulator response

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.05.a.1.a(c)

Question Source: Bank - SSES LOC26R NRC #17 (2015)

Question History: SSES LOC26R NRC #17 (2015)

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 #	205000 A1.01
	Importance Rating	3.3

Shutdown Cooling

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

Proposed Question: #29

A plant cooldown is in progress with the following:

- RHR loop B is operating in Shutdown Cooling mode.
- RHRSW pump B is running.
- RHRSW pump D is secured.
- 10MOV-89B, RHRSW DISCH VLV FROM HX B, is throttled open.
- RHRSW loop B flow is 2500 gpm.
- The Shift Manager has directed raising the cooldown rate.

Which one of the following describes:

(1) the ability to raise the cooldown rate by raising RHRSW pump B flow rate

and

(2) the ability to raise the cooldown rate by starting RHRSW pump D,

in accordance with OP-13D, RHR – Shutdown Cooling?

	Able to Raise RHRSW Pump B Flow?	Able to Start RHRSW Pump D?
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: D

Explanation: The given indications show RHRSW flow is not maximized (up to 4000 gpm per pump is allowed, only currently at 2500 gpm on one pump). Therefore, the cooldown rate may be raised by further opening of 10MOV-89B to raise RHRSW pump B flow. Additionally, the cooldown rate may be raised by starting RHRSW pump D.

A. Incorrect – The cooldown rate may be raised by further opening of 10MOV-89B to raise RHRSW pump B flow. Plausible because RHRSW pump B flow is in the allowable band, just low. RHRSW pump D is allowed to be started. Plausible that only one pump would be allowed to be operated during non-emergency conditions.
B. Incorrect – The cooldown rate may be raised by further opening of 10MOV-89B to raise RHRSW pump B flow. Plausible because RHRSW pump B flow is in the allowable band, just low.
C. Incorrect – RHRSW pump D is allowed to be started. Plausible that only one pump would be allowed to be operated during non-emergency conditions.

Technical Reference(s): OP-13D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13.d

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

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

Shutdown Cooling

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

Proposed Question: #30

The plant is shutdown with the following:

- RHR loop B is operating in Shutdown Cooling mode.
- Reactor coolant temperature is 120ºF and slowly lowering.

Then, a significant leak from RHR pump B suction flange results in the following:

- Reactor water level is 210" and lowering slowly.

If Reactor water level continues to lower, which one of the following identifies the Reactor water level at which an isolation signal will ***first*** be generated that will isolate the leak?

A. 196.5"

B. 177.0"

C. 126.5"

D. 59.5"

Proposed Answer: B

Explanation: The first isolation signal occurs at 177", which causes a SDC isolation (10MOV-17, -18, and -25B).

Note: The question meets the K/A because it presents a malfunction of SDC (system leak) and requires the applicant to determine the effect on Reactor water level (magnitude of level drop before an automatic isolation signal is generated).

A. Incorrect – The first isolation signal occurs at 177". Plausible because 196.5" is the setpoint for a Recirc runback.
C. Incorrect – The first isolation signal occurs at 177". Plausible because 126.5" is the setpoint for HPCI and RCIC start.
D. Incorrect – The first isolation signal occurs at 177". Plausible because 59.5" is the setpoint for LPCI initiation and Group I isolations.

Technical Reference(s): AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10.h

Question Source: Bank - NMP1 2018 NRC #6

Question History: NMP1 2018 NRC #6

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A1.06
	Importance Rating	3.8

High Pressure Coolant Injection

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: System flow

Proposed Question: #31

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

- ST-4N, HPCI Quick-Start, Inservice, and Transient Monitoring Test, is in progress.
- HPCI is running with its discharge aligned to the CSTs.
- HPCI flow and flow controller conditions are as shown in the pictures on the following page.

Then, the following occur:

- The Reactor scrams.
- Reactor water level lowers to a low of 120 inches.

Which one of the following describes the resulting operation of HPCI?

HPCI discharge flow…

A. continues to be aligned to the CSTs and is controlled by the manual demand signal.

B. automatically re-aligns to the Reactor and is controlled by the manual demand signal.

C. continues to be aligned to the CSTs and automatically controls at approximately 4250 gpm.

D. automatically re-aligns to the Reactor and automatically controls at approximately 4250 gpm.


5000
4250
4000
3000
2000
1000
0
GPM
CLOSE
OPEN
AUTO
BAL
MAN
0-100
0-100
HPCI FLOW CNTRL
23FI-108-1
23FIC-108-1
71DCB2


TO
CST
HPCI & RCIC TEST VLV
TO CST
23MOV-24
DC
CLOSE
OPEN
FROM
RCIC
TEST VLV TO CST
23MOV-21
DC
CLOSE
AUTO
OPEN


PMP DISCH VLV
23MOV-20
DC
CLOSE
OPEN
HPCI
PMP
INJ VLV
23MOV-19
DC
CLOSE
AUTO
OPEN
TO
B FDWTR

Proposed Answer: B

Explanation: The initial conditions show HPCI running with approximately 3400 gpm to the CSTs in manual flow control mode. In this alignment, 23MOV-21 and 23MOV-24 are open to allow HPCI discharge flow to go to the CSTs and 23MOV-19 is closed to block HPCI discharge flow from going to the Reactor. When Reactor water level lowers below 126.5 inches, HPCI receives an automatic initiation signal. This causes 23MOV-21 and 23MOV-24 to close and 23MOV-19 to open. This shifts all HPCI flow from the CSTs to the Reactor. However, the HPCI flow controller does NOT automatically shift into automatic on the low Reactor water level signal, therefore HPCI flow control remains in manual.

A. Incorrect – The HPCI test line isolates and the discharge to the Reactor automatically opens. Plausible because this is the initial alignment and it does stay in manual.
C. Incorrect – The HPCI test line isolates and the discharge to the Reactor automatically opens. HPCI continues to operate in manual flow control mode unless operator action is taken. Plausible because the initial lineup is to the CSTs. Also plausible that the automatic initiation signal would ensure design flow.
D. Incorrect – HPCI continues to operate in manual flow control mode unless operator action is taken. Plausible that the automatic initiation signal would ensure design flow.

Technical Reference(s): ST-4N, OP-15, SDLP-23

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.c.4 and 1.05.c.6

Question Source: Bank - 14-2 NRC #17

Question History: 14-2 NRC #17

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 #	209001 K5.04
	Importance Rating	2.8

Low Pressure Core Spray

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

Proposed Question: #32

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

- Annunciator 09-3-1-1, CORE SPRAY HDR A PIPE BREAK DETECTOR ALARM, alarms.
- The alarm is determined to be due to a valid Core Spray line break.

Which one of the following identifies the location of the Core Spray line break?

A. Between the pump and the injection valve (14MOV-11A).

B. Between the injection valve (14MOV-11A) and the injection line check valve (14AOV-13A).

C. Between the injection line check valve (14AOV-13A) and the core shroud.

D. Inside the core shroud.

Proposed Answer: C

Explanation: This alarm is caused by a change in pressure due to a leak between the injection line check valve, 14AOV-13A, and the core shroud.

Note: The question meets the K/A because one aspect of the Core Spray heat removal mechanism is injection through the sparger inside the Core Shroud. The question tests an operational implication of this concept through the alarm and indications of a break that would cause injection to not be routed through the sparger, thereby affecting Core Spray's ability to properly remove heat from the fuel. The question is not testing the adequate core cooling concept of Spray Cooling to avoid overlap with Question #90.

A. Incorrect – This alarm would not detect a leak in this location because both the injection valve and the injection line check valve prevent a pressure change here from affecting the leak detection instrument. Plausible because this leak location would disrupt Core Spray flow in the same way as a leak in the correct location.
B. Incorrect – This alarm would not detect a leak in this location because the injection line check valve prevents a pressure change here from affecting the leak detection instrument. Plausible because this leak location would disrupt Core Spray flow in the same way as a leak in the correct location.
D. Incorrect – This alarm would not detect a leak in this location because there would be no pressure change (this section of piping is already pressurized to the same pressure as surrounding environment). Plausible because this section of piping is directly connected to the piping that is monitored by the leak detection system.

Technical Reference(s): ARP 09-3-1-1, FM-23A, SDLP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.05.a.13

Question Source: Bank – NMP2 2017 Cert #1

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 #	211000 A3.05
	Importance Rating	4.1

Standby Liquid Control

Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Flow indication: Plant-Specific

Proposed Question: #33

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

- Reactor power is 15%.
- SRVs are cycling to control Reactor pressure.
- The Standby Liquid Control (SLC) keylock switch is taken to START SYS A.

Which one of the following describes an indication of proper SLC flow to the Reactor?

A. SLC pump discharge pressure of 1175 psig.

B. SLC pump discharge pressure of 1500 psig.

C. Squib circuit continuity meter reading 4 mA.

D. Squib circuit continuity meter reading 2 A.

Proposed Answer: A

Explanation: The SRVs will control Reactor pressure ~1135 psig. With SLC pump discharge pressure of 1175 psig, there is evidence of proper flow to the Reactor.

Note: While there is no direct flow indication in the Control Room, Operators use discharge pressure and squib indications to determine the status of flow. The question meets the K/A by testing this knowledge of discharge pressure and squib indications to determine the status of flow.

B. Incorrect – A pressure of 1500 psig indicates that the explosive valves failed to fire and flow is through the relief valves back to the SLC tank. Plausible because SLC pumps are positive displacement pumps that are capable of producing this pressure and this pressure is above Reactor pressure.
C. Incorrect – This is not indication of proper squib firing, and therefore not proper indication of flow to the Reactor. Plausible because this is the normal current through the squibs.
D. Incorrect – This is not indication of proper squib firing, and therefore not proper indication of flow to the Reactor. Plausible because this is the approximate firing current for the squibs, but should go to 0 A upon successful firing.

Technical Reference(s): OP-17, SDLP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11 1.05.c.1

Question Source: Bank - NMP1 2010 NRC #19

Question History: NMP1 2010 NRC #19

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 #	212000 K4.02
	Importance Rating	3.5

Reactor Protection

Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: The prevention of a reactor SCRAM following a single component failure

Proposed Question: #34

The plant is operating at 40% power when Turbine Stop Valve (TSV) 3 drifts fully closed.

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

Based on the Turbine Stop Valve Closure scram signal, …

A. NO half scram signals occur.

B. a half scram occurs on RPS A, only.

C. a half scram occurs on RPS B, only.

D. a full scram occurs.

Proposed Answer: A

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 TSVs cause a half scram, but 3 TSVs must close to cause an actual Reactor scram. In this case, TSV 3 has drifted far enough to trip its scram position switches, but no half scram will occur without another TSV drifting.

B. Incorrect – No half scram is received. Plausible because TSV 3 has drifted far enough to trip its scram position switches.
C. Incorrect – No half scram is received. Plausible because TSV 3 has drifted far enough to trip its scram position switches.
D. Incorrect – A full scram does not occur. Plausible because TSV 3 has drifted far enough to trip its scram position switches.

Technical Reference(s): SDLP-05

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.09.f

Question Source: Modified Bank – 16-1 NRC #10

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K2.01
	Importance Rating	2.5

Intermediate Range Monitor

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

Proposed Question: #35

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

A. 24 VDC Battery Bus

B. 125 VDC Battery Bus

C. 120 VAC UPS Bus

D. 120 VAC Instrument Bus

Proposed Answer: A

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

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

Technical Reference(s): OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.04.a

Question Source: Bank - 17-1 NRC #26

Question History: 17-1 NRC #26

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A4.01
	Importance Rating	3.3

Intermediate Range Monitor

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

Proposed Question: #36

A plant startup is in progress with the following:

- The Reactor Mode Switch is in ***START & HOT STBY***.
- IRM A indicates 85 on Range 6.

Which one of the following describes if IRM A is causing a half scram and the required manipulation of the IRM A range switch, in accordance with OP-65, Plant Startup and Shutdown?

IRM A is…

A. causing a half scram. Place IRM A on Range 5.

B. causing a half scram. Place IRM A on Range 7.

C. NOT causing a half scram. Place IRM A on Range 5.

D. NOT causing a half scram. Place IRM A on Range 7.

Proposed Answer: D

Explanation: IRM A is not causing a half scram because it is indicating <118.5/125. IRM A indication must be lowered back to the range of 5-75 per OP-65. This is accomplished by going to Range 7.

A. Incorrect – IRM A is below the scram setpoint of 118.5. Plausible because IRM A is above the range of 5-75, and right at the upscale rod block setpoint. IRM A must be taken to Range 7, not 5. Plausible because this is an easily confused concept.
B. Incorrect – IRM A is below the scram setpoint of 118.5. Plausible because IRM A is above the range of 5-75, and right at the upscale rod block setpoint.
C. Incorrect – IRM A must be taken to Range 7, not 5. Plausible because this is an easily confused concept.

Technical Reference(s): ARP 09-5-2-52, OP-65

Proposed references to be provided to applicants during examination: None

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

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 #	215004 A3.04
	Importance Rating	3.6

Source Range Monitor

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

Proposed Question: #37

A plant startup is in progress with the following:

- The Reactor Mode Switch is in ***START & HOT STBY***.
- All IRMs are on Range 1.
- All SRMs are partially withdrawn.
- A malfunction of the SRM drive control circuitry results in the following indications:
    - SRM A: 60 cps
    - SRM B: 130 cps
    - SRM C: 7×10^3 cps
    - SRM D: 3×10^4 cps

Which one of the following describes the status of the SRM rod block?

The SRM rod block is…

A. NOT received.

B. received due to an SRM Hi Flux signal.

C. received due to an SRM Downscale signal.

D. received due to an SRM Detector Not Full In signal.

Proposed Answer: D

Explanation: The SRM Detector Not Full In rod block occurs when an SRM is not fully inserted with < 100 cps, associated IRMs on range 1 or 2, and the Mode Switch is not in RUN. This prevents control rod withdrawal with RMCS under the given conditions due to SRM A.

A. Incorrect – A control rod withdrawal block is active due to SRM A being <100 cps and partially withdrawn. Plausible because this would be correct if SRM A were fully inserted or >100 cps.
B. Incorrect – SRM Hi Flux rod block occurs with SRM counts above 10^5 cps. All SRMs are less than 10^5 cps. Plausible because SRMs B and D are indicating fairly high.
C. Incorrect – SRM Downscale rod block occurs with SRM counts at 3 cps or less. All SRMs are above 3 cps. Plausible because SRM A is causing a rod block because counts are too low for its position.

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

Question Source: Bank - 14-1 NRC #5

Question History: 14-1 NRC #5

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 #	215005 2.1.28
	Importance Rating	4.1

Average Power Range Monitor/Local Power Range Monitor

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

Proposed Question: #38

Which one of the following describes the way LPRMs are assigned to the APRMs?

Each APRM receives inputs from LPRMs in ___(1)___ core quadrants and from ___(2)___ detector levels.

	(1)	(2)
A.	only 2	only 2
B.	only 2	all 4
C.	all 4	only 2
D.	all 4	all 4

Proposed Answer: D

Explanation: Each APRM receives LPRM inputs from all 4 core quadrants and all 4 detector levels.

A. Incorrect – Each APRM receives LPRM inputs from all 4 core quadrants and all 4 detector levels. Plausible because this would still provide averaging of power in different parts of the core and would make APRMs better at sensing uneven flux distribution.
B. Incorrect – Each APRM receives LPRM inputs from all 4 core quadrants. Plausible because this would still provide averaging of power in different parts of the core and would make APRMs better at sensing uneven flux distribution.
C. Incorrect – Each APRM receives LPRM inputs from all 4 detector levels. Plausible because this would still provide averaging of power in different parts of the core and would make APRMs better at sensing uneven flux distribution.

Technical Reference(s): RAP 7.4.03, OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.05.a.3.a

Question Source: Bank - SSES LOC27 NRC #7 (2015)

Question History: SSES LOC27 NRC #7 (2015)

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A2.08
	Importance Rating	3.0

Reactor Core Isolation Cooling

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 cooling

Proposed Question: #39

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

- RCIC is being started for surveillance testing.
- 13MOV-132, OIL CLR WTR SUPP, fails closed.
- The following annunciators alarm:
    - 09-4-1-5, RCIC TURB CPLG END BRG TEMP HI
    - 09-4-1-15, RCIC TURB GOV END BRG TEMP HI
- The associated RCIC turbine bearing temperatures are 200ºF and rising slowly.

Which one of the following describes:

(1) the cooling water source that is blocked by the failure of 13MOV-132, and

(2) the required action, in accordance with the Alarm Response Procedures?

A. (1) RBCLC
(2) Manually trip RCIC

B. (1) RBCLC
(2) Verify automatic RCIC trip

C. (1) RCIC Pump Discharge
(2) Manually trip RCIC

D. (1) RCIC Pump Discharge
(2) Verify automatic RCIC trip

Proposed Answer: C

Explanation: 13MOV-132 supplies cooling water from the RCIC pump discharge (~16 gpm) to the RCIC oil cooler. RCIC bearing high temperatures result in control room alarms, but do not result in an automatic trip of RCIC. The associated alarm response procedures require a manual trip of RCIC.

A. Incorrect – 13MOV-132 supplies cooling water from the RCIC pump discharge. Plausible because RBCLC supplies cooling water to many important loads in the Reactor Building.
B. Incorrect – 13MOV-132 supplies cooling water from the RCIC pump discharge. Plausible because RBCLC supplies cooling water to many important loads in the Reactor Building. RCIC does not automatically trip due to high bearing temperatures, but must be manually tripped. Plausible because this does cause an alarm and RCIC has many automatic trips.
D. Incorrect – RCIC does not automatically trip due to high bearing temperatures, but must be manually tripped. Plausible because this does cause an alarm and RCIC has many automatic trips.

Technical Reference(s): ARPs 09-4-1-5, 09-4-1-15, OP-19, FM-22A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.05.a.4

Question Source: Modified Bank - 16-1 NRC #86

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

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.

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

Automatic Depressurization

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: #40

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. Plausible because there are almost enough SRVs available.
B. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 7 SRVs unavailable, only 4 remain available. Plausible because there are almost enough SRVs 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. Plausible because MSIVs are currently closed and enough SRVs are open to lower Reactor pressure fairly rapidly.
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. Plausible because MSIVs are currently closed and enough SRVs are open to lower Reactor pressure fairly rapidly.

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 - 16-1 NRC #6

Question History: 16-1 NRC #6

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 #	223002 K3.19
	Importance Rating	2.8

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Containment atmosphere sampling

Proposed Question: #41

The plant is operating at 100% power when a malfunction causes a PCIS Group II isolation signal on one Division II sub-channel.

Which one of the following describes the resulting status of the H2/O2 monitors?

A. Both H2/O2 monitors remain un-isolated.

B. Both H2/O2 monitors isolate due to closure of inboard isolation valves.

C. Both H2/O2 monitors isolate due to closure of outboard isolation valves.

D. One H2/O2 monitor isolates and one H2/O2 monitor remains un-isolated.

Proposed Answer: A

Explanation: PCIS Group II has a “2 out of 2 once” logic arrangement. For H2/O2 monitors, having both Division II sub-channels activate results in the “B” H2/O2 monitor isolating and the “A” H2/O2 monitor remaining in service. With only one Division II sub-channel activate, both H2/O2 monitors remain in service.

B. Incorrect – Both H2/O2 monitors remains un-isolated. Plausible because this would be correct for most PCIS Group II systems given two Division II failures.
C. Incorrect – Both H2/O2 monitors remains un-isolated. Plausible because this would be correct for most PCIS Group II systems given two Division I failures.
D. Incorrect – Both H2/O2 monitors remains un-isolated. Plausible because this would be correct for two Division II failures.

Technical Reference(s): SDLP-16C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 K6.05
	Importance Rating	3.0

Safety Relief Valves

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: #42

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.

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. Plausible because this is the answer that would be chosen if the vacuum breaker connected back to the Reactor, which would function to break vacuum in the SRV discharge piping.
B. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Reactor Building. Plausible because this is the answer that would be chosen if the vacuum breaker connected to the Reactor Building air space, which would function to break vacuum in the SRV discharge piping.
C. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Torus. Plausible because this is the answer that would be chosen if the vacuum breaker connected to the Torus air space, which would function to break vacuum in the SRV discharge piping.

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 – 16-1 NRC #26

Question History: 16-1 NRC #26

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 #	259002 K1.16
	Importance Rating	3.4

Reactor Water Level Control

Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: HPCI: Plant-Specific

Proposed Question: #43

Which one of the following identifies the Reactor water level instrumentation that is used for automatic initiation and tripping of HPCI?

A. Wide range level instruments are used for both of these functions.

B. Narrow range level instruments are used for both of these functions.

C. Wide range level instruments are used for automatic initiation and narrow range level instruments are used for the high level trip.

D. Narrow range level instruments are used for automatic initiation and wide range level instruments are used for the high level trip.

Proposed Answer: C

Explanation: HPCI automatic initiation uses wide range level transmitters 02-3LT-72A(B)(C)(D). HPCI high level trip uses narrow range level transmitters 02-3LT-83C(D).

A. Incorrect – HPCI high level trip uses narrow range level transmitters. Plausible because wide range is used for many other functions, including the HPCI automatic initiation.
B. Incorrect – HPCI automatic initiation uses wide range level transmitters. Plausible because narrow range is used for many other functions, including the HPCI high level trip.
D. Incorrect – HPCI automatic initiation uses wide range level transmitters. Plausible because narrow range is used for many other functions, including the HPCI high level trip. HPCI high level trip uses narrow range level transmitters. Plausible because wide range is used for many other functions, including the HPCI automatic initiation.

Technical Reference(s): SDLP-02B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02B 1.05.c

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 A3.01
	Importance Rating	3.2

Standby Gas Treatment

Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: System flow

Proposed Question: #44

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

- Annunciator 09-3-2-40, RX BLDG VENT RAD MON HI-HI, is received.
- Reactor Building Ventilation exhaust radiation monitors indicate:
  - 17RM-452A: 3×10^4 cpm
  - 17RM-452B: 5×10^3 cpm

Which one of the following describes the status of the Standby Gas Treatment (SGT) system?

A. Both trains of SGT remain in standby.

B. SGT train A auto-initiates. SGT train B remains in standby.

C. SGT train B auto-initiates. SGT train A remains in standby.

D. Both trains of SGT auto-initiate.

Proposed Answer: B

Explanation: 17RM-452A is above the hi-hi setpoint of 1 x 10^4 cpm and 17RM-452B is below the hi-hi setpoint. Each of these radiation monitors causes auto-initiation of their respective SGT train when above the hi-hi setpoint. Therefore, only SGT train A auto-initiates and SGT train B remains in standby.

A. Incorrect – SGT train A auto-initiates due to hi-hi radiation. Plausible because only a single instrument is above the hi-hi setpoint.
C. Incorrect – SGT train B remains in standby. SGT train A auto-initiates due to hi-hi radiation. Plausible that there would be a distinction between above and below refuel floor radiation monitors and which train automatically starts (the given monitors are below refuel floor).
D. Incorrect – SGT train B remains in standby. Plausible because a single instrument above hi-hi does cause an initiation.

Technical Reference(s): ARP 09-3-2-40, OP-20, AOP-15, SDLP-01B

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank - 14-1 NRC #7

Question History: 14-1 NRC #7

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 #	261000 2.1.30
	Importance Rating	4.4

Standby Gas Treatment

Ability to locate and operate components, including local controls.

Proposed Question: #45

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

- Standby Gas Treatment (SBGT) train A is in service.
- Annunciator 09-75-1-16, SGT SYS A ACT CHAR TEMP HI, alarms.
- An operator in the field has confirmed that there is an active fire in the SBGT train A charcoal filter.
- Initiation of fire suppression from the Fire Protection Panel has failed.
- The Unit Supervisor has directed initiation of fire suppression to the SBGT train A charcoal filter at the local breakglass station.

Which one of the following describes the method of fire suppression that is to be initiated and the location of the local breakglass station?

	Method of Fire Suppression	Location of Local Breakglass Station
A.	Water	Reactor Building el. 272’
B.	Water	Reactor Building el. 300’
C.	CO_2	Reactor Building el. 272’
D.	CO_2	Reactor Building el. 300’

Proposed Answer: A

Explanation: The method of fire suppression to the SBGT train A charcoal filter is water spray. The local breakglass station is located on the south wall of Reactor Building el. 272’.

B. Incorrect – The local breakglass station is located on the south wall of Reactor Building el. 272’. Plausible because this is a valid Reactor Building elevation that does contain SBGT-related components (flow instrumentation).
C. Incorrect – The method of fire suppression to the SBGT train A charcoal filter is water spray. Plausible because introduction of water to a charcoal filter can initiate an undesirable exothermic reaction.
D. Incorrect – The method of fire suppression to the SBGT train A charcoal filter is water spray. Plausible because introduction of water to a charcoal filter can initiate an undesirable exothermic reaction. The local breakglass station is located on the south wall of Reactor Building el. 272’. Plausible because this is a valid Reactor Building elevation that does contain SBGT-related components (flow instrumentation).

Technical Reference(s): OP-20, SDLP-01B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B 1.08.c

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.01
	Importance Rating	3.4

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: All breakers and disconnects (including available switch yard): Plant-Specific

Proposed Question: #46

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

- The Main Generator is connected to the grid through both output breakers.
- Then, a Main Turbine trip occurs.

Which one of the following describes the resulting status of the Main Generator output breakers, 71PCB-10042 and 71PCB-10052, and disconnect 71MOD-10031 three (3) minutes later?

	Output Breakers 71PCB-10042 and 71PCB-10052	Disconnect 71MOD-10031
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: A

Explanation: Both Main Generator output breakers automatically open on the Main Turbine trip. Both output breakers remain open until manual action is taken to reclose them. Main Generator disconnect 71MOD-10031 also automatically opens on the Main Turbine trip 71MOD-10031 also remains open unless manual action is taken.

B. Incorrect – Main Generator disconnect 71MOD-10031 automatically opens on the Main Turbine trip. Plausible that this would remain closed until manual action is taken, as the Main Generator output breakers open and remain open on the Main Turbine trip.
C. Incorrect – Both Main Generator output breakers automatically open on the Main Turbine trip and remain open until manual action is taken to reclose them. Plausible that they would automatically re-close after disconnect(s) open to re-establish the preferred 345KV electrical distribution system alignment. OP-11A does contain manual actions to reclose these breakers for this reason. Main Generator disconnect 71MOD-10031 automatically opens on the Main Turbine trip. Plausible that this would remain closed until manual action is taken, as the Main Generator output breakers open and remain open on the Main Turbine trip.
D. Incorrect – Both Main Generator output breakers automatically open on the Main Turbine trip and remain open until manual action is taken to reclose them. Plausible that they would automatically re-close after disconnect(s) open to re-establish the preferred 345KV electrical distribution system alignment. OP-11A does contain manual actions to reclose these breakers for this reason.

Technical Reference(s): AOP-2, OP-11A Section F

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71C 1.05.b.1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 A2.02
	Importance Rating	2.5

Uninterruptable Power Supply (AC/DC)

Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Over voltage

Proposed Question: #47

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

- A malfunction occurs in the RPS MG Set B voltage regulator.
- Annunciator 09-5-1-15, RPS MG B TROUBLE, alarms.
- Output voltage from RPS MG Set B rises to 150 VAC.
- Voltage stays at this value for 10 seconds and then lowers back to normal.

Which one of the following describes the impact of this voltage transient on RPS and the need to enter AOP-60, Loss of RPS Bus B Power?

	Impact on RPS	AOP-60 Entry
A.	EPAs trip	Required
B.	EPAs trip	NOT required
C.	Transfers to alternate power	Required
D.	Transfers to alternate power	NOT required

Proposed Answer: A

Explanation: With RPS MG Set output voltage >132 VAC for >4 seconds, the overvoltage trip of the EPAs occurs. This results in loss of power to RPS B, which requires entry into AOP-60.

B. Incorrect – AOP-60 entry is required. Plausible because MG Set output voltage quickly returns to normal.
C. Incorrect – EPAs trip. Plausible because RPS does have an alternate power source capable of powering the buses.
D. Incorrect – EPAs trip. Plausible because RPS does have an alternate power source capable of powering the buses. AOP-60 entry is required. Plausible because MG Set output voltage quickly returns to normal.

Technical Reference(s): ARP 09-5-1-15, AOP-60

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.14.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 #	2
	Group #	1
	K/A #	263000 K2.01
	Importance Rating	3.1

DC Electrical Distribution

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

Proposed Question: #48

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 loads automatically transfers to DC Power System B?

A. ADS A Control Power

B. HPCI A Control Power

C. EDG A and C Control Power

D. Core Spray A Control Power

Proposed Answer: A

Explanation: On a loss of DC Power System A, ADS A control power automatically transfers to DC Power System B.

B. Incorrect – HPCI A control power does not transfer to DC Power System B. Plausible because DC Power System A is the power supply, some loads do automatically transfer, and HPCI is an important load.
C. Incorrect – EDG A and C control power does not transfer to DC Power System B. Plausible because DC Power System A is the power supply, some loads do automatically transfer, and EDGs A and C are important loads.
D. Incorrect – Core Spray A control power does not transfer to DC Power System B. Plausible because DC Power System A is the power supply, some loads do automatically transfer, and Core Spray A is an important load.

Technical Reference(s): AOP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.09.a.1, 3, 6, and 13

Question Source: Modified Bank – 14-1 NRC #23

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

DC Electrical Distribution

Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights

Proposed Question: #23

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

- 09-8-1-22, 125VDC BATT CHGR B AC SUPP TROUBLE
- 09-8-1-23, 125VDC BATT B VOLT LO

125 VDC BATT BUS B meter on Panel 09-8 indicates 0 VDC.

Which of the following automatic operations occurs as a result of these conditions?

(1) ADS Logic B transfers to DC Power System A
(2) RCIC Logic B transfers to DC Power System A
(3) HPCI Logic B transfers to DC Power System A

A. (1) only

B. (2) only

C. (3) only

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

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K6.08
	Importance Rating	3.6

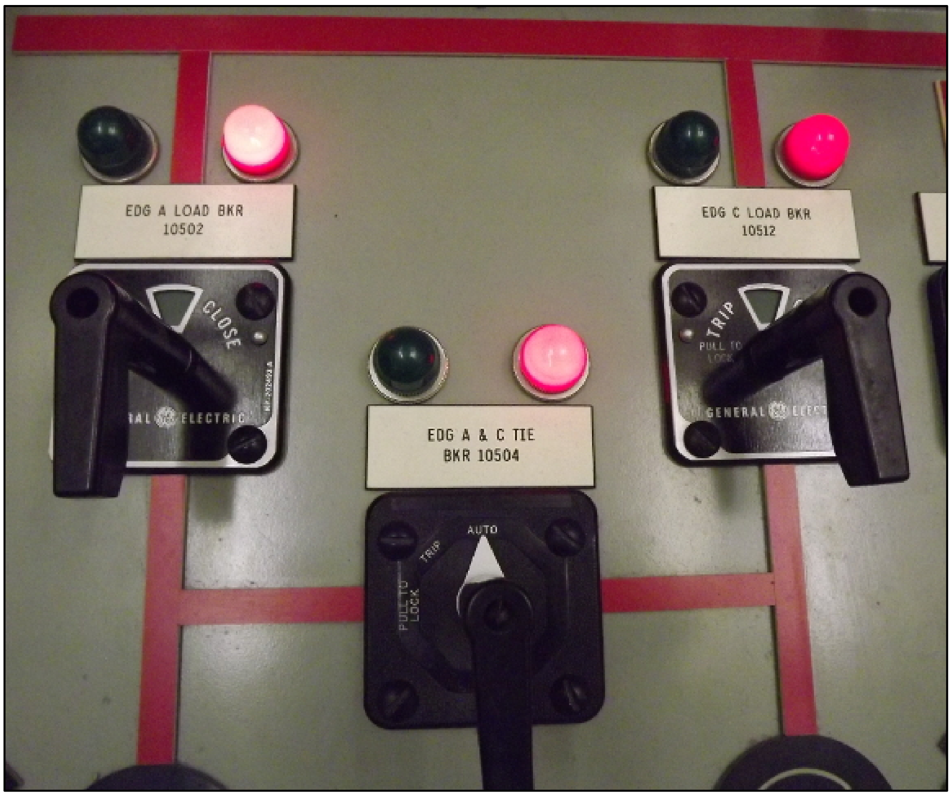
Emergency Generators (Diesel/Jet) EDG

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

Proposed Question: #49

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

- Bus 10300 Normal Supply Breaker 10302 trips.
- Bus 10300 Reserve Supply Breaker 10312 is open and remains open.
- The following picture shows the status of Emergency Diesel Generator (EDG) A and C output and tie breakers **one minute** later:



Which one of the following describes the status of these breakers?

A. All three breakers are in the correct position.

B. The output breakers are in the correct position, but the tie breaker is NOT in the correct position.

C. The tie breaker is in the correct position, but the output breakers are NOT in the correct position.

D. NONE of the breakers are in the correct position.

Proposed Answer: B

Explanation: When breaker 10302 trips with no fault on the 10300 bus, breaker 10312 should close to re-energize the 10300 bus. Since 10312 is open and remains open, the 10300 bus de-energizes. This also causes the 10500 bus to de-energize. Undervoltage on the 10500 bus should cause EDGs A and C to start, their output breakers to close, and their tie breaker to open. The picture shows the output breakers closed as expected, but the tie breaker failed to open as expected.

A. Incorrect – The tie breaker should be open because the EDGs started due to undervoltage. Plausible because the tie breaker is normally closed in the standby lineup.
C. Incorrect – The output breakers should be closed because stem conditions resulted in undervoltage on the 10500 bus. The tie breaker should be open because the EDGs started due to undervoltage. Plausible because this would be correct for a LOCA start of the EDGs without undervoltage.
D. Incorrect – The output breakers should be closed because stem conditions resulted in undervoltage on the 10500 bus. Plausible that the trip of 10302 (rather than undervoltage on the bus alone) would prevent automatic EDG loading.

Technical Reference(s): OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.05.b.3

Question Source: Bank - 14-2 NRC #48

Question History: 14-2 NRC #48

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 #	264000 K5.05
	Importance Rating	3.4

Emergency Generators (Diesel/Jet) EDG

Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Paralleling A.C. power sources

Proposed Question: #50

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

- The regular monthly start and load test is being performed for EDGs A and C per ST-9BA, EDG A and C Full Load Test and ESW Pump Operability Test.

Which one of the following describes how EDGs A and C and the 10500 bus are operated during the loaded portion of this test?

EDG A and C are operated with their governors in…

A DROOP mode. The 10500 bus is disconnected from the 10300 bus.

B. DROOP mode. The 10500 bus remains connected to the 10300 bus.

C. NORMAL mode. The 10500 bus is disconnected from the 10300 bus.

D. NORMAL mode. The 10500 bus remains connected to the 10300 bus.

Proposed Answer: B

Explanation: Before the EDGs are paralleled to the 10500 bus, their governors are placed in DROOP mode to ensure proper load sharing. During the loaded portion of the test, the 10500 bus remains connected to the 10300 bus.

A. Incorrect – During the loaded portion of the test, the 10500 bus remains connected to the 10300 bus. Plausible because the 10300 and 10500 buses disconnect when the EDGs auto-start on UV.
C. Incorrect – Before the EDGs are paralleled to the 10500 bus, their governors are placed in DROOP mode to ensure proper load sharing. Plausible because standby conditions have the EDGs in NORMAL. During the loaded portion of the test, the 10500 bus remains connected to the 10300 bus. Plausible because the 10300 and 10500 buses disconnect when the EDGs auto-start on UV.
D. Incorrect – Before the EDGs are paralleled to the 10500 bus, their governors are placed in DROOP mode to ensure proper load sharing. Plausible because standby conditions have the EDGs in NORMAL.

Technical Reference(s): ST-9BA

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.15.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 #	2
	Group #	1
	K/A #	300000 K4.03
	Importance Rating	2.8

Instrument Air

Knowledge of INSTRUMENT AIR SYSTEM design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water

Proposed Question: #51

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

- Air Compressor B is running.
- Then, cooling water flow to Air Compressor B is lost.

Which one of the following describes the system that supplies Air Compressor B with cooling water and the signal that will trip Air Compressor B?

	Cooling Water System	Trip Signal
A.	RBCLC	Low cooling water flow
B.	RBCLC	High air or oil temperature
C.	TBCLC	Low cooling water flow
D.	TBCLC	High air or oil temperature

Proposed Answer: D

Explanation: Air Compressor B is supplied cooling water from TBCLC. Air Compressor B has trips on high air and oil temperatures, which will result from a loss of TBCLC flow, but not a direct trip on low TBCLC flow.

A. Incorrect – Air Compressor B is supplied cooling water from TBCLC. Plausible because Air Compressor B supplies air to many components in the Reactor Building. Air Compressor B has trips on high air and oil temperatures, but not a direct trip on low TBCLC flow. Plausible because flow automatically initiates on compressor start, and loss of flow will lead to a trip.
B. Incorrect – Air Compressor B is supplied cooling water from TBCLC. Plausible because Air Compressor B supplies air to many components in the Reactor Building.
C. Incorrect – Air Compressor B has trips on high air and oil temperatures, but not a direct trip on low TBCLC flow. Plausible because flow automatically initiates on compressor start, and loss of flow will lead to a trip.

Technical Reference(s): OP-39, ARP 09-6-2-18

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.c.3

Question Source: Bank – NMP1 2013 Cert #52

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 2.4.49
	Importance Rating	4.6

Instrument Air

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question: #52

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

- An air leak develops in the Reactor Building.
- Operators have located the leak and are attempting to isolate it.
- Instrument Air header pressure is 95 psig and stable.
- Scram Air header pressure is 58 psig and stable.
- Annunciator 09-5-2-3, ROD DRIFT, is in alarm.
- One control rod has drifted in from position 12 to 00.
- NO other control rods have drifted.

Which one of the following describes the current need for a manual Reactor scram, in accordance with AOP-12, Loss of Instrument Air, and AOP 27, Control Rod Drift?

A. Both of these AOPs currently require a manual Reactor scram.

B. AOP-12 currently requires a manual Reactor scram, but AOP-27 does NOT.

C. AOP-27 currently requires a manual Reactor scram, but AOP-12 does NOT.

D. NEITHER of these AOPs currently requires a manual Reactor scram.

Proposed Answer: B

Explanation: AOP-12 entry is required due to lowering air pressures. AOP-27 entry is required due to the control rod drift. The Override of AOP-12 requires a manual Reactor scram due to the single control rod drift. AOP-27 does not currently require a manual Reactor scram, but would if a second control rod drifted.

A. Incorrect – AOP-27 does not currently require a manual Reactor scram. Plausible because it would if a second control rod drifted.
C. Incorrect – AOP-27 does not currently require a manual Reactor scram. Plausible because it would if a second control rod drifted. AOP-12 does require a manual Reactor scram. Plausible because air pressures are stable, Operators are still attempting corrective action, and only one rod has drifted.
D. Incorrect – AOP-12 does require a manual Reactor scram. Plausible because air pressures are stable, Operators are still attempting corrective action, and only one rod has drifted.

Technical Reference(s): AOP-12, AOP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.15.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K6.05
	Importance Rating	2.8

Component Cooling Water

Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Motors

Proposed Question: #53

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

- RBCLC pumps A and B are running.
- RBCLC pump C is tagged out for maintenance.

Then, RBCLC pump A trips on motor overcurrent. RBCLC pump B trips on motor overcurrent one minute later.

Which one of the following describes the response of the Emergency Service Water (ESW) system, if any, and the need for a manual Reactor scram, in accordance with AOP-11, Loss of Reactor Building Closed Loop Cooling?

A. ESW remains in standby. A manual Reactor scram is required.

B. ESW remains in standby. A manual Reactor scram is NOT required.

C. ESW automatically starts and injects into the RBCLC system. A manual Reactor scram is required.

D. ESW automatically starts and injects into the RBCLC system. A manual Reactor scram is NOT required.

Proposed Answer: C

Explanation: With RBCLC pump C tagged out and both RBCLC pump A and B tripped, RBCLC pressure will lower below 40 psig. This causes ESW to automatically inject into RBCLC. A manual scram is required by AOP-11 due to trip of all RBCLC pumps.

A. Incorrect – With no RBCLC pumps running, RBCLC pressure lowers below 40 psig. This causes ESW to automatically inject into RBCLC. Plausible because ESW does not automatically start based on RBCLC pump trips alone.
B. Incorrect – With no RBCLC pumps running, RBCLC pressure lowers below 40 psig. This causes ESW to automatically inject into RBCLC. Plausible because ESW does not automatically start based on RBCLC pump trips alone. A manual scram is required by AOP-11 due to trip of all RBCLC pumps. Plausible that an emergency power reduction would first be performed and equipment temperatures monitored before requiring a scram.
D. Incorrect – A manual scram is required by AOP-11 due to trip of all RBCLC pumps. Plausible that an emergency power reduction would first be performed and equipment temperatures monitored before requiring a scram.

Technical Reference(s): AOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.05.c.5 and 1.15.a

Question Source: Bank - 14-1 NRC #11

Question History: 14-1 NRC #11

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 #	201001 2.2.12
	Importance Rating	3.7

CRD Hydraulic

Knowledge of surveillance procedures.

Proposed Question: #54

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

- Monthly surveillance test ST-20C, Control Rod Operability for Fully Withdrawn Control Rods, is being performed.

Which one of the following describes the general test method used in this test?

Each tested control rod is…

A. determined to be operable by administrative means only.

B. kept at position 48 while a coupling check is performed.

C. inserted one notch and then returned to position 48. NO coupling check is required.

D. inserted one notch and then withdrawn to position 48 while performing a coupling check.

Proposed Answer: D

Explanation: ST-20C both exercises the control rod by inserting it one notch, and then returning it to position 48 while performing a coupling check.

A. Incorrect – Each control rod is moved one notch and a coupling test is performed. Plausible because control rods are checked by only administrative means in ST-40D.
B. Incorrect – The control rod is also moved one notch. Plausible that the control rod would not be moved to minimize unnecessary reactivity changes.
C. Incorrect – A coupling check is also performed. Plausible a coupling check was already performed on the rod when it was initially positioned at 48, and uncoupling while stationary at position 48 is highly unlikely.

Technical Reference(s): ST-20C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.13.j

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 #	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: #55

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 position 08.

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 – 16-1 NRC #38

Question History: 16-1 NRC #38

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 #	201002 K1.05
	Importance Rating	3.4

Reactor Manual Control

Knowledge of the physical connections and/or cause-effect relationships between REACTOR MANUAL CONTROL SYSTEM and the following: Rod worth minimizer: Plant-Specific

Proposed Question: #56

A plant startup is in progress with the following:

- APRMs indicate 5% and stable.
- The Rod Worth Minimizer (RWM) NORMAL BYPASS keylock switch is in NORMAL.
- An Operator is withdrawing the first control rod in a group (22-31).
- The group withdraw limit is 24.
- The Operator erroneously withdraws control rod 22-31 to position 26.
- Then, the Operator selects the next control rod in the same group (30-31).

Which one of the following describes the ability to move control rods 30-31 and 22-31 with the Reactor Manual Control System (RMCS)?

	Control Rod 30-31	Control Rod 22-31
A.	Can be moved	Can be inserted, only
B.	Can be moved	Can be withdrawn and inserted
C.	CANNOT be moved	Can be inserted, only
D.	CANNOT be moved	Can be withdrawn and inserted

Proposed Answer: C

Explanation: With Reactor power below 10% and the RWM NORMAL BYPASS keylock switch in NORMAL, the RWM is enforcing the programmed rod sequence. Control rod 22-31 has been withdrawn beyond the withdraw limit. This results in a withdraw error and withdraw block for control rod 22-31. Control rod 22-31 can be inserted with RMCS to correct the error. Until the withdraw error is corrected, all other control rods have withdraw and insert blocks imposed.

A. Incorrect – Control rod 30-31 withdraw and insert are blocked. Plausible because only one control rod positioning error exists, and in some cases multiple are allowed to exist before resulting in a block.
B. Incorrect – Control rod 30-31 withdraw and insert are blocked. Plausible because only one control rod positioning error exists, and in some cases multiple are allowed to exist before resulting in a block. Control rod 22-31 withdraw is blocked. Plausible because the control rod is only withdrawn a single notch past the group limit and no other errors exist.
D. Incorrect – Control rod 22-31 withdraw is blocked. Plausible because the control rod is only withdrawn a single notch past the group limit and no other errors exist.

Technical Reference(s): OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.05.b.4, 1.05.b.5

Question Source: Bank - 9/12 NRC #55

Question History: 9/12 NRC #55

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 #	202001 K5.10
	Importance Rating	2.8

Recirculation

Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION SYSTEM: Motor generator set operation: Plant-Specific

Proposed Question: #57

The plant is operating at 100% power when a leak causes fluid drive oil level in RWR MG set A to lower.

Which one of the following describes the resulting behavior of Recirculation loop A flow?

Recirculation loop A flow…

A. Rises

B. Lowers

C. Remains stable due to automatic lock of the scoop tube

D. Remains stable due to automatic adjustment of scoop tube position

Proposed Answer: B

Explanation: The RWR MG set drive motor is coupled to the generator through a fluid drive that is used to vary the speed of the generator. The amount of oil in the fluid drive working circuit determines the speed of the generator. As oil is removed from the working circuit, the generator slows down. The amount of oil removed from the working circuit is normally controlled by scoop tube position. However, with a leak causing oil level to lower, the scoop tube is no longer in control. With less oil coupling the two machines, generator speed lowers, which cause Recirculation loop A flow to lower.

A. Incorrect – Recirculation loop A flow lowers, not rises. Plausible because speed does change and this would be the answer if the operation of the fluid drive coupling was misunderstood.
C. Incorrect – Recirculation loop A flow lowers. Plausible because there is an automatic scoop tube lockup on multiple conditions, including low oil pressure, which could be an issue if conditions degraded such that oil pressure was lost.
D. Incorrect – Recirculation loop A flow lowers. Plausible if candidate believed there was a feedback circuit in the scoop tube positioning controls that attempted to maintain constant generator speed.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.05.a.9

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 #	204000 K4.06
	Importance Rating	2.6

Reactor Water Cleanup

Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Maximize plant efficiency (use of regenerative heat exchanger)

Proposed Question: #58

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

- Preparations are underway for a plant shutdown.
- RWCU Blowdown is to be established to Radwaste for post-maintenance testing.

Which one of the following describes the effect of establishing RWCU Blowdown flow on NRHX outlet temperature and plant efficiency?

	NRHX Outlet Temperature	Plant Efficiency
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: B

Explanation: Establishing RWCU Blowdown flow diverts some of the water that would otherwise pass through the regenerative heat exchanger. This leads to less cooling of the incoming coolant to the RWCU system, which raises NRHX outlet temperature. This also rejects heat from returning to the Reactor coolant system, increasing makeup from the relatively cold Hotwell. The additional heating that must be done on this makeup water reduces plant efficiency.

A. Incorrect – Plant efficiency lowers. Plausible because less cooling occurs on the incoming water to RWCU, which could be thought of as less loss of heat in a portion of the plant cycle.
C. Incorrect – NRHX outlet temperature rises. Plausible if the Blowdown flow path and cause/effect relationship is not understood. Plant efficiency lowers. Plausible because less cooling occurs on the incoming water to RWCU, which could be thought of as less loss of heat in a portion of the plant cycle.
D. Incorrect – NRHX outlet temperature rises. Plausible if the Blowdown flow path and cause/effect relationship is not understood.

Technical Reference(s): OP-28

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12 1.13.c

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

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

Rod Position Information

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

Proposed Question: #59

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

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

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

Annunciator…

A. 09-5-2-3, Rod Drift, alarms when the control rod time times out.

B. 09-5-2-3, Rod Drift, alarms when the next control rod is selected.

C. 09-5-2-4, Rod Overtravel, alarms when the control rod time times out.

D. 09-5-2-4, Rod Overtravel, alarms when the next control rod is selected.

Proposed Answer: A

Explanation: Annunciator 09-5-2-3, Rod Drift, alarms when the control rod time times out because the control rod fails to indicate at an even position (position 48 reed switch is open) and does indicate at an odd position (position 47 reed switch is closed).

B. Incorrect – The annunciator alarms as soon as the timer times out, not later when the next rod is selected. Plausible that the alarm would be blocked while a rod is selected and has been moved since odd reed switch positions are expected to be picked up during this evolution.
C. Incorrect – Annunciator 09-5-2-4, Rod Overtravel, does not alarm. Plausible because this would be correct for a different combination of reed switch malfunctions (48 open, 49 closed).
D. Incorrect – Annunciator 09-5-2-4, Rod Overtravel, does not alarm. Plausible because this would be correct for a different combination of reed switch malfunctions (48 open, 49 closed). The correct annunciator alarms as soon as the timer times out, not later when the next rod is selected. Plausible that the alarm would be blocked while a rod is selected and has been moved since odd reed switch positions are expected to be picked up during this evolution.

Technical Reference(s): ARPs 09-5-2-3 and 09-5-2-4

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03G 1.05.a.3

Question Source: Bank - SSES LOC28 NRC #32 (2016)

Question History: SSES LOC28 NRC #32 (2016)

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 #	215001 A2.07
	Importance Rating	3.4

Traversing In Core Probe

Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract during accident conditions: Mark-I&II

Proposed Question: #60

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

- Traversing In-Core Probe (TIP) scans are in progress with TIP System 1 detector in the core.
- A coolant leak develops in the Drywell.
- Drywell pressure is 3 psig and rising.
- TIP System 1 detector fails to retract automatically and manually.

Which one of the following describes the resulting operation of the TIP System 1 shear valve?

The TIP System 1 shear valve…

A. automatically closes immediately.

B. automatically closes after a time delay.

C. must be manually closed using a switch at the TIP Room.

D. must be manually closed using a switch on Control Room panel 09-13.

Proposed Answer: D

Explanation: The TIP shear valves are provided as a backup to the normal primary containment isolation feature. The normal primary containment isolation feature automatically retracts the detector and then closes the associated ball valve. If the detector fails to retract, the ball valve cannot close, thus the shear valve is provided to cut through the detector cable and isolate the containment penetration. No automatic shear valve actuation is provided. The operator must detect the failure and manually actuate the shear valve. The control switch for this action is on Control Room Panel 09-13.

A. Incorrect – The shear valve does NOT automatically close. Plausible that it would close after a time delay on failure to retract, as this is the design purposes of the valve.
B. Incorrect – The shear valve does NOT automatically close. Plausible that it would close after a time delay on failure to retract, as this is the design purposes of the valve.
C. Incorrect – The shear valve is fired from Control Room Panel 09-13. Plausible because this is an infrequently used control and some manual actions can be done from TIP rooms.

Technical Reference(s): AOP-15, SDLP-07F

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07F 1.05.b.1, 1.05.b.2

Question Source: Bank – 9/12 NRC #57

Question History: 9/12 NRC #57

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 #	215002 A4.03
	Importance Rating	2.8

Rod Block Monitor

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

Proposed Question: #61

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

- A hardware problem has occurred on Rod Block Monitor (RBM) A.
- The hardware malfunction is equivalent to the RBM A mode switch being out of OPERATE.

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

RBM A is…

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

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

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

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

Proposed Answer: A

Explanation: The RBM is auto-bypassed since power is below 30%. Therefore, no rod blocks are received.

B. Incorrect – The RBM is auto-bypassed since power is below 30%. Plausible because this would be correct if power was >30% and the hardware failure was not causing the INOP trip.
C. Incorrect – RBM A is auto-bypassed. Plausible because this would be correct if power were >30%.
D. Incorrect – RBM A is auto-bypassed. Plausible because this would be correct for a RWM INOP trip during a startup.

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

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank - 17-1 NRC #35

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

RBM

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

Proposed Question: #35

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

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

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

RBM A is…

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

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

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

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

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 A4.04
	Importance Rating	3.0

RHR/LPCI: Torus/Suppression Pool Cooling Mode

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

Proposed Question: #62

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

- RHR A is being placed in Torus Cooling in preparation for HPCI testing.
- An Operator is about to start RHR pump A.

Which one of the following describes the operation of 10MOV-16A, MIN FLOW VLV, during this evolution?

10MOV-16A is initially…

A. open and closes when RHR loop A flow reaches approximately 1500 gpm.

B. open and closes when RHR loop A flow reaches approximately 6500 gpm.

C. closed, opens when RHR pump A starts, and closes when RHR loop A flow reaches approximately 1500 gpm.

D. closed, opens when RHR pump A starts, and closes when RHR loop A flow reaches approximately 6500 gpm.

Proposed Answer: A

Explanation: The normal standby lineup has 10MOV-16A fully open prior to start of RHR pump

A. 10MOV-16A automatically closes when RHR loop A flow reaches approximately 1500 gpm.

B. Incorrect – 10MOV-16A automatically closes when RHR loop A flow reaches approximately 1500 gpm. Plausible because 6500 gpm is the minimum required flow to be established with one RHR pump running.
C. Incorrect – 10MOV-16A is initially open. Plausible that the normal standby lineup would have the valve closed and it would only open when a pump was running with low flow.
D. Incorrect – 10MOV-16A is initially open. Plausible that the normal standby lineup would have the valve closed and it would only open when a pump was running with low flow. 10MOV-16A automatically closes when RHR loop A flow reaches approximately 1500 gpm. Plausible because 6500 gpm is the minimum required flow to be established with one RHR pump running.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.05.a.1.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 #	2
	K/A #	223001 A3.04
	Importance Rating	4.2

Primary Containment and Auxiliaries

Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: Containment / drywell response during LOCA

Proposed Question: #63

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 – 16-1 NRC #32

Question History: 16-1 NRC #32

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 K3.02
	Importance Rating	3.2

Reactor Condensate System

Knowledge of the effect that a loss or malfunction of the REACTOR CONDENSATE SYSTEM will have on following: CRD hydraulics system

Proposed Question: #64

The plant is operating at 100% power when all Condensate pumps trip.

Which one of the following describes the impact of the loss of all Condensate pumps on CRD?

CRD…

A. loses a backup source of water but retains the normal source.

B. loses the normal source of water but automatically swaps to the backup source.

C. loses the normal source of water and must be manually swapped to take suction from the Torus.

D. loses the normal source of water and must be manually swapped to take suction from the Condensate Storage Tanks.

Proposed Answer: B

Explanation: CRD normally takes suction from the Condensate pump discharge header through a pressure control valve. When all Condensate pumps trip, pressure in this header is lost. CRD suction automatically swaps to the backup source (Condensate Storage Tanks) through a check valve.

A. Incorrect – The discharge of the Condensate pumps is the normal source of water to the CRD pumps. Plausible that the Condensate Storage Tanks would be the normal source with the discharge of the Condensate pumps as a backup.
C. Incorrect – CRD suction automatically swaps to the backup source through a check valve. Plausible that the Torus could be manually aligned as a backup, as it is a large volume of water used for emergencies.
D. Incorrect – CRD suction automatically swaps to the backup source through a check valve. Plausible that transfer to the Condensate Storage Tanks would be manual, not automatic.

Technical Reference(s): OP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.10.a

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 #	241000 K1.31
	Importance Rating	3.1

Reactor/Turbine Pressure Regulating

Knowledge of the physical connections and/or cause-effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: Turbine protection

Proposed Question: #65

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 (>225oF) 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 – 16-1 NRC #30

Question History: 16-1 NRC #30

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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

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

Proposed Question: #66

A tagout is being developed for a water system with the following conditions:

- Highest system water temperature: 300ºF
- Highest system water pressure: 100 psig

Which one of the following describes the need to provide dual valve isolation, in accordance with OP-AA-109-101, Clearance and Tagging?

In accordance with OP-AA-109-101, dual valve isolation should…

A. NOT be provided.

B. be provided, due to pressure only.

C. be provided, due to temperature only.

D. be provided, due to both pressure and temperature.

Proposed Answer: C

Explanation: OP-AA-109-101 states that dual valve isolation should be provided when isolating from an energy source having a temperature greater than 200ºF or pressure greater than 500 psig. Therefore, this system should have dual valve isolation based on a temperature of >200ºF, but not based on pressure.

A. Incorrect – This should have dual valve isolation based on a temperature of >200ºF.
B. Incorrect – This should have dual valve isolation, but based on temperature, not pressure. The highest system water pressure is below the threshold of 500 psig.
D. Incorrect – This should have dual valve isolation, but based on temperature, not pressure. The highest system water pressure is below the threshold of 500 psig.

Technical Reference(s): OP-AA-109-101

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 44.09.d

Question Source: Bank – 14-2 NRC #67

Question History: 14-2 NRC #67

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.44
	Importance Rating	3.9

Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Proposed Question: #67

A refueling outage is in progress with the following:

- The core is fully loaded with fuel.
- The following evolutions are planned on the next shift:

  (1) The Refuel Bridge will be re-arranging irradiated fuel bundles in the Spent Fuel Pool.

  (2) The Refuel Bridge will be replacing control rods in the Reactor core.

Which one of the following describes the requirement for maintaining continuous communications between a licensed operator in the Control Room and the Refuel Bridge during these evolutions, in accordance with OSP-66.001, Management of Refueling Activities?

Continuous communications between a licensed operator in the Control Room and the Refuel Bridge is…

A. required during both of these evolutions.

B. NOT required during either of these evolutions.

C. required during evolution (1), but NOT during evolution (2).

D. required during evolution (2), but NOT during evolution (1).

Proposed Answer: D

Explanation: OSP-66.001 requires continuous communication between a licensed operator in the Control Room and the Refuel Bridge during the performance of core alterations. Replacing control rod blades in the core during evolution (2) is a core alteration. Moving irradiated fuel bundles during evolution (1) is not a core alteration and does not require continuous communication with the Control Room.

A. Incorrect – Continuous communication between a licensed operator in the Control Room and the Refuel Bridge is not required during evolution (1). Plausible because this evolution does involve movement of irradiated fuel.
B. Incorrect – Continuous communication between a licensed operator in the Control Room and the Refuel Bridge is required during evolution (2). Plausible because this evolution does not involve fuel and control rod blade replacements are always done in defueled cells.
C. Incorrect – Continuous communication between a licensed operator in the Control Room and the Refuel Bridge is not required during evolution (1). Plausible because this evolution does involve movement of irradiated fuel.

Technical Reference(s): OSP-66.001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2015 Cert #67

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.37
	Importance Rating	4.3

Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question: #68

Given the following individuals:

(1) A qualified, non-licensed Equipment Operator who has been selected for license class but has not yet begun formal license class training.

(2) An instant SRO candidate who is performing the pre-license class plant familiarization guide.

(3) An RO license candidate who is completing the in-plant OJT phase of license class.

Which one of the following identifies the individual(s) that may be allowed to perform reactivity manipulations in the plant under the guidance of a licensed Reactor Operator, in accordance with OP-AA-300, Reactivity Management, and OP-AA-103-103, Operation of Plant Equipment?

A. (3) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: OP-AA-300 and OP-AA-103-103 allows an RO to direct a non-licensed individual to perform reactivity manipulations under their guidance in certain circumstances:

- The non-licensed individual shall be an active participant in an ongoing Licensed Operator Training class.
- The individual shall have met the knowledge requirements as defined in the Licensed Operator Training program.

The knowledge requirements in the Licensed Operator Training program include completion of related Generic Fundamentals and reactivity control system training. Since individual (3) is the only one having completed those requirements to get to the in-plant phase of license class, they are the only individual allowed to perform reactivity manipulations under instruction.

B. Incorrect – Individual (1) is not yet allowed to perform reactivity manipulations. Plausible because they are qualified to perform non-licensed activities and will eventually be allowed to perform under instruction reactivity manipulations prior to being licensed.
C. Incorrect – Individual (2) is not yet allowed to perform reactivity manipulations. Plausible because they are in a preparatory phase of the licensed operator training program and will eventually be allowed to perform under instruction reactivity manipulations prior to being licensed.
D. Incorrect – Individual (1) is not yet allowed to perform reactivity manipulations. Plausible because they are qualified to perform non-licensed activities and will eventually be allowed to perform under instruction reactivity manipulations prior to being licensed. Individual (2) is not yet allowed to perform reactivity manipulations. Plausible because they are in a preparatory phase of the licensed operator training program and will eventually be allowed to perform under instruction reactivity manipulations prior to being licensed.

Technical Reference(s): OP-AA-300, OP-AA-103-103

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - PB 2017 NRC #66

Question History: PB 2017 NRC #66

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 #	2.2.14
	Importance Rating	3.9

Knowledge of the process for controlling equipment configuration or status.

Proposed Question: #69

Which one of the following combinations of tags is NOT allowed to be simultaneously applied to a single component, in accordance with OP-AA-109-101, Clearance and Tagging?

A. Two Danger Tags

B. A Danger Tag and an Information Tag

C. A Special Condition Tag and a Danger Tag

D. A Special Condition Tag and an Information Tag

Proposed Answer: C

Explanation: OP-AA-109-101 prohibits simultaneous placement of a Danger Tag and a Special Condition Tag on a single component.

A. Incorrect – Two Danger Tags may be simultaneous applied on a single component. Plausible because the procedure does contain a restriction for this pairing (positions cannot conflict).
B. Incorrect – A Danger Tag and an Information Tag may be simultaneous applied on a single component. Plausible because the procedure does contain a restriction for this pairing (positions cannot conflict).
D. Incorrect – A Special Condition Tag and an Information Tag may be simultaneous applied on a single component. Plausible because the procedure does contain restrictions for this pairing (positions cannot conflict and Information Tag cannot obstruct Special Condition Tag).

Technical Reference(s): OP-AA-109-101

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 44.05

Question Source: Bank - NMP2 2017 Cert #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 #	2.2.39
	Importance Rating	3.9

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

Proposed Question: #70

The plant is operating at 100% power.

Which one of the following conditions requires action in one hour or less, in accordance with Technical Specifications?

A. MAPRAT at 1.1

B. Drywell pressure at 1.4 psig

C. Torus water temperature at 111^{o}F

D. RCS unidentified leakage at 10 gpm

Proposed Answer: C

Explanation: With Torus water temperature at 111ºF, Technical Specification 3.6.2.1 requires immediate action to scram the Reactor.

A. Incorrect – With MAPRAT at 1.1, the APLHGR thermal limit is SAT. Plausible because Technical Specification 3.2.1 does have a 2 hour requirement based on MAPRAT.
B. Incorrect – With Drywell pressure at 1.4 psig, Technical Specification 3.6.1.4 is SAT. Plausible because this Technical Specification does require action within 1 hour if Drywell pressure exceeds 1.95 psig.
D. Incorrect – With RCS unidentified leakage at 10 gpm, Technical Specification 3.4.4 is not met, but only requires action within 4 hours. Plausible because Technical Specification 3.4.4 is not met and does require action within a fairly short amount of time.

Technical Reference(s): Technical Specifications 3.2.1, 3.4.4, 3.6.1.4, 3.6.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.16

Question Source: Bank - SSES LOC28R NRC #54 (2017)

Question History: SSES LOC28R NRC #54 (2017)

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 #	2.2.22
	Importance Rating	4.0

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #71

The plant experienced a failure to scram with the following:

- The peak Reactor pressure during the transient was 1350 psig.
- The lowest Reactor water level during the transient was 20”.

Which one of the following describes whether the Reactor pressure and Reactor water level Safety Limits were exceeded or NOT, in accordance with Technical Specifications?

	Reactor Pressure Safety Limit	Reactor Water Level Safety Limit
A.	Exceeded	Exceeded
B.	Exceeded	NOT exceeded
C.	NOT exceeded	Exceeded
D.	NOT exceeded	NOT exceeded

Proposed Answer: B

Explanation: The Reactor pressure Safety Limit (1325 psig) was exceeded. The Reactor water level Safety Limit (> top of active fuel, or 0”) was not exceeded.

A. Incorrect – The Reactor water level Safety Limit (> top of active fuel, or 0”) was not exceeded. Plausible because level is much lower than normal and near top of active fuel.
C. Incorrect – The Reactor pressure Safety Limit (1325 psig) was exceeded. Plausible because peak pressure is just slightly above the limit. The Reactor water level Safety Limit (> top of active fuel, or 0”) was not exceeded. Plausible because level is much lower than normal and near top of active fuel.
D. Incorrect – The Reactor pressure Safety Limit (1325 psig) was exceeded. Plausible because peak pressure is just slightly above the limit.

Technical Reference(s): Technical Specification 2.0

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02D 1.16

Question Source: Bank - 2010 NRC #12

Question History: 2010 NRC #12

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

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

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

Proposed Question: #72

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

- Movement of radioactive material in the Reactor Building causes Area Radiation Monitor (ARM) #25, RX BLDG EL 272 EAST HCU AREA, to go into alarm high.
- Annunciator 09-3-1-40, RX BLDG ARM RAD HI, is received and acknowledged.

Which one of the following describes how Annunciator 09-3-1-40 will respond to the following two independent conditions?

If a second ARM input exceeds its high alarm setpoint, then Annunciator 09-3-1-40 ___(1)___.

If ARM #25 returns to normal, then Annunciator 09-3-1-40 ___(2)___.

	(1)	(2)
A.	re-flashes	can be reset before depressing ARM RESET pushbutton(s)
B.	re-flashes	CANNOT be reset before depressing ARM RESET pushbutton(s)
C.	does NOT re-flash	can be reset before depressing ARM RESET pushbutton(s)
D.	does NOT re-flash	CANNOT be reset before depressing ARM RESET pushbutton(s)

Proposed Answer: D

Explanation: Annunciator 09-3-1-40 will actuate upon receipt of a high alarm condition from any of 16 Area Radiation Monitors (ARMs). However, once the annunciator is in, it will not re-flash for any subsequent high alarm conditions from any of the other ARMs. Additionally, once the high radiation condition clears, the high alarm trip is sealed in until the RESET pushbutton is depressed on the corresponding ARM trip unit on control room panel 09-11. Until this RESET pushbutton is depressed, both the ARM trip unit amber HIGH light and Annunciator 09-3-1-40 will be sealed in.

A. Incorrect – The annunciator does not have re-flash capability. Plausible because other annunciators do have re-flash capability and it would be helpful to identify additional areas developing high radiation. The annunciator cannot be reset until the ARM RESET pushbutton is depressed. Plausible because other annunciators clear without the need for manual action at the tripping device.
B. Incorrect – The annunciator does not have re-flash capability. Plausible because other annunciators do have re-flash capability and it would be helpful to identify additional areas developing high radiation.
C. Incorrect – The annunciator cannot be reset until the ARM RESET pushbutton is depressed. Plausible because other annunciators clear without the need for manual action at the tripping device.

Technical Reference(s): ARP 09-3-1-40, 1.78-98, UFSAR Fig 7.13-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.14.d.5

Question Source: Bank - 9/12 NRC #73

Question History: 9/12 NRC #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.41
	Importance Rating	2.9

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #73

Which one of the following describes the ***lowest*** Emergency Action Level (EAL) at which the full Emergency Response Organization (ERO) is required to be staffed?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: B

Explanation: The Alert EAL is the lowest level at which the full ERO is required to be staffed.

A. Incorrect – The Alert EAL is the lowest level at which the full ERO is required to be staffed. Plausible because precautionary staffing of the ERO is allowed at the Unusual Event level.
C. Incorrect – The Alert EAL is the lowest level at which the full ERO is required to be staffed. Plausible because the full ERO is staffed at the Site Area Emergency level also.
D. Incorrect – The Alert EAL is the lowest level at which the full ERO is required to be staffed. Plausible because the full ERO is staffed at the General Emergency level also.

Technical Reference(s): EP-AA-112-100-F-50

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2010 NRC #72

Question History: NMP1 2010 NRC #72

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 #	2.4.3
	Importance Rating	3.7

Ability to identify post-accident instrumentation.

Proposed Question: #74

Which one of the following instruments is NOT required Post Accident Monitoring (PAM) Instrumentation, in accordance with Technical Specifications?

A. Torus water level

B. Torus water temperature

C. Average power range monitor

D. Containment high range radiation monitor

Proposed Answer: C

Explanation: Of the given instruments, only average power range monitors (APRMs) are not required by TS 3.3.3.1, PAM Instrumentation.

A. Incorrect – Torus water level instrumentation is required by TS 3.3.3.1. Plausible because this instrument does not directly monitor the Reactor.
B. Incorrect – Torus water temperature instrumentation is required by TS 3.3.3.1. Plausible because this instrument does not directly monitor the Reactor.
D. Incorrect – Containment high range radiation instrumentation is required by TS 3.3.3.1. Plausible because this instrument is not widely used by operators.

Technical Reference(s): Technical Specification 3.3.3.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2009 Cert #72

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.26
	Importance Rating	3.1

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

Proposed Question: #75

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

- Maintenance personnel report a fire has developed in the East Electric Bay.
- A high temperature alarm has been received for this zone on the Fire Protection Panel.
- NO other adverse effects from the fire have been observed.

Which one of the following describes the operational response required for these conditions in accordance with FPP-1.2, Fire Fighting, and AOP-28, Operation During Plant Fires?

A. Dispatch the full Fire Brigade. Inserting a manual Reactor scram is required now.

B. Dispatch the full Fire Brigade. Inserting a manual Reactor scram is NOT yet required.

C. Dispatch the Fire Brigade Leader to assess the need for full Fire Brigade activation. Inserting a manual Reactor scram is required now.

D. Dispatch the Fire Brigade Leader to assess the need for full Fire Brigade activation. Inserting a manual Reactor scram is NOT yet required.

Proposed Answer: B

Explanation: FPP-1.2 contains guidance for Control Room Operator response to a report from personnel of a fire in the plant. This guidance requires dispatching the full Fire Brigade, NOT just the Fire Brigade Leader. AOP-28 contains criteria for when a Reactor scam is required based on a fire. All four of the following need to be met for a Reactor scram to be required based on this fire:

- Serious fire in progress (E-Plan Alert or worse)
- Reactor in mode 1, 2, or 3
- Ionization alarm at FPP, actuation of fire suppression system, and/or verbal report of fire
- Unexplained EPIC or annunciator alarm, unexplained loss of equipment, and/or unexplained actuation of equipment

The given conditions only satisfy the 2nd and 3rd bullets, therefore a Reactor scram is NOT yet required.

A. Incorrect – Since no unexplained alarms, equipment loss, or equipment actuation have occurred, a Reactor scram is NOT required per AOP-28 yet. Plausible because this would be correct if additional degradation of plant conditions were given.
C. Incorrect – FPP-1.2 requires full Fire Brigade activation due to the verbal report of a fire. Plausible that the Fire Brigade Leader would first be used to confirm the fire and decide where to stage the full Fire Brigade. Since no unexplained alarms, equipment losses, or equipment actuations have occurred, a Reactor scram is NOT required per AOP-28 yet.
D. Incorrect – FPP-1.2 requires full Fire Brigade activation due to the verbal report of a fire. Plausible that the Fire Brigade Leader would first be used to confirm the fire and decide where to stage the full Fire Brigade.

Technical Reference(s): FPP-1.2, AOP-28

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Bank – 14-1 NRC #39

Question History: 14-1 NRC #39

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 #	295021 2.1.23
	Importance Rating	4.4

Loss of Shutdown Cooling

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

Proposed Question: #76

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

- RHR loop B is in a Shutdown Cooling Lineup.
- No RWR pumps are running.
- Irradiated fuel movement is in progress.
- Maintenance is in progress which has the potential for draining the Reactor vessel.
- Then, an inadvertent isolation of Shutdown Cooling occurs.
- The isolation CANNOT be immediately restored.
- Reactor cavity water temperature is 80°F.

Which one of the following describes the need for a Refuel Floor evacuation and required control of the maintenance which has the potential for draining the Reactor vessel, in accordance with AOP-30, Loss of Shutdown Cooling?

	Refuel Floor Evacuation	Maintenance Which Has the Potential For Draining the Reactor Vessel
A.	NOT required	May continue
B.	NOT required	Must be secured
C.	Required	May continue
D.	Required	Must be secured

Proposed Answer: B

Explanation: AOP-30 attachment 2 describes the required Reactor water level control strategy. It requires securing and restoring from any maintenance which has the potential for draining the Reactor vessel. There is no requirement within AOP-30 to perform a Refuel Floor evacuation.

A. Incorrect – AOP-30 requires securing and restoring from any maintenance which has the potential for draining the Reactor vessel. Plausible because there is no evidence that Reactor water level control is an issue.
C. Incorrect – There is no requirement within AOP-30 to perform a Refuel Floor evacuation. Plausible because loss of SDC is a degradation of shutdown safety, will result in rising temperatures of the Reactor coolant and the Refuel Floor environment is in direct communication with the Reactor coolant. AOP-30 requires securing and restoring from any maintenance which has the potential for draining the Reactor vessel. Plausible because there is no evidence that Reactor water level control is an issue.
D. Incorrect – There is no requirement within AOP-30 to perform a Refuel Floor evacuation. Plausible because loss of SDC is a degradation of shutdown safety, will result in rising temperatures of the Reactor coolant and the Refuel Floor environment is in direct communication with the Reactor coolant.

Technical Reference(s): AOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Modified Bank - 14-1 NRC #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Inadvertent Containment Isolation

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

Proposed Question: #83

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

- RHR loop B is in a Shutdown Cooling Lineup.
- No RWR pumps are running.
- Reactor water level is 210 inches and stable.
- Maintenance is in progress which has the potential for draining the Reactor vessel.
- Then, an error during I&C testing results in an inadvertent PCIS Group 2 isolation.
- The isolation CANNOT be immediately restored.

Which one of the following describes the response that is required to be directed?

A. Verify isolations per AOP-15, Isolation Verification and Recovery. AOP-30, Loss of Shutdown Cooling, does NOT need to be entered.

B. Verify isolations per AOP-15, Isolation Verification and Recovery. Enter AOP-30, Loss of Shutdown Cooling. Raise Reactor water level to between 234.5 and 270 inches or start an RWR pump. Secure and restore from the maintenance which has the potential for draining the Reactor vessel.

C. Verify isolations per AOP-15, Isolation Verification and Recovery. Enter AOP-30, Loss of Shutdown Cooling. Secure and restore from the maintenance which has the potential for draining the Reactor vessel. Reactor water level does NOT need to be raised NOR does a RWR pump need to be started.

D. Verify isolations per AOP-15, Isolation Verification and Recovery. Enter AOP-30, Loss of Shutdown Cooling. Raise Reactor water level to between 234.5 and 270 inches or start an RWR pump. The maintenance which has the potential for draining the Reactor vessel may continue as long as Reactor water level control is adequate.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.03
	Importance Rating	3.8

Refueling Accidents

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Airborne contamination levels

Proposed Question: #77

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

- ISFSI cask loading activities are in progress.
- An irradiated fuel bundle is dropped.
- 17RM-456A, Refuel Floor Exhaust Radiation Monitor, indicates $3x10^4$ cpm and stable for over an hour.
- 17RM-456B, Refuel Floor Exhaust Radiation Monitor, indicates $8x10^3$ cpm and stable for over an hour.

Which one of the following identifies whether an Unusual Event and/or Alert is required to be declared, in accordance with EP-AA-1014 EAL Wallboard?

	Declare an Unusual Event?	Declare an Alert?
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: B

Explanation: The given airborne contamination level on 17RM-456A is above the Hi-Hi setpoint ($1x10^4$ cpm) for the Refuel Floor Exhaust radiation monitors and due to damage to irradiated fuel. Therefore, Alert EAL RA2 is met and must be declared. Unusual Event RU2 is NOT met, despite the unplanned ARM reading rise in Table R2, because it is not due to lowering water level.

A. Incorrect – Alert EAL RA2 is met and must be declared. Plausible because only one of the Refuel Floor Exhaust rad monitors is above the threshold and Alert EAL RA1 is NOT applicable for the given radiation monitors.
C. Incorrect – Alert EAL RA2 is met and must be declared. Plausible because only one of the Refuel Floor Exhaust rad monitors is above the threshold and Alert EAL RA1 is NOT applicable for the given radiation monitors. Unusual Event RU2 is NOT met, despite the unplanned ARM reading rise in Table R2, because it is not due to lowering water level. Plausible because it would be met if the radiation level was due to lowering water level.
D. Incorrect – Unusual Event RU2 is NOT met, despite the unplanned ARM reading rise in Table R2, because it is not due to lowering water level. Plausible because it would be met if the radiation level was due to lowering water level. Also plausible for a novice candidate because in many cases both a UE and Alert classification are simultaneously met, while declaration is only made on the higher emergency level. Additionally, there is nothing on the part 1 notification form that precludes declaration of both a UE and Alert simultaneously, it is a required knowledge on the part of the SRO to only declare the higher emergency level.

Technical Reference(s): ARPs 09-3-2-29 and 09-3-2-40, EP-AA-1014

Proposed references to be provided to applicants during examination: Hot EAL Matrix

Learning Objective: LP-AOP 1.12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:
TRH 1/30/20 – Added to choice D plausibility statement based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.02
	Importance Rating	3.2

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: Cooling water temperature

Proposed Question: #78

The plant is operating at 100% power with a lake temperature of 86ºF 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).

A. Incorrect – TS 3.7.2 requires Condition entry because lake temperature is >85°F. Plausible because this would be correct if lake temperature were just slightly lower.
B. Incorrect – TS 3.7.1 Condition entry is not required. Plausible because elevated lake temperature does impact RHRSW performance. TS 3.7.2 requires Condition entry because lake temperature is >85°F. Plausible because this would be correct if lake temperature were just slightly lower.
D. Incorrect – TS 3.7.1 Condition entry is not required. Plausible because elevated lake temperature does impact RHRSW performance.

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: Bank – 16-1 NRC #90

Question History: 16-1 NRC #90

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 #	295030 2.2.38
	Importance Rating	4.5

Low Suppression Pool Water Level

Knowledge of conditions and limitations in the facility license.

Proposed Question: #79

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

- It is discovered that the surveillance for Torus water level (Surveillance Requirement (SR) 3.6.2.2.1) was last performed 36 hours ago.
- The Surveillance Frequency Control Program gives the frequency for SR 3.6.2.2.1 as 24 hours.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the status of LCO 3.6.2.2, Suppression Pool Water Level, in accordance with Technical Specifications?

A. LCO 3.6.2.2 must be declared NOT met immediately.

B. Complete SR 3.6.2.2.1 within a maximum of 2 hours from the time of discovery or then LCO 3.6.2.2 must be declared NOT met.

C. Complete SR 3.6.2.2.1 within a maximum of 6 hours from the time of discovery or then LCO 3.6.2.2 must be declared NOT met.

D. Complete SR 3.6.2.2.1 within a maximum of 24 hours from the time of discovery or then LCO 3.6.2.2 must be declared NOT met.

Proposed Answer: D

Explanation: Surveillance Requirement (SR) 3.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, also 24 hours), whichever is greater. Therefore up to 24 hours are allowed to perform the missed surveillance before being required to declare the LCO not met.

Note: NUREG 1020 ES-401 D.2.a requires Tier 3 questions to maintain their focus on plantwide generic K/As and not become an extension of Tier 2. However, there is no opposite requirement for Tier 2 questions to not test knowledges that can be applied to multiple Tier 2 systems. Therefore, while this question tests a generic SR, it still satisfies the K/A (Low Suppression Pool Water Level and SR 3.6.2.2.1 – 24 hours) and tests at an appropriate level.

A. Incorrect – SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the missed surveillance is over the 25% grace time allowed by SR 3.0.2.
B. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (also 24 hours). Plausible because 2 hours is the completion time of LCO 3.6.2.2 Condition A.
C. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (also 24 hours). Plausible because 6 hours is the grace time that would be allowed under SR 3.0.2 (25% of 24 hours).

Technical Reference(s): Technical Specifications 3.6.2.2 and SR 3.0.3

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2.2

Learning Objective: SDLP-16A 1.18

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:
TRH – Added note regarding Tier 2 / Tier 3 requirements, based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295037 EA2.01
	Importance Rating	4.3

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor power

Proposed Question: #80

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.
- Two SRVs are open.
- Torus temperature is above the Boron Injection Initiation Temperature.
- Reactor pressure is 1138 psig and steady.
- Reactor water level is 100” and slowly lowering.
- RCIC and HPCI are NOT running.

Which one of the following describes the current value of Reactor power and the required control of Reactor water level, in accordance with EOP-3, Failure to Scram?

Reactor power is…

A. ≤10%. Reactor water level must be lowered further.

B. ≤10%. Reactor water level may be maintained at the current level.

C. >10%. Reactor water level must be lowered further.

D. >10%. Reactor water level may be maintained at the current level.

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 2 SRVs = 1.8×10^6 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. Two SRVs multiplied by 8.2% is ~16%. With HPCI and RCIC not running, there are no other steam loads since the MSIVs are shut. With Reactor power >2.5%, Reactor water level >0", Torus temperature above BIIT, and an SRV open, Reactor water level must be lowered further.

A. Incorrect – Reactor power is >10%. Plausible because this would be correct if only one SRV was open.
B. Incorrect – Reactor power is >10%. Plausible because this would be correct if only one SRV was open. Reactor water level must be lowered further. Plausible because this would be correct if Reactor power was <2.5%, Reactor water level was <0", Torus temperature was below BIIT, or if no SRVs were open.
D. Incorrect – Reactor water level must be lowered further. Plausible because this would be correct if Reactor power was <2.5%, Reactor water level was <0", Torus temperature was below BIIT, or if no SRVs were open.

Technical Reference(s): FSAR Table 4.4-1 and Chapter 16.9.2, EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11d EO-1.07

Question Source: Modified Bank – NMP1 2017 NRC #89

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

SRVs

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #89

The plant is operating at 100% power when spurious isolation of both Main Steam lines results in the following:

- The Reactor Mode Switch has been placed in SHUTDOWN.
- Multiple control rods remain withdrawn.
- APRMs are unavailable.
- One Emergency Cooling loop is in service.
- The other Emergency Cooling loop has failed to initiate.
- One ERV is open.
- Reactor pressure is 1080 psig and stable.
- Reactor water level is 72” and stable.

Which one of the following describes the required control of Reactor water level, in accordance with the Emergency Operating Procedures, and the highest Emergency Action Level (EAL) that is met or exceeded?

Reactor water level…

A. must be lowered. The highest EAL met or exceeded is an Alert.

B. must be lowered. The highest EAL met or exceeded is a Site Area Emergency.

C. may be maintained at the current level. The highest EAL met or exceeded is an Alert.

D. may be maintained at the current level. The highest EAL met or exceeded is a Site Area Emergency.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.03
	Importance Rating	2.9

Partial or Complete Loss of D.C. Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Battery voltage

Proposed Question: #81

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

- 125 VDC Station Battery Charger 71BC-1B is removed from service for maintenance.
- Then, Temporary Battery Charger 71BC-9 is placed in service on Battery Bus B.
- The associated Battery Bus voltage is 131 VDC and stable.

Which one of the following identifies the current need for Condition entry into Technical Specification 3.8.4, DC Sources - Operating, and Technical Specification 3.8.7, Distribution Systems - Operating?

	Technical Specification 3.8.4, DC Sources - Operating	Technical Specification 3.8.7, Distribution Systems - Operating
A.	Condition entry required.	Condition entry required.
B.	Condition entry required.	Condition entry NOT required.
C.	Condition entry NOT required.	Condition entry required.
D.	Condition entry NOT required.	Condition entry NOT required.

Proposed Answer: B

Explanation: TS 3.8.4 requires both the battery terminal voltage ≥127.8 VDC and the 125 VDC battery charger in service and capable of supplying ≥270 amps. While the temporary battery charger may maintain bus voltage, it does not satisfy the requirements of TS 3.8.4. Therefore, TS 3.8.4 Condition A must be entered. TS 3.8.7 remains satisfied because the associated bus is still energized to the appropriate voltage and capable of supplying the required loads.

A. Incorrect – TS 3.8.7 condition entry is not required. Plausible because the current capacity of Battery Bus B is much lower with 71BC-9 in service (150 amps) and entry would be required if voltage lowered as a result.
C. Incorrect – TS 3.8.4 condition entry is required. Plausible because the temporary battery charger is placed in service and is maintaining bus voltage. TS 3.8.7 condition entry is not required. Plausible because the current capacity of Battery Bus B is much lower with 71BC-9 in service (150 amps) and entry would be required if voltage lowered as a result.
D. Incorrect – TS 3.8.4 condition entry is required. Plausible because the temporary battery charger is placed in service and is maintaining bus voltage.

Technical Reference(s): OP-43A, Technical Specifications 3.8.4 and 3.8.7

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B EO-1.16

Question Source: Bank - 17-1 NRC #81

Question History: 17-1 NRC #81

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.4.20
	Importance Rating	4.3

High Offsite Radioactivity Release Rate

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

Proposed Question: #82

The plant has experienced an accident with the following:

- The Primary Containment Pressure Limit (PCPL) is being challenged.
- Primary Containment venting would be necessary to prevent exceeding PCPL.

Which one of the following describes:

(1) whether defeating interlocks is allowed to vent the Primary Containment

and

(2) whether exceeding offsite radioactivity release rate limits during venting is allowed,

in accordance with EOP-4, Primary Containment Control?

	(1) Defeating Interlocks	(2) Exceeding Offsite Radioactivity Release Rate Limits
A.	Allowed	Allowed
B.	Allowed	NOT allowed
C.	NOT allowed	Allowed
D.	NOT allowed	NOT allowed

Proposed Answer: A

Explanation: EOP-4 allows both defeating interlocks and exceeding offsite radioactivity release rate limits during venting to avoid exceeding PCPL.

B. Incorrect – Exceeding offsite radioactivity release rate limits during venting is allowed to avoid exceeding PCPL. Plausible because there are other situations in the EOPs where exceeding release rate limits is not allowed.
C. Incorrect – Defeating interlocks is allowed for venting to avoid exceeding PCPL. Plausible because there are other situations in the EOPs where defeating interlocks is not allowed.
D. Incorrect – Defeating interlocks is allowed for venting to avoid exceeding PCPL. Plausible because there are other situations in the EOPs where defeating interlocks is not allowed. Exceeding offsite radioactivity release rate limits during venting is allowed to avoid exceeding PCPL. Plausible because there are other situations in the EOPs where exceeding release rate limits is not allowed.

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e EO-4.07

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 #	295033 EA2.03
	Importance Rating	4.2

High Secondary Containment Area Radiation Levels

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: †Cause of high area radiation

Proposed Question: #83

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

- A seismic event has occurred.
- Multiple barrels of radioactive waste have spilled in the Reactor Building.
- 18RIA-051-25, East HCU Area ARM, is in alarm and indicates upscale (>1x10^3 mr/hr).
- 18RIA-051-26, West HCU Area ARM, is in alarm and indicates upscale (>1x10^3 mr/hr).

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 ARM indications require entry into EOP-5, Secondary Containment Control. The ARMs are indicating that two Max Safe radiation levels have been exceeded (>10^3 mr/hr). Since the high area radiation levels are from leaking barrels of radioactive material, this 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 radiation levels were caused by a primary system discharge.

A. Incorrect – EOP-5 requires a Reactor shutdown. Plausible because the source of the radiation levels is not connected to the Reactor. Also plausible because this would be correct if only one area exceeded the Max Safe value.
C. Incorrect – A Reactor scram is not required. Plausible because this would be correct if the radiation levels were caused by a primary system discharge and only one was above Max Safe.
D. Incorrect – A Reactor scram and an emergency RPV depressurization is NOT required. Plausible because this would be correct if the radiation levels were caused by a primary system discharge.

Technical Reference(s): EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11f EO-1.07

Question Source: Modified Bank - 16-1 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

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.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295014 2.1.7
	Importance Rating	4.7

Inadvertent Reactivity Addition

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

Proposed Question: #84

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

- A control rod drifts from position 00 to 48 and CANNOT be re-inserted.
- Reactor power reaches a maximum of 102% during the transient.
- Operators have lowered Reactor power to 75%.
- MCPR reached a low of 1.05 during the transient.
- MCPR is now 1.25.

Which one of the following describes the need for a further Reactor power reduction and the need to make an ENS notification to the NRC, in accordance with Technical Specifications and LS-AA-1110, Exelon Reportability Reference Manual?

Further Reactor power reduction is __(1)__. ENS notification to the NRC is __(2)__.

	(1)	(2)
A.	required	required
B.	required	NOT required
C.	NOT required	required
D.	NOT required	NOT required

Proposed Answer: A

Explanation: MCPR <1.08, even for just a short period of time, is a violation of Safety Limit 2.1.1.2. Given the Safety Limit violation, Technical Specifications require inserting all insertable control rods within 2 hours. LS-AA-1110 requires ENS notification of the NRC due to a shutdown required by Technical Specifications.

B. Incorrect – ENS notification of the NRC is required. Plausible because there is no specific 10 CFR 50.72 reporting requirement for a Safety Limit violation.
C. Incorrect – Further power reduction is required. Plausible because the required power reduction in AOP-27 has been satisfied and MCPR raised enough to restore compliance with the Safety Limit.
D. Incorrect – Further power reduction is required. Plausible because the required power reduction in AOP-27 has been satisfied and MCPR raised enough to restore compliance with the Safety Limit. ENS notification of the NRC is required. Plausible because there is no specific 10 CFR 50.72 reporting requirement for a Safety Limit violation.

Technical Reference(s): Technical Specification 2.0, LS-AA-1110

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02G 1.14

Question Source: Bank - SSES LOC27 NRC #85 (2015)

Question History: SSES LOC27 NRC #85 (2015)

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1 & 2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295010 AA2.01
	Importance Rating	3.8

High Drywell Pressure

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: †Leak rates

Proposed Question: #85

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

- A leak has developed in the Drywell.
- Input to the Drywell Floor Drain sumps has risen from 0.3 gpm to 3.8 gpm in the last hour.
- Drywell pressure is 1.9 psig and stable.
- Drywell temperature is 130ºF and stable.

Note: Assume the leak rate remains constant and NO further information becomes available about the source of the leak.

With one of the following describes the need for a plant shutdown if this leakage remains at this level, in accordance with Technical Specifications?

Technical Specifications…

A. do NOT require a plant shutdown.

B. require a plant shutdown. The plant must be placed in Mode 3 within a maximum of 12 hours.

C. require a plant shutdown. The plant must be placed in Mode 3 within a maximum of 16 hours.

D. require a plant shutdown. The plant must be placed in Mode 3 within a maximum of 40 hours.

Proposed Answer: C

Explanation: Technical Specification 3.4.4 requires ≤ 2 gpm rise in unidentified leakage in a 24 hour period. Since unidentified leakage has risen by 3.5 gpm in a single hour, the Technical Specification limit has been exceeded. This requires entry into Condition B, which allows 4 hours to attempt to reduce the leakage. After this 4 hours, Condition C must be entered, which allows 12 hours to enter Mode 3, for a total of 16 hours.

A. Incorrect – A plant shutdown is required. Plausible because unidentified leakage remains below the limit of 5 gpm. Also plausible because this would be correct if additional information was given that the source of the leakage was not service sensitive type 304 or type 316 austenitic stainless steel.
B. Incorrect – A maximum of 16 hours is allowed. Plausible because this would be correct if evidence were given that this is pressure boundary leakage.
D. Incorrect – A maximum of 16 hours is allowed. Plausible because this is the correct time for entering Mode 4.

Technical Reference(s): Technical Specification 3.4.4

Proposed references to be provided to applicants during examination: Technical Specification 3.4.4

Learning Objective: SDLP-02A 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 #	1
	K/A #	262001 A2.06
	Importance Rating	2.9

AC Electrical Distribution

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

Proposed Question: #86

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

- A report has been received of smoke coming from MCC-253.
- The Unit Supervisor has directed opening the supply breaker to de-energize MCC-253.

Which one of the following identifies a load that will lose power as a result of this operation and the correct implementation of Technical Specifications (TS)?

	Load	TS Implementation
A.	Standby Gas Treatment Fan A	Enter TS 3.8.7, Distribution Systems – Operating, Condition A
B.	Control Room Emergency Ventilation Fan A	Enter TS 3.8.7, Distribution Systems – Operating, Condition A
C.	Standby Gas Treatment Fan A	Enter TS 3.6.4.3, Standby Gas Treatment (SGT) System, Condition A
D.	Control Room Emergency Ventilation Fan A	Enter TS 3.7.3, Control Room Emergency Ventilation Air Supply (CREVAS) System, Condition A

Proposed Answer: D

Explanation: De-energizing MCC-253 results in loss of power to Control Room Emergency Ventilation Fan A. Technical Specification 3.8.7 bases only require condition entry for loss of upstream electrical buses (L-25 or 10500), but not for just loss of the MCC. Rather, individual system Technical Specifications are entered for this power loss, including Technical Specification 3.7.3 Condition A for loss of one Control Room Emergency Ventilation Fan.

A. Incorrect – Standby Gas Treatment Fan A does not lose power. Plausible because this would be correct for loss of MCC-151. Technical Specification 3.8.7 bases do not require condition entry for loss of this MCC. Plausible because this would be correct for loss of the upstream electrical board.
B. Incorrect – Technical Specification 3.8.7 bases do not require condition entry for loss of this MCC. Plausible because this would be correct for loss of the upstream electrical board, L-25.
C. Incorrect – Standby Gas Treatment Fan A does not lose power. Plausible because this would be correct for loss of MCC-151.

Technical Reference(s): AOP-18B, Technical Specification 3.8.7 and bases, Technical Specification 3.7.3

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	203000 2.1.31
	Importance Rating	4.3

RHR/LPCI: Injection Mode

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

Proposed Question: #87

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

- The Reactor is scrammed.
- A cooldown is commenced.
- Injection from Core Spray and RHR is terminated and prevented.
- The RHR grayboot connectors are disconnected per EP-5, Termination and Prevention of RPV Injection.
- RHR A is aligned as shown on the next page.
- RHR B is NOT available.

Then, the coolant leak degrades and results in the following:

- Reactor water level is 80” and slowly lowering.
- Reactor pressure is 750 psig and slowly lowering.
- Drywell pressure is 14 psig and slowly lowering.
- The alternate level control leg of EOP-2, RPV Control, has been entered.
- Both Core Spray systems are available.
- All Condensate Booster pumps are available.

Which one of the following describes:

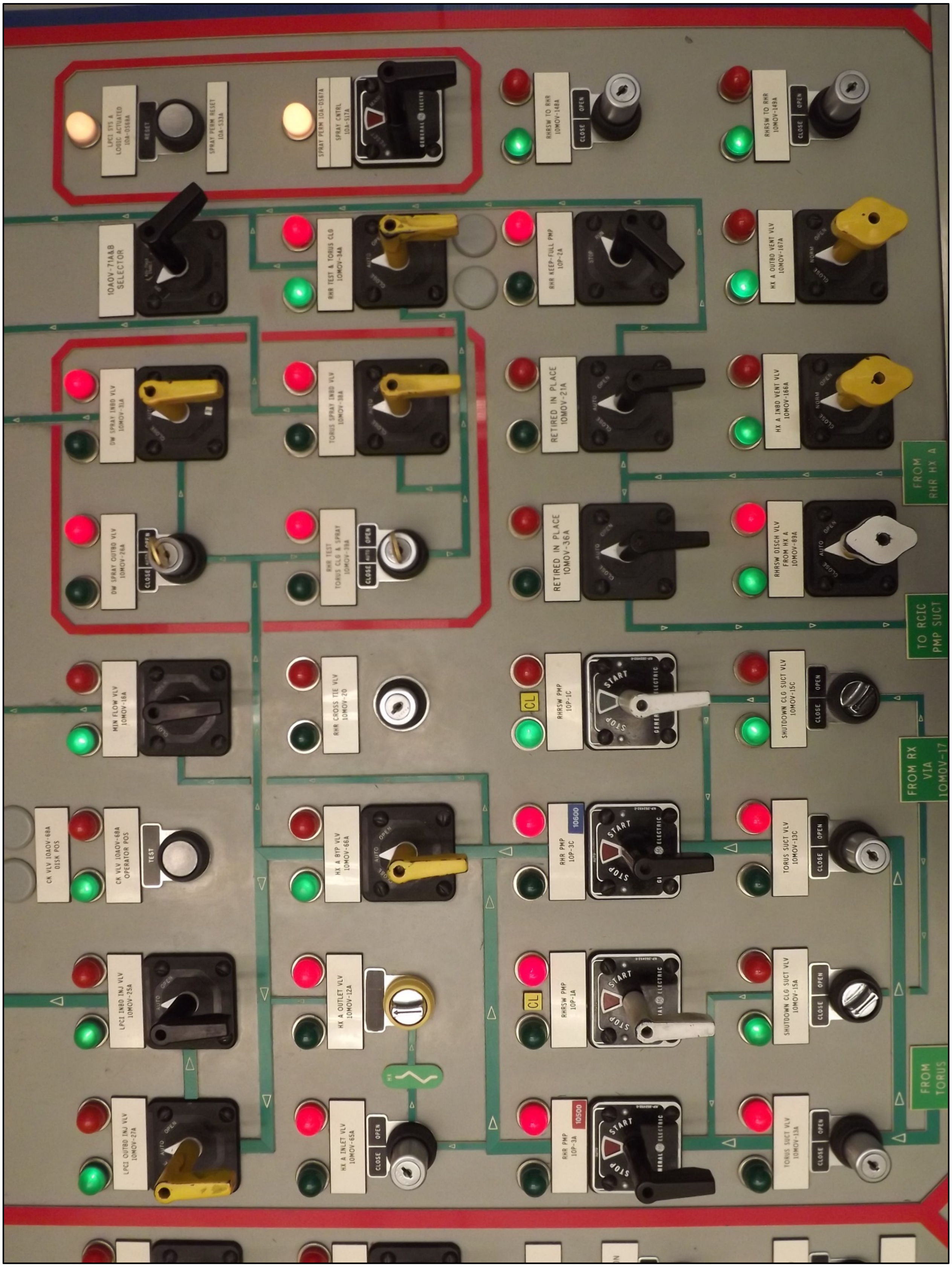
(1) the required control of RHR A sprays

and

(2) the required control of 10MOV-27A, LPCI OUTBD INJ VLV,

in accordance with the Emergency Operating Procedures?

	RHR A Sprays	10MOV-27A
A.	May be left in the current lineup	May be left in the current position
B.	May be left in the current lineup	Must be repositioned
C.	Must be realigned	May be left in the current position
D.	Must be realigned	Must be repositioned


10AOV-71A&B SELECTOR
RETIRED IN PLACE 10MOV-21A
RETIRED IN PLACE 10MOV-36A
FROM RHR HX A
TO RCIC PMP SUCT
FROM RX VIA 10MOV-17
FROM TORUS

Proposed Answer: A

Explanation: The given indications show that the RHR A sprays are in service and 10MOV-27A is closed as part of an earlier terminate/prevent. Reactor water level is lowering but still above the top of active fuel. The alternate level control leg of EOP-2 is applicable, as well as EOP-4. Until Reactor water level drops to top of active fuel, there is no requirement to realign 10MOV-27A for injection. Additionally, there is no requirement to take RHR A out of sprays prior to reaching top of active fuel.

B. Incorrect – There is no requirement to realign 10MOV-27A for injection. Plausible because this may become required if Reactor water level lowers further.
C. Incorrect – There is no requirement to take RHR A out of sprays. Plausible because Drywell pressure has already been significantly lowered by sprays and Reactor water level is continuing to lower, which may require LPCI injection.
D. Incorrect – There is no requirement to take RHR A out of sprays. Plausible because Drywell pressure has already been significantly lowered by sprays and Reactor water level is continuing to lower, which may require LPCI injection. There is no requirement to realign 10MOV-27A for injection. Plausible because this may become required if Reactor water level lowers further.

Technical Reference(s): EOP-2, EOP-4, EP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 A2.01
	Importance Rating	3.8

Standby Liquid Control

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

Proposed Question: #88

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

Time (minutes)	Condition(s)
0	• Reactor power is 25%. • Reactor pressure is 920 psig.
3	• Standby Liquid Control (SLC) pump A is started. • Initial SLC tank level is 90%.
30	• SLC pump A trips.
32	• An Operator attempts to start SLC pump B. • SLC pump B fails to start. • All APRMs now indicate downscale. • All IRMs indicate mid-scale on range 3. • Multiple control rods remain fully withdrawn.

Note: Assume SLC pump A operated at exactly the design flow rate.

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

A Reactor cooldown is currently…

A. allowed based on Reactor power indications, only.

B. allowed based on the amount of boron injected, only.

C. NOT allowed because Reactor power is too high.

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

Proposed Answer: D

Explanation: SLC pump A injected for 27 minutes at 50 gpm, which results in injection of 1350 gallons of boron solution. This will result in a final tank level of approximately 63% (a 27% reduction). This is greater than the hot shutdown boron weight (26% of tank level), but less than the cold shutdown boron weight (45% of tank level). With some boron injected, EOP-3 does not allowed cooldown until the cold shutdown boron weight is injected. Therefore, cooldown is not currently allowed due to the amount of boron injected.

A. Incorrect – Reactor cooldown is not currently allowed. Plausible because a substantial amount of boron has been injected and power has lowered significantly.
B. Incorrect – Reactor cooldown is not currently allowed. Plausible because a substantial amount of boron has been injected and power has lowered significantly.
C. Incorrect – The reason is not Reactor power. Plausible because Reactor power is still not fully downscale on the IRMs (although APRMs are downscale, IRMs are still indicating above Range 1). Also plausible because this decision is made based on Reactor power if no boron is injected yet.

Technical Reference(s): OP-17, EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.07

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 1/30/20 – Added “only” to end of A and B and added to C justification based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215005 A2.07
	Importance Rating	3.4

Average Power Range Monitor/Local Power Range Monitor

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation flow channels flow mismatch

Proposed Question: #89

A plant startup is in progress with the following:

- The Reactor Mode Switch is in ***START & HOT STBY***.
- Recirculation flow unit A is failed upscale and bypassed.
- Then, Recirculation flow unit C fails upscale.

Given the following:

- 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: A

Explanation: With both Recirculation flow unit A bypassed and C failed upscale, APRMs A, C, and E are provided a non-conservative Recirculation flow signal. This makes the flow biased scram function of these APRMs inoperable. With the Reactor Mode Switch in START/HOT STBY, TS 3.3.1.1 requires 2 operable APRMs per trip system to initiate an upscale trip ≤15%. The flow biased scram is not required to be operable. Therefore, TS 3.3.1.1 entry is not required. 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%. The flow biased rod block is not required to be operable. Therefore, TRM 3.3.B entry is not required.

B. Incorrect – TS 3.3.1.1 entry is not required. Plausible because this would be correct if the Reactor Mode Switch was in RUN.
C. Incorrect – TRM 3.3.B entry is not required. Plausible because this would be correct if the Reactor Mode Switch was in RUN.
D. Incorrect – TS 3.3.1.1 entry is not required. Plausible because this would be correct if the Reactor Mode Switch was in RUN. TRM 3.3.B entry is not required. Plausible because this would be correct if the Reactor Mode Switch was in RUN.

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: Modified Bank – 16-1 NRC #90

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

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

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	209001 2.4.9
	Importance Rating	4.2

Low Pressure Core Spray

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

Proposed Question: #90

A plant startup is in progress when a loss of coolant accident results in the following:

- Reactor water level is -19" and slowly lowering.
- Reactor pressure is 210 psig and slowly lowering.
- Core Spray pumps are injecting.
- Core Spray A flow is 3500 gpm.
- Core Spray B flow is 4200 gpm.
- NO other injection systems are available.
- NO SRVs have been opened.
- The Alternate Level Control leg of EOP-2, RPV Control, is being executed.

Which one of the following describes the required action, in accordance with EOP-2?

A. Exit the EOPs and enter the SAOGs.

B. Enter the Emergency RPV Depressurization leg of EOP-2.

C. Rapidly depressurize the Reactor using Turbine Bypass Valves.

D. Minimize steam loads and attempt to stabilize Reactor pressure.

Proposed Answer: B

Explanation: Reactor water level is less than 0” and Emergency RPV Depressurization has not yet been performed, as evidences by no SRVs being opened and the given place in EOP-2. Reactor pressure is low enough for some Core Spray injection based on the low starting pressure of the Reactor startup and the loss of coolant accident, however more Core Spray flow is available after Reactor pressure is lowered further. Based on the current position in EOP-2 and the given plant conditions, the required action is to enter the Emergency RPV Depressurization leg of EOP-2.

A. Incorrect – Exiting the EOPs and entering the SAOGs is not currently required. Plausible because this would be correct if an Emergency RPV Depressurization had already been performed and the given combination of Reactor water level and Core Spray flows was still present.
C. Incorrect – Entering the Emergency RPV Depressurization leg of EOP-2 is required, not rapid depressurization with TBVs. Plausible because the allowance to rapidly depressurize with TBVs is given in EOP-2 and would likely result in greater Core Spray flow.
D. Incorrect – Entering the Emergency RPV Depressurization leg of EOP-2 is required. Plausible because this is the strategy that would be employed if NO injection sources were available in the Steam Cooling leg.

Technical Reference(s): EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.07

Question Source: Bank - NMP1 2018 NRC #90

Question History: NMP1 2018 NRC #90

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 #	290002 2.2.25
	Importance Rating	4.2

Reactor Vessel Internals

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question: #91

Technical Specification 3.4.2, Jet Pumps, requires all jet pumps to be operable.

Which one of the following describes the basis behind this requirement, in accordance with Technical Specifications?

A. Prevent oscillations in core flow.

B. Prevent high vibration levels in the Reactor.

C. Ensure MCPR thermal limit assumptions are met.

D. Ensure core flooding to 2/3 core height during an accident.

Proposed Answer: D

Explanation: TS 3.4.2 bases state the concern with an inoperable jet pump is a lower core flooding elevation, which could adversely affect the water level in the core during the re-flood phase of a LOCA.

A. Incorrect – The basis is related to core flooding elevation, not core flow oscillations. Plausible because an inoperable jet pump could lead to abnormal Recirc and core flow characteristics, which would also be undesirable.
B. Incorrect – The basis is related to core flooding elevation, not vibration levels in the Reactor. Plausible because an inoperable jet pump could lead to higher vibration levels in the Reactor, which would also be undesirable.
C. Incorrect – The basis is related to core flooding elevation, not MCPR assumptions. Plausible because this is part of the basis for other inoperable components (pressure regulator, TBVs).

Technical Reference(s): Technical Specification 3.4.2 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS02 1.05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 1/30/20 – Replaced question based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	234000 A2.01
	Importance Rating	3.7

Fuel Handling Equipment

Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Interlock failure

Proposed Question: #92

A refueling outage is in progress with the following:

- The Reactor Mode Switch is in ***REFUEL***.
- A fuel bundle is latched on the Refuel Bridge main hoist.
- The Refuel Bridge is located over the center of the Reactor.
- The Refuel Bridge main hoist is in the normal-up position.

Then, the following occurs:

- Control rod 02-43 is inadvertently given a withdraw signal.
- The control rod block interlock fails and control rod 02-43 withdraws to position 06.

Which one of the following describes:

(1) whether another interlock exists that should prevent lowering the fuel bundle into the core,

and

(2) whether fuel movements into the core are allowed to continue with the failed control rod block interlock, in accordance with Technical Specifications?

	(1) Does another interlock exist that should prevent lowering the fuel bundle Into the core?	(2) Are fuel movements into the core allowed to continue with the failed control rod block interlock?
A.	No	Yes, provided that all rods, including 02-43, are fully inserted and a control rod block is inserted.
B.	No	No, the interlock must be fixed before any fuel can be moved into the core.
C.	Yes	Yes, provided that all rods, including 02-43, are fully inserted and a control rod block is inserted.
D.	Yes	No, the interlock must be fixed before any fuel can be moved into the core.

Proposed Answer: C

Explanation: The fuel movement will still be interrupted by an interlock before the fuel bundle can be placed into the core. This interlock prevents lowering the main hoist while over the core with a fuel bundle with a control rod not fully inserted. Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock, but only if all control rods are fully inserted and a control rod block is inserted.

A. Incorrect – The fuel movement will still be interrupted by an interlock before the fuel bundle can be placed into the core. Plausible because not all interlocks have such a backup.
B. Incorrect – The fuel movement will still be interrupted by an interlock before the fuel bundle can be placed into the core. Plausible because not all interlocks have such a backup. Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock. Plausible because one of the Required Actions is to immediately suspend in-vessel fuel movements.
D. Incorrect – Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock. Plausible because one of the Required Actions is to immediately suspend in-vessel fuel movements.

Technical Reference(s): OP-66A, Technical Specification 3.9.1

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank – 17-1 NRC #31

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Fuel Handling Equipment

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

Proposed Question: #31

A refueling outage is in progress with the following:

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

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

This fuel movement would first be interrupted by an interlock…

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

B. after the Refuel Bridge begins moving towards the core, but before it gets above the core.

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

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

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	202002 2.2.40
	Importance Rating	4.7

Recirculation Flow Control

Ability to apply Technical Specifications for a system.

Proposed Question: #93

The plant is operating at 95% power with the following:

- Recirculation pump A flow has spuriously drifted lower.
- Recirculation pump A loop flow is 32.3 Mlbm/hr and stable.
- Recirculation pump B loop flow is 38.0 Mlbm/hr and stable.

Considering the following Technical Specifications:

- 3.4.1, Recirculation Loops Operating
- 3.4.2, Jet Pumps

Which one of the following identifies which of these Technical Specifications (TS), if any, currently require Condition entry?

A. NEITHER TS 3.4.1 NOR TS 3.4.2 requires Condition entry.

B. TS 3.4.1 requires Condition entry, but TS 3.4.2 does NOT.

C. TS 3.4.2 requires Condition entry, but TS 3.4.1 does NOT.

D. Both TS 3.4.1 and TS 3.4.2 require Condition entry.

Proposed Answer: B

Explanation: The given flows show greater than 5% Recirculation loop jet pump flow mismatch with total core flow >70% of rated. This in UNSAT per Surveillance Requirement 3.4.1.2 and requires Condition entry in Technical Specification 3.4.1, but not 3.4.2.

A. Incorrect – Technical Specification 3.4.1 Condition entry is required. Plausible because mismatch is <10% and would not require Condition entry if total core flow was <70% of rated.
C. Incorrect – Technical Specification 3.4.1 Condition entry is required. Plausible because mismatch is <10% and would not require Condition entry if total core flow was <70% of rated. Technical Specification 3.4.2 Condition entry is NOT required. Plausible because other jet pump / Recirculation conditions do required entry into this related Technical Specification.
D. Incorrect – Technical Specification 3.4.2 Condition entry is NOT required. Plausible because other jet pump / Recirculation conditions do required entry into this related Technical Specification.

Technical Reference(s): OP-27, Technical Specifications 3.4.1 and 3.4.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.16

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Proposed Question: #94

Redacted due to containing Confidential / Proprietary information

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.41
	Importance Rating	3.7

Knowledge of the refueling process.

Proposed Question: #95

A refueling outage is in progress.

Which one of the following Refuel Floor activities must be directly supervised by a licensed Senior Reactor Operator (SRO), in accordance with OSP-66.001, Management of Refueling Activities?

A. Cleaning Recirc jet pumps in the Reactor vessel.

B. Loading a new fuel bundle into the Spent Fuel Pool.

C. Removing a control rod from a core cell containing fuel.

D. Moving LPRM strings from the core to the Spent Fuel Pool.

Proposed Answer: C

Explanation: Removing a control rod blade from a core cell that contains fuel meets the definition of a Core Alteration. All Core Alterations must be directly supervised by a licensed Senior Reactor Operator.

A. Incorrect – This does not required direct supervision by an SRO. Plausible because Recirc jet pumps are reactivity related components, so improper maintenance could have a negative effect on reactivity management once the plant is returned to power.
B. Incorrect – This does not required direct supervision by an SRO. Plausible because loading a new fuel bundle into the Reactor does require direct supervision by an SRO.
D. Incorrect – This does not required direct supervision by an SRO. Plausible because LPRMs are a specific exception discussed in OSP-66.001.

Technical Reference(s): OSP-66.001

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 73.03

Question Source: Bank – Peach Bottom 2017 NRC SRO #24

Question History: Peach Bottom 2017 NRC SRO #24

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.7
	Importance Rating	3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question: #96

Planning is underway for a Special Test or Evolution.

Given the following items:

(1) Assignment of a specific Coordinator for the evolution.
(2) Infrequent Plant Activity briefing with participation by a Senior Line Manager.
(3) Continuous oversight of evolution by either the Plant Manager or Site Vice President.

Which one of the following identifies which of these items is specifically required for the Special Test or Evolution, in accordance with OP-AA-108-110, Evaluation of Special Tests or Evolutions?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: OP-AA-108-110 requires assignment of a specific Coordinator (called the Special Test or Evolution Coordinator) and conduct of an Infrequent Plant Activity briefing with participation by a senior line manager. OP-AA-108-110 requires ensuring appropriate station management involvement is present during the evolution, but does not specifically require continuous oversight by the Plant Manager or Site Vice President.

B. Incorrect – OP-AA-108-110 requires conduct of an Infrequent Plant Activity briefing with participation by a senior line manager. Plausible that a different level of briefing, such as the Heightened Level of Awareness brief, would be required, or that lower level management involvement would be enough. OP-AA-108-110 requires ensuring appropriate station management involvement is present during the evolution, but does not specifically require continuous oversight by the Plant Manager or Site Vice President. Plausible that the highest level of station management would be required due to the risk involved with these evolutions and past industry events.
C. Incorrect – OP-AA-108-110 requires assignment of a specific Coordinator (called the Special Test or Evolution Coordinator). Plausible that normal Operations leadership (SM/US) would suffice due to their normal high level of responsibility for station operations. OP-AA-108-110 requires ensuring appropriate station management involvement is present during the evolution, but does not specifically require continuous oversight by the Plant Manager or Site Vice President. Plausible that the highest level of station management would be required due to the risk involved with these evolutions and past industry events.
D. Incorrect – OP-AA-108-110 requires ensuring appropriate station management involvement is present during the evolution, but does not specifically require continuous oversight by the Plant Manager or Site Vice President. Plausible that the highest level of station management would be required due to the risk involved with these evolutions and past industry events.

Technical Reference(s): OP-AA-108-110

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP2 2014 Cert #95

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.18
	Importance Rating	3.9

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

Proposed Question: #97

The plant is shutdown for a refueling outage.

Which one of the following describes the minimum requirements for review of the outage risk assessment against actual plant conditions, in accordance with AP-10.09, Outage Risk Assessment?

At least once per…

A. shift by a minimum of two (2) individuals

B. shift by a minimum of one (1) individual

C. day by a minimum of two (2) individuals

D. day by a minimum of one (1) individual

Proposed Answer: A

Explanation: AP-10.09 requires the Safety Shutdown Manager to verify the current plant conditions are in compliance with the outage risk assessment at least once each shift. AP-10.09 also requires a Shift Manager to independently review this verification at least once each shift. Therefore, at least two (2) individuals must perform the review.

B. Incorrect – AP-10.09 requires the review by two (2) individuals. Plausible because this is review of an already developed plan.
C. Incorrect – AP-10.09 requires the review at least once per shift. Plausible because many reviews / surveillances are daily.
D. Incorrect – AP-10.09 requires the review at least once per shift. Plausible because many reviews / surveillances are daily. AP-10.09 requires the review by two (2) individuals. Plausible because this is review of an already developed plan.

Technical Reference(s): AP-10.09

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 41.03b

Question Source: Bank – 14-2 NRC #95

Question History: 14-2 NRC #95

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 #	2.3.12
	Importance Rating	3.7

Knowledge of radiological safety principles pertaining to licensed operator duties, such as 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 RP-AA-460-001, Controls for Very High Radiation 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. Radiation Protection Manager and either the Site Vice President or Plant Manager.

Proposed Answer: D

Explanation: RP-AA-460-001 provides the required controls for personnel entry to a VHRA. Specific approval is required from both the Radiation Protection Manager and either the Plant Manager or Site Vice President.

A. Incorrect – Specific approval is required from both the Radiation Protection Manager and either the Plant Manager or Site Vice President. Plausible because this would be typical for lower level issues.
B. Incorrect – Specific approval is also required from either the Plant Manager or Site Vice President. Plausible because this would be typical for lower level issues.
C. Incorrect – Specific approval is also required from either the Plant Manager or Site Vice President. Approval from the Shift Manager is not explicitly required. Plausible because this would be typical for lower level issues.

Technical Reference(s): RP-AA-460-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – 16-1 NRC #98

Question History: 16-1 NRC #98

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:
TRH 1/30/20 – Replaced question based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.18
	Importance Rating	4.0

Knowledge of the specific bases for EOPs.

Proposed Question: #99

The plant is in a refueling outage with the following:

- The plant is in Mode 5.
- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- The Spent Fuel Pool area radiation monitor, 18RIA-051-12, has exceeded the Maximum Normal reading but is below the Maximum Safe reading.

Which one of the following describes the entry requirement for EOP-5, Secondary Containment Control?

A. EOP-5 entry is required to limit radioactivity release to and from the Reactor Building.

B. EOP-5 entry is NOT required because the Reactor coolant system is in a low energy state.

C. EOP-5 entry is NOT required because the cause of the entry condition is known and it is NOT a primary system discharging into the Reactor Building.

D. EOP-5 entry is NOT required because the bases do NOT address a dropped fuel bundle and assume such an accident is handled using AOP-44, Dropped Fuel Assembly.

Proposed Answer: B

Explanation: The given area radiation monitor condition meets an EOP-5 entry condition. However, EOP bases specifically do NOT require the EOPs to be entered if Reactor coolant temperature is less than 212°F and a Reactor startup or shutdown is NOT in progress. Since the plant is in Mode 5, Reactor coolant temperature is less than 212°F, and a Reactor startup or shutdown is NOT in progress. Therefore EOP-5 entry is NOT required.

Note: While the question deals with EOP entry, it rises to the SRO level because it is not straight memory of EOP entry condition, but also knowledge of bases behind EOP entry requirements ("knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures").

A. Incorrect – Since the plant is in Mode 5, Reactor coolant temperature is less than 212oF and a Reactor startup or shutdown is NOT in progress. Therefore EOP-5 entry is NOT required.
C. Incorrect – EOP-5 bases state that the purpose of the high area radiation level entry condition is to detect and deal with a leak from a primary system discharging into the Reactor Building. However, there is no guidance to NOT enter the EOP if this cause can be ruled out.
D. Incorrect – EOP-5 bases do NOT specifically cover a dropped fuel bundle, however there is no guidance to NOT enter the EOP due to a dropped fuel bundle. If the dropped fuel bundle occurred in the Spent Fuel Pool and caused this high radiation level while the plant was operating at power, EOP-5 entry would be required. AOP-44 would then be executed in parallel with EOP-5.

Technical Reference(s): EOP-5, MIT-301.11F, EP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11F 1.04

Question Source: Bank - 14-1 NRC #80

Question History: 14-1 NRC #80

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 #	2.4.30
	Importance Rating	4.1

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Proposed Question: #100

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

- A plant transient with multiple system malfunctions occurs.
- Plant conditions result in the Shift Manager declaring a Site Area Emergency (SAE).

Which one of the following describes the time requirement for notifying the County, State, and NRC, in accordance with EP-CE-114-100, Offsite Notifications?

The County and State must be notified within ___(1)___ of the SAE declaration.

The NRC must be notified immediately after notification of the County and State, not to exceed ___(2)___ from the SAE declaration.

	(1)	(2)
A.	15 minutes	1 hour
B.	15 minutes	30 minutes
C.	30 minutes	1 hour
D.	30 minutes	30 minutes

Proposed Answer: A

Explanation: The County and State must be notified through transmittal of the Part 1 notification within 15 minutes of the emergency declaration. The NRC must be notified through transmittal of the NRC Event Notification Worksheet immediately after notification of the County and State and not later than 1 hour after the emergency declaration.

B. Incorrect – The NRC notification is required within a maximum of 1 hour, not 30 minutes, such as if the two notifications each were given a 15 minute maximum.
C. Incorrect – The County and State notifications are required within 15 minutes of declaration, not 30 minutes. 30 minutes is the maximum time from when conditions requiring declaration are present in the Control Room (15 minutes max for declaration + 15 minutes max for notification).
D. Incorrect – The County and State notifications are required within 15 minutes of declaration, not 30 minutes. 30 minutes is the maximum time from when conditions requiring declaration are present in the Control Room (15 minutes max for declaration + 15 minutes max for notification). The NRC notification is required within a maximum of 1 hour, not 30 minutes, such as if the two notifications each were given a 15 minute maximum.

Technical Reference(s): EP-CE-114-100

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.1 EO 4.01

Question Source: Bank – 14-2 NRC #98

Question History: 14-2 NRC #98

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments: