1 ID: RO-C-01700-E15-1 Points: 1.00

Given the following Unit 1 conditions:

- The Unit has entered a refueling outage and is in Mode 5.
- East Train RHR is OPERABLE and secured.
- West Train RHR is in service.
- RCS average temperature is 110°F.
- All S/G NR levels are 10%.
- RCS drain procedure is in progress.
- RCS is slightly above Mid Loop.

Subsequently:

- West RHR Pump TRIPS.

Which ONE of the following describes the Technical Specifications required completion time

and action for this condition?

A. Immediately initiate action to restore required S/G secondary side water levels to within limits.

B. Within 1 hour of West RHR Pump trip, close all containment penetrations providing direct access from containment atmosphere to outside atmosphere and take actions to restore one RHR loop to operation.

C. Within 1 hour of West RHR Pump trip initiate action to restore one RHR loop to operable status and in operation.

D. Immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM and take actions to restore one RHR loop to operable status and in operation.

Answer: D

Answer Explanation:

A. Incorrect Plausible since the applicant could misinterpret the given conditions by concluding that Tech Spec 3.4.7 (Loops Filled) applies. The action is in Tech Spec 3.4.7, but it is the incorrect Tech Spec.

B. Incorrect Plausible since the applicant could misinterpret the given conditions by concluding that Tech Spec 3.9.5 applies. This Tech Spec does include "Conditions" that match the conditions given in the stem (loss of an RHR train), but the applicability is incorrect. Though there is no "Completion Time" of 1 hour in this specification, it is plausible, since this Completion Time is extremely common throughout the Tech Specs.

C. Incorrect Plausible since the Tech Spec is correct, AND the action is indeed one of the Required Actions for this specification, but the Completion Time is NOT 1 hour; it is "Immediately".

D. Correct In a loops not filled condition and Mode 5 the correct Tech Spec entry and applicability is for Tech Spec. 3.4.8 (Loops not filled) with no RHR loop in operation. Per the Tech Spec the action is correct and has an immediate completion time.

Comments:

References:

Tech Spec 3.4.8, RCS Loops - Mode 5, Loops Not Filled

KA - 000025AK1.01

Loss of RHR System

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

RO - 3.9 SRO – 4.3

KA Match:

The KA is matched because the applicant must determine that a loss of RHR has occurred, and then determine the operational implications of that condition by selecting which Tech Spec applies, including Required Action and Completion Time.

Cognitive Level: H 3

Higher cognitive level since the question involves numerous mental steps, including analysis of given conditions, and application of a Tech Spec involving loops filled vs. not filled conditions.

Question Source: NEW

Associated objective(s):

(RO-C-01700-E15) State the LCOs and actions required in one hour or less for the following TS: 3.4.6, 3.4.7, 3.4.8, 3.5.2, 3.5.3, 3.9.4, 3.9.5.

2	ID: CM-7828B	Points: 1.00

Given the following conditions on Unit 2:

- Steam Gen 1/2/3/4 SF>FWF Flow Mismatch Alarms - LIT.
- Reactor Coolant System T_{avg} is 574°F and rising.
- Rods are stepping in.
- Main Steam flows are 3.6 x 10^6 lbm/hr and stable.
- Main Feedwater flows are 2.1 x 10^6 lbm/hr and rising.
- Steam Generator pressures are 720 psig and lowering.

Which of the following describes the cause and required action to be taken for the above conditions?

A. A Steam Line Break exists. Perform a Reactor Trip and Main Steamline Isolation.

B. UPC-101, Bypass Steam Header Pressure has failed HIGH. Place Steam Dump control selector switches in OFF.

C. A Feed Line Break exists. Perform a Reactor Trip and Main Feedwater Isolation.

D. MPC-253, Turbine First Stage Impulse Pressure has failed LOW. Perform actions for a failed Turbine First Stage Impulse Pressure transmitter.

Answer: C

Answer Explanation:

A. Incorrect — If a steam break existed RCS temperature would be lowering. Plausible as the alarms are associated with feed as well as steam.

B. Incorrect — Plausible as UPC-101 is an input to steam dump (in the Steam Pressure Mode). However, in this condition, (higher) power level, Tavg Mode is used.

C. Correct — Based on the conditions presented a FW break has occurred. Steam flow is indicating at the 97 to 98% power. A Reactor Trip and Feedwater isolation is warranted.

D. Incorrect — If MPC-253 failed low the alarms would come in (Steam flow higher than calculated power)but SF/FWF mismatch would not be this high. Plausible as this failure would drive some of the alarms being received.

Comments:

Reference:
RO-C-EOP07

K/A: 000054.AK1.01
Loss of Main Feedwater
Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): MFW line break depressurizes the S/G (similar to a steam line break).
RO – 4.1 SRO – 4.3

KA Match
Question matches KA as it requires knowledge of the operational impacts for the given plant conditions and the actions needed to identify and address the Feed line break (Feed Water Isolation).

Cognitive Level: H 3
Higher cognitive level since the question involves numerous mental steps, including analysis of given conditions.

Question BANK - REPEAT
Previous Use: NRC 2014
Original Question:

Associated objective(s):

(RO-C-EOP07-E4) Discuss the plant response for small, intermediate and large secondary side breaks, assuming the plant is initially steady state at full power with all systems in automatic and operating as designed with no operator actions implemented. Include in the description trends for the following parameters as the plant stabilizes and an explanation as to why these trends occur.

a. Steam and feed flow to faulted/intact SGs
b. Faulted and intact SG pressures
c. Faulted and intact SG levels
d. RCS TAVG, PZR level, WR pressure, Subcooling, 1ST STAGE IMP Pressure, and Rx Power
e. Containment pressure

3	ID: CM-39982	Points: 1.00

The following plant conditions exist:

- Unit 2 tripped from 100% power.
- A loss of all FW and AFW has occurred.
- The operating crew has **just** entered FR-H.1, Loss Of Secondary Heat Sink.
- The RO identifies that the East CCP has tripped and that the West CCP will NOT Start.

Which ONE of the following describes the consequences of the CCP failures?

A. RCS Bleed and Feed cooling must be established immediately to ensure sufficient SI flow for RCS heat removal.

B. A Red Path on the Core Cooling Critical Safety Function will develop due to loss of RCS inventory with no available makeup.

C. The RCS will not depressurize quickly enough to ensure sufficient SI flow for RCS heat removal, so additional RCS openings (vent paths) will have to be established.

D. RCS Bleed and Feed cooling must NOT be initiated and secondary depressurization to inject condensate pump flow must be immediately initiated.

Answer: A

Answer Explanation:

A. Correct — Step 2 of OHP-4023-FR-H.1 requires that Bleed and Feed be immediately initiated if the CCPs are not available. If it is known that no CCP is available for bleed and feed, then all RCPs are stopped and bleed and feed is initiated immediately. The RCS will have to depressurize below the shutoff head of the SI pumps before any ECCS injection will occur, so the bleed and feed must be initiated prior to the loss of heat removal capability of the SGs.

B. Incorrect — Although a red condition on Core Cooling may eventually occur, there is available makeup with SI.

C. Incorrect — Bleed and Feed is started early enough so that the RCS can depressurize far enough to allow SI injection. Additional openings are used if all PORVs are NOT available.

D. Incorrect — Action to align condensate pumps may still be taken, but not as a contingency to Bleed and Feed.

Comments:

References:
12-OHP-4023-FR-H-1 Loss of Heat Sink

KA – 00WE05-EK1.1
Loss of Secondary Heat Sink
Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink): Components, capacity, and function of emergency systems.
RO – 3.8 SRO – 4.1

KA Match:
Question matches KA as it questions candidate on the concept of the capacity of the ECCS system to provide heat removal capability during Loss of Heat sink conditions.

Cognitive Level: F 2

Question BANK
Previous Use: NRC 2006
Original Question:

Associated objective(s):

(RO-C-EOP11-E7) For a specified set of conditions, determine if initiation of RCS bleed and feed is necessary.

4 ID: RO26-0023 Points: 1.00

Unit 2 is operating at 100% power.

Given the following indications on RCP #21:

- RCP SEAL 1 LEAKOFF FLOW HIGH alarm is received.
- RCP No. 1 SEAL LEAKOFF FLOW indication is off scale high on the HIGH range.
- RCP SEAL 2 STANDPIPE LEVEL HIGH alarm is LIT.

Which ONE of the following has occurred AND what action is required IAW procedures?

A. The No. 1 and No. 2 SEALS have failed and an immediate reactor trip is required.

B. The No. 1 and No. 2 SEALS have failed and a controlled reactor shutdown is required.

C. The No. 2 SEAL has failed and continued monitoring of RCP conditions is required.

D. The No. 3 SEAL has failed and continued monitoring of RCP conditions is required.

Answer: A

Answer Explanation:

A. Correct #1 Seal has high flow (with > 6 gpm). With the Meter off scale High a Trip of the RCP is required. This requires a reactor trip and RCP shutdown.

B. Incorrect. Plausible because seal failures do require a shutdown but #1 Failure with Seal L/O > 6 gpm requires reactor trip.

C. Incorrect Plausible because indications do show a # 2 seal failure but #1 Seal is also failed.

D. Incorrect Plausible if candidate believes Standpipe alarm is due to #3 Seal failure.

Comments:

Reference:
02-OHP-4022-002-001, Malfunction of a Reactor Coolant Pump

KA - 000015 AK2.07
Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following:
RCP seals
RO - 2.9 SRO - 2.9

K/A Match
Question matches KA as the question requires knowledge of the RCP malfunction based on seal indications and the required actions.

Cognitive Level: H 3

Question BANK
Previous Use: RO22 Audit, RO24 AUDIT, RO26 Audit
Original Question:

Associated objective(s):

(RO-C-AOP0140412-E3) Explain the procedural mitigation strategy for a Malfunction of a RCP

5 ID: CM-0884 Points: 1.00

In Unit 2, the function of "Low T_{avg}" at 554°F, coincident with permissive P-4 (Reactor Trip), is to generate a:

A. Main Steam Line Isolation signal to prevent excessive reactivity due to a rapid RCS cooldown.

B. Feedwater Isolation signal to prevent an excessive RCS cooldown due to overfeeding of the Steam Generators.

C. Main Turbine Trip signal to prevent an excessive cooldown of the Steam Generators and the RCS.

D. Feedwater Conservation signal to ensure an equal distribution of water to the Steam Generators.

Answer: B

Answer Explanation:

A. Incorrect	Main Steam Line Isolation is from Hi Steam flow with lo-lo T_{ave}, Steam Line delta-p, Low Steam Line pressure, or containment pressure hi-hi.
B. Correct	As power is raised,2 OHP-4021-001-006, Step 4.63 Checks for temperature above 554°F to ensure operator is aware that P-4 / Feedwater Isolation with temperature below 554°F is now enabled.
C. Incorrect	Low T_{avg} with P-4 is not a turbine trip signal.
D. Incorrect	Feedwater Conservation is generated by a Train A/B signal for loss of Feedwater Pumps with AFW Pump in AUTO.

Comments:

References:
RQ-C-KNOW

KA - 000007 EK2.02
Reactor Trip - Stabilization
Knowledge of the interrelations between a reactor trip and the following:
Breakers, relays and disconnects
RO - 2.6 SRO - 2.8
CFR - 41.7 / 45.7
EPE.007.EK2.02

KA Match:
Question matches KA as it is asking the relationship between Reactor Trip breakers and Feed Water Isolation relays.

Cognitive Level: F 2

Question BANK
Previous Use: NRC 2007
Original Question:

Associated objective(s):

(RO-C-NOP7-E1) Given a procedural step, Precaution and Limitation, Note, or Caution associated with power escalation (including referenced procedures), explain the basis of the procedure step, Precaution and Limitation, Note, or Caution.

6	ID: RO-C-00800-E6-8	Points: 1.00

Given the following:

- Unit ~~2~~ 1* has experienced a LOCA
- RCS T_h is 414 °F and slowly lowering
- RCS Pressure is 275 psig and slowly lowering
- Bus T11A is de-energized due to a fault
- ECCS is still operating in its design injection phase

Besides the available charging pump, which of the following describes the injection status and approximate flow rate expected for these conditions?

A. SI Pump injecting at ~ 540 GPM
RHR Pump NOT injecting

B. SI Pump injecting at ~ 540 GPM
RHR Pump injecting at ~ 3000 GPM

C. SI Pump injecting at ~ 820 GPM
RHR Pump injecting at ~ 3000 GPM

D. SI Pump injecting at ~ 820 GPM
RHR Pump NOT injecting

Answer: A

* This typographical error was corrected ~ 25 minutes after the examination had begun; all applicants were present and were informed of the change at the time of discovery, following receipt of concurrence for the correction from the NRC Chief Examiner for this exam.

Answer Explanation:

A. Correct — The SI pump will start injecting at ~ 1566 Psig and has a run out of ~ 700 gpm. The RHR will not start injecting until pressure drops to ~ 200 psig.

B. Incorrect — This is the correct SI flow but RHR will not be injecting. Plausible since RHR would be placed in shutdown cooling mode with this temperature and pressure.

C. Incorrect — The SI pump will start injecting at ~ 1566 Psig and has a run out of ~ 700 gpm. Plausible if both SI pumps were running. The RHR will not start injecting until pressure drops to ~ 200 psig. This is the normal RHR flow in the Shutdown Cooling mode.

D. Incorrect — Plausible if both SI pumps were running and RHR injection (cooldown allowed) pressure confused.

Comments:

References:
RO-C-00800, ECCS

KA - 000011-EK2.02
Large Break LOCA
Knowledge of the interrelations between the Large Break LOCA and the following : Pumps
RO - 2.6 SRO - 2.7

KA Match
Question requires candidate to determine which ECCS pumps are injecting during the LOCA and the approximate expected flow.

Cognitive Level: H 3
The candidate must apply knowledge of the ECCS pump capacities and the pump laws to determine approximate flow rates based on RCS pressures.

Question NEW
Previous Use: New
Original Question:

Associated objective(s):

(RO-C-00800-E6) Given system flow, pressure, and temperatures, determine the ECCS status or alignment.

7 — ID: CM-1583 — Points: 1.00

U1 is operating at 100% power.

- The West CCW pump is OOS to inspect and repair packing damage. Expected time to complete the repairs is 10 hours.
- The RO notices amps rising rapidly on the East CCW pump, as well as lowering discharge flow. He stops the pump and places it to Pull-to-Lock.
- The crew then trips the U1 reactor and enters E-0.
- While Unit 1 is taking the necessary corrective actions, Unit 2 receives a call from Unit 1 to assist by cross-tying one of their systems.

Which one of the following best describes the system cross-tie action Unit 2 is being asked to take?

A. CCW to maintain cooling flow to the Centrifugal Charging Pumps.

B. ESW to maintain cooling flow to the Emergency Diesel Generators.

C. ECCS to maintain Safety Injection and Centrifugal Charging Pump flow to the Reactor Coolant System.

D. CVCS to maintain seal injection flow to the Reactor Coolant Pumps.

Answer: D

Answer Explanation:

A. Incorrect Although cooling is lost to the CCPs, the correct action is to cross-tie CVCS, not CCW.

B. Incorrect Although the EDGs do need ESW, the reactor trip did not result in a loss of ESW.

C. Incorrect Although the CCPs and SI pumps are important following a reactor trip, ECCS is not cross-tied for this event.

D. Correct The event is a loss of CCW. In accordance with OHP-4022-016-004, the opposite unit will be asked to cross-tie CVCS in order to provide seal injection to the RCPs.

Comments:

Reference:
OHP-4022-016-004 Loss of Component Cooling water

KA - 000026-AK3.03
Loss of Component Cooling Water
Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW.
RO – 4.0 SRO – 4.2

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of actions required in response to a Loss of CCW in accordance with station procedures.

Cognitive Level: H 3

Question BANK
Previous Use: Bank
Original Question:

Associated objective(s):

(RO-C-AOP0560412-E3) Explain the procedural mitigation strategy for a Loss of CCW

8 ID: NRCAUDIT07-0232 Points: 1.00

Given the following:

- The PH A Group of pressurizer heaters are out of service.
- A Heatup of the RCS (currently at 485°F) is in progress.
- Pressurizer pressure is currently at 700 psig with all PH C Group backup heaters ON and both pressurizer spray valves THROTTLED to 20% open.

A transformer fault causes a loss of power to the PH C Group of pressurizer heaters.

What actions are required regarding the pressurizer spray valves and RCS Heatup?

Pressurizer Spray valves:

A. may remain throttled at 20%. The heatup may continue because pressure will continue to rise with the RCS heatup.

B. should be closed and the heatup stopped. Closing the spray valves will help to limit the depressurization.

C. should be closed. The heatup may continue because pressure will continue to rise with the RCS heatup.

D. may remain throttled at 20% if pressurizer level is raised to maintain pressure. The heatup should be stopped because pressure cannot be raised.

Answer: B

Answer Explanation:

A. Incorrect If the sprays are left open, pressure will lower and subcooling will be lost.

B. Correct Leaving the sprays open will cause the pressurizer to cooldown and pressure to lower. Closing the spray valves will help to limit the depressurization. The heatup needs to be stopped to prevent losing RCS Subcooling.

C Incorrect RCS pressure will not rise with the heatup. While increasing temperature will cause some expansion, the cooling from spray bypass and ambient losses will cause a decrease in pressure.

D Incorrect Increasing pressurizer level will not dramatically increase pressure. The cooling from spray flow will cause a decrease in pressure and a loss of subcooling.

Comments:

References:
RO-C-00202 Pressurizer and Pressure Relief system

KA - 000027 AK3.01
Pressurizer Pressure Control (PZR PCS) Malfunction
Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions:
Isolation of PZR spray following loss of PZR heaters
RO - 3.5 SRO - 3.8
CFR - 41.5 / 41.10 / 45.6 / 45.13
APE.027.AK3.01

KA Match:
Question matches KA as the candidate is required to know the actions required to take on a loss of PZR heaters and the reasons actions are required.

Cognitive Level: H 2

Question BANK
Previous Use: NRC 2002
Original Question:

Associated objective(s):

(RO-C-00202-E13) Describe the operation of the Pressurizer Pressure Control System, both in automatic and manual.

9 ID: NRCAUDIT07-0258A Points: 1.00

Given the following conditions:

- Unit 2 control room operators are responding to a SGTR with a loss of offsite power.
- 2-OHP-4023-E-3, Steam Generator Tube Rupture is in progress.
- All SG pressures are 1032 psig.
- ALL SG PORVs are partially open.
- The Unit Supervisor has just reached Step 3 which requires flow to be isolated from the ruptured SG.

Which ONE of the following describes the actions required for the SG PORV's per 02-OHP-4023-E-3, and the associated reasons?

A. Place the RUPTURED SG PORV controller to manual and close to minimize radiological releases.

B. Change the RUPTURED SG PORV controller setpoint to 1040 psig to minimize radiological releases and prevent challenges to the SG safety valves.

C. Place ALL SG PORV controllers to manual and close to minimize radiological releases.

D. Change ALL SG PORV controller set points to 1040 psig to minimize radiological releases and prevent challenges to the SG safety valves.

Answer: B

Answer Explanation:

A Incorrect	Placing the ruptured SG PORV to manual and closed is undesirable as it may result in opening of the SG safety valve.
B Correct	The setpoint on the ruptured SG PORV is increased to 1040 psig from 1025 psig. This minimizes radiological releases and ensures the PORV is maintained available to prevent challenging the code safety valves.
C Incorrect	While placing ALL SG PORVS to Manual and close will minimize the potential for releases (both from ruptured and contaminated water in the other SGs). This is not desired since it may challenge the safety valves.
D Incorrect	While raising ALL SG PORV set points to 1040 psig will minimize the potential for releases (both from ruptured and contaminated water in the other SGs) while limiting the challenges to the safety valves this is not desired since the RCS temperature and therefore pressure may remaining higher causing a greater leak rate.

Comments:

References:
12-OHP-4023-E-3 Stem Generator Tube Rupture

KA - 000038 EK3.02
Steam Generator Tube Rupture (SGTR)
Knowledge of the reasons for the following responses as they apply to the SGTR:
Prevention of secondary PORV cycling
RO - 4.4 SRO - 4.5
CFR - 41.5 / 41.10 / 45.6 / 45.13

EPE.038.EK3.02
KA Match:
Question matches KA as candidate is questioned on guidance in SGTR procedure to prevent unnecessary PORV cycling and to minimize radiation release.

Cognitive Level: F 3

Question MODIFIED
Previous Use: NRC 2002 Modified
Original Question:

Associated objective(s):

(RO-C-EOP08-E18) For the E-3 series procedures and the ECA-3 series procedures discuss the basis or reason for all Steps.

10 **ID: CM-40367** **Points: 1.00**

The Unit Supervisor has just entered 2-OHP-4023-ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

- The Turbine Driven Aux Feedwater Pump is the only AFW pump running.
- No AFW flow adjustments have been made since AFW auto started after the trip.
- The plant trip occurred 20 minutes ago.
- RCS cold leg temperatures have lowered by 120^{O}F and are still lowering.

Which ONE of the following actions is required to be taken with the AFW system?

A. Lower AFW flow to 60,000 PPH per steam generator.

B. Isolate AFW flow to all steam generators until the cooldown stops.

C. Isolate AFW flow to #21 SG and #24 SG.

D. Lower AFW flow to 25,000 PPH per steam generator.

Answer: D

Answer Explanation:

A - Incorrect This is the typical minimum value (240 x10^3 pph total)
B - Incorrect Flow is maintained to each SG to minimize thermal shock.
C - Incorrect Flow is maintained to each SG to minimize thermal shock.
D - Correct ECA-2.1 Step 2 directs the Crew to reduce FW Flow to 25x10^3 pph to each SG if the cooldown rate is > 100^{0}F/hr.

Comments:

References:

02-OHP-4023-ECA-2.1 Uncontrolled Depressurization of All Steam Generators, step 2

KA - 00WE12 EA1.1
Uncontrolled Depressurization of all Steam Generators
Ability to operate and/or monitor the following as they apply to the Uncontrolled Depressurization of all Steam Generators: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 3.8 SRO - 3.8
CFR - 41.7 / 45.5 / 45.6

KA Match:
Question matches KA as candidate must know manual control actions taken during an Uncontrolled Depressurization of all Steam Generators.

Cognitive Level: H 3

Question BANK
Previous Use: Bank
Original Question:

Associated objective(s):

(RO-C-EOP07-E8) Explain AFW flow control during an uncontrolled depressurization of all SGs to include:

a. The four (4) reasons for controlling AFW flow
b. Minimum AFW flow rate to each SG and the basis for this minimum limit

11	ID: RO26-0111	Points: 1.00

The plant was in Mode 1. Reactor trip and Safety Injection have occurred. Due to high Aux Building radiation levels, the crew has entered 2-OHP-4023-ECA-1.2, LOCA Outside Containment. Actions have been taken in an attempt to isolate the break.

Given the following plant conditions:

- PRZ level is off scale low
- SI flow is 0 GPM
- RCS pressure is 1700 psig and rising
- Aux Building Radiation Monitors are in alarm

Which ONE of the following describes the status of the leak based on the requirements of 2-OHP-4023-ECA-1.2?

The leak is:

A. isolated based on RCS pressure rising.

B. isolated based on SI flow of 0 GPM.

C. NOT isolated based on PRZ level indication not rising.

D. NOT isolated based on Aux Building radiation monitor indication.

Answer: A

Answer Explanation:

A. Correct RCS pressure is the required parameter for determination of isolation.
B. Incorrect SI flow would be 0 if RCS pressure is above shutoff head of the SI Pump.
C. Incorrect PRZ level is not used, but it will rise after a while when RCS inventory is restored.
D. Incorrect Aux Building radiation is used as an entry condition to the procedure.

Comments:

References:

2-OHP-4023-ECA-1.2 LOCA Outside Containment

KA - 00WE04 EA1.2
LOCA Outside Containment
Ability to operate and/or monitor the following as they apply to the LOCA Outside Containment: Operating behavior characteristics of the facility
RO - 3.6 SRO - 3.8
CFR - 41.7 / 45.5 / 45.6
4.5.E04.EA1.2

KA Match:
Question matches KA as candidate is asked to monitor and analyze parameters to determine if the LOCA outside Containment is isolated based on facility procedural requirements.

Cognitive Level: F 3

Question BANK
Previous Use: NRC 2010
Original Question:

Associated objective(s):

(RO-C-EOP09-E36) For each of the E-1 Series procedures, discuss the basis or reason for all Steps.

12	ID: NRCAUDIT07-0693A	Points: 1.00

The following conditions exist:

- Charging, letdown, and PRZ level control system are in automatic.
- 2-QRV-161 Letdown Orifice Valve is Open
- Letdown Hx Outlet Flow QFI -301 73 gpm
- Charging Header Flow QFI -200 85 gpm
- Total seal flow to RCPs QFI -210 to 240 32 gpm

The controlling PRZ level channel fails high to an indicated 100% level.

Which of the following describes the short term effect on total RCP seal injection flow, assuming NO operator action?

Total seal injection flow:

A. raises to about 50 gpm.

B. lowers to about 19 gpm.

C. remains about 32 gpm.

D. lowers to 0 gpm.

Answer: B

Answer Explanation:

A. Incorrect The charging flowpath is not isolated (QRV-200 is still open). This is the minimum charging flow value.

B. Correct With the Pressurizer level channel failed high the charging flow control valve QRV-251 will close to the minimum flow position of ~ 47 gpm. This will cause seal injection flow to lower accordingly (47/78 =0.60 x 32 = 19.3).

C. Incorrect The charging flowpath is not isolated (QRV-200 is still open) so flow to the seals will lower as total charging flow lowers.

D. Incorrect The minimum flow setting on QRV-251 controller will keep some flow going to the seals.

Comments:

References:

RO-C-00202 Pressurizer and Pressure Relief system, SOD-00202-003 Pressurizer Level Control System

KA - 000022 AA1.02
Loss of Reactor Coolant Makeup
Ability to operate and/or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS charging low flow alarm, sensor, and indicator
RO - 3.0 SRO - 2.9
CFR - 41.7 / 45.5 / 45.6
APE.022.AA1.02

KA Match:
Question matches KA as candidate is asked to predict changes in monitored parameters in the CVCS system as they apply during a Loss of Reactor Coolant Makeup situation.

Cognitive Level: H 3

Question MODIFIED
Previous Use: NRC 2006
Original Question:

Associated objective(s):

(RO-C-00202-E5) Describe the control signal flowpath for the Pressurizer Level Control System, starting with the control inputs and passing sequentially through the following components until the final control or protection output is developed:

a. Pressurizer Level Channels
b. Charging Flow Instrument (QFI-250)
c. Master Level Controller
d. Pressurizer Heaters
e. CCP Discharge flow controller
f. Charging Flow Control Valve QRV-251
g. CVCS Letdown Isolation Valves
h. Reactor Protection System (RPS)

13	ID: AOP0550412-E1-11	Points: 1.00

Unit 1 is operating at 100% power when the following conditions are noted

Ann. 120, drop 9 Battery Charger 1CD1 Failure is received (In service Battery Charger).

Ann. 120, drop 29 CRID 1 Inverter Abnormal is received.

Ann. 120, drop 30 CRID 2 Inverter Abnormal is received.

Ann. 120, drop 101 Batter 1CD Voltage Alarm High or Low is received.

Which ONE of the following describes the cause of the alarms and the plant alignment 30 seconds later?

A. A loss of 600VAC has occurred; CRID Bus 1&2 will be de-energized. No backup source available.

B. A loss of 600VAC has occurred; CRID Bus 1&2 will be de-energized awaiting manual transfer to Alternate source.

C. A loss of 250 VDC has occurred; CRID Bus 1&2 will be powered directly from Alternate Source.

D. A loss of 250 VDC has occurred; CRID Bus 1&2 will be powered by inverters from Alternate Source.

Answer: D

Answer Explanation:

A. Incorrect – A loss of 250 VDC has occurred. CRID Inverter Automatically aligns to Alternate source (600VACVital) on a loss of DC Voltage.
B. Incorrect – A loss of 250 VDC has occurred. CRID Inverter Automatically aligns to Alternate source (600VACVital) via the Static switch on a loss of DC Voltage.
C. Incorrect – First part is correct. When CRID Inverter aligns to Alternate source, the 600 volt supply still uses the Inverter to reduce the 600 volt output of CVT to 118 volts
D. Correct – First part is correct. CRID Inverter Automatically aligns to Alternate source (600VACVital) on a loss of DC Voltage.

Comments:

References:
2-OHP-4024-220, Drop 29, RO-C-AOP-D13

KA - 000058 AA2.01
Loss of DC Power
Ability to determine and interpret the following as they apply to the Loss of DC Power:
That a loss of dc power has occurred; verification that substitute power sources have come on line
RO - 3.7 SRO - 4.1
CFR - 41.7 / 41.10 / 43.5 / 45.13
APE.058.AA2.01

KA Match:
Question matches KA as it requires the knowledge of the CRID Inverter (120 VAC Vital Power) response to a loss of 250 VDC and to monitor the status of the inverter following transfer.

Cognitive Level: F 3

Question MODIFIED
Previous Use: NRC 2008
Original Question:

Associated objective(s):

(RO-C-AOP0550412-E1) Identify the event and predict the response of the plant to a Loss of a 250VDC Bus, including final plant configuration.

14	ID: AOP0590412-E1-1	Points: 1.00

Units 1 and 2 are at 100% power.

The East Essential Service Water (ESW) Pumps are running on both Units.

The following alarms are received on Panel 204:

Drop 56 East ESW Header Pressure Low
Drop 53 East ESW Pump Discharge Pressure Low

Unit 2 Flow	(GPM)
WFA-702 East Flow	2800
WFA-704 East Return Flow	2000
WFA-706 West Flow	2200
WFA-708 West Return Flow	2200

The Unit 1 Flow gauges read 2300 GPM.

Which ONE of the following indicates the cause of these alarms and the appropriate remedial action?

Note: 12-OHP-4021-019-001 - Operation of the Essential Service Water System
01-OHP-4022-019-001 - ESW System Rupture
02-OHP-4022-019-001 - ESW System Rupture

A. The Unit 2 East Strainer is clogged and automatic backwash has failed to actuate; use 12-OHP-4021-019-001 to manually backwash the strainer.

B. The Unit 2 East Strainer is stuck in Backwash Mode; use 12-OHP-4021-019-001 to place the strainer controls in manual and select the standby strainer.

C. There is a leak in the Unit 2 East Essential Service Water Header; use 2-OHP-4022-019-001 to isolate the leak.

D. There is a leak in the cross tie to the Unit 1 West Essential Service Water Header; use 01-OHP-4022-019-001 to isolate the leak.

Answer: C

Answer Explanation:

A. Incorrect — The East ESW pump header pressure is low, and the East flow is > than the return Flow indicating a header leak.

B. Incorrect — Plausible if the flow transmitter was prior to the strainer

C. Correct — The U2 East return flow rate is lower than the inlet flow. This indicates a leak somewhere on the U2 East Header.

D. Incorrect — The U1 West flow rates are lower but this is acceptable as long as the inlet and return flows are the same. Plausible since the Units are cross tied and the West flow is lower.

Comments:

References:
02-OHP-2024-204 Drops 53 & 56, 02-OHP-4022-019-001 ESW System Loss / Rupture

KA - 000062 AA2.02
Loss of Nuclear Service Water
Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss
RO - 2.9 SRO - 3.6
CFR - 41.7 / 41.10 / 43.5 / 45.13
APE.062.AA2.02

KA Match:
The question involves a lowering ESW pressure (Loss of ESW) and identification that there is a leak in a header and identifying which one (Cause of the loss).

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-AOP0590412-E1) Identify the event and predict the response of the plant to ESW System Loss or Rupture, including final plant configuration.

15 **ID: CM-39773** **Points: 1.00**

Unit 2 is currently in Mode 3 following refueling. While preparing to start a plant heatup, a lightning strike causes a loss of offsite power. The crew identifies that NO Vital 4KV Bus is energized.

The following conditions exist:

- DG2AB and DG2CD are available but NOT running
- DG Incomplete Starting Sequence alarms are NOT Lit

The crew will initially attempt to restore power to the vital buses by:

A. actuating SI to start diesel generators.

B. energizing the buses from Emergency Power.

C. manually starting the diesel generators.

D. energizing the buses from Reserve Power.

Answer: C

Answer Explanation:

A. Incorrect Plausible because an SI signal would start the EDGS; however the procedure directs a start attempt via the C/S.
B. Incorrect Plausible because EP will be directed if the first attempt to start the available non-running EDGS does not restore power.
C. Correct The first attempt to restore power is to manually start any available non-running EDGs.
D. Incorrect Plausible because Step 9 will direct restoration of power from any available source, including Reserve Power.

Comments:

References:

2-OHP-4023-ECA-0.0 Loss of All AC Power

KA - 000055 EA2.03
Loss of Offsite and Onsite Power (Station Blackout)
Ability to determine and interpret the following as they apply to a Station Blackout:
Actions necessary to restore power
RO - 3.9 SRO - 4.7
CFR - 41.7 / 41.10 / 43.5 / 45.13
EPE.055.EA2.03

KA Match:
Question matches KA as it requires the candidate to determine the appropriate actions to restore power following a Station Blackout.

Cognitive Level: F 2

Question BANK
Previous Use: RO25 AUDIT
Original Question:

Associated objective(s):

(RO-C-EOP14-E7) For each of the ECA-0 series procedures, identify the Major Action Categories and discuss the bases for each.

16	ID: CM-1003B	Points: 1.00

Given the following conditions:

- Unit 1 is at 75% power
- UNIT OR SYSTEM FREQ HIGH OR LOW alarm is LIT
- Grid problems have resulted in frequency dropping to 57.9 Hz

Which ONE of the following represents the proper course of action?

A. Reduce generator load until frequency is restored to at least 59.5 Hz.

B. Manually trip the reactor immediately as frequency is below 58.2 Hz.

C. Transfer the safeguards busses to their Emergency Diesel Generators and maintain current system load.

D. Reduce power to approximately 50 MWe within 30 minutes, and then open the generator output breakers.

Answer: B

Answer Explanation:

A. Incorrect Unlimited operating time is permitted as long as frequency is greater than 59 Hz. Below 59 Hz power must be reduced and unloaded in 30 minutes. Below 58.2 Hz a trip is required.

B. Correct Rx Trip is required if Frequency is < 58.2 Hz per 1.OHP.4024.121, Drop 03

C. Incorrect The Safety Busses are not the only concern, the non-safety loads, and generator/grid stability are concerns at low frequencies. Below 58.2 Hz a trip is required.

D. Incorrect. Unlimited operating time is permitted as long as frequency is greater than 59 Hz. Maximum operating period is 30 minutes when frequency is less than 59 Hz but greater than 58.2 Hz. Below 58.2 Hz a trip is required.

Comments:

References:
OHP 4024.121, Drop 03

KA - 000077 2.2.44
Generator Voltage and Electric Grid Disturbances
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.
RO - 4.2 SRO - 4.4
CFR - 41.10 / 43.5 / 45.12
APE.077.GEN

KA Match:
Question matches KA as candidate must evaluate plant conditions given a grid disturbance and determine actions required to stabilize the plant / system.

Cognitive Level: F 4

Question MODIFIED
Previous Use: RO28 Audit
Original Question:

Associated objective(s):

(RO-C-0821040401-E3) explain the procedural mitigation strategy for Abnormal Electrical Grid Voltage and Frequency

17 **ID: RO-C-AOP0490412-T1-2** **Points: 1.00**

A malfunction of the Control Air 100# regulator on Unit One has caused the 100# Control Air header pressure to lower. The crew has entered 1-OHP-4022-064-001, Control Air Malfunction. With Control Air header pressure at 78# the crew response to this condition is to:

A. close the Ring Header Isolation Valves, notify Unit 2 to trip the reactor, enter E-0 Reactor Trip or Safety Injection.

B. have Unit Two start Unit Two CAC using 2-OHP-4022-064-001, Control Air Malfunction.

C. start Unit One CAC using 1-OHP-4021-064-001, Operation of Plant and Control Air Systems.

D. trip the reactor, enter E-0 Reactor Trip or Safety Injection.

Answer: D

Answer Explanation:

A. Incorrect Plausible as directed to verify at 85#, Unit 2 will perform reactor trip based on their control air header pressure.
B. Incorrect Plausible as this will be done by Unit Two if their Control Air header pressure reaches 90#
C. Incorrect Plausible as action is RNO for CAC not running at 90# but not using the NOP
D. Correct Procedure directs manual trip at 80#

Comments:

References:
1-OHP-4022-064-001 Control Air Malfunction

KA - 000065 2.4.8
Loss of Instrument Air
Emergency Procedures/Plan
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
\RO - 3.8 SRO - 4.5
CFR - 41.10 / 43.5 / 45.13
APE.065.GEN

KA Match:
Question matches KA as it requires the knowledge of how to use Loss of Control Air (Abnormal Operating Procedure) in conjunction with the Emergency Operating Procedures.

Cognitive Level: F 3

Question New
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP01-E25) Describe the Rules of Usage associated with the use of AOPs and ARPs during the implementation of the EOPs.

18 ID: CM-0756 Points: 1.00

Why does ECA-1.1, Loss of Emergency Coolant Recirculation, direct depressurizing the Steam Generators to 670 psig?

A. To accomplish a rapid injection of accumulator water.

B. To prepare for a controlled accumulator injection to keep the core covered.

C. To prevent inadvertent nitrogen addition to the RCS.

D. To prevent a challenge to Reactor Coolant system integrity.

Answer: B

Answer Explanation:

A. Incorrect	Accumulators don't inject until RCS Pressure is <650 psig. Since the RCS is assumed to be saturated RCS pressure will NOT lower to less than 670 psig.
B. Correct	The SGs are depressurized to lower RCS pressure and prepare for a controlled accumulator injection.
C. Incorrect	The depressurization to 670 psig is to allow controlled Accumulator injection. Later depressurization stopped at 90 psig to prevent nitrogen injection.
D. Incorrect	T_{sat} for 670 psig is ~500°F which is well above the temperatures for RCS integrity concerns. Operator may think that since the pressure is so low a challenge to integrity may occur.

Comments:

Reference: 12-OHP-4023-ECA-1.1 Loss of Emergency Coolant Recirculation, Steps 28 & 29 Background.

KA - 000E11-2.1.20
Loss of Emergency Coolant Recirculation
Ability to interpret and execute procedure steps.
RO – 4.6 SRO – 4.6

KA Match -
Question matches KA as it requires the knowledge that the reason for SG Depressurization of the SGs in ECA-1.1, Loss of Emergency Coolant Recirculation is to make accumulators available for injection in preparation for cooling the core.

Cognitive Level: F 3

Question BANK

Previous Use: NRC 2008

Original Question:

Associated objective(s):

(RO-C-EOP09-E36) For each of the E-1 Series procedures, discuss the basis or reason for all Steps.

19 **ID: 2008NRC-0279** **Points: 1.00**

Given the following plant conditions:

- A Small Break LOCA occurred 12 hours ago.
- Containment pressure is 1.2 psig.
- Containment air temperature is 215ºF.
- OHP-4023-FR-Z-3, Response to High Containment Radiation Level, is entered.

Which ONE of the following verifications is a major action category of OHP-4023-FR-Z-3?

A. Both Containment Recirculation Fans are running.

B. Upper and Lower Containment Ventilation Fans are running.

C. Containment Ventilation Isolation has occurred.

D. Control Room Ventilation System is in ISOLATE.

Answer: C

Answer Explanation:

A. Incorrect Plausible because CEQ fans are run during a LOCA but they are run to prevent localized hydrogen concentrations however they are NOT addressed within OHP-4023-FR-Z-3.

B. Incorrect Plausible since these fans do provide some air filtration however they are shutdown during accident conditions.

C. Correct Step 1 of the procedure ensures that CVI has occurred.

D. Incorrect Plausible since the Isolate position is used to maintain a habitable Control Room atmosphere during accident conditions but it is not addressed in this procedure.

Comments:

References:
12-OHP-4023-FR-Z-3 Response to High Containment Radiation Level

KA - 00WE16 EK1.3
High Containment Radiation
Knowledge of the operational implications of the following concepts as they apply to the High Containment Radiation: Annunciators and conditions indicating signals, and remedial actions associated with the High Containment Radiation
RO - 3.0 SRO - 3.3
CFR - 41.8 / 41.10 / 45.3
4.5.E16.EK1.3

KA Match:
Question matches KA as it requires candidate to know the major action categories for procedures dealing with High Containment Radiation levels

Cognitive Level: F 3

Question BANK
Previous Use: NRC 2008
Original Question:

Associated objective(s):

(RO-C-EOP13-E6) For each of the FR-Z series procedures, identify the Major Action Categories and discuss the bases for each.

20 **ID: NRCAUDIT07-1052** **Points: 1.00**

Given the following conditions:

- 2-OHP-4023-FR-C.2 (Response to Degraded Core Cooling) is in progress
- All four Reactor Coolant Pumps are in service
- NESW Supply to the RCP Motor Air Coolers has been isolated.

What are the Major Action Categories associated with implementation of 02-OHP-4023-FR-C.2?

A. 1) Shutdown of All RCPs.
   2) Controlled Secondary Depressurization.

B. 1) Reinitiating High Pressure Safety Injection.
   2) Controlled Secondary Depressurization.

C. 1) Shutdown All RCPs.
   2) RCS Depressurization to Refill the Pressurizer.

D. 1) Reinitiating High Pressure Safety Injection.
   2) RCS Depressurization to Refill the Pressurizer.

Answer: B

Answer Explanation:

A. Incorrect	RCPs are shut down following the secondary plant depressurization.
B. Correct	The major actions of FR-C.2 are to reinitiate High Head SI and to perform a controlled secondary plant depressurization to cool down and depressurize the RCS to inject the SI accumulators.
C. Incorrect	RCPs are shut down following the secondary plant depressurization. The goal is only to get enough water into the RCS to keep the core covered, not to refill the pressurizer.
D. Incorrect	The goal is only to get enough water into the RCS to keep the core covered, not to refill the pressurizer.

Comments:

References:
RO-C-EOP10, 12-4023-FR-C-2 Response to Degraded Core Cooling

KA - 00WE06 EK1.2
Degraded Core Cooling
Knowledge of the operational implications of the following concepts as they apply to the Degraded Core Cooling: Normal, abnormal and emergency operating procedures associated with Degraded Core Cooling
RO - 3.5 SRO - 4.1
CFR - 41.8 / 41.10 / 45.3
4.5.E06.EK1.2

KA Match:
Question matches KA as candidate is required to know the major action categories for the procedure for degraded Core Cooling.

Cognitive Level: F 3

Question BANK
Previous Use: BANK RO25 Audit
Original Question:

Associated objective(s):

(RO-C-EOP10-E12) For each of the FR-C and FR-I series procedures identify the Major Action Categories and discuss the bases for each.

21	ID: CM-7870	Points: 1.00

Which ONE of the following is the expected response to a High Radiation Signal (RED) on the ERS-7401, Unit 1 Control Room Area Radiation Monitor?

A. Only the Unit 1 East and West Pressurization fans start and dampers align.

B. Either the Unit 1 East or West Pressurization fan starts and dampers align depending on which Control Room Air Handling Unit is in service.

C. Only the Unit 1 and Unit 2 East Pressurization fans start and dampers align.

D. Both the Unit 1 and Unit 2 East and West Pressurization fans start and dampers align.

Answer: A

Answer Explanation:

A. Correct	12-OHP-4024-139, drop 14 states that both the East and West Pressurization fans will start and the associated dampers align.
B. Incorrect	Both the East and West Pressurization fans will start and the associated dampers align. One Pressurization fan must be manually stopped.
C. Incorrect	Both the Unit 1 East and West Pressurization fans will start and the associated dampers align. A SI signal on either Unit starts All Pressurization fans on both units. One Pressurization Fan on each Unit is manually stopped.
D. Incorrect	Both the Unit 1 East and West Pressurization fans will start and the associated dampers align. A SI signal on either Unit starts All Pressurization fans on both units. One Pressurization Fan on each Unit is manually stopped.

Comments:

References:
12-OHP-4024-139, drop 14

KA - 000061 AK2.01
Area Radiation Monitoring (ARM) System Alarms
Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following:
Detectors at each ARM system location
RO - 2.5 SRO - 2.6
CFR - 41.7 / 45.7
APE.061.AK2.01

KA Match:
Question matches KA as the candidate must know how the Unit 1 and Unit 2 control room radiation levels are monitored and automatic actions controlled through the detector alarms.

Cognitive Level: F 3

Question BANK
Previous Use: NRC 2006, RO28 Audit
Original Question:

Associated objective(s):

(RO-C-02801A-E8) Given the following Control Room Ventilation System components, describe the conditions that will cause the component to trip, automatically/manually start and/or automatically/manually reposition.

a. Pressurization fans
b. Chiller packages
c. Remotely operated dampers (ACR-DA-1, 1A, 2, 2A, 3)

22	ID: NRCAUDIT07-0162	Points: 1.00

The second step of OHP-4023-ES-0-3, Natural Circulation Cooldown With Steam Void in Vessel, directs the operator to try to start a Reactor Coolant Pump in accordance with OHP-4023-SUP-010, Starting Reactor Coolant Pump(s).

Assuming that RVLIS indicates less than full, why is Pressurizer Level required to be raised to 87% prior to starting a RCP?

A. Ensure that the RCP will have sufficient Net Positive Suction Head for the given plant conditions.

B. Ensure that when the RCP is started a subsequent decrease in Pressurizer Level will not uncover the heaters and/or result in a loss of pressure control.

C. Ensure that there is sufficient mass in the Pressurizer to fill the vessel head should a bubble form, to prevent allowing saturated fluid from entering the SG tubes.

D. Ensure that the Pressurizer will be able to maintain RCS pressure high enough to prevent Nitrogen injection from the Accumulators.

Answer: B

Answer Explanation:

A. Incorrect NPSH requirements for the RCP are met by verifying proper seal Delta P.

B. Correct Per OHP-4023-SUP-010, Starting Reactor Coolant Pump(s), Pressurizer level is raised to >87%, subcooling to at least 62°F, and Pressurizer heaters are energized to saturate the Pressurizer if the vessel is NOT full. This is done to prevent emptying the Pressurizer.

C. Incorrect Based on OHP-4023-ES-0-3, Natural Circulation Cooldown With Steam Void in Vessel, a void / steam bubble already exists in the vessel. With a RCP started, any saturated fluid will be quickly condensed.

D. Incorrect Nitrogen from the Accumulators will NOT inject until RCS pressure is <140 psig and the liquid volume of the accumulators has been injected. This is below the pressure at which a RCP may be operated.

Comments:

References:

12 OHP-4023-SUP-010 starting reactor Coolant pumps

KA - 00WE10 EK2.2
Natural Circulation with Steam Void in Vessel with/without RVLIS
Knowledge of the interrelations between the Natural Circulation with Steam Void in Vessel with/without RVLIS 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
RO - 3.6 SRO - 3.9
CFR - 41.7 / 45.7
4.5.E10.EK2.2

KA Match:

Question matches KA as candidate is required to demonstrate knowledge on RVLIS indications and how it relates to plant conditions with a steam void in the reactor head while using natural circulation.

Cognitive Level: F 3

Question BANK
Previous Use: NRC 2002
Original Question:

Associated objective(s):

(RO-C-EOP03-E2) Discuss the conditions that should be established before RCP restart when the EOPs are in effect.

23 **ID: RO-C-AS19-E3-1** **Points: 1.00**

A fire has been reported in the Unit 2 Control Room Cable Vault (CRCV).

Which ONE of the following describes the actions the Control Room Operator must take?

A. 1. Verify automatic water spray into the CRCV.
2. Verify personnel out of the CRCV.
3. Manually actuate CO2 discharge into the CRCV.
4. After seven (7) minutes, manually re-actuate CO2 discharge.

B. 1. Verify automatic Halon actuation into the CRCV.
2. Verify personnel out of the CRCV.
3. Manually actuate water spray into the CRCV.
4. After seven (7) minutes, manually re-actuate water spray.

C. 1. Verify automatic CO2 actuation into the CRCV.
2. Verify personnel out of the CRCV.
3. Manually actuate Halon into the CRCV.
4. After seven (7) minutes, manually re-actuate Halon.

D. 1. Verify automatic Water spray actuation into the CRCV.
2. Verify personnel out of the CRCV.
3. Manually actuate Halon discharge into the CRCV.
4. After seven (7) minutes, manually re-actuate Halon.

Answer: A

Answer Explanation:

A. Correct Water spray does exist in Unit 2. CO2 is actuated manually as described.
B. Incorrect Water spray actuates automatically in Unit 2. Halon actuation is Automatic and is verified.
C. Incorrect Halon will automatically actuate, followed by a manual CO2 actuation per 2-OHP-4024-101, Drop 56.
D. Incorrect Water spray actuates automatically in Unit 2. Halon actuation is Automatic and is verified

Comments:

References:
OHP-4024-201, Drop 46,56,57

KA - APE000067 2.2.3
Plant Fire on Site
Knowledge of the design, procedural, and operational differences between units.
RO – 3.8 SRO - 3.9
CFR - 41.5 / 41.10 / 45.6 / 45.13
APE.067.2.2.3

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of differences in plant fire protection procedures and design.

Cognitive Level: F 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-AS19-E3) State the fire protection systems used to protect the control room cable vault in both Unit 1 and 2 and describe how each is activated.

24 **ID: CM-39382** **Points: 1.00**

A Steamline break has occurred on Unit 1 SG #11. The break was isolated and Safety Injection (SI) has just been terminated.

The following plant conditions exist:

- East CCP aligned to VCT with normal charging and letdown in service.
- SI and RHR pumps shutdown.
- RCPs are stopped.
- Pressurizer pressure = 1800 psig and rising.
- Pressurizer level = 64% and rising.
- RCS Core Exit temperature = 503°F and rising.
- Containment pressure = 0.1 psig.

SG	11	12	13	14
Levels (NR)	0%	13%	20%	20%
Pressures (psig)	0	825	830	830

Which ONE of the following actions are required for plant recovery and why?

A. Open SG PORVs to stabilize the heatup to prevent pressurizer overfill.

B. Raise Charging flow to raise the Pressurizer level to 82% to enable RCP start.

C. Reinitiate High Head SI flow to stop the heatup.

D. Close SG PORVs to allow plant to return to normal temperature and pressure.

Answer: A

Answer Explanation:

A. Correct Opening the SG PORVs will stabilize the plant heatup and limit the rise in PRZ level (due to RCS expansion).

B. Incorrect Pressurizer Level is only increased to 82% for an RCP start in the case of RCS voiding. The RCS is adequately subcooled in this situation.

C. Incorrect High head SI flow is not required to stabilize heatup. High Head SI flow will increase RCS Volume and contribute to the likelihood of an over pressurization of the RCS.

D. Incorrect Allowing Temperature and pressure to return to normal is undesirable given these conditions.

Comments:

References:
1-OHP-4023-ES-1.1 SI Termination

KA - 00WE02 EA1.3
SI Termination
Ability to operate and/or monitor the following as they apply to the SI Termination:
Desired operating results during abnormal and emergency situations
RO - 3.8 SRO - 4.0
CFR - 41.7 / 45.5 / 45.6
4.5.E02.EA1.3

KA Match:
Question matches KA as candidate is required to evaluate plant conditions during SI termination and identify actions to be taken.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2004
Original Question:

Associated objective(s):

(RO-C-EOP07-E9) Explain the reason for and importance of minimizing/controlling the amount of RCS heatup during secondary side faults following completion of SG blowdown

25 **ID: CM-39975** **Points: 1.00**

During a power ascension, with reactor power at 48%, Control Bank C - Group 1 rod B-8 drops. Prior to the drop it was at 228 steps. While restoring the rod, a Control Rod Urgent Failure alarm occurs.

Which ONE of the following explains why the alarm actuated?

A. All other Bank C - Group 1 rod lift coil disconnect switches are open.

B. All Bank C - Group 2 rod lift coil disconnect switches are open.

C. The step counter of the pulse to analog (P/A) converter was reset to 0.

D. Group C rod moving with group D rods withdrawn.

Answer: B

Answer Explanation:

A. Incorrect While all other Bank C Group 1 rods lift coils de energized, the Alarm is generated from the failure of Group 2 movement (System monitors current through the lift coils - Since Bank C group 1 rod B-8 still has current the alarm is from group 2). Plausible if the system monitored all rods in the bank and compared to the single moving rod's lift coil current.

B. Correct Since the dropped rod is completely inserted, the lift coil disconnect switches for all operable rods within the affected bank are opened. An Urgent failure will occur when the misaligned rod begins to move. This is caused by the non-movement of the group without the misaligned rod.

C. Incorrect The Urgent Failure circuit scans actual lift coil currents to determine rod movement, not indication of bank position provided by the P/A Converter. Plausible because the P/A Converter is reset by MTI during the recovery, and if improperly reset to zero, alarms would be generated (Rod Bank Lo and Rod Bank Lo-Lo).

D. Incorrect The Urgent Failure circuit does not compare lift coil currents between banks, only between groups within a bank. Plausible because this action would result in a Rod Sequence Violation alarm. In this instance, Group C is moved in the bank select mode.

Comments:

References:
4024-110 drop 26, 4022-012-005 Dropped rod

KA - 000005 AA1.02
Inoperable/Stuck Control Rod
Ability to operate and/or monitor the following as they apply to the Inoperable/Stuck Control Rod:
Rod selection switches
RO - 3.7 SRO - 3.5
CFR - 41.7 / 45.5 / 45.6
APE.005.AA1.02

KA Match:
Question matches KA as candidate is required to understand actions taken for a dropped control rod and evaluate annunciators that result from manipulation of the rod selector switch during recovery.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2006, NRC 2010
Original Question:

Associated objective(s):

(RO-C-AOP0240412-E3) Given a set of plant conditions and the occurrence of an abnormal event, without use of references, explain the procedural mitigation strategy for Dropped Rod, Misaligned Rod, or RPI Failure in accordance with plant procedures, and standards and expectations for performance.

26	ID: RO-C-EC01-E1-3	Points: 1.00

Given the following plant conditions:

- The unit is at 100% power when a fire occurs in the Control Room Cable Vault.
- Due to the large amount of smoke in the Unit 1 Control Room, it is decided by the Shift Manager that the Unit 1 control room must be evacuated.

The Unit Supervisor directs the U2 AFW pump be aligned to supply makeup to the #11 and #14 Steam generators

Which ONE of the following responses below describes the Level Indications available for the #11 Steam Generator?

The SG__(1)__Level Indications are available at__(2)__ local Auxiliary Building location(s)

	(1)	(2)
A.	Narrow Range	a single
B.	Wide Range	a single
C.	Narrow Range	two
D.	Wide Range	two

Answer: D

Answer Explanation:

A. Incorrect The SG Narrow range instruments are used in most EOPs but are not available outside the control room.
B. Incorrect The SG Wide Range Levels are located on the LS-1 and LS-2 Local stations (2 SG on each)
C. Incorrect The SG Narrow range instruments are used in most EOPs but are not available outside the control room.
D. Correct The SG #11 and # 14 Wide Range Levels are located on the LSI-1 and LSI-2 Local stations (2 SG on each) and on LSI-4 (All 4)

Comments:

References:
OP-1/2-99020

KA - 000068 AA2.01
Control Room Evacuation
Ability to determine and interpret the following as they apply to the Control Room Evacuation:
S/G level
RO - 4.0 SRO - 4.3
CFR - 41.7 / 41.10 / 43.5 / 45.13
APE.068.AA2.01

KA Match:
Question matches KA as candidate is required to demonstrate knowledge on indications available for S/G level outside the control room when the control room is evacuated.

Cognitive Level: H 2

Question NEW
Previous Use: New
Original Question:

Associated objective(s):

(RO-C-EC01-E1) State the parameters monitored at each of the following remote operating panels.
a. SG and RCS Loop Panels (LSI-1, 2, 5 and 6).
b. Charging and Letdown Panel (LSI-3).
c. RCS Shutdown Indication Panel (LSI-4).

27	**ID: NRCAUDIT07-0011A**	**Points: 1.00**

Unit 1 has just entered Mode 4 following a maintenance outage to replace a leaky fuel assembly. The East RHR pump suction line begins to leak causing radiation levels to increase.

The following radiation channels have alarmed:

- ERA-7305 U1 East RHR Pump Room – RED
- VRS-1505 Unit Vent Effluent Low Range Noble Gas - RED

Which ONE of the following is true regarding system operations based on these conditions?

A. The Auxiliary Building Supply fans have automatically tripped.

B. The AES Fan Charcoal Filter has automatically aligned.

C. The AES Fan Charcoal Filter must be manually placed in service.

D. The Auxiliary Building Supply fans must be manually tripped.

Answer: C

Answer Explanation:

A. Incorrect Auxiliary Building Supply fans are not automatically tripped. (Fuel Pool area fans are tripped on local radiation)

B. Incorrect The dampers realign for flow through the charcoal filter bed when actuation by the manual selector switch or a Phase B actuation signal occurs.

C. Correct If the alarm actuates on the ERA-7300 pump rooms the Operator is required to place the AES Fan Charcoal Filter test Selector Switch to the CHAR FILT position.

D. Incorrect Auxiliary Building Supply fans are not stopped.

Comments:

References:

12-OHP-4024-139, Drop 11

KA - 000059 2.4.9

Accidental Liquid Radwaste Release

Emergency Procedures/Plan

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

RO - 3.8 SRO - 4.2

CFR - 41.10 / 43.5 / 45.13

APE.059.GEN

KA Match
Question requires operator knowledge of response required for a radiation spill/release via the RHR system.

Cognitive Level: Low

Question BANK - REPEAT
Previous Use: NRC 2004, NRC 2012
Original Question:

Associated objective(s):

(RO-C-02801B-E3) Describe the function of the following Auxiliary Building Ventilation System Major Components:
a. Fans
b. Filters
c. Heaters
d. Charcoal Adsorbers
e. Backdraft Dampers
f. Volume Control Dampers
g. Remotely Operated Dampers

28	ID: NRCAUDIT07-0801	Points: 1.00

The following plant conditions exist:

- The Reactor is operating at 100% Power.
- Loop flow measurement determined the '#4' Reactor Coolant Pump impeller has degraded such that its Reactor Coolant System (RCS) loop flow has lowered by 5% from its original value.
- The other three RCS loop flows remain UNCHANGED.

Based on these conditions, which ONE of the following would be a result of the lowered flow rate in the #4 RCP loop?

A. Delta temperature in the #4 RCS loop at full power will be lower.

B. Demand on the pressurizer variable heaters at 2235 psig will be lower.

C. Steam pressure in the #4 Steam Generator at full power will be higher.

D. The reactor core margin to Departure from Nucleate Boiling will be lower.

Answer: D

Answer Explanation:

A Incorrect Plausible since Delta T is affected however, Delta T would actually be higher in this situation.

B Incorrect Plausible since Loop 3 & 4 RCPs provide PRZ spray. However, based on plant design, the #4 RCP is less effective than the #3 RCP and provides limited flow. Additionally under steady state - 2235 psig conditions, the sprays are closed and bypasses provide minimal flow to keep the lines warm, so therefore not much impact on current draw from variable heaters.

C Incorrect Plausible since Delta T is affected however, Higher Delta T means Tcold would be lower and therefore steam pressure would be lower.

D Correct Putting out the same MWt with a reduced flow rate means reduced heat transfer capabilities and therefore operation closer to DNB.

Comments:

References:
TS Basis - B 3.4.1

KA - 003000 K1.10
Reactor Coolant Pump System (RCPS)
Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems:
RCS
RO - 3.0 SRO - 3.2
CFR - 41.2 to 41.9 / 45.7 to 45.8
SF4.003.K1.10

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of how changes in RCP performance will affect the RCS and the reactor itself.

Cognitive Level: H 3

Question BANK
Previous Use: RO24 Audit
Original Question:

Associated objective(s):

(RO-C-01100-E2) Explain the purposes/functions/bases of the following RPS actuation circuits (Reactor Trips):

a. Manual
b. Power Range (high, low, and rate)
c. Intermediate Range
d. Source Range
e. OT(T
f. OP(T
g. Pressurizer Pressure (high and low)
h. Pressurizer Level
i. Loss of RCS Flow
j. SG Low-Low Level
k. SG Low Level w/ Feed/Steam Flow Mismatch
l. RCP UV
m. RCP UF
n. RCP Breaker Position
o. Turbine Trip (oil pressure and stop valve position)
p. Safety Injection

29 ID: AOP0680412-2-1 Points: 1.00

Given the following:

- Unit 1 is in Mode 3.
- The unit experiences a Loss of NESW.

Which of the following describes impact and the required actions in regards to the Containment Chilled Water System?

A. Containment Chillers have lost cooling water. Stop ALL but 1 RCP and perform a Containment Pressure Relief.

B. Containment Chillers are NOT affected. Lower chill water temperature IAW 1-OHP-4021-028-018, "Operation of the Containment Chilled Water System," to provide increased Containment Cooling.

C. Containment Chillers have lost cooling water. Place the Plate and Frame Heat Exchanger in Service IAW 1-OHP-4021-028-018, "Operation of the Containment Chilled Water System."

D. Containment Chillers are NOT affected. Stop associated RCP if Motor Temperature exceeds 145 ^{o}C.

Answer: A

Answer Explanation:

A. Correct NESW Cools the Chillers and a Loss of NESW directs the RCP to be stopped and a Pressure relief to be started.

B. Incorrect NESW Cools the Chillers. .Plausible if NESW still provided cooling to the RCP Motor Coolers and the chillers didn't lose cooling.

C. Incorrect The Plate and Frame HX would normally be used if the chiller failed but NESW also supplies the Plate and Frame HX.

D. Incorrect NESW Cools the Chillers. Three RCPs are stopped to limit the heat load going into containment.

Comments:

References:

RO-C-02001 Containment Chilled water system, OHP-4022-002-001 Malfunction of a Reactor Coolant Pump

KA - 022000 K1.01
Containment Cooling System (CCS)
Knowledge of the physical connections and/or cause-effect relationships between
the CCS and the following systems:
SWS/cooling system
RO - 3.5 SRO - 3.7
CFR - 41.2 to 41.9 / 45.7 to 45.8
SF5.022.K1.01

KA Match:
Question matches KA as the candidate is required to know the effects of a loss of Service Water on the Containment Cooling system.

Cognitive Level: H 3

Question NEW
Previous Use: NEW
Original Question:

Associated objective(s):

(RO-C-AOP0680412-2) explain the required operator actions to stabilize plant conditions after a Loss of Containment Chilled Water in accordance with plant procedures, and standards and expectations for performance.

30 **ID: RO27AUDIT-55** **Points: 1.00**

Which ONE of the following identifies the effect on the Containment Cooling System as containment pressure starts to rise during a LOCA?

A Phase A Isolation will trip the ______________________________.

A. Instrument Room Vent fans ONLY.

B. Instrument Room Vent fans AND CRDM Vent fans.

C. Lower Containment Vent fans ONLY.

D. Lower Containment Vent fans AND CRDM Vent fans.

Answer: A

Answer Explanation:

A. Correct	Instrument Room Coolers do isolate on a Phase A.
B. Incorrect	Plausible, since one of these components do isolate on a Phase A, and the other isolates on a Phase B.
C. Incorrect	Plausible, since these coolers do isolate on an isolation signal, but it is a Phase B, not Phase A
D. Incorrect	Plausible, since these coolers do isolate on an isolation signal, but it is a Phase B, not Phase A

Comments:

References:

RO-C-02800 Containment Ventilation system

KA - 103000 K1.01
Containment System
Knowledge of the physical connections and/or cause-effect relationships between the Containment System and the following systems:
CCS
RO - 3.6 SRO - 3.9
CFR - 41.2 to 41.9 / 45.7 to 45.8
SF5.103.K1.01

KA Match:
Question matches KA as candidate is required to have knowledge of how changes in containment conditions will affect the Containment Cooling System (ventilation).

Cognitive Level: F 3

Question BANK
Previous Use: RO27 AUDIT
Original Question:

Associated objective(s):

(RO-C-02800-E9) Describe the conditions that will cause the following Containment Ventilation component(s) to trip, automatically and/or manually start, and automatically and/or manually reposition.

a. All Fans
b. All Containment Isolation Valves (Dampers)
c. Air Recirculation/Hydrogen Skimmer Dampers VMO-101/102
d. Containment Ventilation Isolation Signal

31	ID: CM-39490	Points: 1.00

Given the following:

- Unit 2 Plant Air Compressor (PAC) is operating with Unit 1 PAC in Standby.
- Both Units are operating at 100% when a tornado causes a Loss of All Offsite Power.
- Both Units Emergency Diesel Generators started and are supplying their respective buses.

Which ONE of the following describes the impact to the Unit 1 Plant & Control Air Systems due to the loss of power, assuming no operator action?

A. Plant Air Compressor is locked out on a load shed signal.
Control Air Compressor is locked out on a load shed signal.

B. Plant Air Compressor is locked out on a load shed signal.
Control Air Compressor will auto start if pressure lowers below auto start setpoint.

C. Plant Air Compressor will automatically start and load.
Control Air Compressor is locked out on a load shed signal.

D. Plant Air Compressor will automatically start and load.
Control Air Compressor will auto start if pressure lowers below auto start setpoint.

Answer: B

Answer Explanation:

A. Incorrect The Control Air Compressor does NOT receive a load shed (lockout) signal.

B. Correct The Plant Air Compressor (PAC) receives a load shed (lockout) signal on an under voltage condition, during the LOOP, and will NOT automatically restart. The Control Air Compressor (CAC) does NOT receive a load shed signal and will have power available when the Vital 600v block loads are re-energized from the EDGs. The CAC will auto start on low pressure if pressure lowers below the auto start setpoint.

C. Incorrect The Plant Air Compressor (PAC) will NOT start due to a load shed (lockout) signal. The Control Air Compressor does NOT receive a load shed (lockout) signal.

D. Incorrect The Plant Air Compressor (PAC) will NOT start due to a load shed (lockout) signal.

Comments:

References:

OP-2-98701, OP-2-98702, OP-2-98034, RO-C-06401 Compressed Air system

KA - 078000 K2.01
Instrument Air System (IAS)
Knowledge of bus power supplies to the following:
Instrument air compressor
RO - 2.7 SRO - 2.9
CFR - 41.7
SF8.078.K2.01

KA Match:

Question matches KA as candidate is required to demonstrate knowledge of power supplies to stations air compressors.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2002, RO28 AUDIT
Original Question:

Associated objective(s):

(RO-C-06401-E10) Describe how the following items provide a support function for the Compressed Air Systems. Also describe how a loss of this support function affects the operation of the Compressed Air System:

a. PAC Auxiliary Lube Oil Pump
b. Seal Air
c. NESW System
d. 600V AC

32 **ID: NRCAUDIT07-0043** **Points: 1.00**

Unit 2 is performing actions of 2-OHP-4023-E-1, Loss of Reactor or Secondary Coolant, in response to a Large Break LOCA.

On the Reactor Trip power was lost to 600 volt bus C, 2-EZC-C.

The Crew has met the criteria to isolate the SI Accumulators.

Which ONE of the following describes the method required to complete this isolation based on these conditions?

Dispatch operators to energize the Accumulator outlet valves on...

A. 2-EZC-A and 2-EZC-D and then manually close all 4 accumulator outlet valves.

B. 2-EZC-A, 2-EZC-B, and 2-EZC-D and then manually close 3 accumulator outlet valves. Depressurize the remaining accumulator.

C. 2-ABV-A and 2-ABV-D and then manually close all 4 accumulator outlet valves.

D. 2-EZC-B and then manually close 2 accumulator outlet valves. Depressurize the remaining 2 accumulators.

Answer: B

Answer Explanation:

A. Incorrect	Accumulators are powered from All 4 EZC Buses.
B. Correct	The Accumulator Outlet valves are powered from 2-EZC-A, 2-EZC-B, 2-EZC-C, and 2-EZC-D. With power lost to 600 volt C bus, EZC-C, 2-IMO-110 will NOT close. The Crew will need to depressurize Accumulator #1.
C. Incorrect	Accumulators are powered from All 4 EZC Buses.
D. Incorrect	Accumulators are powered from All 4 EZC Buses.

Comments:

References:

02-OHP-4023-E-1, Loss of Reactor or Secondary Coolant

KA - 006000 K2.04
Emergency Core Cooling System (ECCS)
Knowledge of bus power supplies to the following:
ESFAS-operated valves
RO - 3.6 SRO - 3.8
CFR - 41.7
SF3.006.K2.04

KA Match:

Question matches KA as candidate is required to demonstrate knowledge of power supplies to accumulator outlet (ESFAS) vlaves.

Cognitive Level: F 3

Question BANK
Previous Use: NRC 2004
Original Question:

Associated objective(s):

(RO-C-00300-E9) Given the following support system(s), explain how a loss of each of the support systems will affect the operation of the CVCS.

a. Control Air
b. Electrical Power
c. CCW
d. PRZ Level Control System

33 ID: RO-C-EOP09-E35-5 Points: 1.00

Given the following conditions:

- Unit 1 experienced a Large Break Loss of Coolant Accident about 20 minutes ago.
- Emergency Core Cooling System (ECCS) Pumps are running in Cold Leg Recirculation.

The transfer of Containment Spray (CTS) Pumps to recirculation is complete with the East and West CTS Pumps running with discharge pressures fluctuating from 220 to 160 psig.

Which of the following lists the action required per 1-OHP-4023-ES-1.3 Transfer to Cold Leg Recirculation?

A. Throttle closed CTS Pump Discharge Valves until pressure stabilizes and continue with 1-OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation.

B. Open 1-IMO-215 and 1-IMO-225, CTS PUMP SUCTION FROM RWST Valves and continue with 1-OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation.

C. Place CTS Pumps in Pull to Lock and transition to 1-OHP-4023-FR-Z.1, Response to High Containment Pressure.

D. Transition to 1-OHP-4023-ECA-1.3, Sump Blockage Control Room Procedure and shutdown ECCS and CTS pumps as directed.

Answer: D

Answer Explanation:

A. Incorrect Plausible because throttling closed on discharge valve will generally reduce flow and correct cavitation.

B. Incorrect Plausible if thought that opening RWST TO CTS Suctions would assist with eliminating cavitation in the Containment Spray Pumps, however, under no condition in ES-1.3 are these valves opened.

C. Incorrect Stopping the pumps would eliminate the symptoms on the CTS pumps and may delay loss of all suction but the blockage/cause for cavitation needs to be addressed.

D. Correct Per the ES-1.3, Foldout Page and step 6.u, transition is made if both trains are indicating cavitation.

Comments:

References:
1-OHP-4023-ES-1.3 Transfer to Cold Leg Recirculation, Foldout Page and step 6

KA - 026000 K3.02
Containment Spray System (CSS)
Knowledge of the effect that a loss or malfunction of the CSS will have on the following:
Recirculation spray system
RO - 4.2 SRO - 4.3
CFR - 41.7 / 45.6
SF5.026.K3.02

KA Match:
Question matches KA as candidate must determine the impact of a cavitating CTS pump while in recirculation mode.

Cognitive Level: H 3

Question BANK
Previous Use: RO29 AUDIT
Original Question:

Associated objective(s):

(RO-C-EOP09-E35) For each of the E-1 Series procedures, discuss the basis or reason for all Foldout Page Items

34 ID: NRCAUDIT07-0922 Points: 1.00

A Small Break LOCA occurred with a loss of offsite power. The diesel generators have started and all the required loads have sequenced on. Safety injection has been reset and the RHR pumps were stopped as directed in 2-OHP-4023-ES-1.2.

- Offsite Power was restored to Bus T21A & T21B.
- The BOP was directed to shut down the 2AB EDG and inadvertently depressed the Emergency Trip Pushbutton for the 2CD EDG.

Which one of the following describes the plant response and the required actions to restore the EDG and associated equipment to its prior status?

The HEA relay will need to be reset:

A. locally to restart the EDG and re-energize T21C & T21D.
The associated CCP, SI, and RHR pumps will automatically restart.
The crew will need to shut down the RHR pump.

B. locally to restart the EDG and re-energize T21C & T21D.
The associated CCP, SI, and RHR pumps will NOT automatically restart.
The crew will need to start the associated CCP and SI pump.

C. in the control room to restart the EDG and re-energize T21C & T21D.
The associated CCP, SI, and RHR pumps will then automatically restart.
The crew will then need to shut down the RHR pump.

D. in the control room to restart the EDG and re-energize T21C & T21D.
The associated CCP, SI, and RHR pumps will NOT automatically Restart.
The Crew will need to Start the associated CCP and SI pump.

Answer: D

Answer Explanation:

A. Incorrect The HEA is in the Control room & Pumps will NOT automatically Restart. Plausible if the student believes that local action is required to reset the HEA and that ECCS pumps will automatically restart.

B. Incorrect The HEA is in the Control room. Plausible if the student believes that local action is required to reset the HEA.

C. Incorrect The Pumps will NOT automatically Restart. Plausible if student believes that ECCS pumps will automatically restart.

D. Correct The EDG will NOT automatically Restart until the HEA relay is reset. This relay is located in the Control Room. Then the EDG will restart due to the Bus Under voltage signal. The Blackout Loads will restart, but the ESF loads must be manually started.

Comments:

References:

RO-C-03200 Emergency Diesel Generator system, 12-OHP-4023-ES-1.2 Post LOCA Cooldown and Depressurization Caution 1C2 Background

KA - 064000 K3.01
Emergency Diesel Generator (ED/G) System
Knowledge of the effect that a loss or malfunction of the ED/G System will have on the following:
Systems controlled by automatic loader
RO - 3.8 SRO - 4.1
CFR - 41.7 / 45.6
SF6.064.K3.01

KA Match:
Question matches KA as candidate is required to demonstrate knowledge on actions to recover EDG after a loss of EDG.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2007
Original Question:

Associated objective(s):

(RO-C-03200-E10) For the Diesel Engine, describe the conditions for the following:

a. Conditions that will cause the Diesel Engine to automatically/manually trip.
b. Conditions that will prevent the Diesel Engine from an automatic/manual trip.
c. Conditions that will cause the Diesel Engine to automatically/manually start.
d. Conditions that will prevent the Diesel Engine from an automatic/manual start.
e. Conditions that will cause an Incomplete Start signal.
f. Diesel Engine Governor response to changing speed/load.

35	ID: CM-2042A	Points: 1.00

The primary function of the Ice Condenser is to limit peak ______ in the containment to less than the design limit following a Loss of Coolant Accident, or a _____.

A. pressure; Steam Line Break

B. pressure; Loss Of Containment Cooling

C. temperature; Steam Line Break

D. temperature; Loss Of Containment Cooling

Answer: A

Answer Explanation:

A. Correct	The ice bed absorbs thermal energy to limit pressure from a Loss of Coolant accident or Steam line Break.
B. Incorrect	First part is correct. The ice bed absorbs thermal energy to limit pressure from a Loss of Coolant accident or Steam line Break.
C. Incorrect	While Tech Specs do limit Containment temperatures, the concern is Containment leakage due to elevated pressure. Second Part is Correct
D. Incorrect	While Tech Specs do limit Containment temperatures, the concern is Containment leakage due to elevated pressure. The ice bed absorbs thermal energy to limit pressure from a Loss of Coolant accident or Steam line Break.

Comments:

REFERENCES:
UFSAR, RO-C-01000

KA - 025000-K3.01
Ice Condenser System
Knowledge of the effect that a loss or malfunction of the ice condenser system will have on the following: Containment
RO - 3.8 SRO - 3.8

KA Match
Question requires candidate to identify the correct design purpose of the Ice condenser which also determines the parameter effected (higher pressure) due to a loss.

Cognitive Level: F 3

Question: Bank- Modified CM-2042 to fix distracters B & C, which previously listed SGTR and Loss of AC as the events.
Previous Use:
Original Question:

Associated objective(s):

(RO-C-01000-E1) Explain the purpose(s) and/or function(s) of the Ice Condenser System.

36 **ID: 2008NRC-0379** **Points: 1.00**

A reactor trip with a safety injection occurred due to a feed line break on SG#22. The crew is performing actions of 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

Steam Generator Aux Feedwater Flows were indicating as follows:

	SG21	SG22	SG23	SG24
Flow Instrument	FFI-210	FFI-220	FFI-230	FFI-240
Flow	200 KPH	Pegged High	200 KPH	200 KPH

Ann. 214, Drop 9, TDAFP DISCHARGE FLOW HIGH, has just alarmed.

Which ONE of the following responses is correct given these conditions?

A. Do NOT trip TDAFP. Manually throttle AFW Flow to ALL 4 SGs until the Discharge Flow High Alarm clears. This alarm indicates that Aux Feed Flow Retention has FAILED TO ACTUATE.

B. Do NOT trip TDAFP. Verify that AFW Flow to ALL 4 SGs has automatically throttled as expected for Aux Feed Flow Retention. This alarm indicates that Aux Feed Flow Retention has ACTUATED.

C. Trip TDAFP. This alarm indicates that the feed line break is on the Aux Feed Line.

D. Trip TDAFP. This alarm indicates that Aux Feed Flow Retention has FAILED TO ACTUATE.

Answer: B

Answer Explanation:

A. Incorrect The Alarm Indicates that Flow Retention has Actuated and will Automatically Throttle the valves.

B. Correct Upon High AFP flow to a SG (>975 gpm) the flow retention circuit will throttle the AFP valves closed to prevent pump run out. This is an expected alarm given these conditions. The pump should continue to operate after verifying that flow retention is properly operating.

C. Incorrect This alarm is expected for this condition. The AFP should NOT be stopped.

D. Incorrect The alarm indicates that flow retention has actuated. The AFP should NOT be stopped.

Comments:

References:

2-OHP-4024.214 Drop 9, RO-C-05600, Auxiliary Feedwater System

KA - 061000 K4.04
Auxiliary / Emergency Feedwater (AFW) System
Knowledge of AFW System design feature(s) and/or interlock(s) which provide for the following:
Prevention of AFW run out by limiting AFW flow
RO - 3.1 SRO - 3.4
CFR - 41.7
SF4.061.K4.04

KA Match:

Question matches KA as candidate must recognize system design features to prevent pump runout and the indication in the control to identify the feature is active.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2008
Original Question:

Associated objective(s):

(RO-C-05600-E12) Discuss the purpose and operation of the following AFW operating conditions.
a. Flow retention.
b. Flow conservation.

37	ID: RO-C-00202-E16-3	Points: 1.00

Given the following conditions:

- A malfunction has caused a Reactor Trip on Unit 1.
- Pressurizer PRESS CTRL SELECTOR switch is in the Channel 1-2 position.
- 1-NPP-152, Pressurizer Pressure Channel 2, is indicating 2500 psig.
- ACTUAL Reactor Coolant System (RCS) pressure is 1930 psig and lowering.
- Pressure Spray Valves 1-NRV-163 and 1-NRV-164 are closed in AUTO

Pressurizer status is as follows;

- Pressurizer Safety Valves are CLOSED.
- 1-NRV-151, PRZR PORV, is OPEN.
- 1-NRV-152, PRZR PORV, is CLOSED.
- 1-NRV-153, PRZR PORV, is OPEN.
- Pressurizer level is 80 and rising%.

Which of the following ADDITIONAL channels has experienced a failure to create these conditions?

A. 1-NPS-121, Wide Range RCS Pressure Channel 1

B. 1-NPP-151, Pressurizer Pressure Channel 1

C. 1-NPP-153, Pressurizer Pressure Channel 3

D. 1-NPS-153, Pressurizer Pressure Channel 4

Answer: C

Answer Explanation:

A. Incorrect — NRV-151 and NRV 153 will NOT open unless the bistable from Channel 3 NPP-153 also indicates a high condition. Plausible since NPS-121 also feeds NRV-151 for LTOP, but it is not used unless the PORV switches are in LTOP.

B. Incorrect — NRV-151 and NRV 153 will NOT open unless the bistable from Channel 3 NPP-153 also indicates a high condition. Plausible since NPP-151 is the control channel and feeds NRV-152.

C. Correct — NRV-151 and NRV 153 will open only if the bistable from Channel 3 NPP-153 indicates a high condition along with the failure of NPP-152 Channel 2.

D. Incorrect — A failure of NPP-151 will cause the PZR master controller to fail high causing the spray valves to open and sending an open signal to NRV-152. NRV-152 will open from the bistable from Channel 4 NPS-153 also indicates a high condition.

Comments:

References:

SOD-00202-001 & SOD-202-002, RO-C-00202

KA - 010000 K4.03

Pressurizer Pressure Control System (PZR PCS)

Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Over pressure control

RO - 3.8 SRO - 4.1

CFR - 41.7

SF3.010.K4.03

KA Match:

Question matches KA as candidate is required to demonstrate knowledge on PZR PORV actuation signals to provide over pressure control.

Cognitive Level: H 3

Question NEW

Previous Use: New

Original Question:

Associated objective(s):

(RO-C-00202-E16) Given the following Pressurizer and Pressure Relief and Control System components, describe the conditions that will cause the component to trip, automatically/manually start and/or automatically/manually reposition.

a. Pressurizer Spray Valves
b. Pressurizer PORVs
c. Pressurizer Heaters
d. Pressurizer Safeties
e. Charging Flow Control Valve QRV-251

38	ID: AOP0550412-E1-10	Points: 1.00

Which ONE of the following describes the effect on a closed Circulating Water pump breaker if DC control power is lost to the breaker?

A. The breaker immediately trips open and cannot be reclosed until control power is restored.

B. The breaker can be tripped from the Control Room but automatic trip functions are NOT operable.

C. Automatic trips are NOT operable and tripping the breaker from the Control Room is NOT possible.

D. Automatic trips are operable but tripping the breaker from the Control Room is NOT possible.

Answer: C

Answer Explanation:

A. Incorrect	The breaker has stored energy in the spring, but it cannot be released due to the loss of power. Plausible since many signals (RPS) require power to maintain contacts open.
B. Incorrect	The breaker has stored energy in the spring, but it cannot be released due to the loss of power. Plausible since many signals(RPS) require power to maintain contacts open and generate trip on loss of power.
C. Correct	A loss of DC control power will prevent breaker operations with the control switch (and trip functions)
D. Incorrect	While it is true that the spring has stored energy, the spring release mechanism cannot release the spring to cause the trip.

Comments:

References:
OP-1-98401-16

KA - 063000 K4.04
D.C. Electrical Distribution System
Knowledge of D.C. Electrical System design feature(s) and/or interlock(s) which provide for the following:
Trips
RO - 2.6 SRO - 2.9
CFR - 41.7
SF6.063.K4.04

KA Match:
Question matches KA as it requires candidate to demonstrate knowledge of the design features provided by DC power for generating a Trip signal.

Cognitive Level: H 2

Question BANK
Previous Use: NRC 2010
Original Question:

Associated objective(s):

(RO-C-08204-E1) Explain the purpose(s) and/or function(s) of the 250 Volt DC System.

39 **ID: RO26-0066** **Points: 1.00**

The filter on the Iodine Channel (ERS-1303) for the Lower Containment Monitor was changed yesterday.

It is identified that the indication for channel ERS-1303 has shown a slowly rising trend since the change. (Indication is still below the Alert setpoint)

Which ONE of the following describes the operational implications of this condition?

A. This trend is expected as the filter approaches the current containment Iodine levels. The trend will rise at a faster rate when performing a Containment Pressure Relief.

B. This trend is expected as the filter approaches the current containment Iodine levels. The trend will rise at a slower rate when performing a Containment Pressure Relief.

C. This trend is NOT expected. Enter 1-OHP-4022-002-020, Excessive RCS leakage to address the cause of the rise.

D. This trend is NOT expected. Declare the Channel INOPERABLE, and verify that ERS-1400 Lower Containment Monitor, Channel 1403 Iodine is OPERABLE.

Answer: A

Answer Explanation:

A. Correct	Following a filter change the indication will gradually rise as the filter becomes saturated (approaching current Containment Iodine concentrations). When performing a purge the rate of rise will be higher.
B. Incorrect	When performing a purge the rate of rise will be higher.
C Incorrect	Following a filter change the indication will gradually rise as the filter becomes saturated (approaching current Containment Iodine concentrations).
D. Incorrect	Following a filter change the indication will gradually rise as the filter becomes saturated (approaching current Containment Iodine concentrations).

Comments:

References:

RO-C-01350 Radiation Monitoring System

KA - 073000 K5.01
Process Radiation Monitoring (PRM) System
Knowledge of the operational implications of the following concepts as they apply to the PRM System:
Radiation theory, including sources, types, units, and effects
RO - 2.5 SRO - 3.0
CFR - 41.5 / 45.7
SF7.073.K5.01

KA Match:

Question matches KA as it tests the expected response of the iodine process radiation monitor. This includes the theory of operation including types of radiation monitored.

Cognitive Level: H 3

Question BANK
Previous Use: RO26 AUDIT
Original Question:

Associated objective(s):

(RO-C-01350-E7) List the parameters used to verify proper operation of the Radiation Monitoring System.

40	ID: RO27AUDIT-30	Points: 1.00

Given the following conditions on Unit 2:

- A dilution to critical is being performed.
- Initial counts are 10 cps on both Source Range Channels N31 and N32.
- The first batch of primary water results in count rate changing from 10 cps to 20 cps.

- Assume the following:
  - Boron worth of 10 pcm / ppm
  - Initial K_{eff} is 0.98
  - Initial Boron Concentration is 2000 ppm

If the count rate is doubled again, the volume of primary water required for the doubling will be _____(1)_____ than the first batch, and as a result the reactor will then be _______(2)_______.

	(1)	(2)
A.	greater	subcritical
B.	less	subcritical
C.	greater	critical
D.	less	critical

Answer: B

Answer Explanation:

A. Incorrect — See proof below. Plausible if the applicant does not properly recall the relationship between count rate and the change in Keff.

B. Correct — See proof below.

C. Incorrect — See proof below. Plausible, since the applicant may confuse the concept of count rate doubling, where after a doubling occurs, if the SAME AMOUNT OF REACTIVITY IS ADDED AGAIN, THE REACTOR WILL BE CRITICAL.

D. Incorrect — See proof below. Plausible, since the applicant may confuse the concept of count rate doubling, where after a doubling occurs, if the SAME AMOUNT OF REACTIVITY IS ADDED AGAIN, THE REACTOR WILL BE CRITICAL.

Proof;

Assume initial Keff is 0.98.
Using equations ((CR1) (1-Keff 1)) = ((CR2)(1-Keff2)) and r = Keff-1/Kef
((CR1) (1-Keff 1)) = ((CR2)(1-Keff2))
Keff2 = 1- ((10)(1-0.98))/20 = 0.99
After the first batch of primary water was added, Keff was increased to 0.99.
Change in reactivity from 0.98 to 0.99 was +0.010307 DK/K
Assessing the second batch:
((CR1) (1-Keff 1)) = ((CR2)(1-Keff2))
Keff2 = 1- ((20)(1-0.99))/40 = 0.995
After the second batch of primary water was added, Keff was increased to 0.995
Change in reactivity from 0.99 to 0.995 was +0.005076 DK/K
Assume initial boron concentration of 1900 ppm. To change from 0.98 to 0.99 requires +0.010307 DK/K, which equates to 1030 pcm. Using a boron worth thumb rule of 10 pcm/ppm, this is a concentration change of 103 ppm. To dilute from 1900 ppm to 1800 ppm (rounded for illustrative purposes) requires 3400 gallons of primary water.
The second change, from 0.99 to 0.995 requires +0.005076DK/K, which equates to 508 pcm. Using a boron worth thumb rule of 10 pcm/ppm, this is a concentration change of 50 ppm. To dilute from 1800 ppm to 1750 ppm requires approx. 1750 gallons of primary water.
The amount of reactivity to achieve the second doubling was less than the amount of reactivity to achieve the first doubling, and since Keff2 is 0.995, the reactor remains subcritical.
The amount of primary water to dilute to accomplish the second doubling was less than the amount of primary water to accomplish the first doubling.

Comments:

References:
RO-C-GF03 pg. 27-33, explanation from question

KA - 004000 K5.07
Chemical and Volume Control System (CVCS)
Knowledge of the operational implications of the following concepts as they apply to the CVCS:
Relationship between SUR and reactivity
RO - 2.8 SRO - 3.2
CFR - 41.5 / 45.7
SF1.004.K5.07

KA Match:
Question matches K/A as it asks for the operational implications of the concept. This is addressed in the question by, 1.) Having the applicant evaluate how much dilution is needed, and 2.) What is the effect on the reactor.

Cognitive Level: H 3

Question BANK
Previous Use: RO27 AUDIT
Original Question:

Associated objective(s):

(RO-C-GF04-E11) Describe the effects of boration/dilution on reactivity.

41	ID: NRCAUDIT07-0716	Points: 1.00

The following conditions exist:

- Containment pressure instrument Channel #1, 2-PPP-303 (PT-937) declared inoperable.
- Required actions per 2-OHP-4022-013-011 Containment Instrumentation Malfunction have been completed.
- Required Technical Specification Actions have been taken for Channel #1, 2-PPP-303 (PT-937)

Which ONE of the following describes the REMAINING coincidence for the SAFETY INJECTION ACTUATION and CTS ACTUATION?

Remaining Channels to cause actuation / Remaining Channels with INPUT to this function

	SAFETY INJECTION ACTUATION	CTS ACTUATION
A.	2/3	2/3
B.	1/3	2/3
C.	1/2	1/3
D.	2/3	1/3

Answer: A

Answer Explanation:

A. Correct The CTS Actuation Bistable is placed in the BYPASSED condition to prevent inadvertent actuation. This changes the remaining channel coincidence to 2/3 instead of the previous 2/4. Only 3 channels (Channels 2, 3, & 4) feed the SI Actuation This channel does NOT feed SI so the SI coincidence remains at 2/3.

B. Incorrect This channel does NOT feed SI. (True if candidate assumes 4 channels feed SI)

C. Incorrect This channel does NOT feed SI Actuation. CTS is bypassed. (True if candidate assumes this channel does feed SI and that CTS is tripped)

D. Incorrect The CTS is placed in BYPASS. (True if candidate knows this channel does NOT feed SI but assumes CTS is tripped)

Comments:

References:

02-OHP-4022-013-011 Containment Instrumentation Malfunction pg. 2 and Att. D

KA - 013000 K6.01
Engineered Safety Features Actuation System (ESFAS)
Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS:
Sensors and detectors
RO - 2.7 SRO - 3.1
CFR - 41.7 / 45.7
SF2.013.K6.01

KA Match:

Question matches KA as candidate must demonstrate knowledge on containment pressure channels and effects a loss of a channel will have on ESFAS.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2006
Original Question:

Associated objective(s):

(RO-C-01100-E6) For any RPS/ESFAS/AMSAC actuation circuit:

a. State the total number and type of process sensor inputs.
b. State the logic/coincidence of signals needed to actuate the protection system.
c. State the nominal setpoint.
d. Describe any blocking signals/permissive/conditionals.
e. Describe any associated status lights.
f. Describe any associated alarms.
g. Explain what plant actions occur in direct response to the actuation signal.
h. Explain how the sensor is "tripped" or "bypassed" to support T.S. action statement or surveillance requirements.
i. Explain when "bypass" is allowed to be used per T.S. 3.3.1-1 and 3.3.2-2.

42 ID: NRCAUDIT07-0803 Points: 1.00

Given the following plant conditions:

- The Pressurizer Level is 58%
- The plant is being cooled down in preparation for refueling
- Reactor Coolant System (RCS) temperature is 175°F
- RCS pressure is 285 psig
- Charging flow control is in manual
- The East Residual Heat Removal (RHR) train is in the Shutdown Cooling Mode
- The East RHR heat exchanger suddenly develops a 50 gpm tube leak

Based on these conditions and assuming no operator action is taken, what will be the result of this event?

A. Pressurizer Level rises and the RCS cooldown rate rises.

B. Pressurizer Level lowers and the RCS cooldown rate lowers.

C. CCW surge tank level will rise, until overflowing to the Waste Gas Header.

D. CCW surge tank level will lower, until the CCW pumps trip, resulting in a loss of shutdown cooling.

Answer: B

Answer Explanation:

A. Incorrect CCW is at lower pressure than RHR, therefore all parameters are exactly opposite of what would happen. Plausible if candidate assumes CCW pressure is higher.

B. Correct RHR system pressure is higher than CCW system pressure, causing an RCS leak into CCW. With Hot RCS mixing with the CCW the RCS cooldown will not be as effective. With charging in manual a 50 gpm loss from the RCS will result in lowering RCS level

C. Incorrect Plausible because although CCW head tank rises and overflows, the overflow/ drain funnel goes to the AB Drain system versus the waste gas system. The tank is vented to the AB exhaust through the vent/overflow valve CRV-412. CRV-412 may auto close on High radiation. The tank will then be protected by the safety valve which relieves to AB Drain system.

D. Incorrect RHR system pressure is higher than CCW system pressure which is around 80 to 90 psig, therefore leakage will be into the CCW system and head tank level will rise. Plausible if candidate assumes CCW pressure is higher.

Comments:

References:

1-OHP-4022-016-003 CCW In-Leakage Attachment D East RHR Heat Exchanger

KA - 005000 K6.03
Residual Heat Removal System (RHRS)
Knowledge of the effect of a loss or malfunction of the following will have on the RHRS:
RHR heat exchanger
RO - 2.5 SRO - 2.6
CFR - 41.7 / 45.7
SF4.005.K6.03

KA Match:
Question matches KA as candidate must evaluate the effects of a malfunction in the RHR heat exchanger will have on the plant.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2007, RO25 AUDIT
Original Question:

Associated objective(s):

(RO-C-AOP0580412-E1) identify the event and predict the response of the plant to CCW System In-Leakage, including final plant configuration.

43	ID: AOP0450412-E1-8	Points: 1.00

Given the following plant conditions:

- Unit 1 is at 700 MW on the Main Turbine.
- All control systems are in AUTOMATIC.

Which ONE of the following describes the plant response to a trip of the East Main Feed Pump?

As SG water levels start lowering, the Feedwater Regulating Valves open further,

A. the West MFP speed rises and the Main Turbine runs back. The Standby Hotwell Pump and Condensate Booster Pump start due to low Condensate System pressure.

B. the West MFP speed rises but will not maintain SG level. Both the Steam-driven and Motor-driven Auxiliary Feedwater Pumps start when SG levels reach the Low-Low level setpoint.

C. the feedwater header pressure lowers, causing the West MFP speed to rise until it trips on overspeed. Both the Steam-driven and Motor-driven Auxiliary Feedwater Pumps start when SG levels reach the Low-Low level setpoint.

D. the feedwater header pressure starts to lower. The West MFP Speed rises and the Main Turbine runs back. NO pumps auto start.

Answer: D

Answer Explanation:

A. Incorrect The Hotwell and Condensate Booster pumps will not start as Condensate system pressure will not significantly decrease since the total amount of FW flow required does not change.

B. Incorrect The Main FW pumps can supply 60% flow and so a Low-Low level would not be reached and AFW will not start.

C. Incorrect The Main FW pumps can supply 60% flow and so the West FW pump would not trip on overspeed.

D. Correct On the loss of the East Main FW Pump, reduced flow will cause the FW regulating valves open further as the SGs try to maintain normal level & FW flow matched to steam flow. FW Pump Discharge pressure will decrease and the FW pp vs. Steam pressure Delta P will cause the West FW pump speed to increase to restore programmed Delta Pressure. The Main Turbine will runback to 626MW (60%), 700 MW would be ~ 67% power.

Comments:

References:

1-OHP-4022-055-001 Loss of One Main Feed Pump, RO-C-AOP-D14

KA - 059000 A1.03
Main Feedwater (MFW) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW System controls including:
Power level restrictions for operation of MFW pumps and valves
RO - 2.7 SRO - 2.9
CFR - 41.5 / 45.5
SF4.059.A1.03

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of the MFW design operation based on power levels and response of the system pumps and valves.

Cognitive Level: H 2

Question MODIFIED
Previous Use: NRC 2002 Modified
Original Question:

Associated objective(s):

(RO-C-AOP0450412-E1) identify the event and predict the response of the plant to a Loss of one MFW Pump, including final plant configuration.

44 ID: RO-C-01100-E6-31 Points: 1.00

Given the following conditions:

- Unit 2 is at 80% Power.
- 2-NPS-153 Pressurizer Pressure Channel 4 fails low.

Which of the following describes the impact this failure will have on the Over Power (OPDT) and Over Temperature (OTDT) Delta Temperature Trip set points, if any?

A. The OPDT setpoint AND the OTDT setpoint will lower.

B. The OTDT setpoint will lower while the OPDT setpoint is unaffected.

C. The OPDT setpoint will lower while the OTDT setpoint is unaffected.

D. No impact, Pressurizer Channel 4 does NOT feed the OPDT nor OTDT Circuits.

Answer: B

Answer Explanation:

A. Incorrect Channel 4 feeds loop 4 OTDT circuit. The OPDT circuit does NOT receive a Pressure Input.

B. Correct Channel 4 feeds loop 4 OTDT circuit. The OPDT circuit does NOT receive a Pressure Input.

C. Incorrect Channel 4 feeds loop 4 OTDT circuit. The OPDT circuit does NOT receive a Pressure Input.

D. Incorrect Channel 4 feeds loop 4 OTDT circuit. Plausible since channel 4 is not used for Pressurizer Low Pressure SI.

Comments:

References:

OP-2-99034 (K1), OHP-4022-013-009 Pressurize Pressure Instrument Malfunction, Attachment D, TS 3.3.1 Table 3.3.1-1

KA - 012000 A1.01
Reactor Protection System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including:
Trip setpoint adjustment
RO - 2.9 SRO - 3.4
CFR - 41.5 / 45.5
SF7.012.A1.01

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of inputs to Reactor Trip system setpoint calculation.

Cognitive Level: H 3

Question NEW
Previous Use: NEW
Original Question:

Associated objective(s):

(RO-C-01100-E6) For any RPS/ESFAS/AMSAC actuation circuit:

a. State the total number and type of process sensor inputs.
b. State the logic/coincidence of signals needed to actuate the protection system.
c. State the nominal setpoint.
d. Describe any blocking signals/permissive/conditionals.
e. Describe any associated status lights.
f. Describe any associated alarms.
g. Explain what plant actions occur in direct response to the actuation signal.
h. Explain how the sensor is "tripped" or "bypassed" to support T.S. action statement or surveillance requirements.
i. Explain when "bypass" is allowed to be used per T.S. 3.3.1-1 and 3.3.2-2.

45 ID: NRC2010-62 Points: 1.00

Given the following plant conditions:

- -Unit 2 is in MODE 4 at 280^0 F.
- -The Unit 2 East CCW HX is in service with 5000 GPM of ESW Flow.
- -The Unit 2 EAST RHR train in service with 5000 GPM of CCW Flow.
- -All four ESW pumps are in service with crossties open.
- -The DG2CD is running at a constant 3200 KW for a Surveillance test.

Which ONE of the following describes the impact on the listed parameters if the Unit 2 East ESW pump trips?

NOTE: Assume no operator action.

	U2 East CCW Heat Exchanger CCW Outlet Temperature	DG2CD Jacket Water (JW) Heat Exchanger JW Outlet Temperature
A.	RISES	RELATIVELY CONSTANT
B.	RISES	RISES
C.	RELATIVELY CONSTANT	RELATIVELY CONSTANT
D.	RELATIVELY CONSTANT	RISES

Answer: A

Answer Explanation:

A. Correct ESW Flow through the CCW HX is maintained by manually throttling ESW through the CCW HX. When the East ESW pump Trips this will lower flow through the HX (since the flow is manually controlled). The Flow through the DG is maintained constant by the automatic temperature control valve.

B. Incorrect CCW HX temperature will rise due to lower flow but DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student is not aware of DG temperature control valve.

C. Incorrect CCW HX temperature will rise due to lower flow while DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student may think that ESW through CCW is temperature controlled and/or student may assume that the West ESW would supply flow (ESW crossties are between units not trains).

D. Incorrect CCW HX temperature will rise due to lower flow and DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student may think that ESW through CCW is temperature controlled and/or student may assume that the West ESW would supply flow (ESW crossties are between units not trains).

Comments:

References:
RO-C-01900 Essential Service Water system, RO-C-03201 Diesel Generator Auxiliaries, SOD-01900-001

KA - 076000 A1.02
Service Water System (SWS)
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.
RO - 2.6 SRO - 2.6
CFR - 41.5 / 45.5
SF4.076.A1.02

KA Match:
Question matches KA as it tests ability to predict/monitor CCW & DG JW parameters/temperatures associated with reduction in ESW (SWS) cooling.

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2010
Original Question:

Associated objective(s):

(RO-C-01900-E11) Given the following Essential Service Water System components, describe the conditions that will cause the component to trip, automatically/manually start and/or automatically/manually reposition.

a. ESW pumps
b. ESW pump discharge valves
c. ESW pump discharge strainers
d. Component Cooling Water Heat Exchangers
e. Containment Spray Heat Exchangers
f. Diesel Generator Cooling (Normal and Alternate supply)

46 ID: 0020920412-E3-9 Points: 1.00

Given the following conditions on Unit 2:

- Unit 2 was operating at 100% power when the turbine tripped.
- The reactor failed to automatically trip but was manually tripped.
- All other systems operated as expected.
- The Emergency procedures have been performed and the plant stabilized.
- A pressurizer Safety valve momentarily opened and has begun leaking.

Which ONE of the following represents the expected actions that must be taken to restore PRT Pressure to normal limits?

The PRT pressure may be reduced by venting to:

A. the Waste Gas header as long as pressure is less than 20 psig.

B. the Waste Gas header as long as pressure is less than 10 psig.

C. the RCDT as long as pressure is less than 10 psig.

D. the RCDT as long as pressure is less than 20 psig.

Answer: B

Answer Explanation:

A. Incorrect The Waste Gas header vent from the PRT RRV-103 will auto isolate at 10 psig. Plausible since The PRT and RCDT share a connection to the Waste gas system prior to the containment penetration, but this is downstream of RRV-103.

B. Correct The Waste Gas header vent from the PRT RRV-103 will auto isolate at 10 psig.

C. Incorrect The Waste Gas header vent from the PRT RRV-103 will auto isolate at 10 psig. Plausible since The PRT and RCDT share a connection to the Waste gas system prior to the containment penetration, but this is downstream of RRV-103.

D. Incorrect The Waste Gas header vent from the PRT RRV-103 will auto isolate at 10 psig. Plausible since The PRT and RCDT share a connection to the Waste gas system prior to the containment penetration, but this is downstream of RRV-103.

Comments:

References:

2-5128A, 12-5137A, OHP-4024-208 Drop 31

KA - 007000 A2.04
Pressurizer Relief Tank/Quench Tank System (PRTS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Over pressurization of the waste gas vent header
RO - 2.5 SRO - 2.9
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF5.007.A2.04

KA Match:

Question matches KA as it requires the candidate to determine the impact to the PRT of high pressure on the Waste Gas Header (Tank pressure to Waste Gas Header and interlock to prevent Over pressurization of Waste Gas Header) and the required actions to address this situation.

Cognitive Level: H 3

Question NEW
Previous Use: NEW
Original Question:

Associated objective(s):

(RO-C-0020920412-E3) explain the procedural mitigation strategy for a Leaking Pressurizer Power Operated Relief Valve

47 **ID: AOP0420412-E2-4** **Points: 1.00**

Unit 2 Reactor Startup is in progress with Reactor Power at 3% and rising.

The following conditions exist:

- All RCP Motor Bearing Lube Oil CCW flow low annunciators are lit.
- All flows and temperatures are STABLE as follows:

	RCP #21	RCP #22	RCP #23	RCP #24
• Lower Bearing temps	186°F	205°F	188°F	203°F
• Upper Motor Bearing temps	189°F	190°F	185°F	215°F
• Seal Leakoff temps	189°F	180°F	175°F	175°F
• Seal Injection Flow	7 gpm	7 gpm	8 gpm	10 gpm

Which ONE of the following operator actions MUST be taken based upon these conditions?

A. Manually trip the reactor, Enter 2-OHP-4023-E-0, Reactor Trip or Safety Injection, perform immediate actions, and trip the #24 RCP.

B. Manually trip the reactor, Enter 2-OHP-4023-E-0, Reactor Trip or Safety Injection, perform immediate actions, and trip the #22 AND #24 RCP.

C. Do NOT trip the reactor. Trip the #22 RCP and initiate a Shutdown to be in Hot Shutdown in 1 hour.

D. Do NOT trip the reactor. Trip the #24 RCP and close the No. 1 Seal Leak off valve.

Answer: A

Answer Explanation:

A. Correct Upper motor bearing water temperature is > 205°F which requires a trip. The reactor must first be tripped and verified and then the RCP is tripped.

B Incorrect - Plausible since a normal shutdown followed by pump shutdown is required for some RCP failure conditions. #24 RCP Upper bearing temperature has exceeded the trip setpoint.

C Incorrect - Plausible since the plant will not auto trip except, the Plant is not analyzed / licensed to operate with less than 4 RCPs. Some Tech Specs require Hot Standby in 1 hour.

D Incorrect - Plausible since the plant will not auto trip except, the Plant is not analyzed / licensed to operate with less than 4 RCPs. The #21 RCP has high seal leakoff temps but they are < 200 °F with adequate Seal Injection (Trip is required for >185 °F AND loss of seal injection)

Comments:

References:

2-OHP-4022-002-001, Malfunction of a Reactor Coolant Pump (Step 1)

KA - 008000 A2.07

Component Cooling Water System (CCWS)

Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of high or low CCW flow rate and temperature; the flow rate at which the CCW standby pump will start

RO - 2.5 SRO - 2.8

CFR - 41.5 / 43.5 / 45.3 / 45.13

SF8.008.A2.07

KA Match:

Question matches KA as it requires the candidate to determine the consequences of the low CCW flow and the required actions to address this situation.

Cognitive Level: H 3

Question NEW

Previous Use: NEW

Original Question:

Associated objective(s):

(RO-C-AOP0420412-E2) Explain the required operator actions to stabilize plant conditions after Malfunctions of the CCW System prior to formal procedure implementation in accordance with plant procedures, and standards and expectations for performance.

48 ID: RO26-0042 Points: 1.00

Given the following plant conditions in Unit 2:

- The reactor is critical at 11% power just prior to rolling the turbine.
- Manual FW control is maintaining SG Levels.
- RCS temperature is being controlled by steam dumps in automatic in the Steam Pressure Mode.

2-UPC-101 Steam Header Pressure transmitter then fails to 1200 psig.

What will be the initial response of the steam generator levels and the subsequent required actions?

A. Lower due to shrink and then rise due to lower steam release rate. Place Steam Dump Selectors to OFF.

B. Lower due to shrink and then rise due to higher FW flow at the lower pressure. Place Steam Dump in T_{avg} mode of operation.

C. Rise due to swell and then lower due to the higher steam release rate. Place Steam Dump Selectors to OFF.

D. Rise due to swell and then lower due to lower FW flow at the higher pressure. Place Steam Dump in T_{avg} mode of operation.

Answer: C

Answer Explanation:

A. Incorrect This failure will cause steam dumps to open. The SG will swell due to lower SG pressure.

B. Incorrect The SG will swell due to lower SG pressure.

C. Correct The Steam Dumps will be trying to maintain Steam Line pressure. When the indicated pressure fails high the dumps will open, causing SG pressure to be lower. This will cause an initial swell followed by level lowering as the RCS temperature lowers.

D. Incorrect The SG will swell from the lower SG pressure followed by level lowering as the RCS temperature lowers.

Comments:

References:
RO-C-GF21, RO-C-05100 Steam Generating system

KA - 039000 A2.04
Main and Reheat Steam System (MRSS)
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: Malfunctioning steam dump
RO - 3.4 SRO - 3.7
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF4.039.A2.04

KA Match:
Question matches KA as it requires the knowledge of the relationship between SG pressure, RCS Temperature and Shrink and Swell as it relates to SG level.

Cognitive Level: H 3

Question BANK
Previous Use: Modified From NRC 2006
Original Question:

Associated objective(s):

(RO-C-GF21-E19) Describe the basic operation of a U-tube type steam generator.

49 ID: RO-C-08200-E23-1 Points: 1.00

Unit 1 has tripped due to loss of off-site power. Both emergency diesel generators started automatically to provide power to all four safeguards buses. Off-site power has since been restored and the buses 1A &1B are being supplied by TR101AB and buses 1C &1D are being supplied by TR101CD. Both emergency diesel generators have been secured. The operators are in ES-0.1, Reactor Trip Response and the following conditions exist:

- RCS Temp is 547°F
- RCS Pressure is 2200 psig
- SG Narrow Range Levels are 34%

The crew has been directed to restart RCP 13.

Which ONE of the following describes the requirements and/or consequences of starting a RCP in this electrical configuration?

A. The Load Tap Changer Setting must be raised in MANUAL prior to the RCP start. The higher voltage will cause the associated loads to draw MORE amps.

B. The Load Tap Changer Setting will rise in AUTO when the RCP starts since starting the RCP will cause Bus Voltage to Lower. The lower voltage will cause the associated loads to draw LESS amps.

C. The Load Tap Changer must be placed in MANUAL prior to the RCP start to prevent the Bus Voltage from overshooting as RCP starting current lowers.

D. The Load Tap Changer must be left in AUTO prior to the RCP start to maintain the required Bus Voltage as RCP starting current rises and then lowers.

Answer: D

Answer Explanation:

A. Incorrect The Load Tap Changer will try to maintain bus voltage as the starting current of the RCP lowers Bus Voltage while in AUTO. Raising the Bus Voltage would make the associated loads draw LESS amps.

B. Incorrect The Load Tap Changer will try to maintain buss voltage as the starting current of the RCP lowers Bus Voltage. The Lower voltage will make the associated loads draw MORE Amps.

C. Incorrect The Load Tap Changer will try to maintain bus voltage as the starting current of the RCP lowers Bus Voltage while in AUTO it will not overshoot and placing it in manual may allow voltage to drop too low.

D. Correct The Load Tap changer is left in auto and allowed to maintain the bus voltage throughout the RCP start.

Comments:

References:

1-OHP-4023-SUP-010 Starting a Reactor Coolant Pump, 1-OHP-4024-121 Drops 68,78

KA - 062000 A3.01
A.C. Electrical Distribution System
Ability to monitor automatic operation of the A.C. Distribution System, including:
Vital ac bus amperage
RO - 3.0 SRO - 3.1
CFR - 41.7 / 45.5

SF6.062.A3.01

KA Match:

Question matches KA as candidate must demonstrate knowledge on monitoring of bus amperage for RCP start and operation of the Load Tap Changer to maintain bus amperage (voltage) during pump start.

Cognitive Level: H 3

Question NEW
Previous Use: NEW
Original Question:

Associated objective(s):

(RO-C-08200-E23) Describe the consequences of exceeding either bus/equipment voltage limits or current limits.

50 **ID: NRCAUDIT07-0128** **Points: 1.00**

Given the following plant conditions:

- Unit 1 is at 100% power and stable.
- Steam Generator Level Controls are in AUTOMATIC.
- Steam Generator #12 Steam Flow Channel 1, 1-MFC-121, is selected to the Steam Generator Level Control System.

An unidentified calibration error results in Steam Generator #12 Steam Flow Channel 2, 1-MFC-120, indicating 10% low (indicates 90% vs 100% Steam Flow).

When requested by MTI, operators switch the controlling Steam Flow channel to 1-MFC-120.

Assuming all controllers remain in Automatic, when the operator switches the controlling channel the Steam Generator (SG) Level Control System will:

A. initially lower feed flow, then control #12 SG level approximately 10% below program level.

B. not change feed flow to the #12 SG, but Feedwater Delta-P Program will be lowered approximately 10%.

C. initially raise feed flow to #12 SG, then return level to program level.

D. initially lower feed flow to #12 SG, then raise flow to restore level to approximately program level.

Answer: D

Answer Explanation:

A. Incorrect SG level setpoint is fixed and is not impacted by the SF channel. Plausible if student does not understand that SGWLC is level dominant.

B. Incorrect SG FW flow will be affected & FW DP program will only be affected ~2.5% power since all 4 channels are summed. Plausible if student believes that a level dominant SGWLC will keep FF constant regardless of steam flow for a constant level.

C. Incorrect FW flow will not raise and will, in fact, lower. Plausible if student confuses the SF channel relationship in the example.

D. Correct SG FW flow will initially lower to match the lower Steam Flow, as a level deviation error builds in it will raise FW flow to restore level to the desired SG Level Setpoint. (Level Error is dominant to Steam Flow).

Comments:

References:
RO-C-05100 Steam Generating System

KA - 059000 A3.02
Main Feedwater (MFW) System
Ability to monitor automatic operation of the MFW System, including: Programmed levels of the S/G
RO - 2.9 SRO - 3.1
CFR - 41.7 / 45.5

SF4.059.A3.02
KA Match:
Question matches KA as candidate is required to predict response of the Main Feedwater system to maintain programmed level in the S/G when given a situation effecting the control system.

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-05100-E6) Describe the flowpath through the following FRV Flow Control components:

a. SG NR Level Transmitter
b. Level Controller
c. Steam Generator Steam Flow Transmitter
d. Steam Generator Steam Pressure Transmitter
e. Flow Controller
f. FW Flow Transmitter

51 ID: NRCAUDIT07-0807 Points: 1.00

Given the following:

- Unit 2 is in Mode 4 with the West Component Cooling Water (CCW) pump operating.
- Repairs were just completed on the West Residual Heat Removal (RHR) heat exchanger.
- CCW had previously been isolated and drained on the West RHR heat exchanger.
- The Crew restoring the clearance on the West RHR heat exchanger was aware that the CCW side had previously been drained. The crew attempted to fill and vent the CCW side of the heat exchanger.
- When the CCW Inlet to the RHR heat exchanger is opened, CCW system pressure and surge tank level lowers causing the following alarms to annunciate on Panel 204:
  - Drop 88 - West CCW Surge Tank LVL HI OR LOW
  - Drop 94 - West CCW pump Discharge Pressure Low
  - Drop 98 - East CCW Surge Tank LVL HI OR LOW
- Lowest CCW header pressure observed was 79 psig

Which ONE of the following is the expected status of the CCW pumps and Make-up valves? Assume NO operator actions.

	East Pump	West Pump	CRV 410	CRV 411
A.	Running	Stopped	Closed	Closed
B.	Stopped	Stopped	Open	Open
C.	Running	Running	Closed	Closed
D.	Running	Running	Open	Open

Answer: C

Answer Explanation:

Drop 88 - West CCW Surge Tank LVL HI OR LOW------657' 0"
Drop 89 - East CCW Pump Low Pressure Start-up--------80 psig
Drop 94 - West CCW pump Discharge Pressure Low ----85 psig
Drop 98 - East CCW Surge Tank LVL HI OR LOW ------657' 0"

All Distractors contain combinations that are plausible if the candidate does not understand the pump auto starts, trips and surge tank makeup system.

A. Incorrect The West CCW pump would not have tripped.

B. Incorrect The West CCW pump would not have tripped and the makeup valves must be manually opened.
C. Correct The Standby CCW pump will start when pressure is lowered to < 80 psig. CCW makeup must be aligned manually. The Pumps will not trip on Low Surge tank level.
D. Incorrect The East CCW pump would have started on Low pressure and the makeup valve must be manually opened.

Comments:

References:
RO-C-01600 Component Cooling water system

KA - 008000 A4.07
Component Cooling Water System (CCWS)
Ability to manually operate and/or monitor in the control room: Control of minimum level in the CCWS surge tank
RO - 2.9 SRO - 2.9
CFR - 41.7 / 45.5 to 45.8
SF8.008.A4.07

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of the CCW surge tank make up system and the CCW pumps auto start conditions.

Cognitive Level: H 3
Question MODIFIED
Previous Use: AUDIT RO24
Original Question:

Associated objective(s):

(RO-C-01600-E5) Given the following CCW System components, describe the conditions that will cause the component to trip, automatically/manually start, and/or automatically/manually reposition:

a. CCW Pumps
b. Remotely Operated Valves
1) CRV-412
2) CMO-419/429
3) CMO-410/420
4) CRV-445
5) CRV-485
6) CRV-470

52 **ID: CM-7599B** **Points: 1.00**

Unit 2 is in Mode 3 with the Shutdown Bank Rods withdrawn.
Which ONE of the following combinations of breaker positions would cause the rods to fall into the core?

Note: The Motor Generator INPUT Breakers are both closed.

- RTA = Reactor Trip Breaker A
- RTB = Reactor Trip Breaker B
- BYA = Reactor Trip Bypass Breaker A
- BYB = Reactor Trip Bypass Breaker B
- MGN Output = MG North output Breaker
- MGS Output = MG South output Breaker

LEGEND: X = CLOSED, O = OPEN

	RTA	RTB	BYA	BYB	MGN	MGS
A.	X	X	O	O	X	O
B.	X	O	O	X	O	O
C.	O	X	X	O	X	X
D.	X	X	X	O	O	X

Answer: B

Answer Explanation:

A. Incorrect Only one MG set breaker is open. Both are required to be open to trip reactor.
B. Correct RTA and BYB closed will provide flowpath to the rod coils. Both MG set breakers open will trip the reactor.
C. Incorrect RTB and BYA closed will provide flowpath to the rod coils.
D. Incorrect One MG Set with the RTA (or BYA) path and the RTB path will provide the required Power

Comments:

References:
SOD-01200-002, RO-C-1200 Rod Control and Rod Position Indicating system

KA - 012000 A4.07
Reactor Protection System
Ability to manually operate and/or monitor in the control room:
M/G set breakers
RO - 3.9 SRO - 3.9
CFR - 41.7 / 45.5 to 45.8
SF7.012.A4.07

KA Match:
Question matches KA as it requires ability to monitor the MG set and Reactor Trip breakers to determine when the reactor has been successfully tripped.

Cognitive Level: F 2

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-01200-E11) DESCRIBE the operation of the reactor trip breakers and bypass breakers.

53 ID: RO-C-00800-E13-3 Points: 1.00

Given the following conditions:

- Unit 1 is in MODE 3
- RCS pressure = 2235 psig
- RCS temperature = 547°F
- Accumulator #11 Level = 920 ft3
- Accumulator #11 Pressure = 642 psig
- Accumulator #11 Boron Concentration = 2405 ppm

Which ONE of the following describes the status of the #11 Accumulator?

A. All required parameters are within acceptable Tech. Spec. limits.

B. The boron concentration is below acceptable Tech. Spec. concentration.

C. The accumulator level is below acceptable Tech. Spec. level.

D. The accumulator Tech. Spec. is NOT applicable in this condition.

Answer: C

Answer Explanation:

A. Incorrect The Accumulator limits in MODES 1, 2 & 3 (with RCS pressure >1000 psig) per TS 3.5.1 are: Pressure is >/=585 psig and </=658 psig (/w Instrument Uncertainty. is >/=595 psig and </=650 psig) Level is >/=921 ft3 and </=971 ft3 (/w I Instrument Uncertainty. is >/=927.5 ft3 and </=965 ft3). Boron concentration in the accumulator is >/=2400 ppm and </= 2600 ppm. Level is Low at <921 ft3.

B. Incorrect Tech. Spec requirements are met. Boron is >2400 ppm. Plausible if boron concentration requirement is no known.

C. Correct Tech. Spec requirements are NOT met Level is Low at <921 ft3.

D. Incorrect The Accumulator limits apply in MODES 1, 2 & 3 (with RCS pressure >1000 psig). Plausible if accumulator condition of applicability in Mode 3 is not known.

Comments:

References:
TS 3.5.1

KA - 006000 2.2.37
Emergency Core Cooling System (ECCS)
Equipment Control
Ability to determine operability and/or availability of safety related equipment.
RO - 3.6 SRO - 4.6
CFR - 41.7 / 43.5 / 45.12
SF3.006.GEN

KA Match:
Question matches KA because the candidate must know that low level in the accumulator renders that part of the ECCS system INOPERABLE.

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00800-E13) Given ECCS and other plant parameters, determine if any T.S. LCO Action statements are in effect.

54 ID: RO-C-EOP09-E40-17 Points: 1.00

Given the following plant conditions:

- A LOCA has occurred.
- The crew has entered OHP-4023-E-1, Loss of Reactor or Secondary Coolant.
- The following parameters exist:
  - All SG pressures - 730 psig and trending down slowly.
  - All SG levels - being controlled at 40% NR.
  - PRZ level - off-scale high.
  - Containment Pressure - 2.9 psig .
  - RWST level- 30% and trending down slowly.
  - RCS pressure - 375 psig and trending down slowly.
  - Highest CET - 500°F.

Based on these indications, which ONE of the following procedure will the Unit Supervisor direct the crew to enter next?

A. OHP-4023-E-2, Faulted Steam Generator Isolation

B. OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization

C. OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation

D. OHP-4023-FR-Z.1, Response to High Containment Pressure

Answer: C

Answer Explanation:

A. Incorrect SG pressures are trending down because RCS temperature is trending down and AFW flows are maintaining levels.

B. Incorrect RCS Pressure not stable, and low RCS inventory (low reactor vessel level and high PRZ level indicates a large head bubble). This may be the correct direction if the plant did not need to be placed on the Recirc sump.

C. Correct RWST level is at 30%. Entry to OHP-4023-ES-1.3 on low RWST level of 30% is required. Alignment to Recirc sump begins at 20%

D. Incorrect This would be entered as an Orange path but ES-1.3 will take precedence.

Comments:

References:
12-OHP-4023-ES-1.3 Transfer to Cold Leg Recirculation, OHP-4023-E-1 FOP

KA - 005000 2.4.2
Residual Heat Removal System (RHRS)
Emergency Procedures/Plan
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
RO - 4.5 SRO - 4.6
CFR - 41.7 / 45.7 / 45.8
SF4.005.GEN

KA Match:
Question matches KA as candidate must recognize set points associated with EOP entry.

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP09-E40) For each of the E-1 Series procedures, identify the Procedure Transitions.

55 ID: NRCAUDIT07-0581 Points: 1.00

During a Loss of Coolant Accident (LOCA) the operators give a high priority to the Spray Additive Tank LO-LO Level alarm.

Which ONE of the following is the reason for this high priority?

A. The Spray Additive Tank Outlet valves close automatically, but the Eductor Supply valves must be manually closed.

B. The Spray Additive Tank Outlet valves and Eductor Supply valves may NOT have closed automatically. The resulting gas addition may bind the Containment Spray pumps.

C. The Spray Additive Tank Outlet valves and Eductor Supply valves must be closed manually following transfer to cold leg recirculation.

D. The Spray Additive Tank Outlet valves and Eductor Supply valves may NOT have closed automatically. The resulting nitrogen injection would form nitrous acid and lower the Recirculation sump pH.

Answer: B

Answer Explanation:

A. Incorrect	The Spray Additive Tank Outlet valves AND Eductor Supply valves close automatically (Lo-Lo SAT level).
B. Correct	The Spray Additive Tank Outlet valves and Eductor Supply valves close automatically (Lo-Lo SAT level) to help ensure that gas addition does not bind the CTS pumps.
C. Incorrect	The Spray Additive Tank Outlet valves and Eductor Supply valves close automatically (Lo-Lo SAT level).
D. Incorrect	The reason is to limit air binding of the CTS Pumps.

Comments:

References:
RO-C-00900 Containment Spray and Hydrogen Recombiner systems, OHP-4024-105 Drop 3

KA - 026000 2.4.18
Containment Spray System (CSS)
Emergency Procedures/Plan
Knowledge of the specific bases for EOPs.
RO - 3.3 SRO - 4.0
CFR - 41.10 / 43.1 / 45.13
SF5.026.GEN

KA Match:
Question matches the K/A because the candidate must know the importance of the alarm condition as related to bases for procedure.

Cognitive Level: H 3

Question BANK
Previous Use: RO28 AUDIT
Original Question:

Associated objective(s):

(RO-C-00900-E12) Given the following Containment Spray system and/or hydrogen removal or monitoring system components, describe the conditions that will cause the components to trip, automatically/manually start and/or automatically/manually reposition:

a. CTS pump start and trip
b. CTS pump discharge valve opening
c. SAT outlet valve opening
d. SAT outlet valve closure
e. Eductor supply valve closure
f. CTS heat exchanger ESW outlet valve opening

56	ID: RO-C-01350-E4-17	Points: 1.00

Given the following conditions on Unit 1:

- A Steam Generator tube leak is in progress.
- Gland Steam Condenser Vent (SRA-1800) Radiation Monitor is in service.
- A rapid plant shutdown is in progress.

Which of the following describes the flow path of fission products from the condenser?

A. A portion of the fission products will be monitored by SRA-1800 before being transmitted to the atmosphere (sampling flow passes through SRA-1800).

B. All of the fission products will be monitored by SRA-1800 before being transmitted to the atmosphere (entire flowpath passes through SRA-1800).

C. A portion of the fission products will be monitored by SRA-1800 before being transmitted to the Waste Gas System (sampling flow passes through SRA-1800).

D. All of the fission products will be monitored by SRA-1800 before being transmitted to the Waste Gas System (entire flowpath passes through SRA-1800).

Answer: A

Answer Explanation:

A. Correct	A sample line from the GSCV discharge header goes to the Process Monitor.
B. Incorrect	Plausible if believed that the entire discharge from the GSCV was monitored by SRA-1800, however only a portion of the discharge is monitored.
C. Incorrect	Plausible as only a portion of the fission products will be monitored; however, the GSCV discharge cannot be routed to the Waste Gas System for storage and later discharge.
D. Incorrect	Plausible if believed that the entire discharge from the GSCV was monitored by SRA-1800, and if believed that the GSCV discharge can be routed to the Waste Gas System.

Comments:

References:
OP-1-5122

KA - 055000 K1.06
Condenser Air Removal System (CARS)
Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: PRM system
RO - 2.6 SRO - 2.6
CFR - 41.2 to 41.9 / 45.7 to 45.8
SF4.055.K1.06

KA Match:
Question matches KA as it tests knowledge of the Gland steam PRM flow path connections to the CARS.

Cognitive Level: F 2

Question BANK
Previous Use: RO29 AUDIT
Original Question:

Associated objective(s):

(RO-C-01350-E4) Describe the function of the following Radiation Monitoring System Monitors including any automatic actions that occur on a high alarm:
1. VRS-1101/1201 (2101/2201) Upper Containment Area Monitor
2. ERS-1300(2300) Lower Containment Radiation Monitor Train A
3. ERS-1400(2400) Lower Containment Radiation Monitor Train B
4. VRS-1500(2500) Unit Vent Effluent Radiation Monitor
5. MRA-1600/1700(2600/2700) Steam Generator PORV Monitors
6. SRA-1800(2800) Steam Packing Exhauster (GSLO) Monitor
7. SRA-1900(2900) Steam Jet Air Ejector Vent Monitor
8. ERA-7300/8300 AUX BLDG EQUIP ROOM AREAS
9. ERS-7400/8400 CONT ROOM INCORE/AUXB AR
10. ERA-7500 AUX BLDG 673/587 Areas
11. ERA-7600 AUX BLDG 633/650 AREAS
12. RRS-1000 Liquid Waste Effluent Monitor
13. CRS-3301/3401 (4301/4401) East/West CCW Hx Outlet Monitor

57	ID: NRCAUDIT07-0717	Points: 1.00

Unit 1 has experienced a LOCA and Loss of Offsite power.

The following conditions exist:

- Emergency Diesel Generator 1AB failed to start.
- Emergency Diesel Generator 1CD has started and loaded as designed.
- Power has been restored to the Reserve Aux Transformers.
- No buses have been energized from the RATs.

The Unit Supervisor directs you to verify or restore power so a Hydrogen Recombiner may be run.

Which ONE of the following actions is required to enable the associated Hydrogen Recombiner to be operated?

A. Verify that bus T11C has energized 600V Bus 11C and MCC-1-EZC-C for (Train A) Hydrogen Recombiner Number 2.

B. Verify that bus T11C has energized 600V Bus 11C and Close the 11AC crosstie to supply power to Bus 11A and MCC-1-EZC-A for (Train B) Hydrogen Recombiner Number 1.

C. Energize RCP Bus 1B from the RAT to supply power to 600V Bus 11BMC for (Train B) Hydrogen Recombiner Number 1.

D. Energize RCP Bus 1C from the RAT to supply power to 600V Bus 11CMC for (Train A) Hydrogen Recombiner Number 2.

Answer: A

Answer Explanation:

A. Correct Hydrogen Recombiner #1 is powered form MCC-1-EZC-B and Hydrogen Recombiner #2 is powered form MCC-1-EZC-C.
B. Incorrect Hydrogen Recombiner #1 is powered form MCC-1-EZC-B
C. Incorrect Hydrogen Recombiner #1 is powered form MCC-1-EZC-B
D. Incorrect Hydrogen Recombiner #2 is powered form MCC-1-EZC-C

Comments:

References:

RO-C-00900, Containment Spray and Hydrogen Recombiner systems

KA - 028000 K2.01
Hydrogen Recombiner and Purge Control System (HRPS)
Knowledge of bus power supplies to the following: Hydrogen Recombiners
RO - 2.5 SRO - 2.8
CFR - 41.7
SF5.028.K2.01

KA Match:

Question matches KA as candidate must demonstrate knowledge of power supplies to Hydrogen Recombiners to ensure restoration power to appropriate bus.

Cognitive Level: F 2
Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00900-E9) Given the following support systems, explain how a loss of each of the support systems will affect the operation of the Containment Spray system and/or hydrogen removal and monitoring systems:

a. CCW
b. ESW
c. Nitrogen Gas
d. CTS pump and discharge valve power supplies
e. CEQ fans
f. Control Air

58 **ID: CM-8488** **Points: 1.00**

Given the following plant conditions:

- The plant is at 90% power at End of Life (EOL)
- Control Bank 'D' rods are at 207 steps
- ONE of the Control Bank 'D' rods drops, and is indicating it is fully inserted
- The reactor does NOT trip and no operator action is taken

Based on these conditions, power in the affected core quadrant (with the dropped rod) will ____(1)_____ and RCS T_{avg} will ___(2)______.

	(1)	(2)
A.	lower	lower
B.	remain the same	lower
C.	Lower	remain the same
D.	remain the same	remain the same

Answer: A

Answer Explanation:

A. Correct Quadrant power will lower and T_{avg} will lower.

B. Incorrect Radial power will tilt (lower in quadrant with rod), not remain the same. Plausible choice if student confuses axial power with radial power since the second choice (Tavg lowering) is correct.

C. Incorrect Quadrant power will lower, but T_{avg} will also lower (not remain the same). Plausible if student believes that overall power and hence T_{avg} are not impacted.

D. Incorrect Plausible if student confuses axial power with radial power and believes that overall power and hence T_{avg} are not impacted.

Comments:

References:
OHP-4022012-005 Dropped or Misaligned Rod

KA - 001000 K3.02
Control Rod Drive System
Knowledge of the effect that a loss or malfunction of the CRDS will have on the following:
RCS
RO - 3.4 SRO - 3.5
CFR - 41.7 / 45.6
SF1.001.K3.02

KA Match:
Question matches KA as candidate is required to demonstrate knowledge on the effects of a dropped rod on RCS (reactor).

Cognitive Level: F 2

Question BANK
Previous Use: RO24 AUDIT
Original Question:

Associated objective(s):

(RO-C-AOP0240412-E1) Given a set of plant conditions and the occurrence of an abnormal event without operator intervention, without use of references, identify the event and predict the response of the plant to Dropped Rod, Misaligned Rod, or RPI Failure, including final plant configuration.

59 **ID: NRCAUDIT07-0115** **Points: 1.00**

Both Units are responding to a loss of Plant Air (PA) and Control Air (CA) event.

The following conditions exist:

- Both Units' Plant Air Compressors (PAC's) have failed.
- Both Units' Control Air Compressors (CAC's) have failed.
- Both Units' PA Header pressures are 70 psig and lowering.
- Both Units' CA Header pressures are 80 psig and lowering.
- An AEO has been dispatched to start the Back-up Plant Air Compressor (B/U PAC), per 01-OHP-4022-064-001, Control Air Malfunction.

Which ONE of the following statements describes the B/U PAC ability to pressurize both Units' Control Air Systems under these conditions?

A. PRV-10 or PRV-11, Plant Air Header Crosstie Valves to Unit 2, must be jumpered and reopened.

B. PRV-20 or PRV-21, Plant Air Header Crosstie Valves to Unit 1, must be jumpered and reopened.

C. The B/U PAC will discharge to both Units' Control Air Systems.

D. The B/U PAC to Plant Air Receiver crosstie must be manually opened.

Answer: C

Answer Explanation:

A. Incorrect This would supply backup air to Unit 1.
B. Incorrect This would supply backup air to Unit 2.
C. Correct The Backup Air compressor discharges into the section of piping between the Plant Air header crosstie valves. This is also where the Control Air headers are connected allowing the backup air compressor to supply control air.
D. Incorrect There is no crosstie valve to the Plant Air Receivers and Plant Air is isolated from Control Air.

Comments:

References:
SOD-06401-002 Plant air SOD

KA - 079000 K4.01
Station Air System (SAS)
Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following:
Cross-connect with IAS
RO - 2.9 SRO - 3.2
CFR - 41.7
SF8.079.K4.01

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of Plant Air system cross-connects including where the Backup Air compressor is tied into the PA system.

Cognitive Level: H 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-06401-E3) Describe the basic flow path through the plant and control air systems during all modes of operation.

60 **ID: NRC2010-47** **Points: 1.00**

Given the following condition on Unit 2:

- Unit is operating at 100% power when an inadvertent Steam Line Isolation occurred.
- Immediately following the isolation, and resultant plant response, the operators note that all the SG PORVs failed to open.

One minute following the Steam Line Isolation, which ONE of the following describes the maximum expected SG pressures?

A. 1025 psig

B. 1040 psig

C. 1065 psig

D. 1085 psig

Answer: C

Answer Explanation:

A. Incorrect This is the Normal PORV Setpoint and would be the expected pressures of the other SGs if the PORV's had not failed.
B. Incorrect This is the PORV setpoint used in the SG tube rupture procedures for the faulted SG.
C. Correct Following the Steam Line Isolation a Rx trip would be expected due to OTDT. SG pressures would initially surge opening most of the safeties but as the RCS cooled down pressures would stabilize on the lowest safety valve setpoint (1065 psig) due to the reduction in Reactor Power and Decay heat during the initial 30 seconds of the event.
D. Incorrect The pressures may initially surge to this level but would quickly (less than 30 seconds) drop after the Rx trip.

Note - SG Safety valve set points:

SV-1A	1065 psig
SV-1B	1065 psig
SV-2A	1075 psig
SV-2B	1075 psig
SV-3	1085 psig

Comments:

References:
RO-C-05103 Main Steam System

KA - 035000 K6.02
Steam Generator System (S/GS)
Knowledge of the effect of a loss or malfunction of the following will have on the S/GS:
Secondary PORV
RO - 3.1 SRO - 3.5
CFR - 41.7 / 45.7
SF4.035.K6.02

KA Match:
Question matches KA as it requires candidate to demonstrate knowledge of how the loss of a PORV and subsequent SG Stop Valve closure will impact the SG pressure.

Cognitive Level: F 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-05103-E2) Describe the function(s) associated with the following Main Steam System Major Components:

a. Main Steam Leads
b. Main Steam Safety Valves (SV-1, 2, 3)
c. SG PORVs (MRV-213, 223, 233, 243)
d. Steam Supply to the TDAFP (MCM-221,231)
e. SG Stop Valves (MRV-210, 220, 230, 240)
f. Main Steam Dump Valves (MRV-211,212,221,222,231,232,241,242)
g. Main Steam Line Nozzles
h. Turbine Bypass (Equalizing) Header
i. Main Steam System Drains

61 **ID: P384** **Points: 1.00**

The power range nuclear instruments have been adjusted to 100% based on a calculated calorimetric (secondary heat balance). Which one of the following will result in actual reactor power being less than indicated reactor power?

A. The Feedwater temperature used in the calorimetric calculation is higher than actual Feedwater temperature.

B. The Reactor Coolant Pump heat input term is omitted from the calorimetric calculation.

C. The Feedwater flow rate used in the calorimetric calculation is lower than actual Feedwater flow rate.

D. The Steam pressure used in the calorimetric calculation is higher than actual Steam pressure.

Answer: B

Answer Explanation:

A. Incorrect Because the feedwater heat transfer rate value is subtracted in the CTP calculation, an increase in the feedwater heat transfer rate results in a calculated power lower than actual power. If NIs are adjusted down to the calculated value, actual reactor power would be higher than indicated power.

B. Correct If the value for the reactor coolant pump heat input term was omitted from the equation, the indicated power would be greater than the actual power. This is because as the pump operates it adds heat energy to the fluid it is pumping from friction caused by head loss. If the heat the pump was putting into the water entering the core was not subtracted in the calculation it would appear that the reactor core was creating more power than it was.

C. Incorrect The feedwater flow rate has two effects on the power equation. Although both terms are reduced when evaluated in the CTP equation they have opposite effects - but not equal effects. The reduction in the feedwater heat transfer rate would, by itself, cause an increase in the indicated power. The resulting decrease in the steam heat transfer rate would, by itself, cause a decrease in the indicated power. The specific enthalpy of the steam is much greater than the specific enthalpy of the feedwater, therefore the steam heat transfer rate term decreases significantly more than the feedwater heat transfer rate term. So overall the effect is a decrease in the calculated/indicated power.

D. Incorrect The relationship between pressure and specific enthalpy may not be obvious without looking at the steam tables or Mollier diagram. In the region of interest, with reactor at full or nearly full power operations, a higher steam pressure value results in a lower specific enthalpy. Using a lower steam heat transfer rate results in a calculate/indicated power lower than actual power.

Comments:

References:

RQ-C-FUND-6,

KA - 015000 A1.01
Nuclear Instrumentation System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the NIS controls including: NIS calibration by heat balance
RO - 3.5 SRO - 3.8
CFR - 41.5 / 45.5
SF7.015.A1.01

KA Match:
Question matches KA as candidate must demonstrate knowledge of terms used in Heat Balance and the effect on Nuclear Instrumentation adjustments, if terms are improperly applied.

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-GF19-E14) Explain methods of determining core thermal power.

62	ID: RO-C-01200-E22-4	Points: 1.00

Given the following conditions:

- Unit 1 is at 100% power
- A loss of power to CRP-3 occurs

What is the effect on the Rod Position Indication (RPI) system and the procedural action necessary, if any?

A. RPI indications are maintained since backup power supply (via ELSC) **automatically** energized the RPI Cabinets. Operation is allowed since RPI has been restored.

B. RPI indications are lost. Dispatch an operator to **manually** align backup power supply (via ELSC) to energize the RPI Cabinets. Operation is allowed since RPI has been restored.

C. RPI indications are maintained since backup power supply (via ELSC) **automatically** energized the RPI Cabinets. The CRP-3 power feed to RPI must be restored within 4 hours.

D. RPI indications are lost. **Manually** initiate a Reactor Trip and verify Reactor Trip by Wide Range power and negative start-up rate since the Rod Bottom Lights are NOT lit.

Answer: B

Answer Explanation:

A. Incorrect ELSC is the backup power source but it doesn't automatically energize RPI. Operation is allowed once a power source is restored.

B. Correct The indications are initially lost but may be restored by aligning ELSC manually to supply RPI. The IRPI Cabinet Power Supply Transfer Switch provides mechanism to transfer power from normal power supply (CRP-3) to backup power supply (ELSC).

C. Incorrect ELSC is the backup power source but it doesn't automatically energize RPI. Operation is allowed once a power source is restored.

D. Incorrect ELSC is the backup power source but it doesn't automatically energize RPI. A reactor trip is not required. Time to restore the RPI indication is allowed.

Comments:

References:

RO-C-01200 Rod Control and Rod Position Indicating system

KA - 014000 A2.06
Rod Position Indication System (RPIS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of LVDT
RO - 2.6 SRO - 3.0
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF1.014.A2.06

KA Match:
Question matches KA as it tests knowledge of the plant indications and action required for a loss of the RPI due to loss of power.

Cognitive Level: H 3

Question BANK
Previous Use: RO29 AUDIT
Original Question:

Associated objective(s):

(RO-C-01200-E22) IDENTIFY control room indications, controls and alarms associated with the Rod Position Indication System.

63	**ID: NRCAUDIT07-0149A**	**Points: 1.00**

Given the following:

- A reactor trip occurred 20 minutes ago due to a Loss of Offsite Power.
- RCS Pressure is 965 psig.

(Reference provided)

Which ONE of the following sets of indications show that Natural Circulation conditions exist?

	RCS Core Exit Temp	SG Pressure	RCS CL Temp
A.	404°F	870 psig	402°F
B.	500°F	640 psig	496°F
C.	511°F	280 psig	434°F
D.	517°F	539 psig	462°F

Answer: B

Answer Explanation:

A. Incorrect Subcooling is >40°F (138°F) but colder than SG saturation temp of 529°F

B. Correct RCS Pressure = 965 psig = 542°F therefore 42°F subcooling and SG saturation temperature is approximately equal to the RCS cold leg temperature.

C. Incorrect Subcooling is <40°F (31°F)

D. Incorrect Subcooling is <40°F (25°F)

Comments:

References:

02-OHP-4023-SUP-011, Natural Circulation Verification

NOTE: Provide Steam Tables to answer question.

KA - 017000 A3.01
In-Core Temperature Monitor (ITM) System
Ability to monitor automatic operation of the ITM System, including: Indications of normal, natural, and interrupted circulation of RCS
RO - 3.6 SRO - 3.8
CFR - 41.7 / 45.5
SF7.017.A3.01

KA Match:
Question matches KA as candidate must evaluate plant conditions to determine in Natural Circulation conditions exist.

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP03-E6) Define Natural Circulation.

64	ID: CM-0079A	Points: 1.00

Unit One is completing a refueling outage and the following conditions currently exist:

- Defueled at Mid-loop conditions
- RHR pumps secured
- Making preps to refill the RCS and the Lower Reactor Cavity using RWST gravity fill
- A line up to allow gravity fill requires you to open 1-IMO-310,East RHR Pump Suction.

Which ONE of the following conditions will allow the opening of 1-IMO-310, East RHR Pump Suction?

A. ICM-305, East RHR Recirc Sump to E RHR/CTS Pumps is open
IMO-330, East RHR Discharge to Containment Spray is closed
IMO-340, Chg Pumps Suct from E RHR HX is closed

B. RCS pressure is less than or equal to 375 psig
IMO-330, RHR Spy to Upper Cntmt East is open
IMO-215, East CTS pump suction is closed

C. ICM-305, East RHR Recirc Sump to E RHR/CTS Pumps is closed
IMO-330, RHR Spy to Upper Cntmt East is closed
IMO-340, Chg Pumps Suct from E RHR HX is closed

D. RCS pressure is less than or equal to 425 psig
IMO-340, Chg Pumps Suct from E RHR HX is open
IMO-215, East CTS Pump Suction is open

Answer: C

Answer Explanation:

A. Incorrect The associated SUMP Recirc suction needs to be CLOSED.

B. Incorrect RWST Suction is NOT interlocked with RCS pressure. Plausible since RHR to Loop suctions are interlocked to open. IMO-215 must be closed to open ICM-305 but NOT IMO-310.

C. Correct The associated SUMP Recirc suction, RHR Spray Valve, and the Supply to the CCPs must the closed.

D. Incorrect RWST Suction is NOT interlocked with RCS pressure. Plausible since RHR to Loop suctions are interlocked to open. IMO-215 must be closed to open ICM-305 but NOT IMO-310.

Comments:

References:
RO-C-01700 Residual Heat Removal

KA - 002000 A4.07
Reactor Coolant System (RCS)
Ability to manually operate and/or monitor in the control room: Flow path linking the RWST through the RHR system to the RCS hot legs for gravity refilling of the refueling cavity
RO - 2.8 SRO - 3.1
CFR - 41.7 / 45.5 to 45.8
SF4.002.A4.07

KA Match:
Question matches KA since it asks the candidate to recognize interlocks that must be met to operate a valve required to be opened to allow refilling the refueling cavity.

Cognitive Level: F 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-01700-E7) Explain the control interlocks associated with the operation of the following RHR System components.

a. IMO-330/331
b. IMO-340/350
c. IMO-128/ICM-129
d. ICM-305/306
e. IMO-310/320

65	ID: CM-0034A	Points: 1.00

During operations at 95% reactor power and pressurizer level of 53% on Unit 2, the Level Program Controller fails to maximum. Which ONE of the following combination of INDICATIONS does the operator have that the programmed level has failed high?

	CVCS Letdown	Backup Heaters	PRZ Level Trend	Level Deviation Alarm Actuated
A.	isolated	energized	higher	low
B.	in service	deenergized	higher	low
C.	isolated	deenergized	lower	high
D.	in service	energized	lower	high

Answer: B

Answer Explanation:

The Program Level failing high will cause the Level Control system to respond to try to match actual level to program level. In doing so QRV-251 will throttle open while in Automatic. The throttling open will cause actual level to rise resulting in the rising trend. The heaters are controlled by the control and bistable channels and the Master Controller. The Master Controller will see a low level deviation between actual and program level and not energize the back up heaters. As actual level is >5% below program the Deviation alarm will be for a Low Deviation. (Ann 108/208 drop 4)

A. Incorrect Letdown will remain in service, heaters will not be energized based on low level deviation.

B. Correct The Program Level failing high will cause the Level Control system to respond to try to match actual level to program level. In doing so QRV-251 will throttle open while in Automatic. The throttling open will cause actual level to rise resulting in the rising trend. The heaters are controlled by the control and bistable channels and the Master Controller. The Master Controller will see a low level deviation between actual and program level and not energize the back up heaters. As actual level is >5% below program the Deviation alarm will be for a Low Deviation. (Ann 108/208 drop 4)

C. Incorrect Letdown will remain in service and deviation alarm will be low deviation.

D. Incorrect Heaters will not energize based on level deviation, level trend will higher and deviation alarm will be low deviation.

Comments:

References:
RO-C-00202 Pressurizer and Pressure Relief system

KA - 011000 2.1.7
Pressurizer Level Control System (PZR LCS)
Conduct of Operations
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
RO - 4.4 SRO - 4.7
CFR - 41.5 / 43.5 / 45.12 / 45.13
SF2.011.GEN

KA Match:
Question matches KA as candidate must demonstrate knowledge on the PZR Level Control system and effects of a change to one of the inputs to the control system.

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00202-E13) Describe the operation of the Pressurizer Pressure Control System, both in automatic and manual.

66 ID: RO-C-ADM04-E5-1 Points: 1.00

Given the following data obtained from a scheduled surveillance.

From Technical Data Book figure 19.1. - All times in seconds:

VALVE	DIRECTION	MIN STROKE	MAX STROKE	LIMIT
1-IMO-54	CLOSED	5.0	8.2	9.0
1-IMO-54	OPEN	5.2	8.5	10.0

Valve Test results: Valve stroke time Closed: 8.4 seconds
Valve stroke time Open: 8.2 seconds

Which ONE of the following describes the status of 1-IMO-54, Boron Injection Tank Discharge to Cold Leg 4?

A. OPERABLE, stroke times are within surveillance requirements.

B. INOPERABLE, because immediate retesting is not allowed for surveillances.

C. INOPERABLE, if an immediate retest cannot be performed.

D. OPERABLE, if an immediate retest results in the same stroke time.

Answer: C

Answer Explanation:

A. Incorrect Valve is outside MAX time for closed, but under the limit per OHI-4016. Must be declared INOPERABLE if cannot retest immediately.
B. Incorrect OHI-4016 does allow retesting valve for valves exceeding MAX stroke time but not exceeding LIMIT stroke time. (Attachment 33, 3.1.4.h
C. Correct Valve is outside MAX time for closed, but under the limit per OHI-4016. Must be declared INOPERABLE if cannot retest immediately.
D. Incorrect Valve is outside MAX time for closed, but under the limit. Repeating test with same time does not meet OHI-4016 requirements to call valve OPERABLE.

Comments:

References:
OHI-4016 Conduct of Operations: Guidelines, Section 3.1.4, TDB 19.1

KA - 194001 2.1.25
Generic
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc.
RO - 3.9 SRO - 4.2
CFR - 41.10 / 43.5 / 45.12
P2.1.25

KA Match:
Question matches KA as it tests candidate's ability to interpret data from a surveillance test and compare against acceptance criteria.

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM04-E5) Describe the theory and practice of valve testing.

67 **ID: NRCAUDIT07-0965** **Points: 1.00**

Given the following conditions in Unit 2:

- Unit 2 is in MODE 6
- Refueling is in progress
- Source Range Audible Count Rate in containment and Control Room just became INOPERABLE

Which ONE of the following describes the required Technical Specification actions for these conditions?

A. Immediately initiate actions to isolate Unborated water sources to the RCS.

B. Within one hour verify adequate SHUTDOWN MARGIN and suspend all core alterations.

C. No action is required as long as both Source Range Flux Monitors remain OPERABLE.

D. Within 15 minutes, return Control Room Audio Count Rate to OPERABLE and return the containment Audio Count Rate to OPERABLE within one hour.

Answer: A

Answer Explanation:

A. Correct Technical Specification 3.9.2 requires an audible count rate circuit to be operable. If the audible is lost, TS 3.9.2 action C requires the operator to immediately initiate actions to isolate unborated water sources to the RCS.
B. Incorrect If < 2 SR channels are operable, the Action is to immediately suspend Core Alterations.
C. Incorrect The Unborated water sources must be isolated.
D. Incorrect The Unborated water sources must be isolated

Comments:

References:

Tech. Spec. 3.9.2

KA - 194001 2.1.42

Generic

Conduct of Operations

Knowledge of new and spent fuel movement procedures.

RO - 2.5 SRO - 3.4

CFR - 41.10 / 43.7 / 45.13

P2.1.42

KA Match:

Question matches KA as candidate must demonstrate knowledge of actions for a loss of Audible Source Range counts IAW with procedures and Tech Specs.

Cognitive Level: F 3

Question BANK

Previous Use: NRC 2007

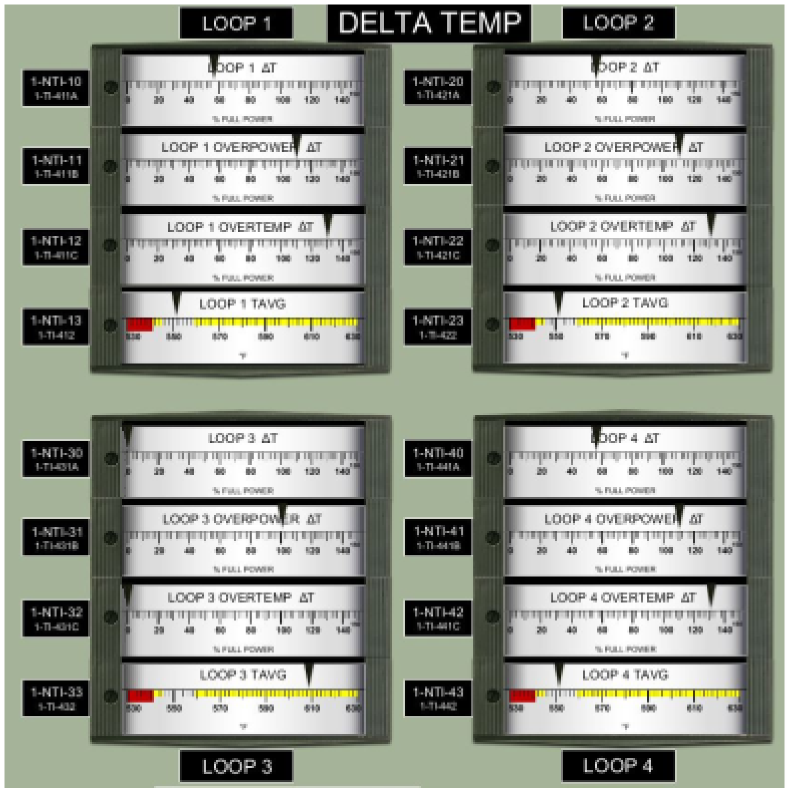
Original Question:

Associated objective(s):

(RO-C-ADM13-E8) Given an abnormal condition, equipment failure or an accident involving nuclear fuel determine what actions the control room team must take.

68	**ID: CM-0010A**	**Points: 1.00**

Which Loop 3 failure would result in the indications depicted below?



A. One T_{HOT} detector has failed high

B. The T_{COLD} detector has failed high

C. One T_{HOT} detector has failed low

D. The T_{COLD} detector has failed low

Answer: B

Answer Explanation:

A. Incorrect — Plausible because T_{AVG} is depicted higher than the other loops, but if a T_{HOT} detector had failed high Delta-T would also be higher.

B. Correct — T_{AVG} is higher than the other loops because $(T_{COLD}+T_{HOT})/2$ is larger, and Delta-T has failed low because $T_{HOT} - T_{COLD}$ is less than 0.

C. Incorrect — Plausible because Delta-T has failed low, but if T_{HOT} had failed low T_{AVG} would also be lower.

D. Incorrect Plausible because both T_{AVG} and Delta-T indications have failed, but the response depicted is the opposite of a T_{COLD} failure.

Comments:

References:

RO-C-00200 Reactor Coolant system

KA - 194001 2.1.45
Generic
Conduct of Operations
Ability to identify and interpret diverse indications to validate the response of another indication.
RO - 4.3 SRO - 4.3
CFR - 41.7 / 43.5 / 45.4
P2.1.45

KA Match:

Question matches KA as candidate is given diverse indication and must evaluate the data to validate plant conditions / failures.

Cognitive Level: F 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00200-E6) Recognize RCS instrumentation which provides an input to SSPS/RPS or other monitoring or interlock functions and describe the functions provided.

69 ID: 2007-0542 Points: 1.00

Which ONE of the following is required to identify/track Tech Spec status of equipment that is made Inoperable for planned maintenance during Modes 1 through 4? (Assume Inoperability will continue through shift turnover)

A. A Control Room Log entry and a Shift Manager Log entry

B. An AR and a Shift Manager Log entry

C. An AR and an Abnormal Position Log entry

D. A Control Room Log entry and an Open Items Log entry

Answer: D

Answer Explanation:

A. Incorrect The SM log does not need to duplicate Control Room Entries & an Open Items Log is required.
B. Incorrect SM log entry not required and an AR is not required for planned maintenance.
C. Incorrect Abnormal Position Log entry not required and an AR is not required for planned maintenance.
D. Correct A Control Log Entry and Open Items Log Entry is required for inoperable TS equipment.

Comments:

References:
OHI-4000 Conduct of Operations: Standards, OHI-4043 Technical Specification Open Items Log

KA - 194001 2.2.23
Generic
Equipment Control
Ability to track Technical Specification limiting conditions for operations.
RO - 3.1 SRO - 4.6
CFR - 41.7 / 41.10 / 43.2 / 45.13
P2.2.23

KA Match:
Question matches KA as candidate must identify requirements to track TS LCO equipment IAW station procedures.

Cognitive Level: F 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM05-E7) Describe the process by which the plant is determined to be in compliance with Tech. Specs.

70 ID: 2008NRC-0640 Points: 1.00

Who ensures that the **scope of a post maintenance test** is adequate to determine that equipment is operable following maintenance activity?

A. The Responsible System Engineer

B. The Work Week Manager

C. The Mechanical Maintenance Supervisor

D. The Shift Manager or Work Control Center SRO

Answer: D

Answer Explanation:

A. Incorrect System engineers are responsible for reviewing work on their systems, but do not determine what testing is required for operability.

B. Incorrect Work Week Manager only co-ordinate the activities that are required to be performed for post maintenance testing.

C. Incorrect The Mechanical Maintenance Supervisor is responsible for documenting PMT's are completed as part of work packages.

D. Correct The SM or Work Control Center SRO both ensures that the scope of a post maintenance test is adequate to determine that equipment is operable following maintenance activity and also authorizes the performance of PMT's.

Comments:

References:

RO-C-ADM04, PMI-2294 Post Maintenance Testing Program,

KA - 194001 2.2.21
Generic
Equipment Control
Knowledge of pre- and post-maintenance operability requirements.
RO - 2.9 SRO - 4.1
CFR - 41.10 / 43.2
P2.2.21

KA Match:

Question matches KA as it tests knowledge of post-maintenance testing responsibilities for equipment OPERABILITY IAW station procedures.

Cognitive Level: F 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM04-E2) Explain in general terms the different programs of testing that are in place at Cook Power Station.

71 ID: NRCAUDIT07-0837 Points: 1.00

Which ONE of the following will AUTOMATICALLY stop the selected Monitor Tank pump during a liquid release to the Unit 2 Circulating Water System?

A. HIGH flow alarm on Liquid Waste Sample Flow channel RFS-1010.

B. ALERT alarm on Liquid Waste Effluent channel RRS-1001.

C. Loss of one Unit 2 Circulating Water pump.

D. HIGH alarm on Liquid Waste Local Area channel RRA-1003.

Answer: A

Answer Explanation:

A. Correct HIGH or LOW sample flow on RFS-1010 will cause the monitor tank pumps to trip. Channel Failure on RRS-1001 or RRR-1002 or a high radiation alarm on RRS-1001 will also cause the pumps to trip.

B. Incorrect An "Alert" alarm will NOT cause the pumps to trip. Plausible if student believes anticipatory alarm will trip monitor tank pump during release.

C. Incorrect A loss of ONE CW pump will NOT automatically cause the pumps to trip, but 2-RRV-286, Liquid Waste Effluent to Unit 2 CW Discharge valve will auto close on the loss of ALL CW pumps on Unit 2. Plausible if student believes CW pump breakers are interlocked with monitor tank pumps.

D. Incorrect High local area alarm will NOT cause the monitor tank pumps to trip. Plausible if student believes a high local area alarm, which could be attributed to tank/pipe fluids, would provide a trip of the monitor tank pump.

Comments:

References:
12-OHP-4024-139 drop 18

KA - 194001 2.3.15
Generic
Radiation Control
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
RO - 2.9 SRO - 3.1
CFR - 41.12 / 43.4 / 45.9
P2.3.15

KA Match:
Question matches KA as it requires operator knowledge of the actions resulting from a radiation monitor alarm.

Cognitive Level: H 3

Question BANK - REPEAT
Previous Use: NRC 2012
Original Question:

Associated objective(s):

(RO-C-02200-E8) Describe the conditions that will cause the following liquid waste disposal component to trip, automatically/manually start and/or automatically/manually reposition.
a. RRV-285

72 ID: NRCAUDIT07-0991A Points: 1.00

Given the following conditions on Unit 1:

- Unit is in Mode 6
- R-5, Spent Fuel Pit Radiation monitor fails HIGH.

Which ONE of the following correctly describes the required Operator actions for the high radiation alarm?

A. Manually trip Supply and Exhaust Fans.
Manually OPEN the Charcoal Filter Outlet Dampers.
Manually CLOSE the Charcoal Filter Bypass Dampers

B. Verify the Supply and Exhaust Fans Automatically trip.
Verify the Charcoal Filter Outlet Dampers are CLOSED.
Verify the Charcoal Filter Bypass Dampers will are OPEN.

C. Verify the Supply Fans Automatically trip.
Manually stop one Exhaust Fan.
Verify the Charcoal Filter Outlet Dampers are OPEN.
Verify the Charcoal Filter Bypass Dampers are CLOSED.

D. Manually trip one Supply Fan.
Verify the Exhaust Fans Automatically start.
Verify the Charcoal Filter Outlet Dampers are CLOSED.
Verify the Charcoal Filter Bypass Dampers are OPEN.

Answer: C

Answer Explanation:

A. Incorrect. All actions are Automatic with exception of Stopping one Exhaust fan.

B. Incorrect Exhaust fans will not trip. Charcoal Filter Outlet dampers open. Charcoal filter bypass dampers are closed.

C. Correct A high radiation alarm on R-5 will cause the dampers to re-align through the charcoal filters, and result in stopping all Spent Fuel Pit supply fans. One Exhaust fan is Manually Stopped per ARP.

D. Incorrect Charcoal Filter Outlet dampers open. Charcoal filter bypass dampers are closed. Supply fans are Automatically stopped

Comments:

References:
1-OHP-4024-138, Drop 5

KA - 194001 2.3.12
Generic
Radiation Control
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.
RO - 3.2 SRO - 3.7
CFR - 41.12 / 45.9 / 45.10
P2.3.12

KA Match:
Question matches KA as it requires the knowledge of the proper charcoal filter alignment for the fuel handling exhaust fans following a radiation alarm on the spent fuel pit area radiation monitor.

Cognitive Level: H 3

Question BANK
Previous Use: RO28 AUDIT
Original Question:

Associated objective(s):

(RO-C-01350-E5) Describe the function of the following Westinghouse Radiation Monitoring System channels including any automatic actions that occur on a high alarm:
1. R-5 Spent Fuel Pit Area Monitor
2. R-6 Nuclear Sampling Room Area Monitor
3. R-8 Waste Drumming Room Area Monitor
4. R-17A/17B East/West CCW Header Monitor
5. R-19 SG Blowdown Sampling Effluent Monitor
6. R-24 SG Blowdown Treatment Effluent Monitor
7. R-20/28 East/West ESW Header Effluent Monitor

73 ID: CM-39232 Points: 1.00

The control room operators are responding to a Steam Generator Tube Rupture. In order to cool down the RCS and establish required subcooling margin, the operators dump steam to the condenser using the intact SG's.

Which ONE of the following describes why this method of RCS cooldown is preferred over dumping steam through the PORV's of the intact SG's?

A. Minimizes radiological releases.

B. Minimizes thermal shock to the reactor vessel.

C. Minimizes shrink experienced by the RCS.

D. Minimizes RCS subcooling requirements.

Answer: A

Answer Explanation:

A. Correct	Dumping steam to the condenser will keep the radioactivity contained within the plant. Prior to the ruptured SG isolation, the leaking RCS would have mixed with the secondary and raised the activity levels of the intact SGs.
B. Incorrect	The magnitude of the cooldown will be the same with both the SG PORVs and the condenser. Using the condenser may actually cooldown the RCS faster.
C. Incorrect	The magnitude of the cooldown will be the same with both the SG PORVs and the condenser. This is directly related to the amount of RCS shrink.
D. Incorrect	RCS Subcooling requirements are the same.

Note: Distracters are plausible since the SG PORV's may provide greater cooling at maximum rate and are typically less controllable than the fine control provided by the steam dumps.

Comments:

References:
12-OHP-4023-E3 Steam Generator Tube Rupture

KA - 194001 2.3.11
Generic
Radiation Control
Ability to control radiation releases.
RO - 3.8 SRO - 4.3
CFR - 41.11 / 43.4 / 45.10
P2.3.11

KA Match:
Question matches KA as it requires the candidate to understand how radiation release is controlled when experiencing a Steam generator Tube Rupture.

Cognitive Level: F 2

Question BANK
Previous Use: 2008 NRC
Original Question:

Associated objective(s):

(RO-C-EOP08-E18) For the E-3 series procedures and the ECA-3 series procedures discuss the basis or reason for all Steps.

74 ID: NRCAUDIT07-1024 Points: 1.00

Given the following events and conditions:

- Unit 2 is in Mode 6 with refueling activities in progress.
- Containment purge is in service
- A fuel element accidentally dropped into the cavity.
- All radiation monitor TRIP / BLOCK switches are in the NORMAL position.
- The Manipulator Crane area radiation monitor has a HIGH alarm.
- ERS-2305 and ERS-2405, Lower Containment Noble Gas Low Range Radiation Monitors, have a HIGH alarm.

Which ONE of the following actions should occur, assuming that operators follow the required procedure steps and systems operate as designed?

A. Containment evacuation alarm sounds automatically.
Containment purge stops automatically.

B. Containment evacuation alarm is manually actuated by the control room operator.
Containment purge stops automatically.

C. Containment evacuation alarm sounds automatically.
Containment purge is stopped manually by the control room operator.

D. Containment evacuation alarm is manually actuated by the control room operator.
Containment purge is stopped manually by the control room operator.

Answer: B

Answer Explanation:

Procedure Actions for OHP 4022.018.004
Step 1 directs the operator to actuate the Containment Evacuation alarm.
For the protection of personnel, it is important to evacuate the affected area until radiation surveys can be completed. Step 2 follows up the alarm with a page announcement notifying all non-essential personnel to evacuate the containment. Steps 5 through 9 verify the containment purge and pressure relief systems are shut down and isolated. This will limit the exposure of personnel outside containment.

A. Incorrect	Containment evacuation alarm is manually actuated by the control room operator.
B. Correct	Step 1 directs the operator to actuate the Containment Evacuation alarm. Containment purge stops automatically.
C. Incorrect	Containment evacuation alarm is manually actuated by the control room operator. Containment purge stops automatically.
D. Incorrect	Containment purge stops automatically.

Comments:

References:

2- OHP 4022.018.004 Irradiated Fuel Handling Accint in Containment Building – Control Room Actions

KA - 194001 2.4.31
Generic
Emergency Procedures/Plan
Knowledge of annunciator alarms, indications, or response procedures.
RO - 4.2 SRO - 4.1
CFR - 41.10 / 45.3
P2.4.31

KA Match:
Question matches KA as candidate is required to demonstrate knowledge of annunciator response procedure associated with an abnormal procedure that leads to an Emergency Plan situation.

Cognitive Level: H 3

Question BANK
Previous Use: RO22 AUDIT
Original Question:

Associated objective(s):

(RO-C-01350-E4) Describe the function of the following Radiation Monitoring System Monitors including any automatic actions that occur on a high alarm:

1. VRS-1101/1201 (2101/2201) Upper Containment Area Monitor
2. ERS-1300(2300) Lower Containment Radiation Monitor Train A
3. ERS-1400(2400) Lower Containment Radiation Monitor Train B
4. VRS-1500(2500) Unit Vent Effluent Radiation Monitor
5. MRA-1600/1700(2600/2700) Steam Generator PORV Monitors
6. SRA-1800(2800) Steam Packing Exhauster (GSLO) Monitor
7. SRA-1900(2900) Steam Jet Air Ejector Vent Monitor
8. ERA-7300/8300 AUX BLDG EQUIP ROOM AREAS
9. ERS-7400/8400 CONT ROOM INCORE/AUXB AR
10. ERA-7500 AUX BLDG 673/587 Areas
11. ERA-7600 AUX BLDG 633/650 AREAS
12. RRS-1000 Liquid Waste Effluent Monitor
13. CRS-3301/3401 (4301/4401) East/West CCW Hx Outlet Monitor

75	ID: CM-1055	Points: 1.00

Following the declaration of an ALERT by the Site Emergency Coordinator, initial notification to the State of Michigan must be completed within __(1)___ minutes, with updates provided a maximum of every ___(2)___ minutes thereafter.

	(1)	(2)
A.	15	15
B.	15	30
C.	60	30
D.	60	60

Answer: B

Answer Explanation:

A. Incorrect Plausible if candidate thinks updates have the same 15 minute requirement as initial notifications.
B. **Correct** PMP-2080-EPP-100 Attachment #8 requires EMD-32A within 15 minutes of Initial classification, classification upgrade, or change in PAR.EMD-32b is required within 30 minutes of previous EMD-32 form.
C. Incorrect Plausible as NRC notification time requirement is 60 minutes, second part is correct for updates.
D. Incorrect Plausible as NRC notification time requirement is 60 minutes.

Comments:

References:
PMP-2080-EPP-100 Emergency Response Attachment #8

KA - 194001 2.4.29
Generic
Emergency Procedures/Plan
Knowledge of the emergency plan.
RO - 3.1 SRO - 4.4
CFR - 41.10 / 43.5 / 45.11
P2.4.29

KA Match:
Question matches KA as it requires knowledge of requirements concerning off-site notifications as contained in station Emergency Planning procedure.

Cognitive Level: F 2

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(ST-C-EP04-E5) State the time limitations for initial offsite notifications and updates to the Berrien County Sheriff's Department, Michigan State Police, and NRC. (SRO/STA/RO)

76 **ID: RO-C-EOP09-E36-48** **Points: 1.00**

Given the following conditions:

- A Steam Line Break has occurred on Unit 2.
- The faulted Steam Generator has stopped depressurizing.
- Normal Charging is being established per 2-OHP-4023-ES-1.1, Safety Injection Termination.

While attempting to control pressurizer level with normal charging, pressurizer level continues to lower.

Which of the following describes the action to be directed by the US?

A. Actuate Safety Injection and transition to 2-OHP-4023-E-1, Loss of Reactor or Secondary Coolant.

B. Actuate Safety Injection and transition to 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

C. Realign the Centrifugal Charging Pump Injection flowpath and transition to 2-OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization.

D. Realign the Centrifugal Charging Pump Injection flowpath and remain in 2-OHP-4023-ES-1.1, Safety Injection Termination.

Answer: C

Answer Explanation:

A. Incorrect. Plausible because once SI is terminated the Foldout Page requires transition to E-1 after starting ECCS Pumps.

B. Incorrect. Plausible because this would be correct if events occurred in ES-0.1, Reactor Trip Response. ES-1.1 requires manual operation and transition to correct procedure for LOCA.

C. Correct. Per ES-1.1, Step 8 RNO the crew is either in the wrong procedure or another event has occurred. Equipment alignment at this point would require transition to ES-1.2 for more appropriate recovery.

D. Incorrect. Plausible because the Charging Pump injection flowpath should be realigned, however, the RNO action of Step 8 requires entry into ES-1.2, Post LOCA Cooldown and Depressurization.

Comments:

References:
12-OHP-4023-ES-1.2 –Post LOCA Cooldown and Depressurization Step #: 13 N1

KA - 000009 EA2.04
Small Break LOCA
Ability to determine and interpret the following as they apply to a small break LOCA:
PZR level
RO - 3.8 SRO - 4.0
CFR - 41.7 / 41.10 / 43.5 / 45.13
EPE.009.EA2.04

KA Match:
Question matches KA as it requires candidate to interpret the PZR level trend during a small break LOCA.

SRO - Question has candidate assess the plant conditions and determine the correct actions based detailed procedural steps (RNO). Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use: RO29 Audit
Original Question:

Associated objective(s):

(RO-C-EOP09-E36) For each of the E-1 Series procedures, discuss the basis or reason for all Steps.

77 **ID: RO-C-EOP14-E9-1** **Points: 1.00**

Given the following:

- A Loss of All AC Power has occurred.
- The control room operators have performed the Immediate Operator Actions and have depressurized the intact S/G's to 290 #'s.
- Bus T21D has been energized from DG2CD.
- Pressurizer Level is off scale low.
- The Crew has transitioned to 2-OHP-4023-ECA-0.2 Loss of Offsite Power Recovery with SI required and are at Step 5 "Load Equipment on the AC Emergency Bus"

Which ONE of the following describes the Unit Supervisor's direction to the operator regarding the North Safety Injection pump and the status of SI injection flow based on RCS Pressure?

A. Leave the pump in Pull to Lock. Flow would NOT be expected.

B. Place the pump in Neutral and allow it to auto start on the SI Signal. Flow would be expected.

C. Place the pump in Neutral and allow it start on the Blackout Timer. Flow would NOT be expected.

D. Manually Start the SI Pump. Flow would be expected.

Answer: D

Answer Explanation:

A. Incorrect Plausible if candidate does not understand why S/G's are depressurized and believes RCS pressure still above injection value.
B. Incorrect The RCS pressure will lower when the SG's are depressurized so SI flow would be present once the pump starts. Plausible since the SI signal would have actuated but was reset earlier in the procedure.
C. Incorrect Plausible if candidate not aware pump is started and does not understand why S/G's are depressurized and believes RCS pressure still above injection value.
D. Correct While performing the S/G depressurization an SI signal is expected and will be reset when received.

Comments:

References:
2-OHP-4023-ECA-0.2 Loss of All AC power Recovery with SI Required

KA - 000056 AA2.39
Loss of Offsite Power
Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Safety injection pump ammeter and flowmeter
RO - 3.5 SRO - 3.6
CFR - 41.7 / 41.10 / 43.5 / 45.13
APE.056.AA2.39

KA Match:
Question matches KA as it requires candidate to determine the alignment of the SI pumps (amps and flow) based on the current status of the detailed procedural actions.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps. [55.43(b)(5)]

Cognitive Level: H 3

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP14-E9) For each of the ECA-0 series procedures, discuss the basis or reason for all Steps.

78 **ID: CM-0050A** **Points: 1.00**

Stable plant conditions exist with Pressurizer Level Channel 1 selected for control. The reference leg for the Channel 1 Pressurizer Level transmitter is suspected of having a leak based on control panel indications. The crew has entered 1-OHP-4022-013-010, Pressurizer Level Instrument Malfunction, and 1-OHP-4022-002-020, Excessive Reactor Coolant Leakage. RCS leakage has been estimated at .5 gpm.

- ❖ All actions have been completed in 1-OHP-4022-013-010, Pressurizer Level Instrument Malfunction up to performing the Attachment for bistable tripping.

Which ONE of the following describes the anticipated instrument and plant response with no operator action and what action will meet the Tech Spec Required Actions based on above conditions?

	Channel 1 PZR LVL Indication	Channel 2/3 PZR LVL Indication	VCT Actual Level
A.	Lowering	Rising	Rising
	Shutdown the plant and be in Mode 3 in 4 hours based on RCS leakage		
B.	Rising	Lowering	Rising
	Place Pressurizer Level Channel in Trip within 6 hours		
C.	Rising	Lowering	Rising
	Shutdown the plant and be in MODE 5 in 6 hours based on RCS leakage		
D.	Lowering	Rising	Lowering
	Place Pressurizer Level Channel in Trip within 1 hour		

Answer: B

Answer Explanation:
Level Trend Explanation: The reference leg leak would result in measured D/P rising which would make indicated level rise. The control system would respond by attempting to lower actual by lowering charging flow and lowering actual level. The reduced charging flow would result in VCT level rising.

A. Incorrect TS 3.4.13 limits Unidentified Leakage to .8 gpm on Unit One. Plausible if candidate believes entry into RCS Leakage is required. This action is Condition D in TS 3.4.13.

B. Correct TS 3.3.1 Condition D requires bistables to be tripped within 6 hours.

C. Incorrect Plausible if candidate believes entry into RCS Leakage TS 3.4.13 is required. Condition D requires Mode 3 in 6 hours and Mode 5 in 36 hours. Candidate may confuse time requirements.

D. Incorrect TS 3.3.1 does have action times with 1 hour completion times

Comments:

References:
TS 3.4.13, OHP-4022-002-020 Excessive Reactor Coolant Leakage, OHP-4022-013-010 Pressurizer Level Instrument Malfunction

KA - 000008 AA2.27
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident:
Effects on indicated PZR pressure and/or level of sensing line leakage
RO - 2.9 SRO - 3.2
CFR - 41.7 / 41.10 / 43.5 / 45.13
APE.008.AA2.27

KA Match:
Question matches KA as candidate must predict indicated and actual PZR level trends with vapor space accident.

SRO - Facility operating limitations in the TS and their bases. Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1). Candidate must identify Tech Spec required actions based on RCS leak size. [55.43(b)(2)]

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00800) Emergency Core Cooling System

79 **ID: RQ-C-4051-1B** **Points: 1.00**

Unit 2 is operating at 100%. The CRID-3 Inverter failed. The Crew attempted to transfer the Alternate Source to Load, but the transfer was not successful. The Crew re-energized CRID-3 by placing the inverter Manual Output Selector to the ALT OUTPUT 2-EZC-C-5AR position.

Subsequently, the CRID-2 inverter automatically swapped to the Alternate Source. Attempts to transfer it back to the Normal Source were not successful.

Which of the following is the most limiting action required based on the conditions listed?

A. Restore one inverter to OPERABLE status within 24 hours.

B. Restore both inverters to OPERABLE status within 8 hours.

C. Restore both inverters to OPERABLE status within 6 hours.

D. Initiate action within 1 hour to be in Mode 3 within 7 hours.

Answer: D

Answer Explanation:

A. Incorrect. Both inverters are inoperable, so this is not the most limiting action. Plausible if the CRID-2 inverter is considered operable because it automatically transferred to the Alternate source. This is the correct action for only one inverter inoperable.

B. Incorrect. This is a correct action if the CRID buses were considered inoperable because they are energized from the alternate source. Plausible if the bases for T.S. 3.8.9 were improperly applied and because both inverters are supplying their respective CRIDs from the alternate source.

C. Incorrect. This would be the correct action if the inverters were from the same train. Plausible if the action is improperly applied or the Train relationship is not properly identified.

D. Correct. Both inverters are inoperable in the listed conditions. No action is identified for two inoperable inverters from opposite trains, therefore Tech Spec 3.0.3 applies since no listed action applies.

Comments:

References:
TS 3.8.7, LCO 3.0.3

KA - 000057 2.2.40
Loss of Vital AC Electrical Instrument Bus
Equipment Control
Ability to apply Technical Specifications for a system
RO - 34 SRO - 4.7
CFR - 41.10 / 43.2 / 43.5 / 45.3
APE.057.GEN

KA Match:
Question matches KA as it presents the candidate with failures (loss of instrument bus and primary inverter feeds from different trains) and requires candidate to determine the applicable Technical Specification and Actions required.

SRO - Facility operating limitations in the TS and their bases. Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1). Candidate must identify Tech Spec required actions based on the Inverters that are inoperable. The candidate must determine use the basis to determine that the inverter failures are not covered by the TS Action statements even though the Instrument buses are energized. [55.43(b)(2)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-08203-E9) Given a description of plant conditions and/or the results of a Surveillance Test determine the following: All applicable TS, TRMs, System Operability and most limiting LCO.

80 **ID: CM-40207B** **Points: 1.00**

Unit 2 was operating at 40% power and experienced a severe Feedwater Break. Steam Generator (SG) 2 has completely depressurized and 2-OHP-4023-E-2, Faulted Steam Generator Isolation, has been entered.

The following conditions exist:

- Reactor Coolant System T_{colds} are 490°F and lowering.
- All Main Feedwater Isolation Valves are closed.
- All SG Stop Valves and Stop Valve Dump Valves are closed.
- All SG Narrow Range Levels are off scale low.
- STEAM GEN 1, 3, and 4 SF > FWF FLOW MISMATCH annunciators are LIT.
- SG 1, 2, 3, 4 STEAM LINE PRESSURE LOW annunciator just alarmed.
- Pressure in SG's 1, 3, and 4 are lowering.

Which ONE of the following procedural transitions will the US direct, if any, based on these conditions?

A. 2-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink.

B. 2-OHP-4023-ECA-2.1, Uncontrolled Depressurization of all Steam Generators.

C. Do NOT transition; remain in 2-OHP-4023-E-2, Faulted Steam Generator Isolation.

D. 2-OHP-4023-FR-H.5, Response to Steam Generator Low Level.

Answer: B

Answer Explanation:

A. Incorrect No indications of (inadequate) AFW flow exists. No reason to enter FR-H.1.

B. Correct 2-OHP-4023-E-2, Step 3 requires a transition to ECA-2.1 in the event that no SG Pressure Boundary is intact. SG 21, 23, and 24 Steam Gen Steam Line Pressure Low annunciators at set at 500 psig which equates to 470 °F T_{sat}. These indications show that the pressure reduction in the SGs is from the cooldown (depressurization) of the SGs rather than the RCS cooldown.

C. Incorrect A Transition is required since all SG pressures are lowering.

D. Incorrect FR-H.5 is a yellow path procedure and is NOT required under these conditions.

Comments:

References:
2-OHP-4023-E-2, Faulted Steam Generator Isolation

KA - 000040 2.4.6
Steam Line Rupture
Emergency Procedures/Plan
Knowledge of EOP mitigation strategies.
RO - 3.7 SRO - 4.7
CFR - 41.10 / 43.5 / 45.13
APE.040.GEN

KA Match:
Question meets the KA because the candidate must recognize that all SGs are faulted (based on alarm status).

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. Procedural selection/actions included for SRO criteria. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2007
Original Question:

Associated objective(s):

(RO-C-EOP07-E17) For E-2 and ECA-2.1 identify the Procedure Transitions.

81 **ID: RO-C-EOP04-E14-4** **Points: 1.00**

Given the following conditions:

- An ATWS has occurred on Unit 1.
- 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS is in progress.
- Emergency Boration is in progress.
- Safety Injection has actuated.
- All Steam Generator pressures are 800 psig and lowering.
- All Reactor Coolant System Cold Leg temperatures are 521ºF and lowering.
- Reactor Power is 9% (on all nuclear instruments) and lowering.

Which of the following mitigation strategies and follow on transition is appropriate for the event?

A. Remain in 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS and isolate faulted steam generators. Transition to 1-OHP-4023-E-0, Reactor Trip or Safety Injection when 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS is complete.

B. Remain in 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS and isolate faulted steam generators. Transition to 1-OHP-4023-E-0, Reactor Trip or Safety Injection when faulted steam generator isolation is complete.

C. Transition to 1-OHP-4023-E-2, Faulted Steam Generator Isolation while performing 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS concurrently. Transition to 1-OHP-4023-E-0, Reactor Trip or Safety Injection when faulted Steam Generator isolation is complete.

D. Transition to 1-OHP-4023-E-2, Faulted Steam Generator Isolation while performing 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS concurrently. Transition to 1-OHP-4023-E-0, Reactor Trip or Safety Injection when 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS is complete.

Answer: A

Answer Explanation:

A. Correct The faulted SG will be isolated in FR-S.1. Transition to E-0 will be made when FR-S.1 is complete.

B. Incorrect Plausible because remaining in FR-S.1 is correct but transition to E-0 after fault isolation is not correct. FR-S.1 must be completed to transition to E-0.

C. Incorrect Plausible because E-2 can be performed in parallel with other EOPs, however, FR-S.1 must remain as the procedure in affect. FR-S.1 must be completed to transition to E-0.

D. Incorrect Plausible because E-2 can be performed in parallel with other EOPs, however, FR-S.1 must remain as the procedure in affect. Transition to E-0 will be made when FR-S.1 is complete.

Comments:

References:

1-OHP-4023-FR-S.1 Response to Nuclear Generation / ATWS

KA - 000029 2.1.7
Anticipated Transient Without Scram (ATWS)
Conduct of Operations
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
RO - 4.4 SRO - 4.7
CFR - 41.5 / 43.5 / 45.12 / 45.13
EPE.029.GEN

KA Match:
Question matches KA as candidate is required to evaluate given conditions and make an operational decision on where / how to proceed.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: F 3

Question BANK
Previous Use: RO29 Audit
Original Question:

Associated objective(s):

(RO-C-EOP04-E14) For each of the FR-S series procedures identify the Major Action Categories and discuss the bases for each.

82 **ID: NRCAUDIT07-0385** **Points: 1.00**

Step 1 of OHP-4023-FR-P.1, Response To Imminent Pressurized Thermal Shock Condition, has the operator check that Residual Heat Removal (RHR) heat exchanger outlet flow is greater than 400 gpm if RCS pressure is less than 300 psig.

Which ONE of the following is the basis for this step?

A. Ensure adequate mixing in the cold leg downcomer region during natural circulation conditions.

B. Prevent core exit temperatures from exceeding the required temperature to place RHR in service.

C. Ensure adequate Low Head Safety Injection cooling prior to accumulator isolation.

D. Prevent implementation of Pressurized Thermal Shock (PTS) actions if a Large Break LOCA has occurred.

Answer: D

Answer Explanation:

A. Incorrect Mixing in the downcomer (beltline) region is a concern for localized PTS. However, it is NOT the basis for verifying the indications in this step.

B. Incorrect RCS temperature must be less than 350ºF to place RHR in service.

C. Incorrect Low Head Safety Injection cooling is NOT required for isolating the accumulators.

D. Correct Step 1 prevents the operator from unnecessarily attempting to address a PTS concern when one does not exist. A Large Break LOCA will cause symptoms which warrant entry into OHP-4023-FR-P-1, Response to Imminent Pressurized Thermal Shock Condition, however, a PTS concern does NOT exist due to the impossibility of re-pressurization.

Comments:

References:

12-OHP-4023-FR-P-1 Response to Imminent Pressurized Thermal shock Condition Background step 1

KA - 00WE08 EA2.1
Pressurized Thermal Shock
Ability to determine and interpret the following as they apply to the Pressurized Thermal Shock: Facility conditions and selection of appropriate procedures during abnormal and emergency operations
RO - 3.4 SRO - 4.2
CFR - 41.7 / 41.10 / 43.5 / 45.13
4.5.E08.EA2.1

KA Match:
Question matches KA as it requires candidate to know the bases for an Emergency Procedure based on plant conditions that governs a decision for procedure use.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP12-E29) For each of the FR-P series procedures discuss the basis or reason for all Steps.

83 **ID: RO-C-EOP9-T1-1** **Points: 1.00**

While performing 1-OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization the crew is at the step to check intact Steam generator levels.

Narrow range levels are #11 – 42%

#12 – 44%

#13 – 43%

#14 – 42%

The BOP is directed to maintain levels between 20 and 50%. After throttling AFW supply valves to all S/G's the BOP reports #3 S/G level is 48% and rising not under his control.

The unit supervisor will:

A. Direct BOP to trip close #3 S/G Stop Valve

B. Transition to E-3, Steam generator Tube Rupture

C. Direct BOP to trip the East MDAFW pump

D. Transition to E-2 Faulted Steam Generator Isolation

Answer: B

Answer Explanation:

A. Incorrect Plausible as Stop Valves are closed once in Tube Rupture procedure
B. Correct RNO for step directs transition to E-3
C. Incorrect Plausible as stopping AFW pump would stop level rise if not ruptured (AFW Supply valve leak by)
D. Incorrect Plausible as a Faulted Steam Generator could cause level rise due to swell.

Comments:

References:

1-OHP-4023-ES-1.2 Post LOCA Cooldown and Depressurization

KA - 00WE03 EA2.1
LOCA Cooldown and Depressurization
Ability to determine and interpret the following as they apply to the LOCA Cooldown and Depressurization: Facility conditions and selection of appropriate procedures during abnormal and emergency operations
RO - 3.4 SRO - 4.2
CFR - 41.7 / 41.10 / 43.5 / 45.13
4.5.E03.EA2.1

KA Match:
Question matches KA as candidate is required to interpret conditions during a post LOCA Cooldown and evaluate for a procedure transition.

SRO - Procedural selection/actions included for SRO criteria. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: H 3

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP09-T1) Upon completion of this lesson the students will provide a conceptual description of LOCAs. This description will include the principle concerns and assumptions associated with various size LOCAs at different locations in the Reactor Coolant System. This description will also include the major actions for each procedure; the reasons for the major actions and loops contained in the procedures; the bases for all critical steps; and the procedure transition criteria encountered in the procedures.

84 **ID: RO26-0176A** **Points: 1.00**

Given the following conditions on Unit 2:

- Unit was at 90% power.
- Control Rods were in AUTO.
- CBD began to step out with no mismatch signal.
- Rods were taken to MANUAL and rod motion ceased.

The following conditions now exist:

- Annunciator Panel 210 Drop 21 NUCLEAR INSTN SYSTEM TILT CMPTR ALARM is LIT
- CBD Bank Demand position is now 222 Steps.
- Group 2 Bank D Rod Position Indicators (RPIs) ALL indicate 222 Steps.
- Group 1 Bank D RPIs indicate as follows:

Rod D4:	205 Steps.
Rod D12:	222 Steps.
Rod M12:	207 Steps.
Rod M4:	222 Steps.

Reactor Engineering has determined that all CBD rods are free to move and has provided the following information:

- R is 1.041
- CFQ is 2.335
- K(Z) is .95 (at 10 feet)
- $F^{W}_{Q}(Z)$_ is 2.174 (at 10 feet)

(Reference Provided)

Which ONE of the following identifies the Technical Specification Action Condition(s) that the US must enter?

A. 3.1.4.A Only

B. 3.1.4.B Only

C. 3.1.4.A and 3.1.4.B Only

D. 3.1.4.B and 3.1.4.D Only

Answer: D

Answer Explanation:

A. Incorrect Rods are free to move, as stated in the stem, thus they are all operable.
B. Incorrect Using the numbers provided with Figure 3.1.4-1, rods which are <14 steps misaligned meet alignment requirements. This is the correct action with only 1 misaligned rod.
C. Incorrect Rods are free to move and are operable.
D. Correct Both D4 and M12 are unacceptably misaligned. D4 is 17 steps off and M12 is 15 steps off.

Comments:
References: TS 3.1.4
(Attachment provided)

KA - 000001 2.4.31
Continuous Rod Withdrawal
Emergency Procedures/Plan
Knowledge of annunciator alarms, indications, or response procedures.
RO - 4.2 SRO - 4.1
CFR - 41.10 / 45.3
APE.001.GEN

KA Match:
Question matches KA as candidate must evaluate alarms and indications for a continuous rod withdrawal and resulting plant conditions.

SRO - Selected as SRO due to application of Tech Specs. Facility operating limitations in the TS and their bases. Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1). [55.43(b)(2)]

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2010 Modified
Original Question:

Associated objective(s):

(RO-C-01200-E24) EXPLAIN the basis for Technical Specification LCOs, Action Statements, and Surveillance Requirements associated with the Rod Position Indication System.

85 **ID: CM-2699** **Points: 1.00**

While In procedure OHP-4023-E-3, Steam Generator Tube Rupture, at step 30, the following plant conditions exist:

- PZR level is 20%.
- Ruptured SG level is 20% and lowering.
- Letdown is in service.
- Natural circulation exists in the RCS.
- Containment conditions are normal.

(Reference Provided)

Which ONE of the following choices lists the actions directed by US based on current conditions?

A. Use auxiliary spray and reduce letdown.

B. Turn on PRZ heaters.

C. Raise charging.

D. Use auxiliary spray and turn on PRZ heaters.

Answer: C

Answer Explanation:

FROM STEP 30 TABLE AND ATTACHMENT C

	Ruptured SG Narrow Range Level		
PZR Level	Rising	Lowering	Off Scale High
< 34%	Raise Charging /Depressurize RCS	Raise Charging	Raise Charging / Maintain RCS and SG Press equal
34-50%	Depressurize RCS	Turn on PZR Heaters	Maintain RCS and SG Press equal
50 - 72%	Depressurize RCS / Reduce Charging	Turn on PZR Heaters	Maintain RCS and SG Press equal
>72%	Reduce Charging	Turn on PZR Heaters	Maintain RCS and SG Press equal

Comments:

References:
01-OHP 4023.E-3, Steam Generator Tube Rupture, Step 30 or Attachment C

KA - 000037 2.1.25
Steam Generator (S/G) Tube Leak
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc.
RO - 3.9 SRO - 4.2
CFR - 41.10 / 43.5 / 45.12
APE.037.GEN

KA Match:
Question matches KA as candidate must use a table from a station procedure to base actions to be taken during a Steam Generator Tube Rupture.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP08-E18) For the E-3 series procedures and the ECA-3 series procedures discuss the basis or reason for all Steps.

86 **ID: NRC2010-16** **Points: 1.00**

Given the following conditions on Unit 2:

- Unit was operating at 100% power when a malfunction of the Control Air dryer occurs.
- The Control Air header rapidly depressurizes and cannot be restored.

Which ONE of the following describes the correct operator response?

The Unit Supervisor will direct the crew to immediately trip the Reactor and implement:

A. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, may be performed concurrently after transitioning to 2-OHP-4023-ES-0.1, Reactor Trip Response.

B. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, is NOT needed since the EOP network may be performed without reliance on Control Air.

C. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, may NOT be performed until completion of 2-OHP-4023-ES-0.1, Reactor Trip Response.

D. 2-OHP-4022-064-002, Loss Of Control Air Recovery, until restoration of Control Air from any source.
Perform 2-OHP-4023-E-0, Reactor Trip or Safety Injection steps as time allows.

Answer: A

Answer Explanation:

A - CORRECT. OHI-4023, Abnormal/Emergency Procedure User's Guide allows Abnormal Procedures to be implemented concurrently with Non-Accident (ES-0.1, 0.2 or 0.3) Emergency Procedures after the immediate actions are complete at US discretion.

B - INCORRECT. Performance of 02-OHP-4023-E-0 is required upon the reactor trip, but the operators must continue to perform 02-OHP-4022-064-002 to address the loss of Control Air.

C - INCORRECT. User's Guide allows Abnormal Procedures to be implemented concurrently with Non-Accident (ES-0.1, 0.2 or 0.3) Emergency Procedures.

D - INCORRECT. The Unit Supervisor should direct action of 02-OHP-4023-E-0, first, NOT as time allows. 02-OHP-4023-E-0 actions take priority over 02-OHP-4022-064-002.

Comments:

References:

OHI-4023 Abnormal / Emergency Procedures Users Guide Attachment 2, Step 3.0.

KA - 078000 A2.01
Instrument Air System (IAS)
Ability to (a) predict the impacts of the following malfunctions or operations on the IAS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions
RO - 2.4 SRO - 2.9
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF8.078.A2.01

KA Match:

Question matches KA as it requires candidate to evaluate the impact and mitigate the consequences of an Air Dryer malfunction.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use: NRC 2004, NRC 2010
Original Question:

Associated objective(s):

(RO-C-AOP0490412-T1) Given a set of plant conditions and the occurrence of Control Air System Malfunctions/Loss of Control Air Recovery, without use of references, identify the event, predict the response of the plant with no operator action, and explain required operator actions to mitigate the event per applicable Annunciator Response Procedures and AOP OHP-4022-064-001 & OHP-4022-064-002.

87 **ID: RO-C-00800-E13-2** **Points: 1.00**

The Tech Spec Chemistry technician has just completed sampling the Unit Two RWST for the weekly sample and reports boron concentration of 2660 ppm.

The BOP reports to you the following for Unit Two RWST:

- Temperature – 76.2^{0}F
- Boron – 2660 ppm
- The BOP notes volume to be 89% (378,560 gal)

You will direct the BOP to take what action to restore the RWST?

Procedure Descriptions
12-OHP-4021-005-001 Boric Acid Transfer System Operation
- Attachment 2 – Unit 2 Blender Transfer to Unit Two RWST, BART, or CVCS HUT's

2-OHP-4021-018-008 Operation of RWST Support Systems
- Attachment 3 – Filling Unit 2 RWST from Unit 1 RWST
- Attachment 4 – Operation of Unit 2 RWST Freeze Protection System

A. Restore level using 2-OHP-4021-018-008 Attachment 3.

B. Restore level using 12-OHP-4021-005-001 Attachment 2.

C. Restore boron concentration using 12-OHP-4021-005-001 Attachment 2.

D. Restore temperature using 2-OHP-4021-018-008 Attachment 4.

Answer: C

Answer Explanation:

A. Incorrect Level is within TS requirements. Plausible if candidate believes level is low. This is not correct choice of procedure if level was actually low.

B. Incorrect Level is within TS requirements. Plausible if candidate believes level is low.

C. Correct RWST requires boron to be 2400 - 2600 ppm, temp $\geq$70 F, and level $\geq$375,000 gallons. Based on the reported results the RWST is Inoperable due to boron being above limit. 12-OHP-4021-005-001 Attachment 2 is the correct choice to restore born concentration to within limits

D. Incorrect Temperature is not low. Plausible as the correct procedure is listed to restore temperature if it was actually low.

Comments:

References:

TS 3.5.4, 12-OHP-005-001 Attachment 2 Unit 2 Blender Transfer to Unit Two RWST

KA - 004000 A2.27
Chemical and Volume Control System (CVCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Improper RWST boron concentration
RO - 3.5 SRO - 4.2
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF1.004.A2.27

KA Match:
Question matches KA as it requires the candidate to determine that the RCS boron Concentration is too high and determine the correct procedural (Tech Spec) actions required.

SRO - Facility operating limitations in the TS and their bases. Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1). [55.43(b)(2)]

Cognitive Level: H 3

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-00800-E13) Given ECCS and other plant parameters, determine if any T.S. LCO Action statements are in effect.

88 **ID: RO-C-01350-E4-18** **Points: 1.00**

Unit Two is returning to full power operations after reducing power to repair a leak on a Main Condenser tube. While holding power at 80% for chemistry, Annunciator panel 211 Drop 49 “PPC RMS U2 CT ALARM OR ABNORMAL” alarm is received. The operators determine 2-SRA-2805, Gland Steam Condenser Vent Low Range Noble Gas, has failed low by observing a White indication on PPC RMS.

(Reference provided)

Release via this pathway may continue provided….

A. Flow rate is estimated at least once per 4 hours

B. Grab samples are taken at least once per shift

C. Grab samples are taken once every 24 hours

D. Two independent samples have been analyzed

Answer: B

Answer Explanation:

A. Incorrect	Plausible as this requirement is listed in the Action above correct action in PMP-6010-OSD-001
B. Correct	PMP-6010-OSD-001 Att. 3.4 Item 5.a directs Action #6.
C. Incorrect	Plausible as this sample frequency is listed in PMP-6010-OSD-001
D. Incorrect	Plausible as this is requirement for Inoperable liquid release path monitor

Comments:

References:

PMP-6010-OSD-001 Off-Site Dose Calculation Manual Attachment 3.4

KA - 073000 A2.02
Process Radiation Monitoring (PRM) System
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
RO - 2.7 SRO - 3.2
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF7.073.A2.02

KA Match:
Question matches KA as candidate is presented with a RMS detector failure and must evaluate the impact and actions required to be taken due to failure.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Process for gaseous/liquid release approvals, i.e., release permits. [55.43(b)(5)]

Cognitive Level: H 3

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-01350-E4) Describe the function of the following Radiation Monitoring System Monitors including any automatic actions that occur on a high alarm:

1. SRA-1800(2800) Steam Packing Exhauster (GSLO) Monitor
2. SRA-1900(2900) Steam Jet Air Ejector Vent Monitor

89 **ID: CM-1192A** **Points: 1.00**

Given the following conditions:

- Unit 2 has entered MODE 4 from MODE 3.
- RHR system has just been placed in service.
- One RHR pump is disassembled for maintenance.
- The running RHR pump trips on motor overload and cannot be restarted.
- No AFW pumps are currently running.

Which ONE of the following describes the preferred heat removal method that should be directed by the US in accordance with 2-OHP-4022-017-001, Loss of RHR Cooling?

A. Start AFW pumps to feed Steam Generators for steaming

B. Initiate RCS Bleed and Feed to control RCS temperature

C. Vent RCS through Pressurizer PORVs to inject accumulators

D. Inject cooler RWST water through one CCP

Answer: A

Answer Explanation:

2-OHP-4022-017-001 requires that on a Loss of RHR with no LOCA occurring that Operators attempt to use Steam Generators as first method of heat removal.

A. Correct Step 10 directs establishing Heat Sink using AFW pumps to establish and maintain S/G level while dumping steam via steam dumps or PORV's.
B. Incorrect Plausible as Bleed and Feed is a method used for Core Cooling in EOP network but this is not preferred means per procedure.
C. Incorrect Plausible as Accumulator injection is used to cool and cover the Core in EOP network but this is not preferred means per procedure.
D. Incorrect Plausible as CCP would inject RWST water but this is not preferred means per procedure.

Comments:

References:
02-OHP-4022-017-001 Loss of RHR Cooling

KA - 061000 2.4.9
Auxiliary / Emergency Feedwater (AFW) System
Emergency Procedures/Plan
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.
RO - 3.8 SRO - 4.2
CFR - 41.10 / 43.5 / 45.13
SF4.061.GEN

KA Match:
Question matches KA as candidate is given shutdown condition with a loss of RHR and asked for actions to mitigate condition.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. [55.43(b)(5)]

Cognitive Level: H 3

Question MODIFIED
Previous Use:
Original Question:

Associated objective(s):

(RO-C-AOP0430412-E3) explain the procedural mitigation strategy for a Loss of RHR Cooling

90 **ID: RO-C-EOP01-E25-2** **Points: 1.00**

Given the following conditions on Unit 1:

- Unit is operating at 100% power.
- Pressurizer (PRZ) level is 68% and rising.
- Reactor Coolant System (RCS) pressure is 1960 psig and lowering.
- Pressurizer Relief Tank (PRT) level is 90% and rising.
- PRT pressure is 50 psig and rising.
- PRZ PORV, NRV-153 is showing dual indication after placing the Control switch to close
- PORV Block Valve, NMO-153 will NOT Close.

Which ONE of the following describes the direction given by the US and basis for direction?

A. Trip the reactor and implement 1-OHP-4023-E-0, Reactor Trip or Safety Injection. The PRT Rupture disc will break since the PRT is NOT designed for continuous discharge.

B. Implement 1-OHP-4022-002-009, Leaking Pressurizer Power Operated Relief Valve, to isolate the PRZ PORV. The PRT Rupture disc has broken since the PRT is NOT designed for continuous discharge.

C. Trip the reactor and implement 1-OHP-4023-E-0, Reactor Trip or Safety Injection. The PRT Rupture disc will NOT break since the PRT is designed for continuous discharge.

D. Implement 1-OHP-4022-002-009, Leaking Pressurizer Power Operated Relief Valve, to isolate the PRZ PORV. The PRT Rupture disc will NOT break since the PRT is designed for continuous discharge.

Answer: A

Answer Explanation:

A. Correct — A reactor trip is warranted since pressure is within 10 psig of the setpoint and lowering. UFSAR Section 4.2.2.3: "The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the 100%-power pressurizer water level setpoint. The tank is not designed to accept a continuous discharge from the pressurizer."

B Incorrect — The reactor needs to be tripped and E-0 performed since pressure is only 10 psig above the trip setpoint (Unit 1 setpoint recently changed from 1875 psig and normal pressure from 2085 to 2235). The Rupture Disc will not break until > 100 psig and then pressure lowers to ~ Containment Pressure.

C Incorrect — The PRT is NOT designed for continuous input, so the rupture disc will break

D Incorrect — The reactor needs to be tripped and E-0 performed since pressure is only 10 psig above the trip setpoint. The tank is not designed to accept a continuous discharge from the pressurizer.

Comments:

References:
1-OHP-4023-E-0 Reactor Trip or Safety Injection, UFSAR Section 4.2.2.3

KA - 007000 2.1.27
Pressurizer Relief Tank/Quench Tank System (PRTS)
Conduct of Operations
Knowledge of system purpose and/or function.
RO - 3.9 SRO - 4.0
CFR - 41.7
SF5.007.GEN

KA Match:
Question matches KA based on requiring the candidate to recognize the open PORV, pending PRT rupture, and know the requirement to trip and determine appropriate the emergency procedure to address the issue.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: F 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-EOP01-E25) Describe the Rules of Usage associated with the use of AOPs and ARPs during the implementation of the EOPs.

91 **ID: RO-C-EOP03-E19-22** **Points: 1.00**

Unit 2 was operating at steady state full power when a loss of off-site power occurred. The following were noted during implementation of .2-OHP-4023-E-0, Reactor Trip or Safety Injection:

- WR Neutron flux is less than 5% and lowering
- RTB is closed
- RTA, BYA, and BYB are open
- All Auxiliary Feedwater Pumps are Running
- The following rods indications were observed prior to the RCP bus transfer:

  B8 – 9 steps
  G2 – 7 steps
  K14 – 9 steps
  N11 – 5 steps
  H8 – 11 steps

- ALL Rod Bottom Lights are NOT Lit

After entry into E-0, will the US stay in E-0 or transition to FR-S-1 and what is the Justification?

A. Transfer to FR-S-1 All rods must be inserted < 10 steps to diagnose whether or not the sub criticality safety function is satisfied.

B. Stay in E-0 As long as Wide Range Neutron Flux is less than 5% and lowering, the reactor is shutdown.

C. Transfer to FR-S-1 Without ALL Trip Breakers and rod bottom lights you cannot meet the criteria for reactor trip and must transition to FR-S-1

D. Stay in E-0 As long as you perform procedure 1-OHP-4022-012-005, Dropped Or Misaligned Rod concurrently with E-0.

Answer: B

Answer Explanation:

A. Incorrect Plausible because rod RPI has been lost without the rods being fully inserted. Incorrect as FR-S.1 is used if WR Flux is not less than 5% or if flux is rising.

B. Correct Even without Rod Bottom Lights, a check that Neutron Flux is lowering and Power <5% confirms that the reactor is shutdown.

C. Incorrect Plausible because RTB B remains closed and rod RPI has been lost without the rods being fully inserted. Incorrect as FR-S.1 is used if WR Flux is not less than 5% or if flux is rising.

D. Incorrect Plausible since 1 rod is >10 steps but it would be addressed in the RX trip response vs. the AOP.

Comments:

References:
2-OHP-4023-E-0 Reactor Trip or Safety Injection

KA - 014000 A2.01
Rod Position Indication System (RPIS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Loss of offsite power
RO - 2.8 SRO - 3.3
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF1.014.A2.01

KA Match:
Question matches KA as it presents candidate with a Loss of Offsite Power, Rod Position Indication, and reactor parameters and requires them to make a decision on how / where to proceed.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: H 2

Question MODIFIED
Previous Use: NRC 2014-99 Modified
Original Question:

Associated objective(s):

(RO-C-EOP03-E19) For each of the E-0 series EOPs discuss the basis or reason for all Steps.

92 **ID: NRCAUDIT07-0736** **Points: 1.00**

Rod Control was in AUTO with Unit 1 power at 79% when MPC-253, Turbine Impulse Pressure Channel 1, failed low. The crew determined that a Turbine Runback was NOT in progress and placed rod control in Manual.

Which method will the US direct the operators to use to "Restore Equilibrium Conditions" in accordance with 1-OHP-4022-012-003, Continuous Control Bank Movement?

A. Initiate boration.

B. Reduce Turbine Load.

C. Insert Control Rods.

D. Initiate dilution.

Answer: B

Answer Explanation:

Per 1-OHP-4022-012-003, Continuous Control Bank Movement, a reduction in steam demand is the preferred method to restore T_{avg} to T_{ref}. The failure of MPC-253 would have caused the control Rods to Insert (T_{ref} lowered).

A. Incorrect	Rods inserted and T_{avg} would be low for the current power level. Plausible if candidate does not understand reactivity conditions.
B. Correct	Reducing Turbine Load would raise T_{avg}.
C. Incorrect	Rods inserted and T_{avg} would be low for the current power level. Plausible if candidate does not understand reactivity conditions.
D. Incorrect	The procedure prefers a reduction in turbine load to correct a low T_{avg}. Plausible because a dilution would raise T_{avg}.

Comments:

References:
01-OHP-4022-012-003 Continuous Control Bank Movement

KA - 045000 A2.11
Main Turbine Generator (MT/G) System
Ability to (a) predict the impacts of the following malfunctions or operations on the MT/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Control problems in primary, e.g., axial flux imbalance; need to reduce load on secondary
RO - 2.4 SRO - 2.9
CFR - 41.5 / 43.5 / 45.3 / 45.13
SF4.045.A2.11

KA Match:
Question matches KA as it requires the candidate to determine the failure impact on the plant evaluating the automatic control systems impact and the detailed procedural knowledge required to restore equilibrium conditions.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use: 2006 NRC
Original Question:

Associated objective(s):

(RO-C-AOP0200412-E3) explain the procedural mitigation strategy for Continuous Control Bank Movement

93 **ID: NRCAUDIT07-0456** **Points: 1.00**

The following plant conditions exist:

- Unit 1 in refueling
- A leak has developed which has caused level to drop in the Spent Fuel Pit
- The Spent Fuel Pit was initially at normal level and area radiation was at 7 mrem / hr
- After 20 minutes the Spent Fuel Pit level has decreased further and area radiation is 18 mrem / hr

Which ONE of the following operator responses will be directed based on these conditions?

A. Begin makeup to the Spent Fuel Pit with the most immediately available source of water whether borated or not.

B. Ensure fuel transfer cart is in containment to permit closure of the transfer tube valve.

C. Close the transfer tube valve to isolate the containment from the Spent Fuel Pit.

D. Ensure Spent Fuel Pit cooling and skimmer system is turned on.

Answer: C

Answer Explanation:

A. Incorrect	Make up will be directed after the isolation and only borated water sources will be used.
B. Incorrect	The transfer cart must be on the Spent Fuel Pit side to close the Transfer Tube valve.
C. Correct	OHP 4022.002.006, "Loss of Refueling Water Level during Refueling Operations" directs the closing of the Transfer Tube Gate valve to isolate the Spent Fuel Pit from the Refueling Cavity.
D. Incorrect	Spent Fuel pit cooling is checked in the Loss of Spent Fuel Pit Cooling not Loss of Level.

Comments:

References:

1-OHP 4022.002.006 Loss of Refueling Water Level During Refueling Operations

KA - 034000 2.4.35
Fuel Handling Equipment System (FHES)
Emergency Procedures/Plan
Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
RO - 3.8 SRO - 4.0
CFR - 41.7 / 41.10 / 43.5 / 45.13
SF8.034.GEN

KA Match:
Question matches KA as candidate must know required local actions given conditions during an emergency while fuel transfer is in progress.

SRO - Fuel handling facilities and procedures. Refuel floor SRO responsibilities. [55.43(b)(7)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-01800-E10) Describe the mitigating strategy for a loss of Spent Fuel Pit Cooling as outlined in 12-OHP-4022-018-001.

94 **ID: RO27AUDIT-94** **Points: 1.00**

Given the following conditions:

- A tornado warning has been issued for the Cook Plant area
- The SRO has entered 12-OHP-4022-001-010, Severe Weather

During procedure implementation with offsite power still available, the SRO will direct the operators to ensure all EDGs are ______(1)_________ to _________(2)___________.

	(1)	(2)
A.	running in parallel with offsite power source	maintain positive control of electrical power to the plant
B.	running and parallel to safeguards bus	allow isolation of safeguards bus from the RCP bus
C.	stopped and in standby	prevent a switchyard fault from damaging the EDG
D.	stopped and in standby	ensure that diesels are only started following verification of no damage

Answer: C

Answer Explanation:

A. Incorrect EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

B. Incorrect EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

C. Correct EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

D. Incorrect EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

Comments:

References:

12-OHP-4022-001-010 Severe Weather

KA - 194001 2.1.6
Generic
Conduct of Operations
Ability to manage the control room crew during plant transients.
RO - 3.8 SRO - 4.8
CFR - 41.10 / 43.5 / 45.12 / 45.13
P2.1.6

KA Match:
Question matches KA because the candidate must determine what direction to give the operators concerning the EDGs during a tornado warning.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK - REPEAT
Previous Use: NRC 2012
Original Question:

Associated objective(s):

(RO-C-ADM06-E5) Given the occurrence of a Natural or Destructive Phenomena Inside a Vital Area or Severe Weather condition, determine the appropriate actions in accordance with applicable plant procedures.

95 **ID: RO-C-ADM07-E2-1** **Points: 1.00**

Manually blocking an alarm is ONLY allowed when the annunciator alarm is considered blocked due to which of the following?

A. Standing alarm with multiple inputs.

B. Plant conditions impact ability of the alarm to perform its' intended function.

C. Nuisance alarm requiring Operator attention to attend to alarm.

D. Alarm is degraded but capable of alerting Operators of an off normal condition.

Answer: C

Answer Explanation:

A. Incorrect See explanation for correct answer.
B. Incorrect See explanation for correct answer.
C. Correct PMP-4043-APC-001 NOTE prior to Section 3.6.8 states manually blocking an alarm is NOT required when the annunciator alarm is considered blocked due to any of the following:

a. Standing alarm with multiple inputs.

b. Plant conditions impact ability of the alarm to perform its' intended function.

c. Alarm is degraded but capable of alerting Operators of an off normal condition.

D. Incorrect See explanation for correct answer.

Comments:

References:

PMP-4043-APC-001 Abnormal Position Control NOTE prior to Section 3.6.8

KA - 194001 2.2.43
Generic
Equipment Control
Knowledge of the process used to track inoperable alarms.
RO - 3.0 SRO - 3.3
CFR - 41.10 / 43.5 / 45.13
P2.2.43

KA Match:

Question matches KA as candidate is required to demonstrate knowledge of process for blocking and tracking inoperable alarms IAW station procedures.

SRO - Facility licensee procedures required to obtain authority for design and operating changes in the facility. Administrative processes for disabling annunciators. [55.43(b)(3)]

Cognitive Level: F 2

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM07-E2) Given PMP-4043-APC-001 describe the requirements to block an alarm.

96 **ID: RO-C-ADM02-E9-2** **Points: 1.00**

While on tour an operator discovered that 12-FP-699, Fire Protection Pump House Recirc to North Fire Protection Water Storage Tank Inlet Shutoff Valve, was missing its seal.

The last operation performed on the Fire Protection system was when the previous shift switched FROM the North FP Water Storage Tank in service TO the South FP Water Storage Tank being in service. The South tank remains in service now.

You should instruct the operator to verify that 12-FP-699 is in the __(1)__ position and to seal it in that position with a seal that is __(2)__.

A. 1. Open
   2. Orange Plastic

B. 1. Open
   2. Red Plastic

C. 1. Closed
   2. Orange Plastic

D. 1. Closed
   2. Red Plastic

Answer: D

Answer Explanation:

A. Incorrect Plausible because 12-FP-699 would be sealed OPEN if the North Fire Protection Water Storage Tank was in service, and Common Unit (12-) components are normally sealed with an orange plastic seal. Incorrect because the South Fire Protection Water Storage Tank is in service, and red plastic seals are used on Fire Protection valves regardless of their Unit designator.

B. Incorrect Plausible because 12-FP-699 would be sealed OPEN if the North Fire Protection Water Storage Tank was in service, and this Fire Protection valve should be sealed with a red plastic seal. Incorrect because the South Fire Protection Water Storage Tank is in service.

C. Incorrect Plausible because 12-FP-699 should be sealed CLOSED with the South Fire Protection Water Storage Tank in service, and Common Unit (12-) components are normally sealed with an orange plastic seal. Incorrect because red plastic seals are used on Fire Protection valves regardless of their Unit designator.

D. Correct PMP-4043-SLV-001, Sealed/Locked Valves, lists the required position for 12-FP-699 as TBD* which requires the US/SRO to determine the correct position for current plant conditions. SLV-001 refers to the Initial Lineup procedure for this valve, 12-OHP-4021-066-001, Fire Protection System (Water) Operation, which directs 12-FP-699 to be SEALED CLOSED when the South Fire Protection Water Storage Tank is in service (Att. 16 Step 4.1; LUS 3 or 5). SLV-001 designates 12-FP-699 as an "FP" valve, requiring a red plastic seal IAW Steps 3.1.6 and 3.1.11.

Comments:

References:
PMP-4043-SLV-001 Sealed / Locked Valves

KA - 194001 2.2.14
Generic
Equipment Control
Knowledge of the process for controlling equipment configuration or status.
RO - 3.9 SRO - 4.3
CFR - 41.10 / 43.3 / 45.13
P2.2.14

KA Match:
Question matches KA as it requires candidate to have knowledge of station procedure on actions for a valve with a missing seal.

SRO - Facility licensee procedures required to obtain authority for design and operating changes in the facility. This is an SRO ONLY question because PMP-4030-SLV-001 requires the US/SRO to determine the correct position of 12-FP-699 according to plant conditions. [55.43(b)(3)]

Cognitive Level: F 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM02-E9) Describe the actions to be taken when a valve seal is found to be broken or missing.

97 **ID: 2008NRC-0603** **Points: 1.00**

While preparing a release permit for a waste monitor tank, it is determined that RFS-1010 (Liquid Waste Effluent Sample Flow) switch failed HIGH. It is estimated that repairs will take at least 3 days.

(Reference Provided)

The release:

A. may NOT be approved until the flow monitor is restored to OPERABLE.

B. may be approved since the effluent monitor (RRS-1001) is OPERABLE.

C. may be approved for up to 30 days provided the flow rate is estimated at least once per 4 hours during the actual release.

D. may be approved after at least two independent samples are analyzed and at least two qualified persons independently verify the discharge valve lineup.

Answer: D

Answer Explanation:

A. Incorrect Plausible because of the desirability of monitoring a liquid release for radioactivity. Delaying the release would be the correct choice if there was no contingency available for monitoring, but the release may continue if 2 samples are taken and the flowpath is dual verified.

B. Incorrect Plausible since RRS-1001 is not directly affected by the failure of RFS-1010, but IAW the Offsite Dose Calculation Manual, RFS-1010 is required for Operability of RRS-1001.

C. Incorrect Plausible because these are the required steps for ODCM Attachment 3.2 Action 4, which would apply to RFI-285, the flow measurement device in the liquid release line immediately downstream of RFS-1010.

D. Correct The operator is required to interpret the impact of RFS-1010 being failed high in order to determine the operability of RRS-1001. After determining that RRS-1000 is INOPERABLE, the appropriate actions of the ODCM must be applied to the liquid release process of OHP-4021-006-004, Datasheet 1 and Attachment 3. A release may continue if 2 samples are taken and the flowpath is dual verified.

Comments:

References:
PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.2

Attachment Provided: PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.2

KA - 194001 2.3.6
Generic
Radiation Control
Ability to approve release permits.

RO - 2.0 SRO - 3.8
CFR - 41.13 / 43.4 / 45.10
P2.3.6

KA Match:
Question matches KA as it requires candidate to demonstrate ability to determine requirements to approve a release with an inoperable flow instrument.

SRO - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. Process for gaseous/liquid release approvals, i.e., release permits. [55.43(b)(4)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-ADM10-E5) Given a copy of the ODCM and a set of plant conditions, verify the limits of operation are satisfied.

98 **ID: 2008NRC-0632** **Points: 1.00**

A LOCA that resulted in significant core damage occurred at 1600 hours. Containment Pressure and Radiation levels were recorded every 30 Minutes as follows:

Time	Radiation (R/Hr)	Pressure (psig)
1600	400,000	6.2
1630	400,000	6.2
1700	400,000	5.6
1730	300,000	5.2
1800	300,000	4.5
1830	90,000	4.0

At 1835 hours, while performing Emergency Operating Procedures, a step is encountered which states "Check PRZ level - GREATER THAN 20% [24% ADVERSE]".

Which ONE of the following describes the required Pressurizer level to be verified and why?

A. 20% because adverse values are no longer required because of the limited integrated dose and pressure reduction.

B. 24% because adverse values must be used until evaluated for lasting effects because the integrated dose limit has been exceeded.

C. 24% because adverse containment exists due to the current containment radiation dose rate.

D. 24% because adverse containment exists due to the current containment pressure.

Answer: A

Answer Explanation:

A. Correct — Adverse containment values are required to be used when containment pressure is >5 psig or >10^5 R/Hr. When pressure lowers to <5 psig normal values may be used as long as the integrated dose is <10^6 R. The integrated dose is low enough to allow normal values to be used.

B. Incorrect — The integrated dose is low enough to allow normal values to be used.

C. Incorrect — The current Dose Rate is <10^5 R/Hr.

D. Incorrect — Pressure is <5 psig.

Comments:

References:
OHI-4023, Abnormal / Emergency Procedure User's Guide, Attachment 2, Step 6,

KA - 194001 2.3.5
Generic
Radiation Control
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personal monitoring equipment, etc.
RO - 2.9 SRO - 2.9
CFR - 41.11 / 41.12 / 43.4 / 45.9
P2.3.5

KA Match:
Question matches KA as it requires candidate to demonstrate ability to use rad monitor readings to determine the condition of containment for using adverse containment values while implementing the EOPs.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. [55.43(b)(3) & (5)]

Cognitive Level: H 3

Question BANK

Previous Use: NRC 2004 Modified question by lowering dose rates so that the integrated dose becomes <105 R changing the correct answer to A.

Original Question:

Associated objective(s):

(RO-C-EOP01-E8) State the conditions that define "Adverse Containment".

99 **ID: RO-C-AS17-T1-2** **Points: 1.00**

The Fire Protection Team Leader has called the Shift Managers office to inform Operations that they will be disabling 2 of the fire detectors in the EDG Ramp hallway for 24 hours to allow painting to occur.

(Reference provided)

Per U2 TRM which choice below lists required actions the US will direct to be taken?

A. Inspect the zone(s) with the inoperable detectors using a fire watch hourly.

B. Establish a continuous fire watch patrol of the affected area.

C. Inspect the zone(s) with the inoperable detectors using a fire watch hourly and verify that Monitoring Program Action Levels have not been exceeded.

D. Verify $\geq$ 1/2 of the total Function A detectors in the zone are OPERABLE and verify no two adjacent fire detectors in the affected area are inoperable

Answer: D

Answer Explanation:

A. Incorrect Action for Function B detector.
B. Incorrect Not required for given conditions of < 14 days
C. Incorrect Action for Function B detector.
D. Correct TRM 8.7.4 Condition A requires: Verify $\geq$ 1/2 of the total Function A detectors in the zone are OPERABLE - AND - verify no two adjacent detectors in the affected zones are inoperable - AND - restore detectors to operable status in 14 days. If not completed Condition C must be entered which requires establishing a continuous fire watch patrol of the affected area

Comments:

References:

2- TRM 8.7.4

Provide U2 TRM 8.7.4

KA - 194001 2.4.25
Generic
Emergency Procedures/Plan
Knowledge of fire protection procedures.
RO - 3.3 SRO - 3.7
CFR - 41.10 / 43.5 / 45.13
P2.4.25

KA Match:
Question matches KA as it requires candidate to have knowledge of procedures (Tech Spec) dealing with fire protection.

SRO - Conditions and limitations in the facility license. Administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc. [55.43(b)(1)]

Cognitive Level: H 3

Question NEW
Previous Use:
Original Question:

Associated objective(s):

(RO-C-AS17-T1) Operate and monitor the Fire Water System in accordance with plant procedures

100 **ID: CM-0980** **Points: 1.00**

During power operations on Unit One, a leak develops on the Reactor Coolant System (RCS). While responding to the leak, the Control Room operators determine that the RCS leak rate is 8 gpm and also discover the following conditions:

- East CCW RMS radiation indication is rising
- West CCW RMS radiation indication is rising
- East CCW Surge Tank level indication is rising
- West CCW Surge Tank level indication is rising

The US should direct the operators to:

A. initiate a controlled shutdown per 1-OHP-4021-001-003, POWER REDUCTION procedure.

B. trip the reactor and go to 1-OHP-4023-E-0, REACTOR TRIP OR SAFETY INJECTION procedure.

C. perform the actions of 1-OHP-4022-016-001, CCW OUT-LEAKAGE/ MALFUNCTION OF THE CCW SYSTEM procedure.

D. perform the actions of 1-OHP-4022-016-003, CCW IN-LEAKAGE procedure.

Answer: D

Answer Explanation:

A. Incorrect The RCS Excessive Leakage procedure will direct a transition to the CCW In-Leakage procedure to attempt to isolate the leak. Plausible because an unisolable leak above TS limit would require a plant shutdown.

B. Incorrect Tripping the reactor would be directed by the procedure for an unisolable leak which exceeded the capacity of the normal charging line. Plausible if the student believes that this is a conservative decision, especially if the CCW information is incorrectly analyzed as a confirmed leak on both CCW headers.

C. Incorrect CCW Out-leakage procedure deals with leaks out of the CCW system or with loss of a CCW pump. Plausible if student believes that this procedure deals with leaks out of the RCS into the CCW system.

D. Correct The RCS Excessive Leakage procedure will direct transition to CCW In-Leakage to attempt to isolate the leak.

Comments:

References:

1-OHP-4022-002-020, Excessive RCS Leakage

KA - 194001 2.4.11
Generic
Emergency Procedures/Plan
Knowledge of abnormal condition procedures.
RO - 4.0 SRO - 4.2
CFR - 41.10 / 43.5 / 45.13
P2.4.11

KA Match:

Question matches KA as it presents a CCW in leakage condition and requires the candidate to identify station procedural requirements.

SRO - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. [55.43(b)(5)]

Cognitive Level: H 3

Question BANK
Previous Use:
Original Question:

Associated objective(s):

(RO-C-AOP0160412-E3) explain the procedural mitigation strategy for Excessive RCS Leakage