

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000007A201	
		Ability to determine and interpret the following as they apply to a reactor trip: - Decreasing power level, from available indications	
	Importance	4.1	4.3

Question # 1

Given:

- Unit 2 was operating at 100% when the Main Turbine tripped, causing a reactor trip.

Which of the following conditions will indicate to the RO that the reactor is tripped while performing the Immediate Actions?

- A. Intermediate Range SUR $-.3$ dpm, Power Range NI 4%.
- B. Source Range SUR $-.3$ dpm, Intermediate Range SUR $-.3$ dpm.
- C. Source Range NI 10^4 cps, Intermediate Range NI 10^{-5} amps, Power Range NI 3%.
- D. Power Range NI 6%, Intermediate Range SUR $-.3$ dpm, Intermediate Range NI 10^{-5} amps.

Answer: A

Explanation/Justification:

A. Correct. Following the prompt drop from 100% power, Intermediate Range will indicate a -1/3 dpm startup rate and Power Range will be less than 6%.

B. Incorrect but plausible because the -1/3 dpm SUR is correct for the Intermediate; however, the source range NIs will not be energized.

C. Incorrect but plausible because indications for source and intermediate ranges are approximately equivalent; however the source range would not be energized. The power range indication is consistent with a reactor trips prompt drop.

D. Incorrect but plausible because expected while power is significantly reduced, it is greater than 5% indication used for reactor trip

Technical References: 2-E-0

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE00 1

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000008K101	
		Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: - Thermodynamics and flow characteristics of open or leaking valves	
	Importance	3.2	3.7

Question # 2

Given:

- Unit 2 is operating at 100% power steady state when a small leak develops on a PZR PORV.
- After the PZR Pressure Control system returns PZR pressure to normal, the following conditions exist:

PZR pressure: 2235 psig

PRT pressure: 5.3 psig

PRT temperature: 90° F

Determine the temperature downstream of the leaking PORV PRIOR to closing the PORV Block Valve.

- A. 162° F
- B. 212° F
- C. 228° F
- D. 240° F

Answer: C

Explanation/Justification:

Constant enthalpy. After being throttled across the PORV, the downstream temperature will be at saturation for the pressure it is at, which is PRT pressure. 5.3 psig + 20 psia.

Can either use the Mollier Diagram to intersect the 20 psia line with the Saturation Curve or go right to 20 psia in the Table 1 - Saturated Steam: Temperature Table and find 228.

Technical References: Steam Tables

Proposed References to be provided: None

Learning Objective

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	0000092107	
		Conduct of Operations - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	
	Importance	4.4	4.7

Question # 3

The unit is at 100% power with all systems aligned for normal operations. Pressurizer level starts to drop and auto makeup to the VCT starts. The CRS enters AOP-LEAK-1 "Sudden Increase in RCS Leakage". 21 Charging Pump is in service and speed increases to the maximum output. The CRS has the ATC start 22 Charging Pump and raise speed to maintain PRZR level. The ATC reports that 22 Charging Pump is now at maximum speed. The 45 gpm Orifice is in service. Pressurizer level is still dropping.

Which of the following actions should be carried out next?

- A. Isolate letdown and start the 23 Charging Pump.
- B. Trip the reactor, initiate SI and GO TO E-0.
- C. Initiate a POP Plant Shutdown to zero power conditions.
- D. Align Charging Pump suction to the RWST.

Answer: B

Explanation/Justification:

A. Incorrect but plausible because isolating letdown would reduce mass out of the RCS and starting 23 charging pump would add mass to the RCS; however, AOP-LEAK-1 directs reactor trip and SI.

B. Correct. If pressurizer level cannot be maintained with 2 charging pumps and 45 gpm orifice AOP-LEAK-1 directs reactor trip and SI.

C. Incorrect but plausible because a plant shutdown would place less of a transient on the RCS; however this is not what the AOP directs.

D. Incorrect but plausible because operating 3 charging pumps will exceed the capability of the VCT makeup system.

Technical References: 2-AOP-LEAK-1
Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPLEK 3

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000011A115	
		Ability to operate and/or monitor the following as they apply to a Large Break LOCA: - RCS temperature and pressure	
	Importance	4.2	4.2

Question # 4

Which of the following are criteria in E-1, Loss of Reactor or Secondary Coolant, used for determining that a LBLOCA has occurred versus a SBLOCA?

- A. RCS pressure \leq 320 psig and VC sump level increasing
- B. VC sump level increasing and R-7, Incore Instrument Room \geq 20 mr/hr
- C. RCS pressure \leq 320 psig and RHR flow \geq 240 gpm
- D. RHR flow \geq 240 gpm and R-7, Incore Instrument Room \geq 20 mr/hr

Answer: C

Explanation/Justification:

Technical References: 2-E-0

Proposed References to be provided: None

Learning Objective I2LP-ILO-SIS01 7

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000022A201	
		Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: - Whether charging line leak exists	
	Importance	3.2	3.8

Question # 5

Given the following condition:

- The plant is operating at 100% power

Subsequently, over a 15-minute period, the Team observes the following indications:

Parameter	Initial	Final
Tave	558.8°F	558.8°F
VCT Level	32%	29%
Pressurizer Level	45%	43%
Containment Radiation Monitor R2	4 mr/hr	33 mr/hr
Non-Regenerative HX outlet temperature	90°F	90°F
Non-Regenerative HX outlet flow	75 gpm	63 gpm
Charging flow	85 gpm	87 gpm
Regenerative HX Charging outlet temperature	468 °F	367 °F
Charging Header pressure	2405 psig	2400 psig
Component Cooling Water Surge Tank level	55%	55%

Which of the following events is occurring?

- Letdown line leak upstream of Regen Heat Exchanger.
- Charging line leak upstream of Regen Heat Exchanger.

- C. Charging line leak downstream of Regen Heat Exchanger.
- D. Letdown line leak downstream of Non-Regen Heat Exchanger.

Answer: A

Explanation/Justification:

This question meets the KA because it requires the candidate to evaluate charging and letdown conditions and determine whether a charging or letdown leak exists.

A Correct . A letdown leak upstream of Regen Hx (RHX) is inside containment resulting in increased Rad levels. NRHX outlet temperature is controlled by controlling CCW to NRXH thus not affected by flow of water through Hx. NRHX outlet flow will decrease due to leak. Charging flow will increase due to reduction in PRZR level. RHX outlet temperature is lower because less hot letdown is flowing through RHX and more charging flow resulting in lower temperature. Charging header pressure difference is not indicative of any leak.

B. With a charging line leak upstream of NRHX outlet flow would not decrease. In addition RHX outlet temperature would increase not decrease due to lower charging flow.

C. A charging line leak downstream of the RHX would not affect the NRHX outlet flow. Also slight increase in charging flow would not result in a 100° change in RHX charging outlet temp.

D. A letdown leak downstream of NRHX would not result in a 100° decrease in RHX charging outlet temperature. In addition the Containment radiation monitor would not increase.

Technical References: 2-AOP-LEAK-1
3-AOP-LEAK-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-CVCS 1

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000025K101	
		Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: - Loss of RHRS during all modes of operation	
	Importance	3.9	4.3

Question # 6

Given:

- Unit 2 is in MODE 5 with CETs at 195°F
- RHR is in service in Shutdown Cooling
- All RHR cooling is lost
- Neither RHR pump can be started

Which of the following conditions will prevent using Steam Generator(s) to remove RCS decay heat IAW 2-AOP-RHR-1, Loss of RHR?

- A. PZR level is 5%.
- B. Narrow Range Level is <0% on all SGs
- C. ALL MSIVs and Bypass valves are shut.
- D. ONLY one SG is intact with feed and bleed capability.

Answer: A

Explanation/Justification:

A. Correct. Pressurizer level must be maintained > 10% for the RCS and SGs to be considered "coupled".

B. Incorrect but plausible because applicant may believe that SG level is required to be a heat sink in AOP-RHR-1; however, only the capability of being feed with release path is required.

C. Incorrect but plausible because MSIVs and bypass valve provide a heat removal path to the condenser, but it is not necessary.

D. Incorrect but plausible because candidate may believe that one SG is not adequate as a heat sink.

Technical References: 2-AOP-RHR-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPRHR 3

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 4

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000027K203	
		Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: - Controllers and positioners	
	Importance	2.6	2.8

Question # 7

Given:

- The plant is operating at 99% power when the controlling Pressurizer Pressure Channel PT-455 fails low.
- The ATC Reactor Operator takes manual control of the Master Pressure Controller to control pressure at 2235 psig.

With NO other operator action taken, determine the PORV functionality for PCV-456 and PCV-455C.

- A. Both PORVs will open automatically when required.
- B. Only PORV PCV-456 will open automatically when required.
- C. Only PORV PCV-455C will open automatically when required.
- D. Neither PORV PCV-455C or PCV-456 will open automatically when required.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because in if the Defeat Switch was in Defeat Channel 1&4 this would be true.

B. Correct.

C. Incorrect. Plausible because one channel is controlled from the controlling channel and the other is not. Candidate must know which PORV is controlled from the controlling channel.

D. Incorrect. Plausible because if the defeat switch is in Defeat 1&4 and channel 457 failed low this would be true.

DWG 9371-F-33733

Technical References: Drawing
Syst Desc 1.4

Proposed References to be provided: None

Learning Objective I2LP-ILO-RCSPZR 5
I3LP-ILO-RCSPZR 5,6

Question Source: Bank

Question History: Unit 2 NRC 2016

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000029K306	
		Knowledge of the reasons for the following responses as they apply to the ATWS: - Verifying a main turbine trip; methods	
	Importance	4.2	4.3

Question # 8

Given:

- Unit 2 was operating at 100% power when a reactor trip signal failed to trip the reactor, and the reactor cannot be tripped.
- The crew has entered 2-FR-S.1 Response to Nuclear Power Generation / ATWS

Which of the following describes the required actions if the Stop Valves fail to close and the reason for this action?

- A. Close MSIVs, IF NOT successful runback the turbine to limit RCS pressure transient for all initiating conditions
- B. Close MSIVs, IF NOT successful runback the turbine to prevent an excessive cooldown
- C. Runback the turbine, IF NOT successful close MSIVs to limit RCS pressure transient for all initiating conditions
- D. Runback the turbine, IF NOT successful close MSIVs to prevent an excessive cooldown

Answer: D

Explanation/Justification:

A. Incorrect but plausible because this is the sequence/method used to verify turbine trip in ECA-0.0. Also tripping the turbine limits the pressure transient for loss of feedwater transients but increases pressure for other transients.

B. Incorrect but plausible because this is the sequence/method used to verify turbine trip in ECA-0.0. The reason for the actions is correct.

C. Incorrect but plausible because the sequence/method used is correct, but tripping the turbine limits the pressure transient for loss of feedwater transients but increases pressure for other transients.

D. Correct. This is the sequence listed in FR-S.1 and reason for the trip.

Technical References: 2-FR-S.1 BG

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPFRS 1

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000038K308	
		Knowledge of the reasons for the following responses as they apply to the SGTR: - Criteria for securing RCP	
	Importance	4.1	4.2

Question # 9

Given:

- A reactor trip with SI has occurred
- The Team has just entered E-3, "Steam Generator Tube Rupture", due to the presence of SGTR symptoms
- The ATC reports that the subcooling criteria for tripping RCPs has been met
- The Team verifies that Safety Injection pumps are running and stop all Reactor Coolant Pumps

The RCPs were stopped to...

- A. reduce flow-induced vibration in the ruptured SG U-tubes, thereby preventing the rupture from increasing in size.
- B. protect against a misdiagnosis of the event, operator error, or the occurrence of a multiple-failure event scenario.
- C. minimize the heat input into the RCS, thereby helping to reduce RCS pressure to equilibrium with secondary pressure.
- D. ensure adequate core cooling. Adequate core cooling cannot be maintained during a SGTR with reactor coolant pumps running.

Answer: B

Explanation/Justification:

A. Incorrect but plausible because candidate may believe that increasing the size of the rupture is the primary concern.

B. Correct

C. Incorrect but plausible because RCPs do add heat to the RCS and cooling down is a strategy for success with a SGTR.

D. Incorrect but plausible because RCPs are tripped during a SBLOCA to minimize mass loss and SG overfill is a concern during SGTR.

Technical References: 2-E-3

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE30 1

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000040K201	
		Knowledge of the interrelations between the Steam Line Rupture and the following: - Valves	
	Importance	2.6	2.5

Question # 10

Given:

- Unit 2 has experienced a Main Steamline break from 100% power.
- The Reactor was tripped, the Main Turbine was tripped, and SI was initiated.
- All MSIVs were shut in E-0, Reactor Trip or Safety Injection.

While isolating the faulted SG in E-2, Faulted Steam Generator Isolation, the operator reports:

- 22 SG level is 10% wide range and lowering.
- 22 SG pressure is 0 psig.
- The remaining SG NR levels are all 5% and rising.
- The remaining SG pressures are 980 and rising.

Which of the following identifies:

- (1) The required actions associated with AFW flow to the faulted SG?
- (2) How steam valves to Turbine Drive AFW pump should be positioned?

- A. (1) Isolate AFW flow to 22 SG
(2) Shut MS-41 (SG 22) ONLY
- B. (1) Isolate AFW flow to 22 SG
(2) Shut MS-41 (SG 22) AND MS-42 (SG 23)
- C. (1) Maintain AFW to 22 SG until at least one SG NR level is >10% in at least one SG.
(2) Shut MS-41 (SG 22) ONLY
- D. (1) Maintain AFW to 22 SG until at least one SG NR level is >10% in at least one SG.

(2) (2) Shut MS-41 (SG 22) AND MS-42 (SG 23)

Answer: A

Explanation/Justification:

On a steamline break, the faulted SG is isolated by, among other things, closing AFW supply valves and steam supply valves to TDAFW pump.

A Correct.

B Incorrect but plausible because applicant may believe that since the MDAFW pumps are operating and capable of feeding the intact SGs, 22 AFW pump is isolated from both steam supply sources. Also item (1) is correct.

C Incorrect but plausible if it is thought that at least one SG >10% is required before isolated AFW to faulted SG.

D Incorrect but plausible if it is thought that at least one SG >10% is required before isolated AFW to faulted SG. Also applicant may believe that since the MDAFW pumps are operating and capable of feeding the intact SGs, 22 AFW pump is isolated from both steam supply sources

Technical References: 2-E-2

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE20 1

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	0000542418	
		Emergency Procedures/Plan - Knowledge of the specific bases for EOPs.	
	Importance	3.3	4

Question # 11

Given:

- The unit was operating at 100% power.
- A malfunction in the lube oil system cause loss of both Main Boiler Feed Pumps
- The unit was manually tripped
- The crew is responding using E-0, Reactor Trip or Safety Injection.
- At step 4 Safety Injection is not required.

Which of the following describes the required AFW flowrate and the basis for this value?

- A. 400 gpm within 30 minutes to prevent a transition to FR-H.1.
- B. 400 gpm within 10 minutes to ensure adequate heat removal.
- C. 760 gpm within 30 minutes to prevent a transition to FR-H.1.
- D. 760 gpm within 10 minutes to ensure adequate heat removal.

Answer: D

Explanation/Justification:

From the Background Document:

If SI is not required, AFW pumps are started to satisfy SPU LONF/LOAC (Loss of normal feedwater/loss of non-vital AC) Analysis TA-03-126 requiring 760 gpm AFW flow within 10 minutes for heat removal if only ONE Motor Driven Pump Auto-Starts.

From NL-04-073:

Using Licensing assumptions, the additional AFW flow at 10 minutes is needed to prevent the pressurizer from going water solid.

A. Incorrect, Plausible because 400 gpm is the correct value if SI is ACTUATED and 30 minutes is the time to establish AFW during Appendix R scenario. Also removal of decay heat and RCP heat is adequate with 400 gpm AFW flow.

B. Incorrect, Plausible because 400 gpm is the correct value if SI is ACTUATED and it is to prevent water relief through the PORVs.

C. Incorrect, Plausible because 760 gpm flowrate is correct and 30 minutes is the time to establish AFW during Appendix R scenario.

D. Correct.

Technical References: 2-E-0 BG

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE00 1

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000055A102	
		Ability to operate and/or monitor the following as they apply to a Station Blackout: - Manual ED/G start	
	Importance	4.3	4.4

Question # 12

A unit trip occurred from 100% power

At step 3 of E-0, Reactor Trip or Safety Injection, the crew transitioned to ECA-0.0, Loss of All AC Power:

- 138 kV and 13.8 kV are unavailable.
- 21, 22, and 23 EDGs are not running.
- All 480 V buses are de-energized.

Which of the following actions is intended to start the emergency diesel generators?

- A. Manually actuate Safety Injection.
- B. Reset 480V Bus Lockout (86) relays.
- C. Depress Blackout logic reset pushbuttons.
- D. Place 480 V bus normal supply breakers in trip pullout.

Answer: A

Explanation/Justification:

- A. Correct. The intent of this action is to send a different start signal to the EDGs.
- B. Incorrect but plausible because the 480V bus lockout relay will prevent auto closure of the EDG breaker, but not auto start of the EDG.
- C. Incorrect but plausible because the Blackout Relay is reset in AOP-138kV-1 when prepared to restore normal power to a 480V bus
- D. Incorrect but plausible because candidate may believe that the normal feed breakers must be open to satisfy undervoltage start logic for the EDG. In addition many breaker switches are placed in Trip Pull Out in ECA 0.0.

Technical References: 2-ECA-0.0

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDGR 1

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000056K301	
		Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: - Order and time to initiation of power for the load sequencer	
	Importance	3.5	3.9

Question # 13

The unit was operating at 100%.

A unit trip occurred

All equipment operated as designed.

5 minutes later a fault occurred on the Station Aux Transformer.

Which of the following correctly states the order of and reasons for the response of Service Water, CCW, and AFW pumps when buses are re-energized?

- A. AFW pumps restart immediately due to SG Shrink (low level)
All SW pumps restart after short time delay to cool EDGs and Turbine Lube Oil
CCW pumps restart later to supply thermal barrier cooling
- B. AFW pumps restart immediately due to SG Shrink (low level)
Essential SW pumps restart after short time delay to cool EDGs
CCW pumps do not restart to prevent bus overload.
- C. CCW pumps restart first to supply thermal barrier cooling
Essential SW pumps restart after short time delay to cool EDGs
AFW pumps restart last to ensure adequate heat sink
- D. CCW pumps do not restart to prevent bus overload
All SW pumps restart after short time delay to cool EDGs and Turbine Lube Oil
AFW pumps restart last to ensure adequate heat sink.

Answer: C

Explanation/Justification:

IPEC has 2 trains of service water, but either train can be “Essential” depending on valve lineup and a switch position. It is plausible to believe that Turbine Lube Oil must be cooled to prevent major system damage (NOTE Unit 3 cools Turbine Lube Oil using Essential Service Water). Since Unit 2 cools Turbine Lube Oil with Non-Essential service water, it is plausible to believe that the Service Water train designated Non-Essential would start to cool Lube Oil.

A. Incorrect but plausible because the AFW pumps have an auto start on low SG level; however the low level start is blocked on a bus undervoltage/SI. Even though no accident exists only, Essential Service Water Pumps will restart after a time delay; also CCW does restart on a Blackout-Unit Trip. The sequence of starting is not correct.

B. Incorrect but plausible because the AFW pumps have an auto start on low SG level; however the low level start is blocked on a bus undervoltage/SI. Essential Service Water Pumps will restart after a time delay; also CCW does not restart on a Blackout within SI. The sequence of starting is not correct.

C. Correct. CCW pumps sequence on at 6 seconds, 9 seconds and 11 seconds. SW pumps start after 15 seconds. AFW pumps sequence on at 25 and 30 seconds.

D. Incorrect but plausible because CCW does not restart on a Blackout with an SI. . Even though no accident exists only, Essential Service Water Pumps will restart after a time delay. AFW pumps do start last. The sequence of starting is correct.

Technical References: Logic Unit 2 Sheet 8

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS01 11

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000057A206	
		Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: - AC instrument bus alarms for the inverter and alternate power source	
	Importance	3.2	3.7

Question # 14

Given:

- Unit 2 is operating at 18% power
- PRZR Level Control is selected to Channel 2
- The following alarm is received in the Control Room: SDF 1-4 "21 OR 22 INVERTER TROUBLE"
- The Team determines that NIS NI Rack C6 (N32, N36 & N42) is deenergized along with numerous control board indications
- An attempt to restore the inverter in accordance with 2-AOP-IB-1, Loss of Power to an Instrument Bus, by pressing the ALTERNATE SOURCE TO LOAD button was unsuccessful

Which one of the following actions is required?

- A. Restore 22 Vital Instrument Bus by aligning to MCC-29A SOLA Transformer; remove inverter from service and restore letdown.
- B. Restore 22 Vital Instrument Bus by bypassing the inverter; control plant parameters manually as necessary and restore letdown.
- C. Trip the reactor; enter E-0, Reactor Trip Or Safety Injection, only if boration rate from RWST exceeds capability to control prior to restoring bus in accordance with AOP-IB-1.
- D. Trip the reactor and enter E-0, Reactor Trip or Safety Injection. Post trip, restore letdown and defeat containment pressure high-high bistable channel 2 in accordance with 2-AOP-IB-1.

Answer: B

Explanation/Justification:

A. Incorrect but plausible because bus 5A has a 2 SOLA transformers. One transformer is from MCC-26A and it supplies 21/21A Instrument bus. The second is from MCC-29A that supplies 23 Instrument Bus not 22. Candidate must remember which Instrument Bus is supplied from MCC-29A.

B. Correct: Per the ARP and AOP the operators will attempt to manually transfer the inverter using the Alternate Source pushbutton. If that does not work the next action is to bypass the inverter which is the only correct answer given.

D. Incorrect: Neither a Trip or defeat of containment pressure bistable is required per AOP or ARP. The bistable fails to un-actuated state for containment pressure high-high. There is a trip required if <P-10 due to loss of IRNI.

Plausible: Containment pressure bistable is affected but if power can not be restored action is to align alternate power from "topaz" inverter not defeat the bistable.

C. Incorrect: Plausible because this is a trip criteria for loss of the 21 IB not 22.

Plausible: Candidate might not recall loss of power effects.

Technical References: 2-AOP-IB-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPIB1 3

Question Source: Bank

Question History: NRC Unit 2 2008

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	0000582420	
		Emergency Procedures/Plan - Knowledge of operational implications of EOP warnings, cautions, and notes.	
	Importance	3.8	4.3

Question # 15

A loss of all AC power occurred 15 minutes ago. The Appendix R Diesel is expected to be available to power 480V busses in 30 minutes. The BOP notes that 21 DC Voltage has dropped from 131V prior to the event to 126V. The team is concerned about a loss of vital AC instrument bus power due to depleting station batteries. Which of the following describes how this condition will be addressed by ECA-0.0, Loss of All AC Power?

- A. Since power will be restored in 30 minutes, no actions will be directed by ECA-0.0 to shed loads.
- B. ECA-0.0 will direct shedding loads using AOP-DC-1, Loss of 125V DC Panel, and AOP-IB-1, Loss of Power to an Instrument Bus.
- C. ECA-0.0 will direct shedding loads using SOP-27.1.6, Instrument Bus, DC Distribution System And PA System Inverter
- D. ECA-0.0 will direct initiation of FSG-004, ELAP DC Bus Load Shed/Management, and FSG-007, Loss of Vital Instrumentation or Control Power.

Answer: B

Explanation/Justification:

For this event an ELAP is not declared since power is expected to be restored within 1 hour. ECA-0.0 will direct use of the AOPs for load shed. There is an option on the foldout page to use the FSGs. However, the candidate has to recall that the note warns against using these actions when not in an ELAP. There is an allowance for SM discretion; however the voltage thresholds in the FSG are much lower than stated in the stem of the question. The simulator shows a 1V drop in DC voltage when the battery charger loses power, so the 5V drop listed in the stem is not unreasonable. This voltage drop makes gives more plausibility to use of FSGs as a distractor. The question tests knowledge of a note in the EOP regarding steps to preclude a loss of vital AC instrument power, so the KA is addressed.

A. Incorrect but plausible since the batteries are designed for 2 hours in these conditions and power will be restored within 60 minutes.

B. Correct answer.

C. Incorrect but plausible since numerous EOPs actions are carried out by SOPs (e.g. MCC reset).

D. Incorrect but plausible if a candidate does not remember the ELAP time requirement or forgets that the note on the foldout precludes use of FSGs for these conditions.

Technical References: 2-ECA-0.0

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPC00 3

Question Source: Bank

Question History: Unit 3 NRC 2017

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	00WE04K101	
		Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment: - Components, capacity, and function of emergency systems	
	Importance	3.5	3.9

Question # 16

Following a reactor trip:

- RCS pressure is 1600 psig and lowering.
- Pressurizer level 10% and slowly lowering.
- Pressurizer spray valves are closed
- PRT parameters are: 50% level, 4 psig, 115°F and steady
- 21-24 steam generator water levels are: 22%, 27%, 24%, 31% and steady.
- AFW flows are 100 gpm, 105 gpm, 110gpm, 120 gpm.
- Primary auxiliary building radiation monitors are rising.
- Plant ventilation radiation monitors are rising.
- Containment pressure and sump levels are normal.

Based on these conditions, which of the following has occurred and what procedure will be used to mitigate it?

- A. Faulted Steam Generator, E-2
- B. Ruptured Steam Generator, E-3
- C. LOCA Outside Containment, ECA-1.2
- D. LOCA Inside Containment, E-1

Answer: C

Explanation/Justification:

A. Incorrect: No indication of steam/feed leak. In particular AFW flow rates and SG levels are responding normally and a steam leak would not result in elevated radiation levels.

Plausible: A loss of secondary coolant could cause a lowering of RCS pressure.

B. Incorrect: No indication of tube rupture. In particular AFW flow rates and SG levels are responding normally

Plausible: A SGTR could cause low RCS pressure and some elevation of secondary radiation levels.

C. Correct: Elevated radiation levels outside the containment and lowering RCS pressure with no evidence of a SGTR leaves a LOCA outside containment.

D. Incorrect: Containment parameters are not indicative of a LOCA inside containment.

Plausible: Large LOCA's can elevate radiation levels outside containment.

This question meets the KA because:

The conditions given indicate that a small break LOCA has occurred outside containment.

RCS pressure 1600 psig is above the shutoff head of SI pumps – the SI pumps are not injecting resulting in lower PRZR level and pressure.

Spray valves closed indicate lowering pressure is NOT due to open spray valve.

PRT conditions are Normal indicating that a PORV/SAFETY valve has not lifted.

AFW flows and SG levels are used to faulted or ruptured SG. The flow rate and SG levels are consistent with a normal trip response.

Technical References: 2-E-0
2-E-1
2-ECA-1.2

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPC10 1

Question Source: Bank

Question History: Unit 2 NRC 2008

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 8

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	00WE05A103	
		Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: - Desired operating results during abnormal and emergency situations	
	Importance	3.8	4.2

Question # 17

Given:

- The unit was operating at 100% power
- The unit was tripped when both Main Boiler Feed Pumps tripped.
- AFW failed to actuate.
- The crew entered FR-H.1, Loss of Heat Sink.
- When conditions were met, the crew initiated Bleed and Feed cooling.

Which of the following gives the desired condition(s) to start the SI flow reduction sequence?

- A. When CETs < 355°F and lowering
- B. When level in at least 1 SG is \geq 10% NR.
- C. When feed flow to at least 1 SG is \geq 200 gpm.
- D. When RCS pressure is < 1500 psig and HHSI flow is indicated.

Answer: B

Explanation/Justification:

A. Incorrect but plausible because 345 is the temperature below which RHR can be placed in service.

B. Correct.

C. Incorrect but plausible because following a normal AFW actuation the SGs are feed at approximately 200 gpm until the operator takes control.

D. Incorrect but plausible because HHSI flow will start at approximately 1500 psig. That will not be an adequate heat sink.

Technical References: 2-FR-H.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPFRH 4

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	00WE11K202	
		Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following: - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	
	Importance	3.9	4.3

Question # 18

The operating crew is performing 2-ECA-1.1 for a loss of emergency coolant recirculation. The crew is depressurizing all intact SGs to inject SI accumulators. In 2-ECA-1.1, how are the accumulators controlled and at what pressure do we stop depressurizing SGs?

- A. Slowly Inject accumulators into the RCS
Stop depressurizing at 170 psig
- B. Rapidly Inject accumulators into RCS
Stop depressurizing at 170 psig
- C. Slowly Inject accumulators into the RCS
Stop depressurizing at 350 psig
- D. Rapidly Inject accumulators into RCS.
Stop depressurizing at 350 psig

Answer: A

Explanation/Justification:

A Correct.

B. Incorrect but plausible some EOPs rapidly depressurize to rapidly inject accumulators. ECA-1.1 slowly injects the accumulators. The stopping pressure is correct.

C Incorrect but plausible because ECA-1.1 does slowly inject the accumulators, but 350 is not the correct value however it is the correct value to put RHR in service

D Incorrect. Incorrect but plausible some EOPs rapidly depressurize to rapidly inject accumulators. ECA-1.1 slowly injects the accumulators. 350 is not the correct value however it is the correct value to put RHR in service

Technical References: 2-ECA-1.1
2-ECA-1.1 BG
Proposed References to be provided: None

Learning Objective I2LP-ILO-SIS01 2

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 8

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	000005K305	
		Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod: - Power limits on rod misalignment	
	Importance	3.4	4.2

Question # 19

Given:

- Unit 2 was performing a power reduction from 100% to 80% power.
- During the power reduction it was identified that a single control rod was misaligned from its group demand counter by more than the permissible amount in Tech Spec Table 3.1.4-1.
- A power reduction to 74% was performed to comply with LCO 3.1.4 Rod Group Alignment Limits, Action B.2.2.

Which of the following describes the bases for performing the Tech Spec required power reduction?

The power reduction was performed to ...

- A. prevent radial flux from exceeding the Tech Spec limit for QPTR.
- B. prevent Xenon oscillations from developing as power production shifts in the core.
- C. ensure that Peaking Factors will not cause core design criteria to be exceeded.
- D. ensure that safety analyses assumptions for SDM are preserved.

Answer: C

Explanation/Justification:

The K/A asks for knowledge of the reasons for the power limits on rod misaligned, not what the limit is.

A Incorrect but plausible because it is a possible result of having a rod misaligned however the power reduction for QPTR is based on 3% for every % QPTR exceeds 1.00.

B Incorrect but plausible because Xenon is localized concern for a dropped rod; however an oscillation is not expected.

C is correct per (B.3.1.4)

D is incorrect as it is the bases for Rod Insertion Limits. (B3.1.6)B is incorrect as it is not listed in the bases doc as a reason for reducing power .

From TS Basis:

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded.

Technical References: Tech Specs
Proposed References to be provided: None

Learning Objective I2LP-ILO-ICROD 10

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	000028K303	
		Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions: - False indication of PZR level when PORV or spray valve is open and RCS saturated	
	Importance	3.5	4.1

Question # 20

Given:

- A PZR Safety Valve has failed open
- Reactor Trip and Safety Injection have occurred.
- All equipment is operating as designed.
- RCS pressure is currently 1300 psig and lowering.
- The crew is performing actions of 2-E-0, Reactor Trip or Safety Injection
- PZR Level is 100%

Which ONE of the following describes the reason for the pressurizer level indication for this event?

- A. Steam from hot leg creates false indication of solid pressurizer.
- B. Steam Bubble in the Reactor Head forces water into pressurizer.
- C. Safety Injection mass addition greater than steam through safety rapidly fills pressurizer.
- D. Cold safety injection water rapidly expands as it is heated rapidly filling the pressurizer

Answer: B

Explanation/Justification:

A. Incorrect but plausible because at TMI steam flow through the pressurizer held the water in the pressurizer. Indication was real but not a reflections of RCS mass.

B. Correct. The water in the reactor vessel head flashes to steam when saturation conditions are met. IPEC is a Thot plant at approximately 1400 - 1500 psig water is forced into the pressurizer. If saturation conditions exist at the core exit steam will flow to the hot legs and out through the PORVs. The steam flow will hold the water in the pressurizer.

C. Incorrect but plausible because at approximately 1500 psig, the Safety Injection pumps will supply water to the RCS. Even with mass flow into the RCS greater than mass flow out, the level response will not be rapid.

D. Incorrect but plausible because cold water from the RWST will heat up as it enters the RCS. This will not be enough to cause the rapid level change in the pressurizer.

Technical References: 2-ES-0.2 BG

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE10 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	0000322402	
		Emergency Procedures/Plan - Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. Note: The issue of setpoints and automatic safety features is not specifically covered in the systems sections.	
	Importance	4.5	4.6

Question # 21

The unit was operating at 100% power.
PR N44 was removed from service due to erratic operation.

Subsequently:

- PR channel N41 Instrument Power Fuse blew.
- The reactor tripped and the crew responded in accordance with E-0 and transitioned to ES-0.1

Approximately 30 minutes later the STA reported a YELLOW path condition on Subcriticality.

- Intermediate Range SUR is NOT less than -0.2DPM
AND
- Source Range is not energized

At this time:

- A. Remove Power Range channel N41 from service to re-energize Source Range detectors.
- B. Remove both Intermediate Range Control Power fuses to remove P-6 permissive.
- C. Manually reenergize the Source Range NIs by HV MANUAL switches to ON on Source Range drawers.
- D. Initiate emergency boration to lower Intermediate Range power to below P-6 setpoint.

Answer: C

Explanation/Justification:

The loss of 2/4 Power Range NIs trips the P-10 bistable blocking the re-energization of the Source Range NIs.

A. Incorrect but plausible because this is what is done if the Intermediate Range NIs are undercompensated. This action will not work because the malfunction is in the Power Range NIs.

B. Incorrect but plausible because this would cause IR power to go to zero; however the malfunction is in the Power Range NIs.

C. Correct. The failure is the result of 2 Power Range NI P-10 bistables tripped. Two of four PR NI P-10 bistables tripped will prevent SR NIs from automatically re-energizing when IR NIs decrease below P-6. Unit 2 modified the standard Westinghouse SR NI drawers adding a switch that can energize or de-energize the SR NIs.

D. Incorrect but plausible because Boration is the procedural direction if SR SUR is not zero or negative.

Technical References: 2-FR-S.2

Proposed References to be provided: None

Learning Objective I2LP-ILO-ICEXC 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 6

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	000060A101	
		Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste Release: - Area radiation monitors	
	Importance	2.8	3

Question # 22

Given:

- The unit is preparing for refueling.
- R-27 Plant Vent Wide Range Gas Monitor is increasing.
- R-44 Plant Vent Radio Gas is in alarm.
- R-50 Gas Decay Tank Activity is stable.
- R-4 Unit 2 Charging Pump Room is in alarm.
- R-5987 Breaker Service Access is stable.

Which of the following events has occurred?

- A. VCT relief valve lift
- B. VCT vapor space leak
- C. Waste Gas Compressor discharge relief valve lift
- D. Waste Gas Compressor discharge line leak.

Answer: D

Explanation/Justification:

Radiation Monitors R-27, R-44, and R-50 are process radiation monitors. R-4 and R 5987 are area radiation monitors. Discharge of radioactive gas will cause local area radiation monitors to increase.

A. Incorrect but plausible because if the relief valve discharged to the VCT room, R-27 and R-44 would increase; however, the relief valve discharges to the Vent Header.

B. Incorrect but plausible because a tank vapor space leak would cause R-27 and R-44 to increase, but the VCT is located above the charging pumps and gas will not cause R-4 to alarm.

C. Incorrect but plausible because the relief valve discharges to the waste gas vent header not the local area.

D. Correct. A leak on the discharge of the waste gas compressors will discharge gas in the vicinity of R-4.

Technical References: Drawing
Proposed References to be provided: None

Learning Objective I2LP-ILO-GWR01 6
I2LP-ILO-RMS001 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	000068A205	
		Ability to determine and interpret the following as they apply to the Control Room Evacuation: - Availability of heat sink	
	Importance	4.2	4.3

Question # 23

Given:

- The Unit 2 is operating at 100% power.
- 21 AFW Pump is tagged and disassembled.

Subsequently:

- A fire in the 480V Switchgear Room forced evacuation of the control room.
- All 480 V Buses are de-energized.
- The reactor, main turbine, MBFP's and RCP's are tripped.
- All other App R equipment is available including 12FD3.

Which one of the following correctly describes the availability of a secondary heat sink?

- A. ONLY 21 SG and 22 SG are available for use with 22 AFW pump.
- B. ONLY 23 SG and 24 SG are available for use with 22 AFW pump.
- C. ALL SGs are available for use with 22 AFW pump, however only 21 SG and 22 SG will be used.
- D. ALL SGs are available for use with 22 AFW pump, however only 23 SG and 24 SG will be used.

Answer: C

Explanation/Justification:

Duplicated from question no 26611- used stem, modified it to make only 22 AFW pump available. Replaced ALL choices to reflect KA asking about availability of heat sink not maintenance of heat sink.

Aux feed will be established if possible from the control room prior to evac, with 21 or 22 AFW pump supplying 21 and 22 SG's. Attachment 2 Conventional Side RO Actions will further explain how to use AFW to feed only 21 and 22 SGs.

A Incorrect but plausible because the TDAFW pump CAN supply ALL SGs, it is preferred to feed 21 and 22 because they are the only 2 which have Wide Range level at Remote Shutdown Panel.

B Incorrect but plausible because 23 and 24 SG are fed from 23 AFW pump.

C Correct 21 and 22 SGs have indication on Remote Shutdown Panel 23 and 24 do not.

D Incorrect but plausible because all SG are available with 22 AFW pump in service, however 21 and 22 SGs are used.

Technical References: 2-AOP-SSD-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-MFW001 10

Question Source: Modified

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	000076K201	
		Knowledge of the interrelations between the High Reactor Coolant Activity and the following: - Process radiation monitors	
	Importance	2.6	3

Question # 24

The plant is operating at 100% Power.

ALL of the following have occurred:

A plant shutdown is in effect due to RCS Activity greater than Tech Spec limits.

Radiation Monitor R-47 is in alarm

RCV-017, CCW Surge Tank Vent Valve has automatically closed.

Which of the following events could have caused these conditions to occur and what is the appropriate procedure to address the condition?

- A. Tube leak in RCP Seal Return HX
Go to 2 AOP-RCP-1, Reactor Coolant Pump Malfunctions
- B. Tube leak in RCP Seal Return HX
Go to 2-AOP-LICCW-1 Leakage into Component Cooling System
- C. Tube leak in Non-regenerative HX
Go to 2-AOP-CCW-1, Loss of Component Cooling Water
- D. Tube leak in Non-regenerative HX
Go to 2-AOP-LICCW-1 Leakage into Component Cooling System

Answer: D

Explanation/Justification:

The only auto closure signal for RCV-017 is high activity on Radiation Monitor R-47.

A. Incorrect but plausible because an operator may not recall the pressure differences involved. Seal return is from RCS, but it is essentially VCT pressure at this point. In addition AOP-RCP-1 is not the correct procedure to address this condition.

B. Incorrect, but plausible for the same reasons as A. AOP-LICCW-1 is the correct procedure to address this condition.

C. Incorrect but plausible because a leak in the Non-regenerative heat exchanger will cause leakage into CCW and may cause RCV-017 to auto close. Incorrect because AOP-CVCS-1 is not the correct procedure to address this condition.

D Correct. A leak in the Non-regenerative heat exchanger will cause leakage into CCW and may cause RCV-017 to auto close. AOP-LICCW-1 is the correct procedure to address this condition.

Technical References: 2-AOP-LICCW-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPLIC 1

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	00WE08K102	
		Knowledge of the operational implications of the following concepts as they apply to the Pressurized Thermal Shock: - Normal, abnormal and emergency operating procedures associated with Pressurized Thermal Shock	
	Importance	3.4	4

Question # 25

Given:

- A LOCA had occurred from HOT STANDBY conditions, Tave at 547°F, 30 minutes ago
- RCS pressure is 125 psig
- RCS Core Exit TCs read 380°F
- RCS Cold Leg temperatures are all 220°F
- 21 SI Pump is running providing 325 gpm flow
- 21 RHR Pump is running providing 1150 gpm flow
- RWST Level is 13 ft. and trending down
- E-0, Reactor Trip or Safety Injection, Attachment 1 is complete

What is the appropriate action to take in response to the above conditions?

Entry into FR-P.1, Response to Pressurized Thermal Shock Condition, is ...

- A. NOT required since RCS pressure is below 350 psig.
- B. made but NO actions are implemented before returning to procedure in effect.
- C. made and a RCS temperature soak for a ONE hour period will be completed.
- D. made and cooldown will continue within a limit of 50°F in any 60 minute period.

Answer: B

Explanation/Justification:

For these conditions, FR-P.1 will be entered, but the procedure will direct the team to go back to procedure and step in effect.

A. Incorrect but plausible. Since no actual actions are taken in FR-P.1, it is plausible a candidate would think entry is not required.

B. Correct answer.

C. Incorrect but plausible since these are actions in FR-P.1.

D. Incorrect but plausible since these are actions in FR-P.1.

KA match because the question makes the candidate determine that a LBLOCA is occurring and then determine that the actions for the LOCA take precedence over the overcooling event that is also occurring.

Technical References: 2-FR-P.1
2-FR-P.1 BG
3-FR-P.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPFRP 4

Question Source: Bank

Question History: Unit 3 NRC 2006

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	00WE10A202	
		Ability to determine and interpret the following as they apply to the Natural Circulation with Steam Void in Vessel with/without RVLIS: - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	
	Importance	3.4	3.9

Question # 26

A natural circulation cooldown is in progress.

RVLIS is less 75%.

PRZR level is approximately 50%

Which of the following describes actions for pressurizer level and basis for these actions prior to starting an RCP?

- A. Pressurizer level should be lowered to accommodate expansion of void in Vessel Head.
- B. Pressurizer level should be lowered to accommodate expansion of RCS due to mixing all loops.
- C. Pressurizer level should be raised to accommodate contraction of RCS due to mixing all loops.
- D. Pressurizer level should be raised to accommodate collapse of void in Vessel Head.

Answer: D

Explanation/Justification:

This question is RO level because it tests the understanding of the consequences of starting an RCP in voided RCS conditions.

RCPs can be started under a variety of conditions. TS 3.4.12 lists requirements for starting RCPs based on RCS temperature and SG temperature.

A. Incorrect but plausible because depending on RCS and SG conditions, RCS could contract when mixing resulting in lowering pressure which would result in expansion of steam void in head i.e., expansion into PRZR. Significant temperature differences would not be expected during natural circulation.

B. Incorrect but plausible because if SG temperature was greater than RCS temperature the coolant would expand into the PRZR. Significant temperature differences would not be expected during natural circulation.

C. Incorrect but plausible because PRZR level would be raised but not to accommodate contraction due to temperature change.

D. Correct. When the RCP is started relatively cold water from RCP discharge will be circulated to the upper head causing the void to collapse.

Technical References: 2-ES-0.3 BG

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPS00 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	2	
	K/A#	00WE15A103	
		Ability to operate and/or monitor the following as they apply to the Containment Flooding: - Desired operating results during abnormal and emergency situations	
	Importance	2.8	3

Question # 27

Given:

- A large break LOCA has occurred.
- The plant is tripped and ECCS is operating as expected.
- Accumulators have discharged and are isolated.
- The STA reports an ORANGE path on Containment.

Which one of the following describes the required action per 2-FR-Z.2, Response to Containment Flooding, and its purpose?

- A. Secure all water sources from outside of containment to prevent damaging vital components.
- B. Secure all water sources from outside of containment to prevent overloading concrete containment structures.
- C. Isolate source of flooding to prevent damaging vital components.
- D. Isolate source of flooding to prevent overloading concrete containment structures.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because procedure does secure the leaking system not all possible sources.

B. Incorrect. Plausible because procedure does secure the leaking system not all possible sources.

C. Correct.

D. Incorrect. Plausible because isolating the source of the leak (Not all water sources) is correct, overloading concrete is not.

Technical References: 2-FR-Z.2
2-FR-Z.2 BG

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPFRZ 4

Question Source: Bank

Question History: Unit 2 NRC 2004

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 9

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	003000K505	
		Knowledge of the operational implications of the following concepts as they apply to the RCPS: - The dependency of RCS flow rates upon the number of operating RCPs	
	Importance	2.8	3

Question # 28

Given:

- Unit 2 is in MODE 4
- Three RCPs are running

Which of the following describes the expected RCS flow indication?

	Idle Loop Flow Indication	Operating Loop Flow Indication
A.	approximately 0%	approximately 100%.
B.	approximately 0%	approximately 106%.
C.	approximately 18%	approximately 100%.
D.	approximately 18%	approximately 106%.

Answer: D

Explanation/Justification:

Flow in the operating loops increases to 106%. Flow in the idled loop indicates 18%, which is actually -18% (reverse flow). If an RCP is stopped, the high pressure discharge of the other three pumps forces the coolant to flow backwards through the newly idled loop. Due to the reduced backpressure of the idle loop, the flow rate in each of the operating loops increases from 100% to about 106% of rated flow. The idle loop has an equivalent flow of about 18%, in the reverse direction. Total core flow will decrease since only three RCPs are operating, and some of their flow is bypassing the core through the idle loop.

A and B incorrect. If candidate knows there is reverse flow and bottom of scale is zero, then it would be plausible to think that the meter would read downscale, or zero.

C incorrect. Indicated flow in the loops with running pumps is greater than 100%.

D is correct. Indicated flow in the idle loop, is the reverse flow indication of approximately 18% and flow in the operating loops is greater than 100% or approximately 106%

Technical References:

Proposed References to be provided: None

Learning Objective I2LP-ILO-RCS001 5
I2LP-ILO-RCSRCP 5

Question Source: Modified

Question History: Diablo Canyon 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 14

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	004000K509	
		Knowledge of the operational implications of the following concepts as they apply to the CVCS: - Thermal shock: high component stress due to rapid temperature change	
	Importance	3.7	4.2

Question # 29

Given the following conditions:

- The plant is in MODE 3, at normal operating pressure and temperature.
- The crew is initiating auxiliary spray in accordance with 2-SOP-1.4, Pressurizer Pressure Control.

Of the following, which is the LOWEST allowable regenerative heat exchanger outlet temperature for Aux Spray initiation under given conditions?

- A. 300°F
- B. 320°F
- C. 340°F
- D. 360°F

Answer: C

Explanation/Justification:

To prevent thermal shock to the spray nozzle do NOT initiate Aux Spray flow if the temperature difference between PZR vapor space and the outlet of the Regen Hx is > 320°F.

- A. Incorrect but plausible because this would result in an acceptable delta T if RCS T cold was used vice Pressurizer temperature.
- B. Incorrect but plausible because this is the actual delta T limit.
- C. Correct. Pressurizer temperature is approximately 651°F (2235 psig) and subtracts 320 = 331 thus 340 will result in a delta T of 311°
- D. Incorrect but plausible because this will result in an acceptable delta T but it is NOT the minimum acceptable temperature.

Technical References:

Proposed References to be provided: None

Learning Objective I2LP-ILO-RCSPZR 11

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	004000K304	
		Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: - RCPS	
	Importance	3.7	3.9

Question # 30

Given:

- Indian Point 2 has just experienced a loss of all CVCS charging capability while at power.
- The operating crew isolated CVCS letdown flow and has entered the appropriate Technical Specification action statements.

How is RCP operation affected by the loss of charging flow?

- A. RCP operation may continue while restoring RCP Seal Injection, since Thermal Barrier Cooling is in service.
- B. The RCPs must be immediately tripped to prevent damage to the RCP seal packages from uncontrolled heatup.
- C. RCP operation may continue while restoring RCP Seal Injection since the number 2 seal is now a film riding seal.
- D. The RCPs must be immediately tripped to prevent damage to the RCP seal packages via contaminants from the RCS backflowing into the RCP Seals.

Answer: A

Explanation/Justification:

A is correct as the thermal barrier is designed to cool rcs flow through it to the seal package.

B is incorrect but plausible if it is thought high temperature RCS fluid will enter the seal package.

C is incorrect but plausible if it is thought that a number 1 seal failure will occur.

D is incorrect but plausible if it is thought that this true event should cause operators to immediately shutdown RCPs.

Technical References:

Proposed References to be provided: None

Learning Objective I2LP-ILO-RCSRCP 11

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	005000K412	
		Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: - Lineup for piggyback mode with CSS	
	Importance	3.1	3.7

Question # 31

A Large Break LOCA occurred 2 hours ago.

Bus 5A tripped on Over Current

All other systems operated as designed.

The crew transferred to Cold Leg Recirculation using ES-1.3, Transfer to Cold Leg Recirculation.

Which one of the following describes the Containment Spray flowpath?

A portion of the flow from:

- A. 22 Recirc Pump supplies the containment spray header via 23 Safety Injection Pump
- B. 21 RHR Pump supplies the containment spray header via 23 Safety Injection Pump
- C. 22 Recirc Pump supplies the containment spray header directly
- D. 21 RHR Pump supplies the containment spray header directly

Answer: C

Explanation/Justification:

At Indian Point, containment spray pumps take a suction only from the RWST. When the RWST is empty, continued containment spray is provided by securing the spray pumps and diverting some recirculation flow to the Spray Headers. Recirc flow is provided by Recirculation Pumps or RHR pumps. Loss of bus 5A will prevent operation of 21 Recirc Pump.

A. Incorrect but plausible because recirculation flow can be diverted to the suction of the Safety Injection pumps for high head recirculation or hot leg recirculation. Safety injection pumps do not connect to the Containment Spray system.

B. Incorrect but plausible because the loss of 21 Recirc pump may cause applicant to believe that RHR will be used instead of Recirc Pumps. Also, recirculation flow can be diverted to the suction of the Safety Injection pumps for high head recirculation or hot leg recirculation. Safety injection pumps do not connect to the Containment Spray system.

C. Correct. When the RWST is empty, normal spray pumps are secured and recirculation flow is directed to the spray header if necessary. Normally recirculation flow is provided from the recirculation pumps (not RHR).

D. Incorrect but plausible because the loss of 21 Recirc pump may cause applicant to believe that RHR will be used instead of Recirc Pumps. This lineup would be correct if NO recirc pumps were available

Technical References: Drawing
Proposed References to be provided: None

Learning Objective I2LP-ILO-RHR001 6

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	006000K202	
		Knowledge of bus power supplies to the following: - Valve operators for accumulators	
	Importance	2.5	2.9

Question # 32

Given:

- Unit was operating at 100% power
- 23 EDG is OOS for maintenance
- Bus 5A de-energized due to a fault
- SBLOCA occurred resulting in a reactor trip and safety injection.
- Accumulator isolation requirements of ES-1.2, Post LOCA Cooldown and Depressurization are satisfied.

Which of the following is correct for the SI Accumulator isolation actions?

- A. 894 A & C will be closed; 22 and 24 Accumulator will be vented
- B. 894 B & D will be closed; 21 and 23 Accumulator will be vented
- C. All Accumulators will be isolated
- D. All Accumulators will be vented

Answer: D

Explanation/Justification:

Indian Point Unit 2 has 3 Safeguards buses 5A, 2A/3A and 6A. While 2 buses are assumed energized in accident analysis, the emergency procedures will function if one bus is energized. Some safeguards equipment is powered from 2A/3A which remains energized in this question.

When a reactor trip occurs with a loss of power to either bus 5A or 6A, the normal feed breaker for all 480V busses open and when the EDGs are up to speed and voltage the emergency feed breakers close. Bus 5A will remain deenergized due to fault and bus 6A will not energize because 23 EDG is OOS. There is not procedural guidance for energizing 6A for these conditions.

The accumulator isolation valves 894 A & C are powered from MCC26A and 894 B & D from MCC26B. MCC26A is powered from bus 5A and MCC26B is powered from bus 6A. With bus 5A de-energized, MCC26A is de-energized and bus 6a is de-energized from 23 EDG OOS.

All distractors are plausible because the candidate must recall which valves are powered from which MCC and which 480V bus supplies which MCC.

Technical References: 2-ES-1.2
Proposed References to be provided: None

Learning Objective I2LP-ILO-SIS01 6

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 6

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	007000K401	
		Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: - Quench tank cooling	
	Importance	2.6	2.9

Question # 33

Given:

- PRT Temperature is 195°F
- PRT Level is 75%
- PRT Pressure is 8 psig.

How are PRT level and temperature lowered in accordance with 2-SOP-1.6, Pressurizer Relief Tank Operations?

- A. Drain to RCDT pump, vent the PRT.
- B. Drain to RCDT pump, spray the PRT.
- C. Drain to VC sump, vent the PRT.
- D. Drain to VC sump, spray the PRT.

Answer: D

Explanation/Justification:

2-SOP-1.6, PRT Operations, Section 4.5, PRT Temperature Control, says to lower level by draining to VC sumop due to the temperature of the PRT. Makeup is supplied via the 519, 552, and 560 valves, above the expected water line, hence "spray". Each of the distracters contains either the incorrect drain path or makeup method, or both.

Drawing 9321-F-2719 shows relationship between PRT, RCDT, RCDT pumps, containment sump, and Rx sump.

Technical References: 2-SOP-1.6

Proposed References to be provided: None

Learning Objective I2LP-ILO-RCSPZR 2

Question Source: Bank

Question History: Unit 3 NRC Retake 2018

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 3

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	008000A102	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: - CCW temperature	
	Importance	2.9	3.1

Question # 34

The component cooling water temperature was 92°F when RHR was initially placed in service at 0100. The component cooling water system temperature is increasing at 0.5°F per minute due to decay heat.

If the rate of temperature rise remains constant, when will the maximum allowable CCW temperature for the initial phase of RHR operation be reached?

- A. 0136
- B. 0156
- C. 0256
- D. 0306

Answer: B

Explanation/Justification:

The normal maximum CCW temperature is 110°. During the initial phase of RHR operation the maximum CCW temperature is allowed to rise to 120°. CCW also has a maximum pump suction temperature of 155°.

- A. Incorrect but plausible because this would be correct for the normal maximum temperature of 110.
- B. Correct.
- C. Incorrect but plausible because if the candidate confuses suction temperature limit of 150° as the maximum, this would be correct. CCW flow limit to RCPs is 150 gpm.
- D. Incorrect but plausible because this would be correct if the candidate believed that 155° was the maximum limit.

Technical References: 2-SOP-4.1.2
3-SOP-RHR-001

Proposed References to be provided: None

Learning Objective I2LP-ILO-CCW001 5

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	010000K202	
		Knowledge of bus power supplies to the following: - Controller for PZR spray valve	
	Importance	2.5	2.7

Question # 35

The unit is operating at 100% power.

A fault has resulted in the loss of 23 static inverter. Which of the following describes the impact of this loss?

- A. Trip the Reactor.
Automatic and manual control of pressurizer heaters and spray IS NOT available.
- B. Trip the Reactor.
Automatic control of pressurizer pressure is lost.
Manual control of heaters and spray IS available
- C. Establish Stable Plant Conditions
Automatic and manual control of pressurizer heaters and spray IS NOT available.
- D. Establish Stable Plant Conditions
Automatic control of pressurizer pressure is lost.
Manual control of heaters and spray IS available

Answer: A

Explanation/Justification:

The loss of power from 23 Instrument Bus will not DIRECTLY cause a reactor trip; however it results in loss of multiple control systems. Because of the loss of multiple control systems, the AOP directs a reactor trip. For all other instrument bus loses the AOP allows continued operation depending on plant conditions.

A. Correct. 23 Instrument Bus supplies the Pressurizer Master Pressure controller and both pressurizer spray valves (i.e. heaters and spray valves cannot be operated).

B. Incorrect but plausible because candidate may believe the master pressure controller and individual spray valves controllers are powered from different instrument buses.

C. Incorrect but plausible because stabilizing plant conditions is correct for loss of 21, 22, and 24 instrument buses. The pressurizer control is correct.

D. . Incorrect but plausible because stabilizing plant conditions is correct for loss of 21, 22, and 24 instrument buses. Also, candidate may believe the master pressure controller and individual spray valves controllers are powered from different instrument buses.

Technical References: 2-AOP-IB-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPIB1 3

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	012000K301	
		Knowledge of the effect that a loss or malfunction of the RPS will have on the following: - CRDS	
	Importance	3.9	4

Question # 36

Given:

- A startup is in progress.
- The unit is at 6% power.
- Intermediate Range N35 failed low.
- The reactor did not trip.

The crew is performing 2-SOP-13.1, Nuclear Instrumentation System Operation. The Control and Instrument power fuses will be removed.

Which of the following describes the impact on the rod control system?

- A. Protection Relays must be tripped using bistable trip switches to prevent a reactor trip and restore manual rod control BEFORE the Control and Instrument power fuse are removed.
- B. Protection Relays must be mechanically blocked to prevent a reactor trip and restore manual rod control BEFORE the Control and Instrument power fuses are removed.
- C. Protection Relays are bypassed using Level Trip Switches to prevent a reactor trip and restore manual rod control AFTER the Control and Instrument power fuses are removed.
- D. Protection Relays are mechanically blocked to prevent a reactor trip and restore manual rod control AFTER the Control and Instrument power fuses are removed.

Answer: B

Explanation/Justification:

This question meets the KA because the described Intermediate Range malfunction results in a loss of function of reactor trip and rod stop from that channel. The question asks how the reactor trip is prevented and how manual rod withdrawal (impact on control rod drive system) is restored.

A Incorrect but plausible because for most reactor protection functions, bistables are tripped to satisfy coincidence requirements of TS.

B. Correct. When the control power fuses are removed, the reactor trip and bistables will trip, but the reactor will not trip and manual rod withdrawal will NOT be blocked with the physical blocking devices.

C. Incorrect but plausible because the Level Trip Switch will block the reactor trip and remove the rod stop until the control power fuses are removed.

D. Incorrect but plausible because the relays must be mechanically blocked, but this must be done before removing control power fuses.

Technical References: 2-SOP-13.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-ICEXC 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	013000K601	
		Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS: - Sensors and detectors	
	Importance	2.7	3.1

Question # 37

Given the following conditions:

- The plant is at 80% power.
- Containment Pressure transmitter 948A has failed low.

Which of the following statements describes the effect of blocking HI-HI Containment Pressure relays when performing AOP-INST-1, Instrument or Controller Failures?

- A. Blocks the actuation of safety injection from containment high pressure when the bistables are tripped.
- B. Reduces the containment spray actuation coincidence such that 1 out of 2 additional channel from either set greater than the setpoint will cause Containment Spray
- C. Reduces the containment spray actuation coincidence to 1 out of 2 on one set and 2 out of 3 in second set greater than setpoint to actuate Containment Spray.
- D. Blocks the actuation signal from the failed transmitter from causing a Phase B containment isolation signal and Main Steam Isolation when bistables are tripped.

Answer: C

Explanation/Justification:

Containment Spray actuation is from two sets of three transmitters.(948 A, B, C and 949 A, B, C). Spray actuation requires 2 transmitters in each SET greater than setpoint to actuate the signal. Finally, containment spray relays must be energized to actuate the signal. Most other features are fail safe de-energize to actuate. Blocking a relay places it in its energized state.

A. Incorrect but plausible because 948A is an input to VC high pressure SI; however, the bistables are de-energize to actuate and bistables are tripped to reduce the coincidence to 1 out of 2.

B. Incorrect but plausible because for all other ESF signals, AOP-INST-1 actions trip one bistable/relay setting the system up such that one more input will cause actuation. For containment spray this failure only affects one SET.

C. Correct.

D. Incorrect but plausible because high containment pressure will actuate Phase B and Main Steam Isolation however, blocking relays does not affect these features.

Technical References: Logic Unit 2 Sheet 12
Syst Desc 10

Proposed References to be provided: None

Learning Objective I2LP-ILO-ESS001 3

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	022000K101	
		Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: - SWS/cooling system	
	Importance	3.5	3.7

Question # 38

Given the following conditions:

- The plant is operating at 100% power
- 21, 23, and 25 FCUs are in service

Subsequently, a reactor trip and Safety Injection occur.

Which one of the following describes the resulting Containment Cooling lineup?

- A. Only 21, 23, and 25 FCUs are in service. Cooling water flow remains unchanged.
- B. All FCUs are in service. Cooling water flow remains unchanged.
- C. Only 21, 23, and 25 FCUs are in service. Cooling water flow has increased.
- D. All FCUs are in service. Cooling water flow has increased.

Answer: D

Explanation/Justification:

The Fan Cooler Units (FCUs) will only auto start on a Safety Injection (Not a blackout-unit trip signal). In addition the safety injection signal will open TCV-1104 and TCV-1105 to ensure adequate service water cooling flow.

A Incorrect: Plausible because ALL FCUs will be operating not just 21, 23 and 25 FCU were originally in service and student may believe they will restart automatically but not 22 and 24 FCU. Also TCV-1103 is normal temperature control for service water from the FCUs. Applicant may also believe that containment temperature is controlled by 1103 alone.

B Incorrect: Plausible because all FCUs will be in service, but the safety injection signal will open TCV-1104 and TCV-1105 to ensure adequate cooling.

C Incorrect: Plausible because ALL FCUs will be operating not just 21, 23 and 25 FCU were originally in service and student may believe they will restart automatically but not 22 and 24 FCU. Also Cooling water alignment is correct.

D: Correct. All FCUs will auto start on a safety injection signal and 1104 and 1105 will open.

Technical References: 2-E-0

3-E-0

Proposed References to be provided: None

Learning Objective I2LP-ILO-VCCARC 13

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	022000A102	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: - Containment pressure	
	Importance	3.6	3.8

Question # 39

Given:

- A steam line rupture occurred inside containment.
- VC pressure peaked at 12 psig when SG boiled dry.
- 480V Buses 2A/3A failed to re-energize upon the transient.

Which of the following are true for these conditions?

- A. EOP adverse values are required until an engineering evaluation is satisfactory. VC pressure will be reduced using the Containment Fan Cooler Units.
- B. EOP adverse values are required until pressure is < 4psig. VC pressure must be reduced using Containment Pressure Relief System.
- C. EOP adverse values are required until an engineering evaluation is satisfactory. VC pressure must be reduced using Containment Pressure Relief System.
- D. EOP adverse values are required until pressure is < 4psig. VC pressure will be reduced using the Containment Fan Cooler Units.

Answer: D

Explanation/Justification:

A. Incorrect but plausible because adverse values are required and when high radiation conditions exist an engineering evaluation is required to use normal values. It is true that pressure will be reduced by the containment fan coolers.

B. Incorrect but plausible because it is correct that adverse values are used if pressure is greater than 4 psig. The pressure relief system is also plausible because there would be no release of radiation from a steam break accident. Also pressure relief cannot be used because the Containment Ventilation Isolation signal will be actuated on the Safety Injection.

C. Incorrect but plausible because adverse values are required and when high radiation conditions exist an engineering evaluation is required to use normal values. The pressure relief system is also plausible because there would be no release of radiation from a steam break accident. Also pressure relief cannot be used because the Containment Ventilation Isolation signal will be actuated on the Safety Injection.

D. Correct

Technical References: OAP-012
Proposed References to be provided: None

Learning Objective I0LP-ILO-EOPROU 7

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	026000A405	
		Ability to manually operate and/or monitor in the control room: - Containment spray reset switches	
	Importance	3.5	3.5

Question # 40

Given:

- A large break LOCA occurred.
- Containment pressure peaked at 27 psig and started trending down.

Which of the following describes the operation of containment spray reset pushbuttons?

- A. Containment Spray signal cannot be reset until recirc spray flow indication is > 960 gpm
- B. Containment Spray signal cannot be reset until directed in ES-1.3 Transfer to Cold Leg recirculation.
- C. Containment Spray signal can be reset in E-1 when VC pressure is < 17 psig
- D. Containment Spray signal can be reset in E-0 when VC pressure is < 2 psig

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because minimum recirc spray is 960; however, this is not verified until after normal spray flow is secured.

B. Incorrect. Plausible because if not previously reset, containment spray reset is directed in ES-1.3.

C. Correct.

D. Incorrect. Plausible because some safeguards signals (e.g., Safety Injection) can be reset regardless of input signal condition.

Technical References: 2-E-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-CS001 6

Question Source: Bank

Question History: Unit 2 NRC 2016

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	039000A302	
		Ability to monitor automatic operation of the MRSS, including: - Isolation of the MRSS	
	Importance	3.1	3.5

Question # 41

Which of the following signals will close the Main Steam Isolation Valves (MSIVs)?

- A. High Steam Flow coincident with Low Tavg
Manual Phase B Isolation
- B. Steam Line Delta-P
Hi-Hi Containment Pressure
- C. High Steam Flow coincident with Low SG Pressure
High Containment Pressure
- D. High Steam Flow coincident with Low Tavg
Hi-Hi Containment Pressure

Answer: D

Explanation/Justification:

A. Incorrect but plausible because high steam flow with low Tavg will close the MSIVs, but Manual phase B will not.

B. Incorrect but plausible because Steam Line Delta P will generate an SI signal but not close the MSIVs. The Hi-Hi containment will close the MSIVs.

C. Incorrect but plausible because high steam flow with low SG pressure will close MSIVs but high containment pressure will not (High High pressure will close the MSIVs).

D. Correct.

Technical References: Syst Desc 10
Proposed References to be provided: None

Learning Objective I2LP-ILO-MSS001 8

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	059000A302	
		Ability to monitor automatic operation of the MFW System, including: - Programmed levels of the S/G	
	Importance	2.9	3.1

Question # 42

Given:

- The reactor is operating at 96% power,
- The controlling steam pressure channel for steam generator 22 fails high,
- NO operator action is taken.

Which ONE of the following describes the effect this transient will have on level control in steam generator 22?

- A. Level stabilizes above program setpoint, but high-level trip is NOT reached.
- B. Level initially controls above program setpoint, but returns to setpoint over several minutes.
- C. Level stabilizes below program setpoint, but lo-lo level trip is NOT reached.
- D. Level initially controls below program setpoint, but returns to setpoint over several minutes.

Answer: B

Explanation/Justification:

Duplicated from question no 19881

A. Incorrect but plausible because actual level will increase above programmed level, but it will not remain above the programmed level setpoint.

B. Correct

C. Incorrect but plausible because candidate must recall how steam pressure is used in the compensating circuit (direct relationship not inverse).

D. Incorrect but plausible because candidate must recall how steam pressure is used in the compensating circuit (direct relationship not inverse). Also steam generator level will return to programmed level setpoint over time

Technical References: 2-AOP-INST-1

Proposed References to be provided: None

Learning Objective I0SM-LOR-MBFP01 2

Question Source: Modified

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	0590002123	
		Conduct of Operations - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	
	Importance	4.3	4.4

Question # 43

Which of the following is required before stopping 21 and 23 AFW pumps per AOP-FW-1 to ensure they do not auto-restart?

- A. Ensure 21 MBFP Reset Switch is in trip.
- B. Verify adequate MFW flow
- C. Ensure SG levels stable above Lo-Lo level setpoint
- D. Verify 21 MBFP auto stop oil pressure is > 25 psig.

Answer: A

Explanation/Justification:

This question meets the KA because the procedural action is to place Main Feedwater Pump Trip/Reset switches in Trip to allow normal operation of AFW motor driven pumps.

A. Correct (2-AOP-FW-1 Step 102)

B. Incorrect. Plausible because adequate feed flow would maintain SG level above the auto start setpoint

C. Incorrect. Plausible because 9% is the auto start setpoint; however, SG level will not decrease to 9% before the AFW pumps auto start on MBFP trip.

D. Incorrect. Plausible because 28 psig is the auto stop oil setpoint that will cause the MBFP to trip, so the pump trip cannot be reset at 25 psig.

Technical References: 2-AOP-FW-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-MFW001 9

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	061000K602	
		Knowledge of the effect of a loss or malfunction of the following will have on the AFW System components: - Pumps	
	Importance	2.6	2.7

Question # 44

Given:

- Reactor tripped from 25% power.
- Bus 6A has an overcurrent lockout.
- No action has been taken by the crew.

Which ONE of the following describes the performance of the Auxiliary Feedwater system for these conditions?

- A. ALL SGs fed at a total AFW flow of approximately 800 GPM.
- B. ALL SGs fed at a total AFW flow of approximately 400 GPM.
- C. ONLY two SGs fed at a total AFW flow of approximately 800 GPM.
- D. ONLY two SGs fed at a total AFW flow of approximately 400 GPM.

Answer: D

Explanation/Justification:

A. Incorrect but plausible because this would be correct if both motor drive AFW pump started and delivered flow to the Steam Generators. 23 AFW pump would not start due to loss of bus 6A.

B. Incorrect but plausible because the 21 motor driven AFW pump will start with a capacity of approximately 400 gpm; however it will only deliver flow to 21 and 22 SGs.

C. Incorrect but plausible because only 21 and 22 SGs will be fed, but the flowrate is too high.

D. Correct. With bus 6A de-energized 23 AFW pump will not be running. 21 AFW pump will automatically start, but 21 AFW pump only feeds 21 and 22 SGs. Flow is automatically controlled at 200 gpm per SG.

Technical References: Syst Desc 21

Proposed References to be provided: None

Learning Objective I2LP-ILO-MFW001 4

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	061000K601	
		Knowledge of the effect of a loss or malfunction of the following will have on the AFW System components: - Controllers and positioners	
	Importance	2.5	2.8

Question # 45

Given:

- The plant is being started up following a refueling outage
- The plant is at 3.5% power
- 21, and 22 AFW pumps are supplying the Steam Generators
- Steam Generator levels are lowering in 21 and 22 SG and stable in 23 and 24 SG.
- There are no feedwater leaks in the plant

Which of the following conditions could be the cause for the SG level decrease?

- A. The steam supply valves to the steam driven AFW pump (1310A and 1310B) closed.
- B. Reactor power has exceeded the capacity of the AFW system.
- C. FC-1135A-S, 21 AFW pump suction flow transmitter has failed high.
- D. PT-406A, 21 AFW Pump Discharge Pressure Transmitter, has failed low.

Answer: D

Explanation/Justification:

A. Incorrect: Plausible because student may believe the turbine driven AFW pump is being used. If 1310A/B closed the turbine driven AFW pump would slow down resulting in lower level in SG it was supplying.

B. Incorrect: Plausible because this is the power level that main feedwater is placed in service and auxiliary feedwater is secured; however, the capacity of the AFW system is greater than 3 percent.

C. Incorrect but plausible because "LOW" suction flow will cause the recirculation valve to open which would reduce flow to the SG. Failure high will prevent the recirculation valve from opening if necessary.

D. Correct: Low discharge pressure is indication of pump runout and the controller will close the discharge valves.

Technical References: Syst Desc 21.2

Proposed References to be provided: None

Learning Objective I2LP-ILO-MFW001 13p

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 4

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	0620002447	
		Emergency Procedures/Plan - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	
	Importance	4.2	4.2

Question # 46

Given:

- The unit was operating at 100% power
- Severe weather has caused fluctuations in grid voltage.
- 480 V buses are fluctuating between 400 and 418 volts.

- 60 seconds after the fluctuations in grid voltage began, the generator tripped.
- 90 seconds after the generator trip, a manual safety injection signal was initiated due to lowering pressurizer level and pressure.

Assuming voltage conditions do not change, which of the following will occur first

- A. The 480 V bus normal feed breaker will open 9 seconds after the reactor trips.
- B. The 480 V bus normal feed breaker will open 180 seconds after the reactor trips.
- C. The 480 V bus normal feed breaker will open 9 seconds after the SI signal is initiated.
- D. The 480 V bus normal feed breaker will open 180 seconds after the SI signal is initiated.

Answer: C

Explanation/Justification:

Sustained Undervoltage (Degraded grid voltage) will start a 180 second timer. If a safety injection signal occurs, the 9 second timer will start. The timers are in parallel and either will initiate opening normal feed breakers when time expires. If voltage is not restored within 180 second (no SI) the normal bus supply breaker will trip. If voltage is not restore within 9 seconds with an SI the normal feed breaker is tripped

A. Incorrect but plausible because the breaker will open 9 seconds after a SI signal due to sustained undervoltage.

B. Incorrect but plausible because with bus voltage below approximately 421 V the normal feed breaker will open after 180 seconds due to sustained undervoltage.

C. Correct. The sustained undervoltage condition starts one of two timers. A 180 second timer starts if no SI signal is present. A 9 second timer starts if a SI signal is present.

D. Incorrect but plausible because the breaker will open 180 seconds after sustained undervoltage condition occurs without an SI signal.

Technical References: 2-ARP-FCF
Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS01 10
I2LP-ILO-EDS01 9

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	063000K402	
		Knowledge of D.C. Electrical System design feature(s) and/or interlock(s) which provide for the following: - Breaker interlocks, permissives, bypasses and cross-ties	
	Importance	2.9	3.2

Question # 47

A loss of 125 VDC Power Panel 21 has occurred. How can the operators confirm that Power Panel 21 was lost and that the associated ASCO transfer switches functioned properly?

- A. Breaker indication lights for 6.9KV Buses 5, 1, and 2 are ON.
Control Power Indicating Light for 21 EDG is WHITE.
- B. Breaker indication lights for 6.9KV Buses 5, 1, and 2 are OFF.
Control Power Indicating Light for 21 EDG is AMBER.
- C. Breaker indication lights for 6.9KV Buses 5, 1, and 2 are ON.
Control Power Indicating Light for 21 EDG is AMBER.
- D. Breaker indication lights for 6.9KV Buses 5, 1, and 2 are OFF.
Control Power Indicating Light for 21 EDG is WHITE.

Answer: B

Explanation/Justification:

The question is asking what the ASCO transfer switch supplies when the normal source is de-energized. When the transfer switch swaps to emergency supply EDG control power remains while 6.9 kv switchgear does not. It would be possible for the ASCO transfer switch to supply all DC control power to all components.

On a loss of 21 DC PP, the 6.9 KV switchgear will lose power. The ASCO switch will operate to provide alternate control power to 21 EDG as indicated by an amber light.

A. Incorrect. Plausible because candidate may believe that the ASCO transfer switch provides control power to 6.9 buses; it does not. Control Power indicating light for 21 EDG is incorrect but it is plausible a candidate could confuse the light color.

B. Correct answer.

C. Incorrect. Plausible because candidate may believe that the ASCO transfer switch provides control power to 6.9 buses; it does not. Control Power indicating light for 21 EDG is correct.

D. Incorrect. Plausible because the 6.9 KV breakers will lose control power. Control Power indicating light for 21 EDG is incorrect but it is plausible a candidate could confuse the light color.

Technical References: Syst Desc 27.1
 Syst Desc 27.3

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS03 3

Question Source: Bank

Question History: Unit 2 NRC 2016

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	063000K103	
		Knowledge of the physical connections and/or cause-effect relationships between the D.C. Electrical System and the following systems: - Battery charger and battery	
	Importance	2.9	3.5

Question # 48

Which of the following describes the 21 Battery response if Battery Charger 21 is lost with no operator action and DC loads remain constant?

21 Battery voltage will remain above a predetermined value for:

- A. 2 hours and then continue to drop at the same rate.
- B. 2 hours and then drop at a more rapid rate.
- C. 8 hours and then continue to drop at the same rate.
- D. 8 hours and then drop at a more rapid rate.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because 2 hour rating is correct; however, voltage will not decrease at a linear rate.

B. Correct. The 2 hour rating is TS basis and voltage decreases more rapidly as actual value decreases.

C. Incorrect. Plausible because 8 hour rating is expected if load are reduced and voltage will not decrease at a linear rate.

D. Incorrect. Plausible because 8 hour rating is expected if load are reduced but voltage decrease rate is correct.

Technical References: Syst Desc 27.5
 Tech Specs

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS03 11
 I2LP-ILO-EDS03 4

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	064000K102	
		Knowledge of the physical connections and/or cause-effect relationships between the ED/G System and the following systems: - ED/G cooling water system	
	Importance	3.1	3.6

Question # 49

Which one of the following describes operation Diesel Generator Cooling Water Outlet Flow Control valves, on an ESF actuation?

- A. Both valves open fully: each valve allows flow through both the Jacket Water and Lube Oil Heat Exchangers
- B. Both valves open fully; one valve allows flow through the JW Heat Exchanger while the other allows flow through the Lube Oil Heat Exchanger
- C. One valve opens fully; while the other remains closed but will open fully at the HIGH Jacket Water temperature alarm setpoint.
- D. One valve opens fully; while the other remains closed but will open fully at the HIGH Lube Oil temperature alarm setpoint.

Answer: A

Explanation/Justification:

A Correct

B Incorrect: Plausible because both valves open fully, but the flowpath is wrong

C Incorrect: Plausible because FCV 1176 does open fully

D Incorrect: Plausible because FCV-1176 does open fully

Technical References: 2-SOP-24.1
Syst Desc 24

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDSEDG 9
I2LP-ILO-SW001 11

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 4

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	073000A401	
		Ability to manually operate and/or monitor in the control room: - Effluent release	
	Importance	3.9	3.9

Question # 50

Given the following:

- A Waste Distillate Tank has been recirculated for 24 hours.
- Control room indications on R-54 Liquid Waste Distillate Monitor and confirmation sample shows that the water cannot be released .
- Current radiation level of $6.0 \text{ E-5 } \mu\text{Ci/ml}$ (limit $5.0 \text{ E-5 } \mu\text{Ci/ml}$).
- Available dilution flow restricts the release rate to no greater than the administrative limit.

Per 2-SOP-5.1.5, Calculation and Recording of Radioactive Liquid Releases, what action is required to be taken?

- A. The Waste Distillate tank will be recirculated until the radiation decays to an acceptable value.
- B. The Waste Distillate tank will be transferred to the Waste Holdup tank for further processing.
- C. The Waste Distillate tank will be diluted until activity per unit volume reaches acceptable value.
- D. The Waste Distillate tank will be transferred to a Waste Collection tank for further processing.

Answer: D

Explanation/Justification:

A Incorrect: Plausible because given enough time the student may believe radiation levels will decrease

B Incorrect: Plausible because student may know that waste will be transferred to a different tank for processing due to the current radiation levels; however they cannot transfer from the distillate tank to the holdup tank.

C Incorrect: Plausible because students may believe that dilution of the water will produce lower radiation levels

D Correct. The procedure directs transfer to another tank for reprocessing as the preferred action.

Technical References: 2-SOP-5.1.3

Proposed References to be provided: None

Learning Objective I2LP-ILO-LWR001 11

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	073000A202	
		Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Detector failure	
	Importance	2.7	3.2

Question # 51

The plant is operating at 100% power.
R-49, Steam Gen Hi Rad/Trouble alarm has just annunciated.
The detector indication has failed high.

Which of the following will occur as a result of the failure and what actions would the control room take?

Blowdown will isolate:

- A SG Sample will be isolated. Use Rad Bypass to restore SG sample.
- B SG Sample will remain in service. Use Rad Bypass to restore SG blowdown.
- C SG Sample will be isolated. Raise Alarm setpoint to restore SG sample.
- D SG Sample will remain in service. Raise Alarm setpoint to restore SG blowdown.

Answer: A

Explanation/Justification:

The question requires knowledge of location of SG Sample tap (upstream or downstream of blowdown isolation valves. .

A. Correct because SG sample is downstream of the blowdown isolation valves. Rad bypass position on the switches would be necessary to re-establish sample.

B. Incorrect but plausible if candidate believes SG sample tap off upstream of the blowdown isolation valves. Rad bypass would be used to re-establish blowdown.

C. Incorrect but plausible because sample will be isolated and raising the alarm setpoint is allowed in the procedure if activity just exceeds the WARN setpoint. This is used to identify degrading conditions.

D. Incorrect but plausible if candidate believes SG sample tap off upstream of the blowdown isolation valves, and raising the alarm setpoint is allowed in the procedure if activity just exceeds the WARN setpoint. This is used to identify degrading conditions.

Technical References: 2-SOP-7.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-SGSBD 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	076000A102	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: - Reactor and turbine building closed cooling water temperatures	
	Importance	2.6	2.6

Question # 52

Given the following:

- Service Water is aligned for 2 header operations.
- Service Water Header 123 is selected as the essential header.
- Service Water Header 21 22 23 24 25 26 High/Low Pressure alarm annunciated
- 24-25-26 Service Water Pump header pressure is 55 psig and lowering slowly.
- The Conventional NPO reported that TCV-1109 Temperature Control Valve Turbine Hall Closed Cooling has a large leak.
- The leak was isolated by closing TCV-1109 inlet and outlet isolation valves
- The bypass valve around TCV-1109 cannot be opened.

Which of the following describes a consequence of this condition and ARP actions?

- A. Manually trip the unit due to loss of cooling to the Main Boiler Feed Pump Lube Oil Coolers
- B. Unit Shutdown due to loss of cooling to the Condensate Pump Oil Coolers
- C. Unit Shutdown due to loss of cooling to the Generator Hydrogen Coolers
- D. Manually trip the unit due to loss of cooling to the Stator Water Coolers

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because Turbine Hall Closed Cooling provides cooling to the MBFP pedestal; however NOT the lube oil coolers.

B. Correct. Condensate Pump Oil Coolers are cooled from THCC. Loss of cooling will result in increasing temperatures and condensate pump shutdown. It will take some time for the temperatures to exceed limits so a shutdown is correct.

C. Incorrect. Plausible because the candidate may believe that the "clean" (not brackish water from the river) is preferred for cooling generator components; however, the Main Generator Hydrogen Coolers are cooled by Non-Essential Service Water

D. Incorrect. Plausible because the candidate may believe that the "clean" (not brackish water from the river) is preferred for cooling the stator water; however, the Stator Water Cooling System is cooled by Non-Essential Service Water.

Technical References: 2-AOP-FW-1
2-ARP-SEF

Proposed References to be provided: None

Learning Objective I2LP-ILO-SW001 13

Question Source: Bank

Question History: Unit 2 NRC 2012

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	076000K307	
		Knowledge of the effect that a loss or malfunction of the SWS will have on the following: - ESF loads	
	Importance	3.7	3.9

Question # 53

Given:

- The unit is at 100% power.
- Service Water is in 2 header operation
- The NPOs completed valve alignment making Service Water Header 1 2 3 the Essential Header.
- The Mode Selector Switch has the 4 5 6 header selected as Essential
- The plant experienced a unit trip coincident with a loss of 138kV Power.

1. Containment Fan Cooler Units
2. Iso-phase Bus Duct
3. CCW
4. Instrument Air Closed Cooling System
5. EDGs
6. Stator Water Cooling
7. Hydrogen Coolers

Which of the following lists component(s) that will NOT have cooling automatically restored?

- A. 1, 4, 5
- B. 3, 5, 7
- C. 2, 3, 5
- D. 1, 5, 6

Answer: A

Explanation/Justification:

All of the items listed are cooled by either Essential or Non-Essential Service Water. In addition, all of the items can have cooling supplied from either the 1 2 3 header or 4 5 6 header depending on the valve lineup. The Mode Selector Switch determines which pumps automatically start (Essential Service Water Pumps if everything is properly aligned).

The physical valve alignment has essential loads on the 1 2 3 Service Water header (Service Water header supplied by 21, 22, 23 SW pumps). The Mode Selector Switch designates the pump supplying the 4, 5, 6 header as essential (Service Water header supplied by 24, 25, 26 SW pumps). The blackout unit trip signal will start 24, 25, and 26 service water pump. This will result in service water to non-essential loads and no service water to essential loads.

A. Correct. These are loads designated to be Essential loads. With no pumps running in the essential header, these loads are not automatically restored.

B. Incorrect but plausible because candidate may believe that both CCW and H2 Coolers would require cooling following a unit trip with LOOP. Also EDG is a correct answer.

C. Incorrect but plausible because candidate may believe that iso-phase bus duct cooling and CCW would require cooling following a unit trip with LOOP. Also EDG is a correct answer.

D. Incorrect but plausible because candidate may believe that stator water cooling would be required following a unit trip with LOOP. Also Containment Fan Cooler Units and EDGs are correct answers.

A. Correct

B. Incorrect

C. Incorrect

D. Incorrect

Technical References: Syst Desc 24

Proposed References to be provided: None

Learning Objective I2LP-ILO-SW001 4

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 4

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	078000A401	
		Ability to manually operate and/or monitor in the control room: - Pressure gauges	
	Importance	3.1	3.1

Question # 54

The plant is operating at 100% power with the following indications and alarm SJF (1-5) "Instrument Air Low Pressure" just came into alarm.

Based on this, select the choice that best completes the following statement:

"The IA compressors ___(1)___ be running and PI-1144 Instrument Air Header pressure would be expected to indicate ___(2)___ .

- A. (1) will
(2) 95 psig
- B. (1) will
(2) 90 psig
- C. (1) will not
(2) 95 psig
- D. (1) will not
(2) 90 psig

Answer: B

Explanation/Justification:

During normal operation the Instrument Air and Station Air systems are supplied by one of two Unit 1 CENTAC compressors. Both Instrument Air Compressors are maintained in AUTO. In Automatic, both air compressors will start at 95 psig decreasing. The Instrument Air Low Pressure alarm is at 90 psig.

- A. Incorrect but plausible because both Instrument Air Compressors will be running but if the alarm just came in (given) the air pressure will indicate 90 psig,
- B. Correct. Both air compressors start at 95 psig decreasing and Low Pressure Alarm is at 90 psig.
- C. Incorrect but plausible because if candidate does not know the auto start or Low Pressure Alarm setpoints of the Instrument Air System.
- D. Incorrect but plausible because if candidate does not know the auto start setpoints of the Instrument Air compressors. The low pressure alarm setpoint is correct.

Technical References: 2-ARP-SJF

Proposed References to be provided: None

Learning Objective I2LP-ILO-SA01 4

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	103000A203	
		Ability to (a) predict the impacts of the following malfunctions or operations on the Containment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Phase A and B isolation	
	Importance	3.5	3.8

Question # 55

The following plant conditions exist on Unit 2:

- Reactor is at 100% RTP
- A manual Phase A isolation signal was inadvertently actuated on Train A.

Which of the following are direct results of the phase A signal and minimum corrective actions required, and their sequence, to restore letdown flow?

	Results	Actions
A.	201, "Letdown Line Normal Path Isolation" and all orifice isolation valves close.	Reset Phase A Restore Instrument Air to VC Open 201 PCV 135 to manual Open Orifice Isolation Valve PCV-135 to Auto
B.	LCV-459, Letdown Isolation Loop 21 and 201, "Letdown Line Normal Path Isolation" close	Reset Phase A Restore Instrument Air to VC Open 201 Open 459 Open Orifice Isolation Valve PCV 135 to manual
C.	201, "Letdown Line Normal Path Isolation" and all orifice isolation valves close.	Reset Phase A Restore Instrument Air to VC Open 201 Open 459 PCV 135 to manual Open Orifice Isolation Valve

D.	LCV-459, Letdown Isolation Loop 21 and 201, "Letdown Line Normal Path Isolation" close	Reset Phase A Restore Instrument Air to VC Open 201 PCV 135 to manual Open Orifice Isolation Valve PCV-135 to Auto
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Answer: C

Explanation/Justification:

459 does not close directly from the Phase A signal. The valve will close due to loss of instrument air to containment. 201 will close on Train A phase A. All orifice isolation valve will close on either train of phase A.

A. Incorrect but plausible because the Results are correct, but the actions does not include opening 459. Letdown flow will not be restored.

B. Incorrect but plausible because 459 will close but not directly from the phase A signal. Also PCV-135 is placed in manual before opening the orifice isolation valve.

C. Correct. 201 and orifice isolation valves get phase A signals to close. Letdown restoration sequence is correct.

D. Incorrect but plausible because 459 will close but not directly from the phase A signal. Valve sequence in Actions does not include reopening 459.

Technical References: 2-SOP-3.1
Drawing

Proposed References to be provided: None

Learning Objective I2LP-ILO-CVCS 5

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	001000K407	
		Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following: - Rod stops	
	Importance	3.7	3.8

Question # 56

Which of the following malfunctions will cause the Overtemperature ΔT Channel Trip or Rod Stop alarm to annunciate?

- A. PR Channel N41 blown Instrument Power fuse
- B. Loop 22 Thot RTD low
- C. PT-456, PRZR Pressure, fails high
- D. Loop 21 Tcold RTD fails low

Answer: D

Explanation/Justification:

- A. Incorrect. Plausible because NIS does provide an input to OT delta T; however, the input is from delta flux.
- B. Incorrect. Plausible because the candidate must determine how the failure will affect actual delta T and how the calculated setpoint will be affected.
- C. Incorrect. Plausible because candidate must recall that Channel 2 (PT-456) cannot be the controlling channel. If a controlling channel failed high, actual pressure would decrease and the alarm would annunciate when the first OT delta T channel setpoint was reached.
- D. Correct. This failure causes the indicated delta T to fail high. The alarm will annunciate when the calculated setpoint is less than the indicated delta T.

Technical References: Tech Specs
Proposed References to be provided: None

Learning Objective I2LP-ILO-ICRXP 4

Question Source: Bank

Question History: Unit 3 NRC 2010

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	011000K603	
		Knowledge of the effect of a loss or malfunction of the following will have on the PZR LCS: - Relationship between PZR level and PZR heater control circuit	
	Importance	2.9	3.3

Question # 57

A rapid load rejection occurred due to closure of a turbine stop valve. You observe the following:

- pressurizer sprays partially open
- modulating heaters full off
- backup heaters on

Which of the following would cause these indications?

- A. Controlling pressurizer pressure channel failed high.
- B. Pressurizer level increase from the downpower.
- C. Controlling pressurizer level channel failed low.
- D. Pressurizer program level decrease from the downpower.

Answer: B

Explanation/Justification:

- A. Incorrect. Plausible because this instrument failure would lead to sprays fully open, backup heaters off and a rapid reduction in actual pressure. The failure would also cause BU heaters to de-energize NOT energize.
- B. Correct. The load rejection due to closure of stop valve, results in a heatup and insurge into the pressurizer; heaters will reduce pressure drop on subsequent outsurge.
- C. Incorrect. Plausible because the pressurizer heaters would respond this way if the level channel failed high instead of low.
- D. Incorrect. Plausible because programmed pressurizer level will decrease when T_{avg} decreases as a result of control rod insertion. This will result in decrease pressure and potential heater on. The sprays would not be open.

Technical References: Syst Desc 1.4
Proposed References to be provided: None

Learning Objective I2LP-ILO-RCSPZR 4
I3LP-ILO-RCSPZR 3

Question Source: Bank

Question History: Unit 3 NRC 2013

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	017000A101	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM System controls including: - Core exit temperature	
	Importance	3.7	3.9

Question # 58

A Loss of Coolant Accident has occurred; Control Room temperature has increased 15 deg-F above its normally maintained environment. As a result, the temperature in all instrument racks has risen by at least that amount. What impact will this change have upon the incore instrumentation information available to the operators?

- A. CET readings will be falsely high by at least 15 deg-F and can be used for trending purposes up to 2300°F
- B. CET readings will be falsely high by at least 15 deg-F and can be used for trending purposes up to 3300°F
- C. No effect on CET readings as they are temperature compensated and can be used for trending purposes up to 2300°F
- D. No effect on CET readings as they are temperature compensated and can be used for trending purposes up to 3300°F

Answer: C

Explanation/Justification:

Thermocouples develop a potential difference (voltage based on the difference in temperatures between measured variable junction and reference junction. The original design of the system used a reference junction box that was maintained at a fixed temperature using electrical heaters. If the reference junction temperature changed, the indication of the CETs changed by the same amount. Currently, a microprocessor is used to compensate the indication based on the reference junction in the control room. If the RTD used for control room temperature compensation failed either high or low, the thermocouples would indicate incorrectly.

A Incorrect: Plausible because 2300°F is correct; however, there is no effect on CET readings.

B Incorrect: Plausible because 3300°F is close to 2300°F and there is no effect on CET readings.

C Correct the indication is compensated by a microprocessor and 2300° is correct for upper limit

D Incorrect: Plausible because there is no effect on CET readings but 3300°F is is not correct but

Technical References: Syst Desc 14
Proposed References to be provided: None

Learning Objective I2LP-ILO-ICNXC 4
I2LP-ILO-ICNXC 7

Question Source: Modified

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	029000A404	
		Ability to manually operate and/or monitor in the control room: - Containment evacuation signal	
	Importance	3.5	3.6

Question # 59

Given:

- Refueling outage is in progress.
- A Containment Purge is in progress.
- The Nuc NPO initiated 21 Large Gas Decay tank release

Subsequently:

- RCV-014 automatically closed.

Which of the following describes the additional impact of this event?

Containment Purge ___(1)___ be automatically secured and the containment evacuation alarm ___(2)___ sound in the VC.

- A. (1) will
(2) will not
- B. (1) will
(2) will
- C. (1) will not
(2) will
- D. (1) will not
(2) will not

Answer: A

Explanation/Justification:

RCV-014 closing indicates that R-44 has exceeded the ALARM setpoint. R-44 provides protection during gas tank release. It automatically isolates RCV-014 when the ALARM setpoint is reached. It also causes a containment ventilation isolation signal (secure containment purge or pressure relief). It does not however cause a containment evacuation alarm. R-41/42 cause containment ventilation isolation signals AND containment evacuation alarm.

- A. Correct. Radiation Monitor 44 will actuate containment purge isolation at the alarm setpoint. Unlike R41/42, R-44 will NOT cause containment evacuation alarm.
- B. Incorrect but plausible because containment purge will be automatically isolated but the evacuation alarm will not sound.
- C. Incorrect but plausible because candidate may believe that a waste gas decay tank release will not affect containment purge and that purge is only controlled by R41/42.. Candidate may believe that since R-44 could signal problems in containment or PAB that the alarm would sound.
- d. . Incorrect but plausible because candidate may believe that a waste gas decay tank release will not affect containment purge and that purge is only controlled by R41/42. It is correct that R-44 will not actuate containment evacuation alarm.

Technical References: 2-SOP-5.4.3

Proposed References to be provided: None

Learning Objective I2LP-ILO-RMS001 4

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	033000K105	
		Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Pool Cooling System and the following systems: - RWST	
	Importance	2.7	2.8

Question # 60

Given:

- The unit is at 100% power
- Spent Fuel Pit Level alarm is annunciated
- SFP transfer gate is reported leaking into the transfer canal
- SFP level is 92'-3" and lowering as reported by NPO
- SFP boron concentration is 2000 ppm

Which method listed below is appropriate for restoring SFP level?

Fill the SFP with:

- A. Blended Makeup from CVCS to the SFP at maximum possible rate.
- B. RWST purification booster pump limited to a maximum of 100 gpm.
- C. Primary Water (PW) using 723, PW makeup to SFP, and then add boron to the pool.
- D. PW using a hose clamped below the water line to reduce possibility of splashing, and then add boron to the pool.

Answer: B

Explanation/Justification:

Tech Spec level SFP >23' above active fuel (92'-2"), boron concentration 2000ppm

This is was a real event at Unit 2. The boron concentration is at the TS limit. Adding PW will lower the SFP boron concentration and is not allowed because it would result in entering a TS action. The given level is 12" below the alarm setpoint and 1" above the TS limit.

A. Incorrect, Plausible because borated water is required for the conditions given; however Blended Makeup from CVCS cannot be aligned to the SFP.

B Correct

C. Incorrect but plausible because normal make up to the SFP, to compensate for evaporation, is from valve 723. Adding PW at that amount will reduce SFP boron concentration below TS value.

D. Incorrect but plausible because normal make up to the SFP can be added from a hose connected to a PW supply. The hose connection is used at the CRS/SM discretion to augment fill from another source or fill at a more controlled rate. Hose is not allowed to be submerged in pool due to siphon concerns. Adding PW at that amount will reduce SFP boron concentration below TS value

Technical References: 2-SOP-4.3.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-SFP001 8

Question Source: New

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	035000K503	
		Knowledge of the operational implications of the following concepts as they apply to the S/GS: - Shrink and swell concept	
	Importance	2.8	3.1

Question # 61

The plant is operating at 100% steady state power when a load limit failure causes a 100 MWe load rejection. Which of the following correctly describes the initial SG level response?

- A. SG level decreases due to shrink effects of the load rejection
- B. SG level increases due to swell effects of the load rejection
- C. SG level decreases due to a mismatch between Steam Flow and Feed Flow
- D. SG level increases due to a mismatch between Steam Flow and Feed Flow

Answer: A

Explanation/Justification:

A. Correct.

B. Incorrect but plausible because SG level changes due to shrink effect; however level decreases not increases.

C. Incorrect but plausible because level will decrease, but not due to Steam Flow Feed Flow mismatch.

D. Incorrect but plausible because it would seem that SG level will increase initially when steam flow is lowered before feed flow.

Technical References: Syst Desc 1.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-ICSGL 4

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	041000K201	
		Knowledge of bus power supplies to the following: - ICS, normal and alternate power supply	
	Importance	2.8	2.9

Question # 62

The unit was operating at 22% power.

21 DC Distribution Panel Circuit 2 breaker tripped resulting in a loss of power to Channel 1 Low Pressure Steam Dump Valves.

Following a unit trip, Channel 1 LP Steam Dump Lockout Relay will be:

- A. tripped and all low pressure steam dump valves will operate as designed
- B. tripped and half of the low pressure steam dump valves will operate as designed
- C. not tripped and all low pressure steam dump valves will operate as designed
- D. not tripped and half of the low pressure steam dump valves will operate as designed

Answer: C

Explanation/Justification:

The question is asking if DC power is required to trip the 86 relays or if loss of power will cause them to trip. In addition, the question asks how the low pressure steam dumps will operate with this failure. Adding power improved the plausibility of distractors where half of the steam dumps operate. High pressure steam dumps operate based on magnitude of the error signal and at lower power the error signal will be lower. This is not true for low pressure steam dumps

A. Incorrect but plausible because without power the 86 relay will NOT trip. It is true that all steam dump valves will operate as designed.

B. Incorrect but plausible because without power the 86 relay will NOT trip. It is also plausible to believe that ½ of the LP steam dumps are adequate to prevent an overspeed condition.

C. Correct.

D. Incorrect but plausible because the 86 will not trip without power and it is plausible to believe that ½ of the LP steam dumps are adequate to prevent an overspeed condition

Technical References: Syst Desc 18.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-SDSLP 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	1	
	K/A#	045000K301	
		Knowledge of the effect that a loss or malfunction of the MT/G System will have on the following: - Remainder of the plant	
	Importance	2.9	3.2

Question # 63

Given:

- A plant startup is in progress per POP 1.3
- Reactor Power: 15%
- Generator Megawatts: 120 Mwe
- Tavg: 550 °F
- One MBFP in service
- Rod control in manual
- Steam Dump control in automatic PRESSURE mode

Subsequently:

- Condenser Vacuum decreased to 18"hg.

Which of the following describes the status of the plant?

- A. SG level control from Main Feed, steam dumps will be closed, control rods height will be inserted to restore Tavg to 547°.
- B. SG level control from Main Feed, steam dumps will restore Tavg to approximately 547°; control rods height will remain unchanged.
- C. SG level control from Main and Aux Feed, steam dumps will restore Tavg to approximately 547°, control rods height will remain unchanged.
- D. SG level control from Main and Aux Feed, steam dumps will be closed, control rods height will be inserted to restore Tavg to 547°.

Answer: A

Explanation/Justification:

18" of vacuum will cause a turbine trip and block of steam dumps. Power is below P-8 setpoint and the reactor will not trip. SGs will not shrink low enough to start AFW and MFW will remain operating.

A. Correct. SG level control will be in manual. Steam dumps will be closed because low vacuum. Rods will have to be inserted to restore Tavg.

B. Incorrect but plausible because SG level control will be from MFW. Candidate may believe Steam Dumps will control restore Tavg to 547°; with rods in manual temperature will not lower.

C. Incorrect but plausible because the main boiler feed pump would remain in operation, but Candidate may believe AFW will also be in service. Candidate may believe Steam Dumps will control restore Tavg to 547°; however with rods in manual temperature will not lower.

D. Incorrect but plausible because the main boiler feed pump would remain in operation, but Candidate may believe AFW will also be in service. It is correct that the steam dump valves will be closed and rods inserted to control temperature

Technical References: 2-AOP-TURB-1
Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPTUR 3

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	2	
	Group#	2	
	K/A#	055000A303	
		Ability to monitor automatic operation of the CARS, including: - Automatic diversion of CARS exhaust	
	Importance	2.5	2.7

Question # 64

Given:

- A SG tube leak of 150 gpd has been confirmed
- The crew was performing a rapid shutdown per 2-AOP-SG-1, Steam Generator Tube Leak
- A Steam Generator Tube Rupture occurs.
- Reactor Tripped.
- Safety Injection actuated.
- The crew is performing actions of E-0, Reactor Trip or Safety Injection.

During the rapid shutdown, condenser air ejector exhaust will be lined up to ___(1)___ and during the performance of E-0 it will be lined up to ___(2)___?

- A. Turbine Hall Roof.
Turbine Hall Roof.
- B. Containment
Containment.
- C. Turbine Hall Roof
Containment.
- D. Containment
Turbine Hall Roof

Answer: D

Explanation/Justification:

Per NOTE in AOP-SG-1, R-45 will divert to the containment when leakage exceeds 75 gpd. Also this level of leakage requires a Rapid Shutdown. With a leakrate of 150 gpd R-45 has diverted air ejector exhausted to containment.

A note in AOP-SG-1 states that the air ejector exhaust diverting to containment is positive indication of a tube leak of > 75 gpd. Also air ejector exhaust diverts at the Alarm level.

A Incorrect. Plausible because exhaust did not reach the setpoint to divert during the tube leak and the SI (Phase A) will restore exhaust to the turbine hall roof.

B Incorrect. Plausible because exhaust would divert to containment when R-45 reached the Alarm setpoint.; however, after a safety Injection, the containment isolation valves close and exhaust returned to normal.

C Incorrect. Plausible because exhaust did not reach the setpoint to divert during the tube leak but did reach the divert setpoint during the SGTR.

D Correct because the divert setpoint was reached during the tube leak and the SI (Phase A) will restore exhaust to the turbine hall roof.

Technical References: Syst Desc 20

Proposed References to be provided: None

Learning Objective I2LP-ILO-RMS001 3

Question Source: Bank

Question History: Unit 2 2016 NRC

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		
	Group#		
	K/A#	056000A204	
		Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of condensate pumps	
	Importance	2.6	2.8

Question # 65

Given the following conditions:

- The plant is in a startup with reactor power level at 80% - INCREASING
- Main Turbine Generator 870 MWE
- All control systems, except Rod Control, are in automatic
- The following equipment is operating:
 - 21 Condensate Pump
 - 23 Condensate Pump
 - 21 Main Boiler Feed Pump
 - 22 Main Boiler Feed Pump
 - High Pressure Steam Dump system is in Temperature Mode

Which ONE of the following is the expected plant response following a trip of the 23 Condensate Pump, and required action?

- A. 22 Condensate Pump will automatically start.
Defeat auto trip of 22 condensate pump.
- B. Turbine runback does not occur.
Reduce load to less than 700 MWE.
- C. Motor Driven Auxiliary Feed Water Pumps automatically start.
Secure Motor Driven Auxiliary Feed Water Pumps and place in AUTO.
- D. Main Boiler Feedwater Pump in Track & Hold.
Reset Track & Hold and return MBFPs to AUTO.

Answer: A

Explanation/Justification:

A. Correct. 22 Condensate Pump will automatically start on low feed pump suction pressure. 22 condensate will also automatically trip on a reactor trip to minimize loads transferred to Station Aux Transformer, and minimizes water hammer in condensate system. If 22 condensate pump auto starts, the auto trip is defeated.

B. Incorrect but plausible because a turbine runback does occur if a main boiler feed pump trips if turbine load is greater than 85%. Reducing load to less than 700 MWE is a value in 2-AOP-FW-1 if a single heater drain pump trips.

C. Incorrect but plausible because motor driven afw pumps automatically start if a Main Boiler Feed Pump trips. Securing the pumps and restoring automatic control would be correct if they started.

D. Incorrect but plausible because track and hold is a protective feature of the Main Boiler Feed Pump speed control system. The conditions to actuate T&H will not occur, but if it did resetting track and hold is the action taken.

Technical References: 2-AOP-FW-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-CND01 8

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012113	
		Conduct of Operations - Knowledge of facility requirements for controlling vital / controlled access.	
	Importance	2.5	3.2

Question # 66

Personnel are off-loading the core during a refueling outage with the following conditions:

- An attempt was made to place an irradiated fuel assembly in the upender inside Containment.
- Another irradiated fuel assembly was already in the upender.
- Some damage has occurred to at least one of the assemblies and both are stuck in position.
- 10 industry experts (non-badged visitors) have been briefed by RP in the Training Building and now need to be escorted into Containment to get a close-up view of conditions to provide recommendations.

What is the MINIMUM number of escorts required?

- A. 1
- B. 2
- C. 4
- D. 5

Answer: B

Explanation/Justification:

A. Incorrect but plausible because 1 person can escort up to 10 individual in the Protected Area.

B. Correct

C. Incorrect but plausible because if an individual could escort 3 individuals, this would be correct.

D. Incorrect but plausible because if an individual could escort 2 individuals, this would be correct.

Technical References: EN-NS-232

Proposed References to be provided: None

Learning Objective IOWKB-ILO-ADM00 2.1.13

Question Source: Bank

Question History: NA

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012140	
		Conduct of Operations - Knowledge of refueling administrative requirements.	
	Importance	2.8	3.9

Question # 67

While entering a refueling outage, the reactor trip breakers are opened at 2000 on April 1st.

In accordance with Tech Specs, which of the following is the soonest that movement of irradiated fuel assemblies is allowed?

Assume all other conditions which must be met to move fuel are met.

- A. 0800 April 4th
- B. 2000 April 4th
- C. 0800 April 5th
- D. 2000 April 5th

Answer: C

Explanation/Justification:

Minimum time to be subcritical is 84 hours. This is found in Tech Spec Bases 3.9.6 and TRM 3.9A Decay Time Refueling

for refueling cavity water level, as well as in the UFSAR Chapter 9 for Spent Fuel Pool Cooling, and Section 14.2 Accident Analysis for a Fuel Handling Accident.

- A. Incorrect but plausible the time of day is correct but the day is not.
- B. Incorrect but plausible because 72 hours is a common action time in Tech Specs.
- C. Correct. 84 hours is correct per TS Basis and TRM 3.9.A
- D. Incorrect but plausible because 96 hours is a common action time in Tech Specs.

Technical References: FSAR
Tech Specs

Proposed References to be provided: None

Learning Objective I3LP-ILO-POP003 9

Question Source: Bank

Question History: Unit 3 Retake

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012235	
		Equipment Control - Ability to determine	
		Technical Specification Mode of Operation.	
	Importance	3.6	4.5

Question # 68

Given the following conditions:

- A plant startup is in progress
- RCS temperature is 185°
- RCS pressure is 340 psig with a bubble in the pressurizer
- 24 RCP is in service

The Outage Control Center has requested that the 80' air lock be defeated.

The unit is in ____ (1) ____ and the airlock ____ (2) ____ be defeated.

- A. (1) MODE 4
(2) can
- B. (1) MODE 4
(2) can NOT
- C. (1) MODE 5
(2) can
- D. (1) MODE 5
(2) can NOT

Answer: C

Explanation/Justification:

A. Incorrect but plausible because at 340 psig the pressurizer is above 200°F; however, the pressurizer is not part of the RCS for mode considerations.

B. Incorrect but plausible because at 340 psig the pressurizer is above 200°F; however, the pressurizer is not part of the RCS for mode considerations. The candidate may believe that the operation of the RCP will impact the ability to defeat the airlock.

C. Correct.

D. Incorrect but plausible because the unit is in Mode 5 and the candidate may believe that the operation of the RCP will impact the ability to defeat the airlock.

Add to references

Tech Spec Section 1.1 and TS 3.6.2.

Technical References:	Tech Specs
Proposed References to be provided:	None

Learning Objective	I2LP-ILO-VCVCB 9 I3LP-ILO-VCVCB 9
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Question Source:	Bank
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Question History:	Unit 3 NRC 2018 Retake
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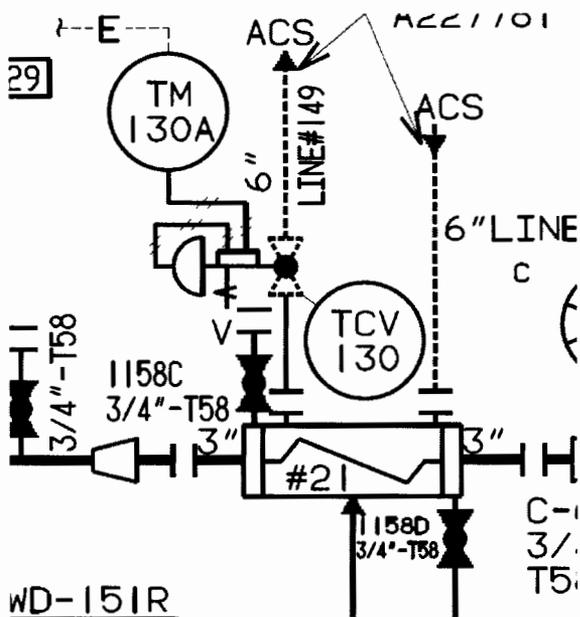
Question Cognitive Level:	Fundamental Knowledge
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10 CRF Part 55 Content:	55.41 (b) 7
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Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012241	
		Equipment Control - Ability to obtain and interpret station electrical and mechanical drawings.	
	Importance	3.5	3.9

Question # 69



Given partial excerpt from the Chemical and Volume Control flow diagram, which of the following describes the significance of broken/dashed lines used for TCV-130?

- A. The valve is included here for information. The actual valve with details appears with solid lines on another drawing.
- B. The valve seat is undercut or valve is mechanically prevented from fully closing. There is always some minimum flow through this flowpath.
- C. Operation of this valve has the potential to cause a change in reactivity.
- D. The interfacing CCW system is classified as a "closed system" for interface LOCA.

Answer: A

Explanation/Justification:

- A. Correct. The "ghosted" component indications are used for information and should not be used for tagging purposes because additional detail may exist on another drawing.
- B. Incorrect. Plausible because some valve are undercut to prevent thermal locking during heatup and cooldown activities. Also some valve have a minimum/maximum opening or closing mechanical stop.
- C. Incorrect. Plausible because TCV-130 has the potential to affect reactivity by causing letdown excessive cooling or heatup.
- D. Incorrect. Plausible because the FSAR classifies some systems as Open or Closed. Isolation valves for all fluid system lines penetrating the containment provide at least two barriers against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis.

Technical References:

Proposed References to be provided: None

Learning Objective I0LP-ILO-BOT001 2

Question Source: Bank

Question History: Unit 2 NRC 2010

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012307	
		Radiological Controls - Ability to comply with radiation work permit requirements during normal and abnormal conditions.	
	Importance	3.5	3.6

Question # 70

You have been directed to perform 2-PT-Q028A, 21 Residual Heat Removal Pump Test. The following area dose rates exist:

- Maximum area dose 7 mr/hr
- Average area dose 3 mr/hr
- Low Dose Area 0.2 mr/hr

In accordance with EN-RP-105, Radiological Work Permits, which of the following choices identifies

- (1) Minimum ACCUMULATED DOSE at which personal protective action(s) is/are required and
 - (2) the personal protective action(s)?
- A. (1) 10 mrem
(2) Exit the radiologically controlled area.
- B. (1) 10 mrem
(2) Move to low dose area and contact RP.
- C. (1) 8 mrem
(2) Exit the radiologically controlled area.
- D. (1) 8 mrem
(2) Move to low dose area and contact RP.

Answer: C

Explanation/Justification:

A. Incorrect but plausible because the RWP states the dose alarm is 10 mrem; however the Stop Work section of the RWP states 80% of the ED dose set point. Also Exiting the area is correct for a dose rate alarm.

B. Incorrect but plausible for the same reason above. Candidate may believe that moving to a low dose area is acceptable if accumulated dose is below limit.

C. Correct

D. Incorrect but plausible because dose is correct, but the candidate may believe that moving to a low dose area is acceptable if accumulated dose is below limit.

Technical References: EN-RP-105
RWP 20162002

Proposed References to be provided: RWP 20162002

Learning Objective

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 12

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012314	
		Radiological Controls - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	
	Importance	3.4	3.8

Question # 71

Given the following conditions:

You are assigning an operator to inspect the reactor vessel o-ring flange.

Dose rate at the flange is 3000 mRem/hr at 30 cm.

The operator has a lifetime TEDE of 750 mRem.

The operator has 200 mRem TEDE for this year.

In accordance with EN-RP-100 (Rad Worker Expectations) and EN-RP-201 (Dosimetry Administration):

The work area must be posted as a (1) Area.

The MAXIMUM time the operator can stay at the flange and not exceed the routine annual administrative guideline for TEDE is (2) minutes.

- A. (1) Locked High Radiation
(2) 16
- B. (1) Locked High Radiation
(2) 36
- C. (1) Very High Radiation
(2) 16
- D. (1) Very High Radiation
(2) 36

Answer: B

Explanation/Justification:

This question was used by McGuire in 2015. Some modifications were made due to procedure differences.

The routine annual administrative guideline for TEDE is 2000 Mrem. $36 \times 3000/60 = 1800$. $1800 + 200 = 2000$. 3000 mRem/hr would be a locked high radiation area.

A. Incorrect but plausible since the candidate could believe there is a 1000 mRem routine limit. It is rare that operators come close to their limit these days, so this is a plausible misconception.

B. Correct answer.

C. Incorrect but plausible this is a much higher dose rate than often encountered. The second part is plausible as discussed for choice A.

D. Incorrect but plausible as discussed for A and C.

Technical References: EN-RP-201

Proposed References to be provided: None

Learning Objective IOLP-ILO-ADM01 1

Question Source: Bank

Question History: McGuire 2015

Question Cognitive Level: Analysis Synthesis Evaluation

10 CRF Part 55 Content: 55.43 (b) 4

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012315	
		Radiological Controls - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
	Importance	2.9	3.1

Question # 72

Given:

- A refueling outage in progress
- Containment integrity is set
- A high burnup fuel assembly with a suspected pin leak is in transfer from the Vapor Containment to the Fuel Building
- A ventilation supply fan trip just tripped, resulting in fuel building pressure becoming more negative
- work on 21 RCP seal resulted in an inadvertent 15 gpm leak onto the floor
- rad workers cleaning containment are staging waste bags by the airlock for removal from containment
- R-2, Containment Area Monitor goes into high alarm

Which of the following is the likely cause of the radiation alarm?

- A. Fuel movement from the Vapor Containment to the Fuel Building.
- B. A drop in Refueling Cavity level due to building pressure differential.
- C. Staging of radioactive cleanup waste by VC airlock.
- D. Coolant leaking from 21 Reactor Coolant Pump seal.

Answer: C

Explanation/Justification:

- A. Fuel movement will not place R-2 in alarm. R-2 is located on the 80 foot level near the containment airlock.
- B. Refueling cavity level during refueling operation prevents the general area radiation level from increasing to the alarm point on R-2
- C. Radioactive material staged near the containment airlock will be detected by R-2.
- D. Increased airborne contamination could increase general area radiation levels; however, it would have been detected by the containment particulate monitor first.

Technical References: 2-ARP-SBF-2
3-ARP-005

Proposed References to be provided: None

Learning Objective I2LP-ILO-RMS001 4

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 12

Comments

Question #73 contains potentially sensitive information and is filed separately as a non-public document.

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012442	
		Emergency Procedures/Plan - Knowledge of emergency response facilities.	
	Importance	2.6	3.8

Question # 74

Given:

- An event has escalated to a Site Area Emergency.
- Emergency Response Facilities have been declared Operational.
- An OFF-SHIFT NPO contacts the Control Room and asks where to report.

Which of the following identifies where you would direct the NPO to report?

- A. Control Room
- B. Operations Support Center
- C. Technical Support Center
- D. Site Assembly Area for evacuation

Answer: B

Explanation/Justification:

When an emergency event is declared, only watchstanding NPOs report to the control room. All available non-watchstanding NPOs will be directed to the OSC. They will be directed to the OSC whether they call in from off-site or on-site.

A. Incorrect. Plausible because watchstanding NPOs report to the Control Room and are dispatched from the control room for events UE or Alert.

B. Correct.

C. Incorrect. Plausible because the OSC and TSC at IPEC are in the same general area. Candidate must know the function of each facility to direct the NPOs to the correct location.

D. Incorrect. Plausible because Site Assembly area is where non-essential personnel are directed. This does not include qualified NPOs.

Technical References: IP-EP-210

Proposed References to be provided: None

Learning Objective I0LP-ILO-ERT001 1

Question Source: Bank

Question History: Unit 3 NRC 2017

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012449	
		Emergency Procedures/Plan - Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	
	Importance	4.6	4.4

Question # 75

Given the following:

Small break LOCA mitigation is in progress in accordance with E-1 "Loss of Reactor or Secondary Coolant".

RHR pumps have been secured in accordance with Step 13 of that procedure.

Containment pressure is 3 psig.

WHICH ONE (1) of the following would require the manual start of the RHR pumps?

- A. RCS pressure is 300 psig.
- B. RCS subcooling margin is 10°F
- C. PZR level cannot be maintained above 4%.
- D. Transfer to Post LOCA Cooldown and Depressurization desired.

Answer: A

Explanation/Justification:

A. Correct.

B. Incorrect but plausible because this does satisfy one of the safety injection reinitiation criteria.

C. Incorrect but plausible because this does satisfy one of the safety injection reinitiation criteria

D. Incorrect but plausible because RHR flow < 240 gpm is a criterion for transition to ES-1.2 Post LOCA Cooldown and Depressurization

Technical References: 2-E-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE10 3
I2LP-ILO-EOPE10 6

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		1
	K/A#	0000152120	
		Conduct of Operations - Ability to interpret and execute procedure steps.	
	Importance	4.6	4.6

Question # 76

Given:

- Reactor Power is 100%
- 23 RCP Seal Return Flow is >6 gpm
- 23 RCP Shaft Vibration is 16 mils and increasing @ 2 mil/hr
- 23 RCP #1 Seal Inlet Temperature is 120°F and stable
- 23 RCP Seal Return Temperature is 135°F and stable
- 23 RCP Stator Winding Temperature is 175°F and stable

The following annunciators are in alarm:

- 23 RCP HIGH VIBRATION
- RCP NO. 1 SEAL RETURN HIGH/LOW FLOW (COMMON)

The CRS has announced entry into AOP-RCP-1.

Which of the following states the proper sequence of actions based on 23 RCP indications?

- A. Trip the reactor, Trip 23 RCP, Initiate E-0, Reactor Trip or Safety Injection, Close 261C 23 RCP Seal Return Valve and 455B RCP 23 Spray Valve.
- B. Trip the reactor, Trip 23 RCP, Close 455B RCP 23 Spray Valve, Go to E-0, Reactor Trip or Safety Injection.
- C. Initiate POP-2,1 (Operation At Greater Than 45% Power) and POP 3.1 (Plant Shutdown From 45% Power), Trip 23 RCP within 8 hours.
- D. Initiate POP-2.1 (Operation At Greater Than 45% Power) and POP 3.1 (Plant Shutdown From 45% Power). 23 RCP Operation can continue.

Answer: A

Explanation/Justification:

A Correct Seal Return Flow > 6 gpm is one of the emergency shutdown criteria in AOP-RCP-1.

B. Incorrect but plausible because the Reactor and RCP should be tripped; however the sequence is not correct (Go To E-0 vs Initiate E-0) and the seal return valve must be closed.

C. Incorrect but plausible because a normal shutdown would be performed if seal return flow is > 5 but < 6 gpm, but it does not direct securing the RCP.

D. Incorrect but plausible because a normal shutdown would be performed if seal return flow is > 5 but < 6 gpm, it does not direct securing the RCP.

Technical References: 2-AOP-RCP-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPRCP 4

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		1
	K/A#	0000262119	
		Conduct of Operations - Ability to use plant computers to evaluate system or component status	
	Importance	3.9	3.8

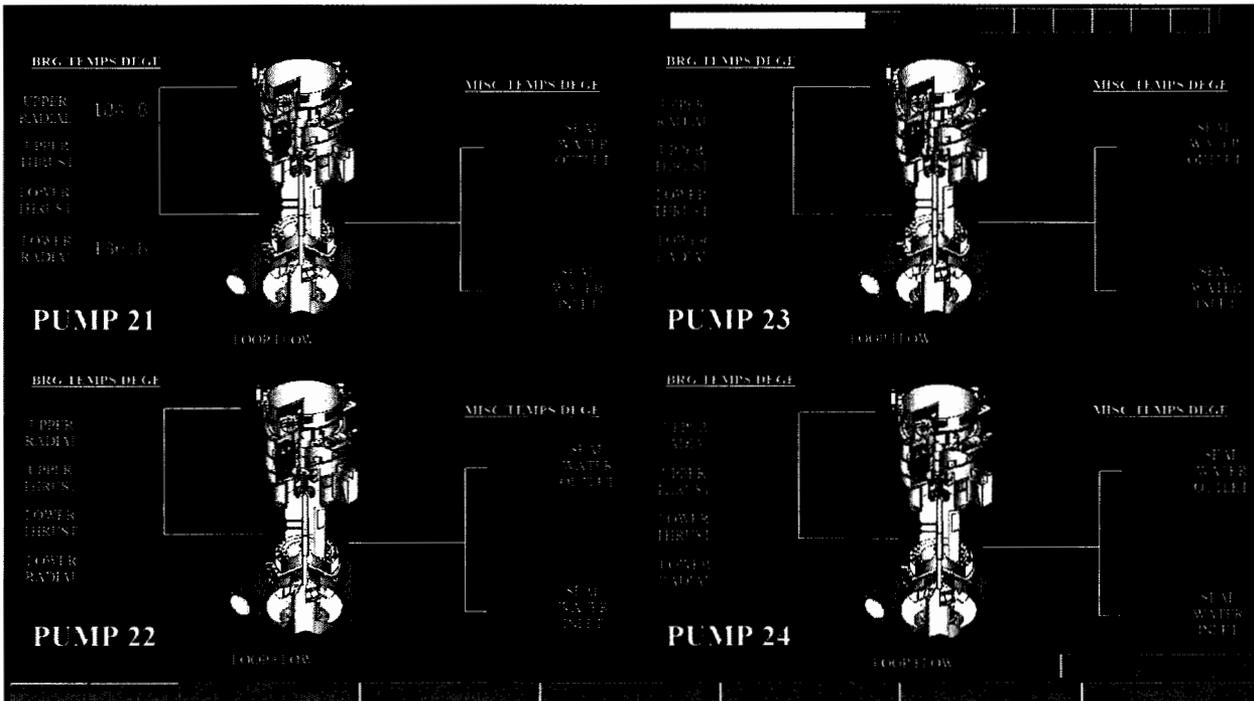
Question # 77

Given:

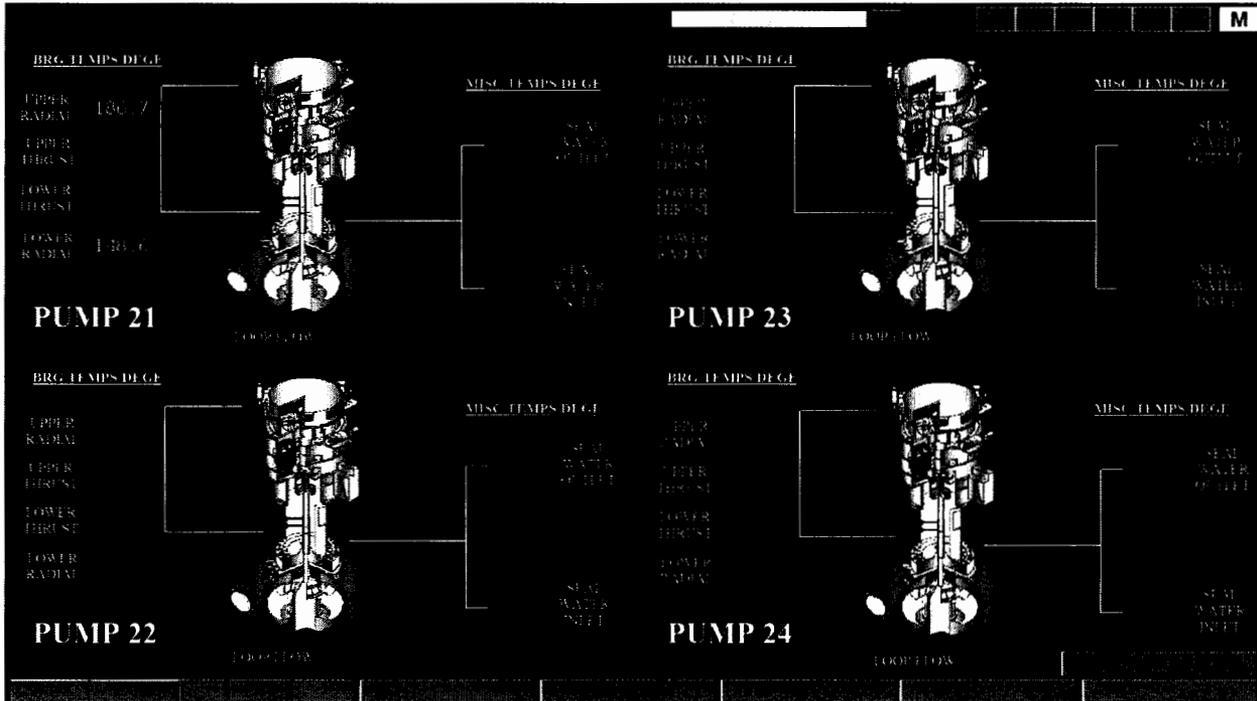
- Unit is at 40% Power.
- CCW Surge Tank level was lowering.

Subsequently:

- NPO initiated makeup to CCW Surge Tank.
- CCW Surge Tank Level is currently stable.
- VC Sump level is increasing.
- RCP BRG TEMP HIGH alarm is annunciated.
- The following screen was observed on PICS.



55 minutes later the following screen was observed on PICS



Which of the following describes the actions for these conditions?

- A. Trip the reactor
Trip all RCPs
- B. Trip the reactor
Trip 21 RCP
- C. Perform a shutdown per POP-3.1 Plant Shutdown from 45%
Trip all RCPs
- D. Perform a shutdown per POP-3.1 Plant Shutdown from 45%
Trip 21 RCP

Answer: D

Explanation/Justification:

This question meets the KA because RCP bearing temperature is only indicated on the plant computer (PICS) system.

A. Incorrect but plausible because the leak can be isolated by closing CCW supply and Return valves. That would isolate all CCW to the RCPs which would require tripping all RCPs.

B. Incorrect but plausible because only 21 RCP is affected, but a reactor trip is not required per AOP-RCP-1 until either CCW is lost for > 5 minutes or 200°F motor bearing temperature.

C. Incorrect but plausible because a plant shutdown is the correct action per AOP-RCP-1, and the basis for the shutdown in the AOP is an 8 hour shutdown. With temperature rising at 3°F/hr 200°F will be exceeded in approximately 4.3 hours.

D. Correct. AOP RCP actions for loss of cooling to the RCPs directs a shutdown with the goal of securing the RCP before 200°F.

Technical References: 2-AOP-CCW-1
2-AOP-RCP-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPCCW 3
I2LP-ILO-AOPRCP 3

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	000056A245	
		Ability to determine and interpret the following as they apply to the Loss of Offsite Power: - Indicators to assess status of ESF breakers (tripped/not-tripped) and validity of alarms (false/not-false)	
	Importance	3.6	3.9

Question # 78

Unit is operating at 100% power
A loss of 138KV off site power occurs.

5 minutes later:

- 480V Lockout lights are energized for all 480V safeguards buses
- 480V Bus 5A indicates 0 Volts
- 52-5A, 480V Bus 5A Normal Feed breaker, position indicating lights are extinguished
- 52-SS5 Sta Serv Xfmr 5 supply breaker, position indicating green light is illuminated
- The following alarms are annunciating:
 - 6900 V Station Service XFMR Breaker Trip (Common) Panel SHF
 - 6900 V Station Aux Breaker Trip 52ST5 52ST6 Panel SHF

Which of the following describes the plant conditions, Sustained UV Safeguard Bus Trip alarm status, and actions to restore power to bus 5A?

- A. 52-5A Breaker is Closed; Alarm is annunciated
Open 5A normal Feed Breaker and Close 21 EDG output breakers in accordance with 2-AOP-138KV-1, Loss of Power to 6.9KV Bus 5 and/or 6.
- B. 52-5A Breaker is Closed; Alarm is clear
Open 5A normal Feed Breaker and Close 21 EDG output breakers in accordance with 2-AOP-480V-1, Loss of Normal Power to Any 480V Bus.
- C. 52-5A Breaker is Open; Alarm is annunciated
Close 21 EDG output breakers in accordance with 2-AOP-138KV-1, Loss of Power to 6.9KV Bus 5 and/or 6.
- D. 52-5A Breaker is Open; Alarm is clear

Close 21 EDG output breakers in accordance with 2-AOP-480V-1, Loss of Normal Power to Any 480V Bus.

Answer: A

Explanation/Justification:

A. Correct.

B. Incorrect but plausible because the bus 5A normal feed breaker is closed; however 2-AOP-480V-1 is not the correct procedure to re-energize the bus.

C. Incorrect but plausible because the bus is de-energized which would support the breaker being open; however, the Station Service Transformer Supply breaker is open preventing the bus from being energized. Also, it is physically possible to simply close the normal supply breaker to re-energize the bus, but the procedure directs opening the normal supply breaker first.

D. Incorrect but plausible because the bus is de-energized which would support the breaker being open; however, the Station Service Transformer Supply breaker is open preventing the bus from being energized. Also, it is physically possible to simply close the normal supply breaker to re-energize the bus, but the procedure is incorrect

Technical References: 2-AOP-138KV-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS01 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		1
	K/A#	000062A204	
		Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: - The normal values and upper limits for the temperatures of the components cooled by SWS	
	Importance	2.5	2.9

Question # 79

Given:

- River Water temperature is 90°
- 1-2-3 is selected as the Essential Service Water Header
- Service Water is in 2 header operations
- 21 and 23 Service Water Pumps are operating
- Containment temperature is 124° and rising at 2°F/hr

Subsequently:

- 22 Service water pump was started and tripped
- The SM declared 22 SW Pump inoperable
- Containment temperature is 132°F and rising

Which of the following is the required action for the above plant conditions?

- A. The crew must immediately enter LCO 3.0.3.
- B. Be in MODE 3 in 6 hours AND MODE 5 in 36 hours.
- C. The crew has 8 hours to restore containment temperature.
- D. The crew has 8 hours to swap Service Water Headers if not complete then enter LCO 3.0.3.

Answer: C

Explanation/Justification:

At IPEC unit 2 Service Water Headers are arranged by supply pump, 21, 22, 23 and 24, 25, 26. Depending on physical valve lineup either header can supply essential or non essential loads. While swapping which header is supplying essential/nonessential loads, both systems are inoperable, but this condition is allowed by TS via a NOTE.

A. Incorrect but plausible because multiple Tech Spec LCOs are entered, but the actions are addressed independently. LCO 3.0.3 does not apply.

B. Incorrect but plausible because this is the action (condition B) for containment temperature IF temperature was not restored after 8 hours (i.e., condition A not met)

C. Correct.

D. Incorrect but plausible because TS 3.7.8 for Service Water System does allow 8 hours to swap headers from non essential to essential without entering LCO 3.0.3.

Technical References: Tech Specs
Proposed References to be provided: None

Learning Objective I2LP-ILO-VCVCB 10

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		1
	K/A#	000065A206	
		Ability to determine and interpret the following as they apply to the Loss of Instrument Air: - When to trip reactor if instrument air pressure is decreasing	
	Importance	3.6	4.2

Question # 80

Given:

- Unit 2 is operating at 15% power when an Instrument Air malfunction occurs.
- The crew enters 2-AOP-AIR-1, Air Systems Malfunction.
- Instrument air header pressure is 75 psig and lowering slowly.
- Letdown isolated
- VCT level lowers to 4%.
- LCV-112B, RWST to Charging Pump Suction LCV, failed to open.

Which of the following describes how the CRS should proceed?

- A. Direct a reactor trip, stop all RCPs, then initiate E-0, Reactor Trip or Safety Injection while continuing in 2-AOP-AIR-1.
- B. Direct a reactor trip and initiate E-0, Reactor Trip or Safety Injection, to stabilize the plant, then transition back to 2-AOP-AIR-1 to address the loss of instrument air when transition out of E-0.
- C. Dispatch NPO to OPEN 288 Manual Suction Stop to Charging Pump, minimize charging pump speed, place excess letdown in service to maintain PRZR level.
- D. Dispatch NPO to OPEN SA-33 SA Unit 3 Tie Stop Valve, Stop running Charging Pump, start manual makeup to restore VCT level.

Answer: A

Explanation/Justification:

A is correct because when VCT level lowers <5%, a reactor trip is directed per AOP-AIR-1 steps 4.12-4.15, which includes reactor trip, stopping RCPs, stopping charging pumps, and isolating letdown (already isolated in stem), then initiating E-0.

B is incorrect but plausible as it directs a reactor trip, but the AOP is performed in parallel with EOPs to restore air regardless of transition.

C is incorrect but plausible because opening 288 is directed when VCT level is less than 8.5, but the reactor is tripped if level decreases to less than 5%. Also, minimizing charging and placing excess letdown in service would help with rising pressurizer level.

D is incorrect but plausible because the candidate may believe opening SA-33 would help restore air pressure. But this valve is closed in this procedure. Stopping the running charging pump is consistent with low VCT level, however it is not directed in the AOP.

Technical References: 2-AOP-AIR-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-SA01 9

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	1	
	Group#	1	
	K/A#	1940012401 <i>077G 2.4.1</i> <i>3/15/19</i>	
		Emergency Procedures/Plan - Knowledge of EOP entry conditions and immediate action steps.	
	Importance	4.6	4.8

Question # 81

Given:

- The plant was operating at 30% power during a plant startup.
- Grid disturbance has caused the Main Generator to trip on over frequency.

Which of the following describe the electric plant conditions and required actions?

- A. All 480V buses are energized from the EDGs
Restore off-site power using AOP-138KV-1, Loss of Power to 6.9KV Bus 5 and/or 6.
- B. All 480V buses are energized from the EDGs
Restore off-site power using AOP-480V-1, Loss of Normal Power to Any 480V Bus.
- C. 480V buses 5A and 6A are energized
480V buses 2A and 3A de-energized.
Restore 480V buses using AOP-480V-1 Loss of Normal Power to Any 480V Bus.
- D. 480V buses 5A and 6A are energized
480V buses 2A and 3A de-energized.
Restore 480V buses using AOP-138KV-1, Loss of Power to 6.9KV Bus 5 and/or 6.

Answer: C

Explanation/Justification:

At unit 2 a loss of any 480V bus will start all EDGs. A reactor trip with a loss of 5A or 6A will open all 480 bus normal feed breakers and close EDG breakers. This includes bus 2A/3A. Also the loss of 5A or 6A does not have to be caused by a loss of offsite power (Over current trip on normal feed breaker).

Over frequency causes a generator trip and lockout of 6.9 buses 1-4 (they will not fast transfer to offsite power due to frequency mismatch). Buses 5 and 6 are not affected by the generator trip. The reactor trips due to turbine trip.

A reactor trip with a loss of 2A/3A will start all EDGs but will NOT open 480V bus normal feed breakers and close EDG breakers. 22 EDG will be running with buses 2A/3A de-energized.

E-0 step 3 is the immediate operator action that checks for 480V bus alignment. Buses 5A and 6A remain energized from offsite power. Buses 2A and 3A will not be energized. The RNO for step 3 directs re-energizing 6.9 KV buses 2 and 3 and 480 V buses 2A and 3A using AOP-480V-1.

A. Incorrect but plausible because grid over frequency locks out only the 1-4 6.9 KV buses from fast transfer to offsite power buses 5 and 6. Candidate may believe that all 6.9 buses de-energize. If all 6.9KV buses were de-energized the 480V buses would be energized from the EDGs. If all 480V buses are energized the RNO for re-energizing buses is not implemented but AOP-480V is used if any 480V bus was de-energized to re-energize for reasons other than loss of 6.9KV bus 5 or 6.

B. Incorrect but plausible because grid over frequency locks out only the 1-4 6.9 KV buses from fast transfer to offsite power buses 5 and 6. Candidate may believe that all 6.9 buses de-energize. If all 6.9KV buses were de-energized the 480V buses would be energized from the EDGs. If all 480V buses are energized the RNO for re-energizing buses is not implemented, but AOP-138KV is used if 6.9 KV bus 5 and/or 6 was de-energized to re-energize 480V buses 5A and/or 6A

C. Correct. 480V buses 2A and 3A will be de-energized. E-0 step 3 RNO is implemented and AOP-480V is the correct procedure to use to re-energize buses.

D. Incorrect but plausible because the bus alignments are correct; however, AOP-138KV will not address losses of only buses 1-4.

Technical References: 2-E-0
Proposed References to be provided: None

Learning Objective I2LP-ILO-EDS01 9

Question Source:	New
Question History:	NA
Question Cognitive Level:	Comprehension
10 CRF Part 55 Content:	55.43 (b) 5
Comments	

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		2
	K/A#	000036A202	
		Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: - Occurrence of a fuel handling incident	
	Importance	3.4	4.1

Question # 82

Given the following conditions:

- Refueling Operations are in progress, Full Core Off-Load is being performed.
- A spent fuel assembly has been latched in the manipulator mast and the assembly has just been lifted clear of the reactor vessel flange.
- Visual Observations indicate Refueling Cavity and Spent Fuel Pool Level are both lowering.
- The local indication in the Refueling Cavity is at 91'

Which of the following describe the correct initial procedural actions based on the above information?

- A. Immediately suspend movement of all irradiated fuel assemblies in accordance with Technical Specifications 3.7.14 and 3.9.6, evacuate all personnel from Containment, initiate Containment Ventilation Isolation (Purge & Pressure Relief Valves Closed), close the fuel transfer canal gate valve and initiate level makeup.
- B. Place the suspended fuel assembly in the containment upender, lower and send to the Fuel Storage Building, then close the fuel transfer canal gate valve, and evacuate all personnel from the Fuel Storage Building (FSB) and Containment VC).
- C. Close the fuel transfer canal gate valve, place the suspended fuel assembly back in the reactor vessel and evacuate non-essential personnel from the Fuel Storage Building (FSB) and Containment (VC).
- D. Place the suspended fuel assembly in the containment upender in the vertical position, evacuate non-essential personnel from the Fuel Storage Building (FSB) and Containment (VC), close the fuel transfer canal gate valve and initiate level makeup.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the student may be focused on the need to immediately suspend movement of irradiated fuel assemblies per the tech spec actions, however, as stated in the tech spec bases; “this does not preclude movement of a fuel assembly to a safe position.” Also plausible because if the assembly is left in the manipulator crane mast with level continuing to lower, actions from the “damaged” fuel assembly section requiring Containment Ventilation Isolation and evacuation are very plausible. The need to makeup is also a subsequent step and therefore additionally plausible.

B. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would never be sent back to the FSB. Transfer cart needs to be on containment side to facilitate gate valve closure. The remaining steps are correct, except that all personnel are not evacuated, only “non-essential”. Plausible because, all personnel are evacuated for a “damaged” assembly.

C. Correct. See steps 4.27 – 4.30 in body of procedure, Attachments 1 & 2.

D. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would be then lowered to the “horizontal” position. Closing the gate valve is correct and the need to makeup is also a subsequent step and therefore additionally plausible.

Technical References: 2-AOP-FH-1
3-AOP-FH-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-FHD001 8
I3LP-ILO-AOPFH1 2
I3LP-ILO-AOPFH1 5

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		2
	K/A#	0000692236	
		Equipment Control - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	
	Importance	3.1	4.2

Question # 83

Given:

- The plant is in MODE 3.
- Personnel airlock interlock mechanism is declared inoperable at 0200.
- Personnel airlock outer door is closed and locked at 0245.

Which of the following describes the TS requirements necessary to permit access to perform repairs on the inoperable interlock mechanism?

- A. Place the plant in Mode 4 prior to entry.
- B. Lock the inner door and post a dedicated individual to perform the interlock function prior to entry.
- C. Lock the inner door and prohibit any entry until after repairs are made.
- D. Restore Containment to Operable status within 4 hours or Be in Mode 4 in 6 hours..

Answer: B

Explanation/Justification:

A. Incorrect but plausible because this is the action if both door are inoperable and one cannot be closed.

B. Correct

C. Incorrect Note 1 to TS 3.6.2 ACTIONS does not prohibit opening of the inner door. The Bases of TS 3.6.2 acknowledges that the containment boundary may not be intact for a short time.

D. Incorrect The air locks have their own TS for inoperability.

Technical References: Tech Specs
Proposed References to be provided: None

Learning Objective I2LP-ILO-VCVCB 4

Question Source: Bank

Question History: DC Cook 2014

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		2
	K/A#	00WE02A202	
		Ability to determine and interpret the following as they apply to the SI Termination: - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	
	Importance	3.5	4

Question # 84

The crew just completed resetting Phase A and B at step 10 in ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

The ATC reports the following SG conditions.

<u>SG</u>	<u>Level</u>	<u>Pressure</u>
SG 21	20% WR	360 psig STABLE
SG 22	19% WR	320 psig DECREASING
SG 23	18% WR	310 psig DECREASING
SG 24	26% WR	380 psig INCREASING

E-2, Faulted Steam Generator Isolation

E-1, Loss of Reactor or Secondary Coolant

Which one of the following actions is correct?

- A. Immediately transition to E-2; a subsequent transition to E-1 will terminate SI with MORE restrictive criteria.
- B. Immediately transition to E-2; a subsequent transition to E-1 will terminate SI using the Foldout Page criteria.
- C. Complete SI Termination in ECA-2.1 then cooldown and place RHR in service per ECA-2.1.

- D. Complete SI Termination in ECA-2.1 then Transition to E-2, a subsequent transition to E-1 will evaluate plant conditons.

Answer: D

Explanation/Justification:

A. Incorrect but plausible because, if the conditions occurred before the crew initiated SI termination actions at step 9 then a transition to E-2 is correct. Also termination criteria in E-1 includes Heat Sink thus MORE restrictive.

B. Incorrect but plausible because, if the conditions occurred before the crew initiated SI termination actions at step 9 then a transition to E-2 is correct. Also termination criteria in E-1 includes termination criteria on the Foldout Page.

C. Incorrect but plausible because ECA-2.1 foldout page directs completing SI Termination (steps 9 – 17) if it has been started then transition to E-2. Also ECA-2.1 does provide actions to cooldown to RHR if a transition out does not occur.

D. Correct. Per the Foldout Page directs “go to E-2” if you are not performing steps 9-17. E-1 will evaluate all plant conditions to identify any additional failures.

Technical References: 2-ECA-2.1

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPC21 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		1
	Group#		2
	K/A#	00WE132435	
		Emergency Procedures/Plan - Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	
	Importance	3.8	4

Question # 85

Given:

- The plant experienced an inadvertent SI.
- 21 SG Pressure is 1130 psig and stable.
- 21 Atmospheric cannot be operated locally.
- Attempts to dump steam from any SG have been ineffective from Control Room.
- The team is in FR-H.2, Response to Steam Generator Overpressure.
- RCS Hot Leg Temperatures are 560⁰F and stable.
- All MSIVs are closed

Which of the following will be directed by FR-H.2 to address these conditions?

- A. Feed 21 SG with cold AFW.
- B. Trip 21 RCP.
- C. Start turbine driven AFW pump.
- D. Locally dump steam using 22, 23, 24 Atmospherics.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because feeding with cold water would lower pressure, however the procedure isolates AFW until a steam release path is established.

B. Incorrect. Plausible because it would reduce heat transfer into the affected SG.

C. Incorrect. The procedure directs dumping steam using 22 AFW pump if it is from affected SG.

D. Correct.

Technical References: 2-FR-H.2

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPFRH 4

Question Source: Modified

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 7

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		1
	K/A#	0060002445	
		Emergency Procedures/Plan - Ability to prioritize and interpret the significance of each annunciator or alarm.	
	Importance	4.1	4.3

Question # 86

Given:

- A small break LOCA occurred.
- 21 SI Pump started and immediately tripped on overcurrent.
- The crew performed the actions of E-0 and transitioned to E-1.
- 23 SI Pump failed to start.

The during the transition brief, the BOP reported that the following alarms were annunciated:

- SAFEGUARDS EQUIPMENT LOCKED OPEN
- SAFEGUARDS VALVE OFF NORMAL POSITION

Which of the following describes why the alarms are annunciating and what actions are required?

- A. The SAFEGUARDS EQUIPMENT LOCKED OPEN is expected due to placing the RHR pump in TPO during performance of E-0.
The SAFEGUARDS VALVE OFF NORMAL POSITION is expected due to automatic repositioning of 851A/851B.
851A or 851B must be opened to restore SI Flow to the RCS.
- B. The SAFEGUARDS EQUIPMENT LOCKED OPEN is expected due to placing the RHR pump in TPO during performance of E-0.
The SAFEGUARDS VALVE OFF NORMAL POSITION is not expected.
Determine which valve has failed and reposition it to its safeguards position.
- C. The SAFEGUARDS EQUIPMENT LOCKED OPEN is expected due to overcurrent trip of 21 SI Pump.
The SAFEGUARDS VALVE OFF NORMAL POSITION is not expected.
Determine which valve has failed and reposition it to its safeguards position.

- D. The SAFEGUARDS EQUIPMENT LOCKED OPEN is expected due to overcurrent trip of 21 SI Pump.
The SAFEGUARDS VALVE OFF NORMAL POSITION is expected due to automatic repositioning of 851A/851B.
851A or 851B must be opened to restore SI Flow to the RCS.

Answer: A

Explanation/Justification:

For a Safety Injection with no complications neither alarm is expected until RHR pump is placed in Trip Pull Out per E-0. The Valve Off Normal alarm is not expected unless a problem exists. Failure of both 21 and 23 SIP will cause both 851A and 851B (discharge valves for 22 SIP) to close. The SIP will not deliver flow to the core until one of the valves is opened and de-energized.

A. Correct. The Safeguards Equipment alarm will annunciate when RHR pump is placed in TPO in E-0 to address strong pump weak pump condition. The trip of either 21 or 23 SIP will reposition 851A or 851B causing the alarm.

B. Incorrect but plausible because the Safeguards Equipment alarm is correct, The Valve Off Normal alarm is expected due to repositioning 851A/B.

C. Incorrect but plausible because Safeguards Equipment alarm is expected but not for the reason given. Valve off normal alarm is expected for the conditions given.

D. Incorrect but plausible because Safeguards Equipment alarm is expected but not for the reason given. The Valve Off Normal alarm and response is correct.

Technical References: 2-E-0

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE00 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		1
	K/A#	012000A201	
		Ability to (a) predict the impacts of the following malfunctions or operations on the RPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Faulty bistable operation	
	Importance	3.1	3.6

Question # 87

Given:

- I&C is performing a surveillance test on Pressurizer High Pressure Reactor Trip.
- The technician reports that the High Pressure Reactor Trip bistable for Channel 455 is stuck at 2465 psig and cannot be adjusted to the correct value.
- The bistable will need to be replaced.
- The new bistable will not be onsite for 5 days.
- All other components satisfied surveillance acceptance criteria.

Tech Spec table 3.3.1-1 for Pressurizer Pressure

7. Pressurizer Pressure

a.	Low	1 ^(e)	4	K	SR 3.3.1.1 \geq 1878 psig SR 3.3.1.7 SR 3.3.1.10
b.	High	1,2	3	E	SR 3.3.1.1 \leq 2416 psig SR 3.3.1.7 SR 3.3.1.10

(1) The faulty bistable value is ____ (1) ____ and ____ (2) ____
(1) (2)

- | | | |
|----|------------------|---|
| A. | non conservative | the entire channel must be removed from service |
| B. | conservative | only the affected bistable must be |

C. non conservative

D. conservative

tripped

only the affected bistable must be
tripped

the entire channel must be removed
from service

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the setpoint is non-conservative (allowable value is 2416 in TS); however, removing the entire channel because of the failure of one bistable is not conservative.

B. Incorrect. Plausible because the TS allowable value is greater than the actual setpoint used in the bistable; the candidate may believe that this value is acceptable. The bistable must be placed in the tripped condition to satisfy TS. It is not necessary to remove the entire channel.

C. Correct. The setpoint is non-conservative (allowable value is 2416 in TS). The bistable must be placed in the tripped condition to satisfy TS. It is not necessary to remove the entire channel.

D. Incorrect. Plausible because the TS allowable value is greater than the actual setpoint used in the bistable; the candidate may believe that this value is acceptable. Also, removing the entire channel because of the failure of one bistable is not conservative.

Technical References: Tech Specs
Proposed References to be provided: None

Learning Objective IOLP-ILO-ITS001 3

Question Source: New

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		1
	K/A#	039000A203	
		Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Indications and alarms for main steam and area radiation monitors (during SGTR)	
	Importance	3.4	3.7

Question # 88

Given:

- The unit was at 100% power.
- A SGTR occurred in 24 SG.
- The crew manually tripped the reactor and actuated SI
- The crew is currently performing the initial cooldown in E-3
- During the cooldown the BOP reports that level in 21 SG is increasing in an uncontrolled manner.

Which of the following identifies the indications to confirm a second tube rupture in 21 SG and what actions are required if a second tube rupture is confirmed?

R-61A, 21 Main Steam N-16 Monitor

R-28, 21 Main Steam Line Monitor

- A. R-61A increasing
R-28 increasing
Stabilize the plant and return to E-3 step 1
Isolate 21 SG
Resume cooldown to target temperature based on 24 SG pressure.
- B. R-28 only increasing
Stop cooldown in progress
Close MSIV for 21 SG
Resume cooldown to target temperature based on 24 SG pressure.

- C. R-61A increasing
R-28 increasing
Stop cooldown in progress
Close MSIV for 21 SG
Initiate cooldown to target temperature based on 21 SG pressure.

- D. R-28 only increasing
Stabilize the plant and return to E-3 step 1
Isolate 21 SG
Initiate cooldown to target temperature based on 21 SG pressure.

Answer: D

Explanation/Justification:

R-61A is main steam line monitor for N-16 gammas. After shutdown it will not indicate. R-28 is the main steam line monitor 21 SG. It will increase when 21 SGTR occurs. E-3 Foldout page directs actions for Multiple Tube Ruptures. Essentially, the procedure is restarted as if first tube rupture did not occur.

A. Incorrect but plausible because R28 will increase but R-61C is N-16 monitor and it will not increase. The cooldown is not resumed but initiated based on 21 SG pressure.

B. Incorrect but plausible because R-28 only will increase, but full isolation of 21 SG is required. The procedure does state that "isolation of Ruptured SG(s) should be completed by closing the MSIV(s).

C. Incorrect but plausible R-28 will increase, but R-61A will not. Initiating a cooldown base in 21 SG pressure is correct.

D. Correct.

Technical References: 2-E-3

Proposed References to be provided: None

Learning Objective I2LP-ILO-EOPE30 4

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		1
	K/A#	062000A212	
		Ability to (a) predict the impacts of the following malfunctions or operations on the A.C. Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Restoration of power to a system with a fault on it	
	Importance	3.2	3.6

Question # 89

Unit 2 is at 100% power when 480V Bus 5A normal feed breaker trips on overcurrent.

- 1) What is the status of the breakers on 480V bus 5A?
- 2) How will power will be restored to Bus 5A in accordance with 2-AOP-480V-1, Loss of Normal Power to Any 480V bus?

- A.
 - 1) All breakers on Bus 5A will be open EXCEPT for MCC-26A and MCC-29/29A
 - 2) Place all Load Breakers in Trip Pull Out
Re-energize the bus from the Station Service Transformer (Normal Feed)
- B.
 - 1) All breakers on Bus 5A will be open EXCEPT for MCC-26A
 - 2) Place all Load Breakers in Trip Pull Out
Re-energize the bus from the Emergency Diesel Generator (Emergency Feed)
- C.
 - 1) All breakers on Bus 5A will be open EXCEPT for MCC-26A and MCC-29/29A
 - 2) Place Non-Safeguards Breakers in Trip Pull Out
Re-energize the bus from the Emergency Diesel Generator (Emergency Feed)
- D.
 - 1) All breakers on Bus 5A will be open EXCEPT for MCC-26A
 - 2) Place Non-Safeguards Breakers in Trip Pull Out
Re-energize the bus from the Station Service Transformer (Normal Feed)

Answer: B

Explanation/Justification:

A. Incorrect but plausible because the breaker for MCC 26A will not trip on undervoltage. The breaker for MCC 29/29A will not trip on a Safety Injection Signal, but it will trip on an undervoltage condition. All load breakers are placed in TPO. The bus is re-energized from the EDG not the normal supply.

B. Correct. because the breaker for MCC 26A will not trip on undervoltage. The breaker for MCC 29/29A will not trip on a Safety Injection Signal, but it will trip on an undervoltage condition.. All load breakers are placed in TPO and the bus is re-energized from the EDG.

C. Incorrect but plausible because the breaker for MCC 26A will not trip on undervoltage. The breaker for MCC 29/29A will not trip on a Safety Injection Signal, but it will trip on an undervoltage condition. Leaving Safeguards breakers in normal would allow an auto start if necessary, but it is NOT desired. Also the bus would be re-energized from the EDG.

D. Incorrect but plausible because the breaker for MCC 26A will not trip on undervoltage and the breaker for all other MCCs will trip. The bus is re-energized EDG not the normal supply.

Technical References: 2-AOP-480V-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOP480 2

Question Source: Bank

Question History: Unti 3 NRC 2010

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		1
	K/A#	1030002244	
		Equipment Control - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	
	Importance	4.2	4.4

Question # 90

A small break LOCA occurred.

Complications with the SI system resulted in no SI flow to core.

VC pressure is 10 psig and slowly lowering

The crew is performing actions of FR-C.1, Response to Inadequate Core Cooling.

The ATC reports VC High Range Radiation Monitors

- R-25 indicates 70 R/hr
- R-26 indicates 62 R/hr.

Which of the following describes the E-Plan classification and basis?

- A. Site Area Emergency based on
RCS Fission Product Barrier (FPP) – Lost
Fuel Clad FPP – Potential Loss
Containment – Intact
- B. Site Area Emergency based on
RCS Fission Product Barrier (FPP) – Lost
Fuel Clad FPP – Lost
Containment – Potential Loss
- C. General Emergency based on
RCS Fission Product Barrier (FPP) – Lost
Fuel Clad FPP –Lost
Containment – Lost
- D. General Emergency based on
RCS Fission Product Barrier (FPP) – Lost

Fuel Clad FPP – Lost
Containment – Potential Loss

Answer: D

Explanation/Justification:

A. Incorrect but plausible because the FPB analysis would be correct for an SAE; however containment is actually potentially lost.

B. Incorrect but plausible the classification is incorrect but the FPB analysis is correct. The candidate must recall that loss of 2 and a potential loss of third is a GE.

C. Incorrect but plausible because the classification is correct based on 3 FPB lost; however, containment is physically intact, but the consequences if it lost would be so severe that it is classified a potential loss.

D. Correct. The classification is Correct and the FPB analysis is correct.

Technical References: IP-EP-AD13

Proposed References to be provided: None

Learning Objective I0LP-ILO-ERT002 1.05

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		2
	K/A#	034000A201	
		Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Dropped fuel element	
	Importance	3.6	4.4

Question # 91

You are notified by the Fuel Handling Supervisor that an irradiated fuel assembly was dropped in the spent fuel pool. The assembly fell into the correct pool location. R-5 and R-44 readings are increasing. R-5 is in ALARM.

Based on these conditions, what actions are required after suspending fuel handling operations in the FSB by AOP-FH-1, Fuel Damage or Loss of SFP/Refueling Cavity Level?

- A. Evacuate ONLY non-essential personnel from the FSB, place FSB ventilation in service and monitor R-44.
- B. Evacuate ONLY non-essential personnel from the FSB, ensure SFP purification is in service with maximum flow and monitor R-44.
- C. Evacuate ALL personnel from the FSB, ensure ALL FSB doors are closed and monitor R-44.
- D. Evacuate ALL personnel from the FSB, ensure SFP purification is in service with maximum flow and monitor R-44.

Answer: C

Explanation/Justification:

A. Incorrect but plausible. All personnel are evacuated and the procedure does not direct placing ventilation in service. However only evacuating non-essential personnel and placing ventilation in service are plausible.

B. Incorrect but plausible. Maximizing purification is plausible, but not in the procedure.

C. Correct answer per AOP-FH-1

D. Incorrect but plausible. Plausible for same reason as B.

Technical References: 2-AOP-FH-1
3-AOP-FH-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-FHD001 8
I3LP-ILO-FHD001 8

Question Source: Bank

Question History: Unit 3 NRC 2013

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		2
	K/A#	068000A204	
		Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Failure of automatic isolation	
	Importance	3.3	3.3

Question # 92

During a release of 14 Waste Distillate Storage Tank (WDST), R-54, Unit 1 Liquid Waste Distillate Monitor, lost power. The NPO reported that the breaker tripped for R-54.

Which of the following states the correct action for this event, and what is required to resume the release?

- A. Ensure the running waste distillate transfer pump is secured.
Obtain 2 independent samples of 14 WDST
Validate release permit
Reset the breaker for R-54
The release can resume if sample is unchanged.
- B. Ensure the running waste distillate transfer pump is secured and discharge valves closed.
Reset the breaker for R-54
Obtain 2 independent samples of 14 WDST
Determine actual volume released from 14 WDST
Validate release permit
- C. Ensure the running waste distillate transfer pump is secured and discharge valves closed.
Declare R-54 inoperable
Obtain 2 independent samples of 14 WDST
Obtain 2 independent verifications of new release permit calculations
Obtain 2 independent verifications of the discharge valve lineup

- D. Ensure the running waste distillate transfer pump is secured.
 - Declare R-54 inoperable
 - Obtain 2 independent verifications of new release permit calculations
 - Obtain 2 independent verifications of the discharge valve lineup
 - Install portable radiation monitor

Answer: C

Explanation/Justification:

A. Incorrect but plausible because stopping the pump is part of terminating release but it does not include closing valves. Obtaining 2 independent samples is also correct and the procedure is correct.

B. Incorrect but plausible because stopping the pump and closing the valves is correct to terminate the leak. 2 independent samples are required but a new release permit is required. Also the procedure is incorrect.

C. Correct.

D. Incorrect but plausible because stopping the pump will stop the release, but it does not include closing valves. Portable monitors are not used for liquid releases. Also the procedure is not correct.

This question satisfies the KA because the candidate must predict the impact of a loss of power to the rad monitor. The release should terminate on a loss of power and thus the operator must take manual actions to satisfy the requirement. That is the second part of the KA.

Technical References:	2-ARP-SAF-1 ODCM
Proposed References to be provided:	None
Learning Objective	I2LP-ILO-RMS001 4
Question Source:	New
Question History:	NA
Question Cognitive Level:	Comprehension
10 CRF Part 55 Content:	55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		2
	Group#		2
	K/A#	0710002238	
		Equipment Control - Knowledge of conditions and limitations in the facility license.	
	Importance	3.6	4.5

Question # 93

Given:

- RCS Activity increased and the crew entered 2-AOP-HIACT-1.
- The crew is stripping fission gasses per 2-SOP-8.1, Reactor Coolant System Chemistry Control.
- During the degas operation, R-50, Gas Decay Tank Activity exceeded the Alarm setpoint.

Which of the following states the ODCM basis for R-50 alarm and required actions?

R-50 Alarm indicates:

- A. In-service Waste Gas Tank combined activity has exceeded 29,761 μCi
Swap in-service Waste Gas Decay Tanks to clear R-50 alarm
- B. Waste Gas Tank combined activity has exceeded 29,761 μCi
Immediately suspend addition of radioactive material to the Gas Decay Tanks
- C. In-service Waste Gas Decay Tank has exceeded 6,000 μCi
Swap in-service Waste Gas Decay Tanks to clear R-50 alarm
- D. Waste Gas Tank combined activity has exceeded 6,000 μCi
Immediately suspend addition of radioactive material to the Gas Decay Tanks

Answer: C

Explanation/Justification:

A. Incorrect but plausible because 29,761 is the Currie limit for all gas decay tank (equivalent Xe-133 not combined), however, R-50 is based on one tank (6,000 Ci). Also the action is correct per ODCM if 29,761 Ci limit is exceeded.

B. Incorrect but plausible because 29,761 is the equivalent Xe-133 limit for all GasDeca Tanks; however, R-50 is based on one tank (6,000 Ci). Also the action is correct per ODCM if 29,761 Ci limit is exceeded.

C. Correct. If the in-service tank exceeds 6,000 equivalent Xe-133, R-50 will alarm and the action per the ARP is to swap tanks.

D. Incorrect but plausible because R-50 is based on exceeding 6,000 Ci but not combined Kr and Xe. The action to swap tanks is also correct.

Technical References: 2-ARP-SAF-1
ODCM

Proposed References to be provided: None

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I2LP-ILO-GWR01 9

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group# K/A#	1940012141 Conduct of Operations - Knowledge of the refueling process.	
	Importance	2.8	3.7

Question # 94

Given:

- The unit is in MODE 6 performing core reload

In accordance with 2-REF-003-GEN Section 3.1, Fuel Movement Requirements – Core Reload,

- 1) Bypassing fuel handling interlocks not covered by a procedure can only be approved by the ____ (1).
- 2) The Refueling SRO ____ (2) required to be inside Containment for control rod latching.

- A. 1. Refueling SRO
2. is
- B. 1. Refueling SRO
2. is NOT
- C. 1. Shift Manager
2. is
- D. 1. Shift Manager
2. is NOT

Answer: A

Explanation/Justification:

This question was Question 95 on the 2015 McGuire NRC Exam

2-REF-003-GEN Section 3.1 states that the RSRO has to give permission to bypass interlocks. The RSRO must in containment for core alterations. Latching control rods is a core alteration.

A. Correct answer.

B. Incorrect but plausible if the candidate does not realize that latching control rods is a core alteration. The first part is correct.

C. Incorrect but plausible. The SM has overall responsibility for the unit, but the RSRO is specified in the procedure. The second part is correct.

D. Incorrect but plausible. The SM has overall responsibility for the unit, but the RSRO is specified in the procedure. The second part is plausible if the candidate does not realize that latching control rods is a core alteration.

Technical References:

Proposed References to be provided: None

Learning Objective I2LP-ILO-FHD001 8

Question Source: Bank

Question History: McGuire 2015

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.43 (b) 6

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group#		
	K/A#	1940012213	
		Equipment Control - Knowledge of tagging and clearance procedures.	
	Importance	4.1	4.3

Question # 95

A worker is signed on as a Work Order Holder on a tagout. Operations needs to restore the system. The Tagout Holder informs operations that all the work is complete and the work order is taken to Complete. The individual worker has left work for day without signing off the tagout.

Which of the following is required to clear the tagout at this time?

- A. Since the work order was taken to Complete, the OM may sign the individual off of the tagout without completing a Tagout Alternate Release.
- B. A Tagout Alternate Release must be performed. This is approved by the OM.
- C. Since the work order was taken to Complete, the SM may sign the individual off of the tagout without completing a Tagout Alternate Release.
- D. A Tagout Alternate Release must be performed. This is approved by the SM.

Answer: D

Explanation/Justification:

This question matches the KA since the candidate has to understand the work order process to know that work order status does not prevent an individual from performing work. Per EN-OP-102, a Tagout Alternate Release must be completed which is approved by the SM.

- A. Incorrect but plausible if the candidate is not sufficiently familiar with the work order/tagout process.
- B. Incorrect but plausible since the procedure could allow the FSS to approve the form.
- C. Incorrect but plausible if the candidate is not sufficiently familiar with the work order/tagout process.
- D. Correct answer.

Technical References: EN-OP-102
Proposed References to be provided: None

Learning Objective I0LP-ILO-ADM01 1

Question Source: Bank

Question History: Unit 3 NRC 2017

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group#		
	K/A#	1940012240	
		Equipment Control - Ability to apply technical specifications for a system.	
	Importance	3.4	4.7

Question # 96

The plant is at 100% power.

- 0830 on December 3, the 21 RHR pump is declared inoperable.
- 2300 on December 5, the 23 HHSI pump is declared inoperable.
- 0215 on December 6, the 21 RHR pump is restored to OPERABLE status.

Which one of the following describes the LATEST time and date to restore the 23 HHSI pump to OPERABLE status without requiring a unit shutdown?

- A. 0830 on December 6
- B. 0830 on December 7
- C. 2300 on December 8
- D. 2300 on December 9

Answer: B

Explanation/Justification:

The original inoperability of 21 RHR pump required return to operability by 0830 on 12/6 (choice A).

23 HHSI becoming inoperable (if alone) would have required operability by 2300 on 12/8 (choice C).

TS 1.3 says to pick the more limiting of either completion time for original entry + 24 hours or subsequent entry completion time. In this case the more limiting time original completion time + 24 hours (choice B).

A. Incorrect but plausible if candidate believes we have to use original required completion time.

B. Correct answer.

C. Incorrect but plausible if the candidate believe that completion time is based solely on what is currently inoperable.

D. Incorrect but plausible if the candidate believes that the 24 hours is added to the longest completion time.

Technical References: Tech Specs

Proposed References to be provided: Tech Spec 3.5.2

Learning Objective I2LP-ILO-RHR001 9

Question Source: Bank

Question History: N/A

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 2

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#	3	
	Group#		
	K/A#	1940012311	
		Radiological Controls - Ability to control radiation releases.	
	Importance	3.8	4.3

Question # 97

Given:

- The unit was operating at 100% power
- A 150 gpm SGTR occurred on 22 SG
- All SG levels are off scale low on narrow range indication
- 21 AFW pump tripped on overcurrent
- 23 AFW pump is running
- The operating crew performed all required actions in E-0, Reactor Trip or Safety Injection and transitioned to E-3, Steam Generator Tube Rupture.

Which of the following statements describes the actions related to 22 AFW pump?

22 AFW pump will:

- A. remain running feeding 21 SG to prevent a dry out condition. An NPO will be dispatched to isolate steam supply from 22 SG.
- B. be tripped to minimize the radioactive release. Pump can be restarted after steam supply from 22 SG is isolated.
- C. remain running feeding 21 and 22 SGs to establish level in 22 SG. An NPO will be dispatched to isolate steam supply from 22 SG.
- D. be tripped to minimize the radioactive release. Pump can be restarted only if necessary for RCS cooldown.

Answer: B

Explanation/Justification:

For this condition 22 AFW pump was started and used to feed 21 and 22 SG. 21 SG level increasing is indication that this was done. In E-3 22 AFW pump is tripped if one motor driven pump is running and the tube rupture is in 22 or 23 SG. When steam supply to 22 AFW pump is isolated from the ruptured SG the pump is restarted.

A. Incorrect but plausible because 21 SG is below 10%, but the pump will be tripped to minimize radiation release. An NPO will be sent to isolate the steam supply from 22 SG to 22 AFW pump to allow restart of the pump.

B. Correct.

C. Incorrect but plausible because adequate level in 22 SG is necessary to perform the cooldown in E-3. An NPO will be dispatched to isolate steam supply from 22 SG after the pump is tripped. The pump can then be restarted

D. Incorrect but plausible because the pump is tripped, but not to prevent excessive cooldown. Additionally, the pump is operated if it is the only AFW pump running and steam supply to the pump from 22 SG is isolated.

Technical References: 2-E-3

Proposed References to be provided: None

Learning Objective I2LP-ILO-MFW001 11
I2LP-ILO-MFW001 15
I2LP-ILO-MFW001 6

Question Source: New

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group#		
	K/A#	1940012312	
		Radiological Controls - 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.	
	Importance	3.2	3.7

Question # 98

Given:

- Unit 2 is at 1% Reactor power coming out of a refueling outage.
- Personnel in containment are accompanied by an RP tech with monitoring equipment.
- Personnel are in containment making adjustments to 23 RCP vibration probes.
- The CRS and SM want to raise power to 2%.

Based on 0-SOP-CB-001, Containment Entry and Egress, what is required regarding this power ascension?

- A. Entry party for the work on the RCP vibration probes will have to leave Containment. When the power increase is complete RP supervision determines when workers can return to the RCP.
- B. Power can be raised since RP tech is continuously monitoring radiation levels; the power ascension does not require additional action per 0-SOP-CB-001.
- C. Power can be raised. However, since there are personnel in the inner crane wall, 0-SOP-CB-001 requires the SM to specifically approve the power ascension.
- D. RP Supervision and entry party shall be notified prior to any planned change in power level. The RP Supervisor shall decide if workers need to exit or move to ALARA area prior to raising power is necessary.

Answer: D

Explanation/Justification:

This situation actually occurred at IP3, which led to the procedural requirement.

A. Incorrect but plausible. It is not unreasonable that 0-SOP-CB-001 would have required removing personnel prior to power ascension, not just moving to a lower dose waiting area.

B. Incorrect but plausible. It would be reasonable to assume that this power change would have minimal effect on dose rates, but this is not true.

C. Incorrect but plausible. The SM is often allowed to authorize items that require slightly greater levels of control and decision making. Based on B above discussion, it is reasonable that a candidate may assume this change will have minimal effect.

D. Correct based 0-SOP-CB-001 step 2.24.1

Technical References: 0-SOP-CB-001

Proposed References to be provided: None

Learning Objective I0LP-ILO-ADM01 4

Question Source: Bank

Question History: Unit 2 NRC 2010

Question Cognitive Level: Fundamental Knowledge

10 CRF Part 55 Content: 55.41 (b) 10

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group#		
	K/A#	1940012408	
		Emergency Procedures/Plan - Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	
	Importance	3.8	4.5

Question # 99

Given the following:

- The plant was operating at 100% power.
- N-16 Radiation Monitor for 22 SG alarms
- NPO reports N-16 indicates 150 GPD leak
- Chemistry confirms the leakrate at approximately 150 GPD
- The operating crew entered 2-AOP-SG-1, Steam Generator Tube Leak
- During the performance of 2-AOP-RSD-1, Rapid Shutdown, the plant was tripped at 250 MWe.

Assuming no further degradation in SG tube integrity and all equipment functions as designed, which of the following describes the expected procedure flowpath for this event?

- A. Perform E-0, Reactor Trip or Safety Injection, and manually initiate Safety Injection
Transition to E-3, Steam Generator Tube Rupture
Transition to ES-3.1, Post SGTR Cooldown Using Backfill
- B. Perform E-0, Reactor Trip or Safety Injection, and manually initiate Safety Injection
Transition to E-3, Steam Generator Tube Rupture
Transition to ES-3.2, Post SGTR Cooldown Using Blowdown
- C. Perform E-0, Reactor Trip or Safety Injection,
Transition to ES-0.1, Reactor Trip Response
Perform actions in 2-AOP-SG-1 in parallel with ES-0.1
Perform plant cooldown using 2-POP-3.3, Plant Cooldown
- D. Perform E-0, Reactor Trip or Safety Injection,
Transition to ES-0.1, Reactor Trip Response
When ES-0.1 is complete return to 2-AOP-SG-1
Perform plant cooldown using 2-AOP-SG-1.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because candidate may believe that once the EOP network is entered, E-3 is the best option. A SG tube leak of this size will not require a safety injection. In addition, a cooldown using backfill would be nearly impossible for a small break.

B. Incorrect. Plausible because candidate may believe that once the EOP network is entered, E-3 is the best option. A SG tube leak of this size will not require a safety injection. In addition, a cooldown using blowdown would be preferable to backfill for a small break.

C. Correct. Since conditions do not exist for a safety injection, a transition to ES-0.1 is correct. Actions in 2-AOP-SG-1 should be performed in parallel to minimize contamination.

D. Incorrect. Plausible because conditions will not exist for a safety injection. A transition to ES-0.1 is correct. However, actions in 2-AOP-SG-1 to minimize the spread of contamination should not wait until ES-0.1 is complete. In addition, 2-AOP-SG-1 does not have cooldown steps.

Technical References: 2-AOP-SG-1

Proposed References to be provided: None

Learning Objective I2LP-ILO-AOPSG1 2

Question Source: New

Question History: Unit 2 NRC 2012

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments

Exam Outline Cross Reference:	Level	RO	SRO
	Tier#		3
	Group# K/A#	1940012430	Emergency Procedures/Plan - Knowledge of which events related to system operations/status that must be reported to internal organizations or external agencies, such as State, the NRC, or the transmission system operator.
	Importance	2.7	4.1

Question # 100

Unit 2 Initial Conditions:

- Reactor Power = 100%

Unit 2 Current Conditions:

- E-0, Reactor Trip or Safety Injection has been initiated
- 21 RCP seal failure exists and the pump has been secured
- RCS leakage is 200 gpm
- CETs = 550°F
- Containment pressure = 15 psi
- Subcooled margin = 50°F
- 1 RCP is operating
- RVLIS dynamic head range = 30%

Based on the current conditions, which one of the following correctly states: (1) the EAL classification required to be made by the shift manager

and

(2) the maximum time for notification of the NRC after the declaration is made?

- A. (1) Alert
(2) 15 minutes
- B. (1) Alert
(2) 1 hour
- C. (1) Site Area Emergency
(2) 15 minutes

- D. (1) Site Area Emergency
- (2) 1 hour

Answer: B

Explanation/Justification:

A. Incorrect but plausible because the alert classification is correct with only one fission product barrier compromised. 15 minutes is plausible because that is the time to complete notifications to the state and counties after declaration,

B. Correct.

C. Incorrect but plausible because candidate may believe that RVLIS indicates a challenge to fuel clad. Fuel clad potential loss combined with RCS loss would be classified as a SAE. 15 minutes is plausible because that is the time to complete notifications to the state and counties after declaration.

D. Incorrect but plausible because candidate may believe that RVLIS indicates a challenge to fuel clad. Fuel clad potential loss combined with RCS loss would be classified as a SAE. The 1 hour notification is correct.

Technical References: IP-EP-120
Steam Tables

Proposed References to be provided: None

Learning Objective I0LP-ILO-ERT002 1.08

Question Source: Bank

Question History: NA

Question Cognitive Level: Comprehension

10 CRF Part 55 Content: 55.43 (b) 5

Comments