

Draft Submittal

**SURRY EXAM**  
**50-280, 50-281/2004-301**

**FEBRUARY 24 - MARCH 2**  
**& MARCH 4, 2004 (WRITTEN)**

1. Senior Reactor Operator Written Exam

**U.S. Nuclear Regulatory Commission  
Site-Specific  
DRAFT SRO Written Examination**

**Applicant Information**

Name:	
Date:	Facility/Unit: Surry Nuclear Plant
Region: II	Reactor Type: W
Start Time:	Finish Time:

**Instructions**

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.88 percent overall, with a 70.80 percent or better on the SRO-only items if given in conjunction with the RO exam; SWO-only exams given alone require an 80.00 percent to pass. You have eight hours to complete the combined examination, and three hours if you are only taking the SRO portion.

**Applicant Certification**

All work done on this examination is my own. I have neither given nor received aid.

\_\_\_\_\_  
Applicant's Signature

**Results**

RO / SRO-Only / Examination Values:      \_\_\_\_ / \_\_\_\_ / \_\_\_\_ Points

Applicant's Scores:      \_\_\_\_ / \_\_\_\_ / \_\_\_\_ Points

Applicant's Grades:      \_\_\_\_ / \_\_\_\_ / \_\_\_\_ Percent

1. 003K4.03 001/2/1/RCP LUBRICATION/MEM 2 5/2 8/N/SR04301/R/MAB/SDR

Which ONE of the following correctly describes the Reactor Coolant Pump (RCP) bearing oil lift system?

- A. The oil lift pump discharge pressure must be greater than 350 psig prior to RCP start. Once the RCP reaches operating speed the thrust runner circulates oil in the upper and lower bearing assemblies.
- B. The oil lift pump discharge pressure must be greater than 300 psig prior to RCP start. Once the RCP reaches operating speed the RCP Oil Lift System supplies the bearing lubrication.
- C. The oil lift pump discharge pressure must be greater than 350 psig prior to RCP start. Once the RCP reaches Operating speed the RCP Oil Lift System supplies the bearing lubrication.
- D. The oil lift pump discharge pressure must be greater than 300 psig prior to RCP start. Once the RCP reaches operating speed the thrust runner circulates oil in the upper and lower bearing assemblies.

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References:

ND-88.1-LP-6, Reactor Coolant Pumps, Rev. 16

Distractor Analysis:

- A. Correct because there is a 350 psig discharge interlock with respective RCP. The Oil Lift Pump ensures adequate lubrication upon RCP start, but once the pump reaches operating speed, the thrust runner acts as an oil pump and circulates oil in the upper and lower bearing assemblies.
- B. Incorrect because pressure interlock is at 350 psig, not 300 psig.
- C. Incorrect because thrust runner circulates oil in upper and lower reservoir, not the Oil Lift System.
- D. Incorrect because pressure interlock is at 350 psig, not 300 psig.

003 Reactor Coolant Pumps

K4.03: Knowledge of RCPs design feature(s) and / or interlock(s) which provide for the following: Adequate lubrication of the RCP.

2. 004A2.17 001/2/1/CVCS/C/A 3.4/3.7/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Operating at 85% power
- Pressurizer pressure control is in its normal configuration
- A Pressurizer Safety Valve is leaking
- 1C-B8, PWZR LO PRESS, annunciates
- 1-AP-31.00, increasing or Decreasing RCS Pressure, has been entered

Which ONE of the following correctly describes the affect on charging flow and an appropriate mitigating action in accordance with 1-AP-31.00?

- A. Charging flow initially increases. Place the PRZR PRESS MASTER CNTRL in MANUAL and increase the demand to try to stop the pressure decrease.
- B. Charging flow initially decreases. Place the PRZR PRESS MASTER CNTRL in MANUAL and increase the demand to try to stop the pressure decrease.
- C. Charging flow initially increases. Place the PRZR PRESS MASTER CNTRL in **MANUAL** and decrease the demand to try to stop the pressure decrease.
- D. Charging flow initially decreases. Place the PRZR PRESS MASTER CNTRL in MANUAL and decrease the demand to try to stop the pressure decrease.

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References:

ND-93.3-LP-5, Pressurizer Pressure Control, Rev. 9

ND-88.3-LP-2, Charging and Letdown, Rev. 10

1-AP-31.00, Increasing or Decreasing WCS Pressure, Rev. 4

Distractor Analysis:

- A. Incorrect because increasing the demand will lower pressure, not increase it.
- B. Incorrect because charging flow will not initially decrease and increasing the demand will lower pressure, not increase it.
- C. Correct because charging flow will initially increase due to the sudden pressure drop in the RCS. Also, decreasing the demand on the controller while in manual will act to try to raise pressure.
- D. Incorrect because charging flow will not initially decrease.

004 Chemical and Volume Control

A2.17: 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: Low PZR pressure.

3. 005K5.02 001/2/1/RHR/MEM 3.4/3.5/B/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- RCS level is 12.5 feet on 1-RC-LI-100A
- RCS level is 12 feet 5 inches on 1-RC-LR-105
- A loss of decay heat removal has occurred and 1-AP-27.00, Loss of Decay Heat Removal Capability, has been entered.
- The RHR system has just been made available.

Which ONE of the following methods per 1-AP-27.00 should be used to sweep air from the RHR lines during a loss of decay heat removal capability if inadequate time exists to completely vent the RHW System prior to boiling in the core?

- A. Refill the RCS to 13.5 feet, verify 10°F subcooling, and run an RHR pump at a flow rate of > 2950 gpm.
- B. Maintain RCS level at 12.5 feet, verify subcooling, and run an RHR pump at a flow of > 2950 gpm.
- C. Maintain RCS level at 12.5 feet, verify subcooling, and run an RHR pump at a flow of < 2950 gpm.
- D. Close RH-MOV-1720A and B, RHR Outlets, then open "A" and "C" Safety Injection Accumulator Isolation MOVs.

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References:

- ND-88.2-LP-1, Residual Heat Removal System Description, Rev. 8
- ND-88.2-LP-2, Operation of Residual Heat Removal System, Rev. 15
- ND-88.2-LP-3, Draindown and Midloop Operations, Rev. 12
- 1-AP-27.00, Loss of Decay Heat Removal Capability, Rev. 10

Distractor Analysis:

- A. Correct because based on procedural Note in 1-AP-27.00, Page 16 or 19, Rev. 10.
- B. Incorrect because RCS needs to be filled to 13.5 feet.
- C. Incorrect because RCS needs to be filled to 13.5 feet. Also flow needs to be greater than 2950 gpm.
- D. incorrect because no procedural guidance exists to support the actions.

Surry ILT Exam Bank Question #275

005 Residual Heat Removal

K5.02: Knowledge of the operational implications of the following concepts as they apply to the RHRs: Need for adequate subcooling.

- Steam Generator levels are 20% and rising
- Subcooling **based** on CETCs is 0 °F
- E-0, Reactor Trip or Safety Injection, has been exited and **Safety** Function Status Trees are being monitored
- RCP Seal Injection flow is 3 gpm to all RCPs
- RCP Seal delta-Ps are all approximately 200 psid
- Source Range Startup Rate is zero
- Attempts to establish HHSI flow have failed

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RHR

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References:

ND-95.3-LP-38, FR-C.1 Response to Inadequate Core Cooling, Rev. 8

FR-C.1, Response to Inadequate Core Cooling, Rev. 18

Distractor Analysis:

- A. Incorrect because  $1.0 \times 10^5$  PPH is well below the MSIV closure setpoint and does not even approach the maximum rate (an entire order of magnitude low).
- B. Incorrect because  $1.0 \times 10^5$  PPH is well below the MSIV closure setpoint and does not even approach the maximum rate (an entire order of magnitude low).
- C. Incorrect because RCPs should be **started even** when normal conditions **not** met.
- D. Correct because procedural guidance exists to support the actions. MSIV closure **will** occur if flow **is** greater than  $1.0 \times 10^6$  PPH. The purpose for the actions **is to** establish low head flow from accumulators and LHSI. RCP support criteria **is** desirable, but not a prerequisite for starting RCPs.

006 Emergency Core Cooling

**K.6.03:** Knowledge of **the** effect of a loss or malfunction on the following will have on ECCS: **Safety** Injection Pumps.

5. 007EK2 02 001/1/1/BREAKER REACTOR TRIP/C/A 2 6/2.8/B/SR04301/R/MAB/SDR

The following conditions exist:

- Unit 1 is at 90% power
- Reactor protection testing **is** in progress
- Reactor Trip Breaker "A" **is closed**
- Reactor Trip Breaker "B" **is open**
- Reactor Trip Bypass Breaker "B" is racked in and closed

Which ONE of the following describes the plant response if reactor trip bypass breaker "A" is racked in and closed?

- A. Both reactor trip bypass breakers "A" and "B" and reactor trip breaker "A" will trip open and the reactor will trip.
- B. Only reactor trip bypass breakers "A" and "B" will trip **open** and the reactor will trip.
- C. Reactor trip breaker "A" will trip open and the plant will remain at 90% power.
- D. Reactor trip bypass breaker "A" will trip open and the plant will remain at 90% power.

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References:

ND-93.3-LP-17, **AMSAC**, Rev. 10

ND-93.3-LP-10, Reactor Protection - General, Rev. 5

Distractor Analysis:

- A. Incorrect because reactor trip breaker "A" will not open.
- B. Correct because this is the correct response per ND-93.3-LP-10.
- C. Incorrect because reactor trip breaker "A" will not open and plant will trip.
- D. Incorrect because the plant will trip.

Surry ILT Bank Question #1667

007 Reactor Trip Stabilization

EK2.02: Knowledge of the interrelationships between a reactor trip and the following: Breakers, relays, and disconnects.

6. 007K5.02 001/2/1/PRT PZR BUBBLE/C/A 3.1/3.4/B/SR04301/R/MAB/SDR

Given the following Unit 1 conditions:

- A heatup is in progress to return to power from a cold shutdown condition
- RCS is filled and vented
- Pressurizer is solid
- A nitrogen blanket has been established on the PRT
- PRT Level = 95%
- Pressurizer Heaters are energized

Which ONE of the following must be accomplished prior to drawing a bubble in the Pressurizer?

- A. Drain the PRT to 60 - 80%.
- B. Verify VCT oxygen concentration **less** than 3%.
- C. Drain the Pressurizer to 22.2%.
- D. Pressurize the **WCS** to 200 - 270 psig on PI-1-403, Nar Range.

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References:

- 1-GOP-1.1, Unit Startup, RCS Heatup from Ambient to 195 Degrees F., Rev. 25
- 1-OP-RC-011, Pressurizer Relief Tank Operations, Rev. 13

Distractor Analysis:

- A. Correct because GOP-1.1 **Step 5.5.4** directs establishment of normal **PWT** level prior to drawing a bubble. OP-RC-011 Step 5.1.1 states the normal PRT level to be 60 - 80%.
- B. Incorrect because GOP-1.1 Step 5.5.6 requirement is to verify VCT oxygen < 2%.
- C. Incorrect because this **is** an action following establishment of drawing a bubble (GOP-1.1, Step 5.5.13).
- D. Incorrect because RCS should be between 300 and 370 psig on PI-1-403.

007 Pressurizer Relief / Quench Tank

K5.02: Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR.

7. 008AA2.06 001/1/1/PRESSURE TRANSMITTER/C/A 3.3/3.6/N/SR04301/R/MAB/SDR

Given the following **Unit 1** conditions:

- Reactor power = 180%
- All other parameters are at normal steady state values
- Subsequently PT-444 fails high

Assuming no operator action is taken, which ONE of the following is correct?

- A. PORV-1455C opens, pressure decreases to 2000 psig, PORV-1455C closes, and pressure stabilizes around 2000 psig.
- B. PORV-1456 opens, pressure decreases to 2000 psig, PORV-1456 closes, and pressure stabilizes around 2000 psig.
- C. PORV-1455C opens, at 2000 psig PORV-1455C closes; however, pressure will continue to decrease causing a reactor trip and safety injection.
- D. PORV-1456 opens, at 2000 psig PORV-1456 closes; however, pressure will continue to decrease causing a reactor trip and safety injection.

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References:

ND-93.3-LP-5, Pressurizer Pressure Control, Rev. 9

Distractor Analysis:

- A. incorrect because both spray valves also open, which causes pressure to continue to decrease.
- B. Incorrect because both spray valves open, which causes pressure to continue to decrease. Also incorrect because PORV-1456 does not open.
- C. Correct because both spray valves open causing a reactor trip on OT-delta-T or Low Pressurizer Pressure, followed by SI.
- D. Incorrect because PORV-1456 does not open.

008 Pressurizer Pressure Control

AA2.03: Ability to determine and interpret the following as they apply to the pressurizer vapor space accident: PORV logic control under low-pressure conditions.

8. 008K1.02 001/2/1/CCW/MEM 3.3/3.4/B/SR04301/R/MAB/SDR

Which ONE of the following correctly describes loads cooled by the Component Cooling Water (CCW) System or subsystem of CCW?

- A. RCP bearing lube oil coolers, neutron shield tank coolers, RCP seal water return cooler, outside recirc spray pump seals.
- B. HHSI pump seals, LHSI pump seals, RHR pump seals, RCP motor air coolers.
- C. RHR pump seals, RCP bearing lube oil coolers, neutron shield tank coolers, HHSI pump seals.
- D. LHSI pump seals, RHR pump seals, RCP motor air coolers, neutron shield tank coolers.

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#### Reference:

ND-88.5-LP-1, Component Cooling, Rev. 19  
ND-88.3-LP-5, Charging System, Rev. 16

#### Distractor Analysis:

- A. Incorrect because outside recirc spray pump seals are not cooled by CC.
- B. Incorrect because LHSI pump seals are not cooled by CC.
- C. Correct because all are cooled by CC or a subsystem.
- D. Incorrect because LHSI pump seals are not cooled by CC.

Requal Bank Question #527

008 Component Cooling

K1.02: Knowledge of the physical connections and / or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS.

9. 008K4 01 001/2/1/COMPONENT COOLING/MEM 3.1/3.3/B/SR04301/R/MAB/SDR

Ten seconds after a Safety Injection occurs, the "A" Component Cooling Pump trips. Which ONE of the following describes the operation of the CC pumps?

- A. The "B" CC pump will not auto start without a required operator action.
- B. The "B" CC pump will auto start 60 seconds after the "A" CC pump trips.
- C. The "B" CC pump will auto start as soon as the "A" CC pump trips.
- D. The "B" CC pump will auto start 50 seconds after the "A" CC pump trips.

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References:

ND-88.5-LP-1, **Component** Cooling Water System, Rev. 19.

Distractor Analysis:

- A. Correct because Auto Start Inhibit due to SI will prevent auto start of the CC pump, but the pump may be manually started at any time.
- B. Incorrect because the Auto Start Inhibit will block the auto start.
- C. Incorrect because the Auto Start Inhibit will block the auto start.
- D. Incorrect because the Auto Start Inhibit will block the auto start.

ILT Bank Question # 537

008 Component Cooling Water System

**K4.81:** Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump.

10. 010A101 001/2/1/BORON PZR SPR Y/C/A 2.8/2.9/N/SRO 301/R/MAB/SDR

The following plant conditions exist:

- Dilution to criticality has just been completed
- Operators note that inadequate proportional heaters appear to be energized
- Pressurizer Pressure is 2230 psig.

Which ONE of the following could result from inadequate Pressurizer Heater output during a dilution to criticality?

(Assume all other controls and components working properly in their normal configuration.)

- A. Boron concentration will be higher in the Pressurizer than in the RCS.
- B. Boron concentration will be lower in the Pressurizer than in the RCS.
- C. Pressurizer and RCS boron concentration will be approximately equal.
- D. The Pressurizer Spray Nozzle will be susceptible to thermal shock.

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References:

1-GOP-1.1, Unit Startup, RCS Heatup From Ambient to 195 Degrees F, Rev. 25

Distractor Analysis:

- A. Correct because RCS boron will be lower due to the dilution. The Pzr will still be at a higher boron concentration until spray flow has created enough out-surge to adequately equalize the boron with the RCS. (Lack of heaters creates lack of sprays.)
- B. Incorrect because boron concentration will be higher in the Pzr.
- C. Incorrect because the lack of heater output will not allow for adequate mixing.
- D. Incorrect because the bypass spray valves are normally open, which is sufficient to prevent thermal shock. (Have utility verify that this is in fact the normal configuration.)

010 Pressurizer Pressure Control

A1.01: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Pzr PCS controls including: PZR and RCS boron concentration.

11. 011EA1.13 001/1/1/SAFETY INJECTION/C/A 4.1/4.2/N/SR04301/R/MAB/SDR

Given the following conditions:

- LOCA has occurred
- RCS subcooling is 0 °F
- RWST Level = 15% and slowly decreasing
- Containment Pressure = 9 psig and decreasing
- Safety Injection Actuation has been reset

Which ONE of the following is the correct action to be taken?

- A. Close Charging Pump Miniflow Recirc Valves. With RWST level at 15%, **push** both RMT pushbuttons for each train if automatic transfer does not occur.
- B. Close Charging Pump Miniflow Recirc Valves. When RWST level reaches 13%, push both RMT pushbuttons for each train if automatic transfer does not occur.
- C. With RWST **level** at 15%, push both RMT pushbuttons for each train if automatic transfer does not occur. **Secure** Containment Spray **Pumps** immediately *following* verification of Phase 1 and 2 RMT.
- D. With RWST level at 13%, push both RMT pushbuttons for each train if automatic transfer does not occur. Secure Containment Spray Pumps immediately following verification of Phase 1 and 2 RMT.

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References:  
ND-95-3-LP-7, E-1 Loss of Reactor or Secondary Coolant, Rev. 14  
1-E-1, Loss of Reactor or Secondary Coolant, Rev. 21  
1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 12

Distractor Analysis:

- A. Incorrect because RMT transfer should not occur at 15%. It will occur at 13.5% and procedurally should be verified at 13%.
- B. Correct because verifications should be made at 13% and manually initiated if needed (directed by ES-1.3).
- C. Incorrect per ES-1.3 spray pumps should not be secured until they show signs of cavitation. Also E-1 does not call for spray to be secured until containment pressure is less than 12 psia. Also, RMT transfer should not occur at 15%.
- D. Incorrect per ES-1.3 spray pumps should not be secured until they show signs of cavitation. Also E-1 does not call for spray to be secured until containment pressure is less than 12 psia.

Surry ILT Bank Question # 872

011 Large Break LOCA

EA1.13: Ability to operate and monitor the following as they apply to a large break LOCA: Safety Injection Components.

12. 011K6.06 001/2/2/CHARGING PRESSURIZER/MEM 2.5/2.8/N/SR04301/R/MAB/SDR

**Due** to a controller failure, the Unit 1 Operator places the Charging Flow Controller to MANUAL to control charging flow. A high Pressurizer Level causes the Operator to try to reduce charging flow to 20 gpm.

Which ONE of the following correctly describes the behavior of FCV-1122 when the Operator attempts to reduce charging flow to 20 gpm?

- A. The Flow Limit Summator no longer limits flow and FCV-1122 can be manually closed to allow 20 gpm flow.
- B. The Flow Limit Summator no longer limits flow, however, FCV-1122 can only be manually closed to allow 25 gpm flow.
- C. The Flow Limit Summator will prevent FCV-1122 from being closed past 25 gpm flow.
- D. The Flow Limit Summator will prevent FCV-1122 from being closed past 30 gpm flow.

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References:

ND-93.3-LP-7, Pressurizer Level Control System, Rev. 6

Distractor Analysis:

- A. Correct because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. Therefore FCV-1122 can be closed manually to any value.
- B. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. Distractor is incorrect because FCV-1122 may be manually closed to any value, even below 25 gpm flow. Distractor is plausible because candidate may not know that FCV-1122 may be throttled to any value with controller in MANUAL.
- C. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. The distractor states that the Flow Limit Summator will limit flow, which is contrary to the fact that it will not limit flow. Distractor is plausible because candidate may not know that the Flow Limit Summator does not function with controller in MANUAL.
- D. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. The distractor states that the Flow Limit Summator will limit flow, which is contrary to the fact that it will not limit flow. Distractor is plausible because candidate may not know that the Flow Limit Summator does not function with controller in MANUAL.

011 Pressurizer Level Control

K6.06: Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Correlation of demand signal indication on charging pump flow valve controller to the valve position.

13. 012A4.04 001/211/RPS TCOLD TEMP/C/A 3.3/3.3/B/SR04301/R/MAB/SDR

"A" Loop Narrow Range Pcold fails low while the reactor is at 100%.

Which ONE of the following will occur?

- A. **Rod** Insertion Limit **Low** and Extra **Low** alarms will be received.
- B. Ch 1 OTDT setpoint will decrease.
- C. "A" Loop Delta **T** Protection Bistable will trip.
- D. The Tavg/ Tref Deviation alarm will be received

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References:

ND-93.3-LP-2, Delta T / **Tavg** Instrumentation System, Rev. 9

ND-93.3-LP-3, Rod Control System, Rev. 14

Distractor Analysis:

- A. Incorrect because failed Tcold **is** filtered out by **Median** Signal Selector.
- B. Incorrect because OTDT setpoint will actually increase.
- C. Correct because Tcold is fed directly to the **RPS** even when failed **low**.
- D. Incorrect because failed Tcold is filtered out by Median Signal Selector.

012 Reactor Protection System

A4.04: Ability to manually operate and / or monitor in the control room: Bistables, trips, resets, and test switches.

14. 012K1.05001/2/1/AMSAC/MEM 3.8/3 9/B/SR04301/R/MAB/SDR

Which ONE of the following lists the method by which AMSAC causes a reactor trip?

- A. Tripping the reactor trip and bypass breaker shunt coils.
- B. Tripping the reactor trip and bypass breakers UV coils.
- C. Tripping the rod drive MG set output breakers.
- D. Tripping the rod drive MG set supply breakers.

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References:

ND-93.3-LP-17, Anticipatory Mitigating System Actuating Circuitry, Rev. 10.

Distractor Analysis:

- A. Incorrect because this does not occur.
- B. Incorrect because this does not occur.
- C. Incorrect because this does not occur.
- D. Correct because this is as stated in ND-93.3-LP-17, Rev. 10.

012 Reactor Protection System

K1.05: Knowledge of the physical connections and / or cause-effect relationships between the RPS and the following systems: ESFAS

15. 013A3-02-001/2/1/SAFETY INJECTION/MEM 4 1142/N/SR04301/R/MAB/SDR

Which ONE of the following correctly states automatic actions that would occur given a Unit 1 low Pressurizer Pressure Safety Injection Signal being present for 5 minutes?

- A. Hydrogen Analyzer Heat Tracing energizes AND Containment Vacuum Pumps trip.
- B. Pressurizer Liquid Sample (SS-TV-100A) receives a close signal AND Motor Driven AFW Pumps start after a 45 second time delay.
- C. Accumulator Nitrogen Relief Lines (SI-TV-101A,B) receive a close signal AND Primary Drain Transfer Tank Vents (VG-TV-109A/B) receive a close signal.
- D. Main Steam Trip Valves (MS-TV-101A/B/C) receive a close signal AND Seal Water Return Valve (MOV-381) receive a close signal.

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#### References:

ND-91-LP-2, Safety Injection System Description, Rev. 16

ND-91-LP-2, Safety Injection System Operations, Rev. 15

P&ID 11448-FM-068A, Flow/Valve Operating Numbers Diagram Feedwater System

Sur9 Power Station Unit 1, Rev. 57

#### Distractor Analysis:

- A. Incorrect because SI signal must be present for 8 minutes for heat trace to energize.
- B. Incorrect because MDAFW Pump starts after 50 sec delay.
- C. Correct because both get a close signal on any SI Signal.
- D. Incorrect because MSTVs only get a close signal on a High Steam Flow SI Signal.

#### 013 Engineered Safety Features Actuation

**A3.82:** Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment.

16. 013K3.01.001/2/1/HOT LEG RECIRC/MEM 4.4/4.7/M/SR04301/R/MAB/SDR

Which ONE of the following could occur if ES-1.4, Transfer to Hot Leg Recirculation, is performed 20 hours after the start of a Large Cold Leg Break **LOCA**?

- A. Debris from the In-Core sump could block coolant **flow** by blocking the **lower** core plate.
- B. Reflux cooling could be lost due to boron precipitation in the hot **leg** nozzles.
- C. Fouling of core heat transfer surfaces due to the dilution of boric acid.
- D. Reduction in size of **the** incore coolant flow channels due to boron precipitation.

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#### References:

ND-95.3-LP-11, ES-1.4, Transfer To Hot beg Recirculation, Rev. 8

ES-1.4, Transfer To Hot beg Recirculation, Rev. 4

#### Distractor Analysis:

- A. Incorrect because debris in the sump **will** not block water discharged from the **S** pumps.
- B. Incorrect because boron precipitation is a concern in the **core**, not **the** hot legs.
- C. Incorrect because fouling of core heat transfer surfaces is a result of boron precipitation, *not* dilution.
- D. Correct because boron precipitation is a concern when boil-off continues and when core temperature decreases. The standard time for transfer to hot leg recirc is 8 hours, not 28 hours, as stated in **the** stem.

013 Engineered Safety Features Actuation

**M3.01:** Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Surry Requal Bank Question #299

17. 014A2 05 001/2/2/RPIS ROD POSITION/MEM 3.9/4.1/B/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- A Small Break LOCA has occurred
- Automatic Safety Injection has occurred
- 1-E-0, Reactor Trip or Safety Injection, has been implemented
- The CRO observes the Rod Position Indication as displaying Control Rods on the bottom of the reactor core, with the exception of three Control Rods.

Which ONE of the following actions is procedurally required as a result of this finding by the CRO?

- A. Continue with 1-E-0, Reactor Trip or Safety Injection.
- B. Emergency borate while proceeding through 1-E-0, Reactor Trip or Safety Injection
- C. Manually insert control rods while proceeding through 1-E-0, Reactor Trip or Safety Injection.
- D. Go directly to 1-FR-S.1, Response to Nuclear Power Generation/ ATWS, Step 1.

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#### References:

1-FR-S.1, Response to Nuclear Power Generation/ ATWS, Rev. 18

1-E-0, Reactor Trip or Safety Injection, Rev. 46

#### Distractor Analysis:

- A. Correct because E-0 should be entered upon Reactor Trip per the rules of EOP usage.
- B. Incorrect because if emergency boration is needed, it will be directed by FR-S.1.
- C. Incorrect because if manual rod insertion is needed, it will be directed by FR-S.1.
- D. Incorrect because FR-S.1 should only be entered as directed by E-0 (or if E-0 has been completed then an Orange or Red path).

#### Qf4 Rod Position Indication

A2.05: 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: Reactor Trip.

Surry ILT Bank Question #1037

18. 015K4.07 001/2/2/PERMISSIVE/C/A 3.7/3.8/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Reactor Power is 5%
- Turbine First Stage Impulse Pressure PT-446 is selected
- Power Range Nuclear Instrument N-41 fails high
- PT-446 fails high

Which ONE of the following correctly describes the impacts of the failures?

- A. Control Rods **do** not move. The Reactor Protection System At-Power Trips are enabled due to the N-41 failure.
- B. Control Rods step out at 72 steps per minute. The Reactor Protection System At-Power Trips are enabled due to the N-41 failure.
- C. Control Rods do not move. The Reactor Protection System At-Power Trips are enabled **due** to the PT-446 failure.
- D. Control Rods step out at 72 steps per minute. The Reactor Protection System At-Power Trips are enabled due to the PT-446 failure.

Surly

References:

ND-93.3-LP-16, Permissive/Bypass/Trip Status Lights, Rev. 8

Surly Simulator Malfunction Cause and Effects, Rev. 6, Malfunction **MMS-14**

Distractor Analysis:

- A. Incorrect because Tref will go to 574 °F, which will cause rods to **step** out at max rate of 72 steps/min. Also incorrect because 2/4 PR NIs must be above 10% to enable At Power Trips.
- B. Incorrect because 2/4 PR NIs must be above 10% to enable At Power Trips.
- C. Incorrect because Tref will go to 574 °F, which will cause rods to step out at max rate of 72 steps/min.
- B. Correct because Tref will go to 574 °F, which **will** cause rods to step out at max rate of 72 steps/min and only 1/2 Turbine First Stage PTs need to be above 10% to enable At Power Trips.

015 Nuclear Instrumentation

K4.07: Knowledge of NIS design feature(s) and / or interlock(s) provide for the following: Permissives.

19. 016A4.01 001/2/2/PIMP IMPULSE/C/A 2.9/2.8/B/SR04301/R/MAB/SDR

The following condition exists:

- Unit 1 at 100% reactor power
- All systems and equipment functions as designed
- All protection channel III's are selected
- First stage impulse pressure channel IV fails low

Which **ONE** of the following would occur initially without operator action?

- A. **AMSAC** would be operationally disabled after 60 seconds.
- B. Steam Dumps would all open.
- C. FRVs would control SG level at no load level.
- D. MOV-CP-100, Condensate Polishing Building Bypass Valve, would open.

---

Surry

References:

ND-93.3-LP-17, Anticipatory Mitigating System Actuating Circuitry (**AMSAC**), Rev. 10  
ND-93.3-LP-9, Steam Dump Control System, Rev. 10  
ND-93.3-LP-8, SG Water Level Control System, Rev. 6

Distractor Analysis:

- A. incorrect because this would occur after 360 seconds.
- B. Incorrect because Channel III is selected.
- C. Incorrect because Channel III is selected.
- D. Correct because, as stated in ND-93.3-LP-9, CP-100 will open in anticipation of the upcoming increase in feedwater flow that will occur during load rejection.

016 Non-Nuclear instrumentation

A4.01: **Ability** to manually operate and / or monitor in the control room: NNI channel select controls.

Surry Requal Bank Question #279

20. 022AKL01 001LL1/RCP SEALS/C/A 2.8/3.2/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Reactor trip has occurred due to a loss of all AC power
- Power has been restored
- The following Reactor Coolant Pump parameters are present for all RCPs:
  - No. 1 Seal Water Outlet Temperatures are 225 °F
  - Lower Seal Water Bearing Temperatures are 220 °F
- The Shift Supervisor directs the operators to restore cooling to the RCP seals per 1-AP-9.02, Loss of RCP Seal Cooling.

Which ONE of the following correctly states the requirements for restoring cooling to the RCP seals and why?

- A. Do not establish seal injection flow or component cooling flow to the thermal barrier heat exchanger because the No. 1 Seal Water Outlet Temperatures are too high. Seal cooling should be restored by cooling the RCS using natural circulation.
- B. Do not establish seal injection flow or component cooling flow to the thermal barrier heat exchanger because the Lower Seal Water Bearing Temperatures are too high. Seal cooling should be restored by cooling the RCS using natural circulation.
- C. Slowly establish seal injection flow to minimize RCP thermal stresses, followed by slowly introducing component cooling flow to the thermal barrier heat exchanger to limit introduction of steam into the CC system.
- D. Slowly establish component cooling flow to the thermal barrier heat exchanger to limit introduction of steam into the CC system, followed by slowly introducing seal injection flow to minimize the RCP thermal stresses.

Surry

References:

1-AP-9.02, Loss of RCP Seal Cooling, Rev. 8.  
ND-88.1-LP-6, Reactor Coolant Pumps, Rev. 16.

Distractor Analysis:

- A. Incorrect because AP-9.02 (Caution page 7) states if No. 1 Seal Water Outlet Temp is > 235 °F then Seal Inj and CCW to Thermal Barrier H.X. should not be restored. instead N.C. should be used to cool the seals.
- B. Incorrect because AP-9.02 (Caution page 7) states if Lower Seal Water Bearing Temperature is > 225 °F then Seal Inj and CCW to Thermal Barrier H.X. should not be restored. instead N.C. should be used to cool the seals.
- C. Incorrect because CC flow should be established prior to seal injection flow.
- D. Correct as stated in 1-AP-9.02 NOTE prior to step 7 and CAUTIONS prior to steps 9 and 15.

822 Loss of Rx Coolant Makeup

AK1.01: Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: Consequences of thermal shock to RCP seals.

21. 022G2.4.22 001/2/1/SAFETY FUNCTIONS/C/A 3.0/4.0/B/SR04301/R/MAB/SDR

Unit 1 has tripped and Safety Injection has actuated due to a Large Break Loss of Coolant Accident (LOCA).

Many complications have occurred.

**The crew** has exited E-0, Reactor Trip or Safety Injection. **The Shift Technical Advisor** has started to monitor Critical Safety Function Status Trees and reports:

- Subcriticality - Orange Path
- Heat Sink - Yellow Path
- Core Cooling - Orange Path
- Containment - Red Path

Which ONE of the following states the correct procedure transition?

- A. FR-S.1, Response to Nuclear Power Generation/ATWS, based on Subcriticality Orange **Path**.
- B. FR-H.1, Response to Secondary **Heat** Sink, based on Heat Sink Yellow Path.
- C. FR-C.1, Response to Inadequate Core Cooling, based on Core Cooling Orange Path.
- D. FR-Z.1, Response to High Containment Pressure, based on Containment Red Path.

Sur9

References:

ND-95.3-LP-26, Critical Safety Function Status Trees, Rev. 5

Distractor Analysis:

- A. Incorrect based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Subcriticality Orange Path does not take priority over any Red Path.
- B. Incorrect based on **the** rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Heat Sink Yellow Path does not take priority over Containment Red Path.
- C. Incorrect based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). Core Cooling Orange Path does not take priority over Containment Red Path.
- D. Correct based on the rules of use for **safety function status trees** (ND-95.3-LP-26 Page 15). The Containment Red Path takes priority over the other paths. Only knowledge of safety function priority rules are needed to answer this question.

022 Containment Cooling

G2.4.22: **Knowledge** of the bases for prioritizing safety functions during abnormal and emergency operations.

Turkey Point Bank Question TP03301

22. 022K3.02 001/2/1/CONTAINMENT PRESSURE/C/A 3.0/3.3/B/SR04301/R/MAB/SDR

Unit 2 is operating at 100% power with Chilled CC in service to containment.  
2-CD-REF-1A trips due to a fault.

Which ONE of the following describes the effect on Unit 2 containment parameters?

- A. Indicated partial pressure will increase. Containment temperature will decrease.
- B. Indicated partial pressure will increase. Containment temperature will increase.
- C. Indicated partial pressure will decrease. Containment temperature will decrease.
- D. Indicated partial pressure will decrease. Containment temperature will increase.

Surry (Utility should add noun names to equipment in the stem.)

References:

ND-88.5-LP-1, Component Cooling, Rev. 19

Distractor Analysis:

- A. incorrect because partial pressure will decrease due to loss of chilled CC.
- B. incorrect because partial pressure will decrease due to loss of chilled CC.
- C. Incorrect because containment temperature will increase due to a loss of chilled CC.
- D. Correct because partial pressure will decrease and containment temperature will increase due to a loss of chilled CC.

Bank Question # 544

022 Containment Cooling

K3.02: Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment Instrument Readings.

23. 026A2.07 00 1/21/CONTAINMENT SPRAY/C/A 3.6/3.9/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- A Large Break LQCA has occurred inside containment
- Safety Injection has actuated
- Containment Pressure peaked at 28 psia
- Current Containment Pressure is 15.8 psia
- "1A", "2A" and "2B" Recirculation Spray Pumps are operating
- "1B" Recirculation Spray Pump tripped on Overload (OL)
- 1A-E7, RS PP 1A VIB, annunciates and the alarm cannot be cleared

Which ONE of the following states the correct operator action for these conditions?

- A. Secure Recirculation Spray Pump "1A" using the handswitch in the control room.
- B. Place the Recirc Spray Pump 1A in PTL, then secure Recirculation Spray Pump "1A" locally at the breaker (14H4).
- C. Reset CLS, then place the handswitch for Recirculation Spray Pump "1A" in PTL.
- D. Allow Recirculation Spray Pump "1A" to operate, but monitor vibrations closely.

Surry

References:

ND-91-LP-5, Containment Spray System, Rev. 13

ND-91-LP-6, Recirculation Spray System, Rev. 9

1-RM-A7, RS/SW HX A ALERT/FAILURE, Rev. 5

Distractor Analysis:

- A. Incorrect because with CLS present, the handswitch in the control room cannot be used to secure the pump. Containment pressure must be less than 12 psia **B** reset CLS. Pressure currently **is** 15.8 psia.
- B. Correct because local operation of the breaker will stop the pump. In addition, the ARP will have **the** operator place the handswitch in PTL, but the lesson plan (ND-91-LP-6 Page 6) states that the pump cannot be secured from the control room with CLS present. Furthermore, the ARP gives guidance to secure the distressed pump as long as two other RS Pumps are operating. The stem states that two other pumps are operating ("2A" and "2B").
- C. Bncorrect because **the** CLS cannot be reset until containment pressure is less than 12 psia.
- D. Incorrect because the ARP gives guidance to **secure** the distressed pump as long as two other RS Pumps are operating. The stem states that **two** other pumps are operating ("2A" and "2B").

026 Containment Spray

A2.07: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, **use** procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Loss** of containment spray suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature (exceeded cavitation, voiding), or sump level below cutoff (interlock) limit.

Note:

The ARP states that high vibration alarms may be caused by cavitation of the pump. Cavitation could be caused by high water temp, low water level, etc.

24.02AK3.02001/I/1/CCW SAFETY INJECTION/MEM 3.5/3.7/M/SR04301/R/MAB/SDR

A High Steam Flow Safety Injection Signal is received.

Which ONE of the following correctly describes the response of the Component Cooling Water System components?

- A. TV-CC-109A and B (CC Isolation Valves from RHR) close and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trip Valve) remain as-is.
- B. TV-CC-109A and B (CC Isolation Valves from RHR) remain as-is and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trig Valve) remain as-is.
- C. TV-CC-109A and B (CC Isolation Valves from RHR) close and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trip Valve) close.
- D. TV-CC-109A and B (CC Isolation Valves from RHR) remain as-is and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet **Flow** Outside Trip Valve) close.

Surry

References:

88-05-01, Component Cooling Water System, Rev. 19

Distractor Analysis:

- A. Correct because lesson plan states CC-109 closes on Phase I and 110 closes on Phase III isolation.
- B. Incorrect because lesson plan states CC-109 closes on Phase I and 110 only closes on Phase III isolation.
- C. Incorrect because lesson plan states CC-109 closes on Phase I and 110 only closes on Phase III isolation.
- D. Incorrect because lesson plan states CC-109 closes on Phase I and 110 closes on Phase III isolate.

026 Loss of Component Cooling

AK3.02: Knowledge of the reasons for the following responses as they apply to Loss of Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS.

The loss of CCW occurs in part of the system due to the ESFAS isolation of TC-CC-109A/B.

25. 026G2.4.46 001/2/1/CONTAINMENT SPRAY/C/A 3.5/3.6/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- A Large Break LOCA has occurred
- Safety Injection has actuated
- Containment Pressure peaked at 27 psia
- RCS subcooling is 0 °F
- Steam Generator bevels are 22% and slowly rising
- **RWST** emptied while performing ES-1.3, Transfer to Cold Leg Recirculation
- ES-1.3, Transfer to Cold Leg Recirculation, has been completed and the crew has transitioned back to E-1, Loss of Reactor or Secondary Coolant
- All equipment operated normally

Which ONE of the following alarms is consistent given the above plant conditions?

- A. 1E-A1, HI-HI CTMT PRESSCLS CH-1
- B. 1B-B1, CS PP 1A LOCKOUT OW OL TRIP
- C. 1A-D7, RS PP 1A LOCKOUT OW OL TRIP
- D. 1B-F6, CTMT INST AIR HDR LO PRESS

Surry

References:

- 1-E-1, Loss of Reactor or Secondary Coolant, Rev. 21
- 1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 12
- 1E-A1, HI-HI CTMT PRESSCLS CH-1, Rev. 0
- 1B-B1, CS PP 1A LOCKOUT OR OL TRIP, Rev. 0
- 1A-D7, RS PP 1A LOCKOUT OR **Ob** TRIP, Rev. 0
- 1B-F6, CTMT INST AIR HDR LO PRESS, Rev. 1
- ND-91-LP-5, Containment Spray System, Rev. 13
- ND-91-LP-6, Recirculation Spray System, Rev. 9

Distractor Analysis:

- A. Incorrect because containment pressure **is** now less than the setpoint, which **is** known by CLS having been reset. **As** a part of going to Cold Leg Wecirc, CLS and SI must be reset.
- B. Correct because **1-ES-1.3** has been completed and the RWST **has** been emptied; therefore, **the** CS Pumps would be placed in PPL due to the lack of a suction source (cavitation). Placing **the** CS Pumps in PTL yields **1B-B1** for CS Pump **1A**.
- C. Incorrect because Outside Recirc Spray Pump **1A** would be placed in AUTO when stopped.
- D. Incorrect because CLS and SI must have been reset prior to completion of **1-ES-1.3** and instrument air would have been restored to containment.

026 Containment Spray

G2.4.46: Ability to verify that alarms are consistent with plant conditions.

26. 027A4.03 001/2/2/IODINE/MEM 3.3/3.2/B/SR04301/R/MAB/SDR

Which ONE of the following describes the operation of the Iodine Filtration Fans (1-VS-F-3A/3B)?

- A. Automatically start on a Hi-Hi CLS.
- B. Automatically start on a containment gas high alarm.
- C. Automatically stop on a Hi-Hi CLS signal.
- D. Must be manually started under all conditions.

Surry

References:

ND-88.4-LP-6, Containment Ventilation, Rev. 5

Distractor Analysis:

- A. Incorrect because fans are only manually operated.
- B. Incorrect because fans are only manually operated.
- C. incorrect because fans are only manually operated.
- D. Correct because fans are only manually operated.

027 Containment Iodine Removal

A4.83: Ability to manually operate and / of monitor in the control room: CIRS fans.

Question Status:

Surry Bank ILT Question #741

27. 027AK3 01.001/1/1/PRESSURIZER SPRAY/C/A 3 513 8/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- The Reactor is at 100% Power.
- A malfunction in the Pressurizer Heater Control Circuit has resulted in Proportional Heaters being de-energized.
- A small amount of leakage in the Pressurizer Auxiliary Spray Valve is occurring.
- Pressurizer Pressure is 2215 psig and slowly lowering.

1-AP-31.00, Increasing or Decreasing RCS Pressure, has been entered.

Which ONE of the following states the correct position of the normal sprays and backup heaters?

- A. Normal sprays are OFF (valves closed) and backup heaters are ON.
- B. Normal sprays are ON (valves open) and backup heaters are OFF.
- C. Normal sprays are OFF (valves closed) and backup heaters are OFF.
- D. Normal sprays are ON (valves open) and backup heaters are ON.

Surry

References:

ND-93.3-LP-5, Pressurizer Pressure Control, Rev. 9

1C-B8, PRZR LO PRESS, Rev. 1

1-AP-31.00, Increasing or Decreasing RCS Pressure, Rev. 6

Distractor Analysis:

- A. Incorrect because backup heaters do not energize until 2210 psig.
- B. Incorrect because spray valves do not start to open until 2255 psig.
- C. Correct because backup heaters do not energize until 2210 psig and spray valves do not open until 2255 psig.
- D. Incorrect because backup heaters do not energize until 2210 psig.

027 Pressurizer Pressure Control System Malfunction

AK3.01: Knowledge of the reasons for the following responses as they apply to pressurizer pressure control malfunctions: Isolation of PZR spray following loss of PZR heaters.

28.028G2.2.12001/2/2/HYDROGEN RECOMBINER/C/A 3 0/3.4/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- The plant is at 50% power
- 1-PT-37.2, Electric Hydrogen Recombiner, is about to be performed to determine the reference power that would be used in the event that the Recombiners are used following a **LOCA**.

Which ONE of the following correctly states 1-PT-37.2 limitations that are applicable during the performance of this test?

- A. At no time should the heater temperature be allowed to exceed 1400°F as monitored by the highest thermocouple reading **AND** containment hydrogen concentration must be verified to **be** less than 0.75%.
- B. At no time should the heater temperature be allowed to exceed 1400 °F as monitored by the highest thermocouple reading **AND** containment hydrogen concentration must be verified to be less than 1.00%.
- C. At no time should the heater temperature be allowed to exceed 1300 °F as monitored by the highest thermocouple reading **AND** containment hydrogen concentration must be verified to be **less** than 0.75%.
- D. At no time should the heater temperature be allowed to exceed 1300 °F as monitored by the highest thermocouple reading **AND** containment hydrogen concentration must be verified to be less than 1.00%.

Surry

References:

1-PT-37.2, Electric Hydrogen Recombiner, Rev. 9

Distractor Analysis:

- A. Correct because these are both requirements listed on Section 4.0 of 1-PT-37.2. The unit *is* at power, therefore 4.3 states that containment hydrogen concentration must be verified less than 0.75% (being at power and making the operator determine if 4.3 applies is part of what makes the question C/A). Section 4.2 states that heater temperature must remain less than 1400°F at all times.
- B. Incorrect because verifying containment hydrogen less than 1% is not the correct requirement. Plausible because applicant may not know that the requirement is 0.75%, vice 1.0%.
- C. Incorrect because verifying the highest temperature less than 1300 °F **is** not the correct requirement. True - if **the** operator ensures temperature is less than 1300 °F, then he has also ensured that it is less than 1400°F, but **this** question tests the knowledge of the requirement, not simply a method for meeting the requirement. Plausible because applicant may not know **the** temperature requirement.
- D. Incorrect because of reasons in C and D distractor analysis.

028 Hydrogen Recombiner and Purge Control  
G2.2.12: Knowledge of surveillance procedures.

29. 029EK3.09 001/1/1/ATWS AFW/MEM 3.7/4.0/M/SR04301/R/MAB/SDR

Which ONE of the following describes **the** reason why charging pump suction are manually aligned *to* the **WWST** during an ATWS vice manually initiating a Safety Injection?

- A. Prompt operator action **will** ensure the most direct method of borating into the RCS and manual alignment of charging pump suction to **the** RWST prevents compounding the problem **by** charging **the** RCS solid via Safety Injection.
- B. Prompt operator action will ensure the most direct method of borating into the RCS and initiation of **SI** would reduce the possible paths for emergency boration and add to an RCS overpressure condition if one exists.
- C. Manual initiation of Safety Injection **would** delay the addition of borated water to the RCS and complicate the recovery actions. Alignment of charging pump suction to the RWST **is** the most direct method of borating **the** RCS.
- D. Manual initiation of **SI** would result in the undesirable trip of Main Feedwater Pumps and alignment of Charging Pump suction to the **RWST** is the most direct method of borating the RCS.

Surry

References:

ND-95.3-LP-36-DRR, FR-S.1 Response to Nuclear Power Generation/ ATWS, Rev. 10  
FR-S.1, Response to Nuclear Power Generation/ ATWS, Rev. 15

Distractor Analysis:

- A. Incorrect because **the** concern with initiating **SI** is not creating a solid plant condition, but with reducing the probability of maintaining a secondary heat sink because **MFV** pumps will **trip** upon **SI** initiation.
- B. Incorrect because the concern with initiating **SI** is not creating a high **RCS** pressure condition, but with reducing the probability of maintaining a secondary heat sink because **MFV** pumps will trip upon **SI** initiation.
- C. Incorrect because manual initiation **would** not delay addition of borated water. The concern is with reducing the probability of maintaining a **secondary** heat sink because **MFV** pumps will trip upon **SI** initiation.
- D. Correct because **per** ND-95.3-LP-36-DRR, FR-S.1 Response to Nuclear Power Generation / ATWS, both of these statements accurately reflect the basis for Step 4.

029 ATWS

**EK3.09:** Knowledge of the reasons for the following responses as they apply to the ATWS: Opening centrifugal charging **pump** suction valves from RWST.

Modified ILT Bank Question # 3390

30. 032AA1.01 001/1/2/SOURCE INTERMEDIATE/C/A 3.1/3 4/B/SR04301/R/MAB/SDR

The following conditions exist:

- Present time is 1428 hours
- Reactor tripped at 1405 hours
- All Rod Bottom Lights are lit
- N-35 reading is  $2 \times 10^{-10}$  amps
- N-36 reading is  $4 \times 10^{-11}$  amps
- Source Range is not energized
- Power level prior to trip was 90%

Which ONE of the following describes the correct actions given the above parameters?

- A. When both IR channels read  $< 5 \times 10^{-10}$  amps, verify source range channels energized.
- B. Place the **source** range trip bypass switches in the **NORMAL** position.
- C. **Energize** the source range channels **by** depressing the source range manual reset pushbuttons.
- D. Transfer NR-45 to one **source** range and one intermediate range channel.

Surry

References:

ND-93.2-LP-3, Intermediate Range NIs, Rev. 10.

Distractor Analysis:

- A. Incorrect because SR energizes at  $2/2$  IR  $< 5 \times 10^{-11}$  amps.
- B. Incorrect because SR should already be energized in the **NORMAL** position and this action would not energize the SR.
- C. Correct because IR are under-compensated and SR must be manually energized.
- D. Incorrect because SR should both be energized.

032 Loss of Source Range NI

AA1.01: Ability to operate and / or monitor the following as they apply to loss of source range nuclear instrumentation: Manual restoration of power.

31. 033AA2.04 001/1/2/OVERLAP NI/MEM 3.2/3.6/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Critical approach has just been completed.
- Reactor is **stable** at the Point of Adding Heat.
- One Intermediate Range (IR) Nuclear Instrument (NI) is suspected of displaying inaccurate indications.

Which ONE of the following correctly describes the expected Power Range (PR) NI and the known operable IR NI indications for the above conditions to verify that the suspect IR NI is in fact falsely indicating?

- A. IR =  $2.5 \times 10^{-8}$  Amps; PR between 0.2 and 1 %
- B. IR =  $2.0 \times 10^{-6}$  Amps; PR between 0.2 and 1 %
- C. IR =  $1.0 \times 10^{-8}$  Amps; PR < 0.2 %
- D. IR =  $1.0 \times 10^{-5}$  Amps; PR < 0.2 %

Surry

References:

ND-93.2-LP-4, Power Range NIs, Rev. 16

1-GOP-1.4, Unit Startup, HSD to 2% Reactor Power, Rev. 29

Distractor Analysis:

- A. Incorrect because  $2.5 \times 10^{-8}$  Amps is about where critical data is taken (too low).
- B. Correct based on above two references: ND-93.2-LP-4 (H/T-4.3) & 1-GOP-1.4 (Page 29 CAUTION).
- C. Incorrect because  $1.0 \times 10^{-8}$  Amps is about where critical data is taken (too low).
- D. Incorrect because  $1.0 \times 10^{-5}$  Amps is above the POAH and should correspond to about 2% power.

033 boss of Intermediate Range NI

AA2.04: Ability to determine and interpret the following as they apply to the loss of intermediate range nuclear instrumentation: Satisfactory overlap between source-range, intermediate-range, and power-range instrumentation.

32. 034A4 01.001/2/2/RADIATION MONITOR/C/A 3.3/3.7/N/SR04301/R/MAB/SDR

Unit 1 is in a refueling outage when the following events occur:

- Purge Isolation Valves (MOV-VS-100A, B, C, and D) Close
- Unit Purge Supply Fans (4A and 4B) Trip
- Containment Instrument Air Suction Valves (TV-IA-101 A/B) Close

Which ONE of the following radiation monitors could have caused **these** actions?

A. Process Vent Particulate and Gas Monitors (RM-RI-101 / 102)

5. RM-I61 (Containment High Range Gamma Monitor)

C. RM-162 (Manipulator Crane Monitor)

D. RM-163 (Reactor Containment Area Monitor)

Surry

References:

ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 5

Distractor Analysis:

- A. incorrect because RM-RI-101 / 102 **do** not cause these actions.
- B. Incorrect because **RM-I61** does not cause these actions.
- C. Correct per ND-93.5-LP-1.
- 5. incorrect because RM-163 does not cause these actions.

034 Fuel Handling Equipment

**A4.01:** Ability to manually operate and / or monitor in the control room: Radiation Levels.

33. 03SA3.01001/2/2/STEAM GENERATOR/C/A 4.0/3.9/N/SR04301/R/MAR/SDR

The following Unit 1 conditions exist:

- Plant is stable at 75% Power
- "A" SG Steam Line FT-MS-475 is selected for Steam Generator level control
- "A" SG Steam line PT-MS-475 fails high

Which ONE of the following correctly describes the impact on the "A" Steam Generator Level control?

- A. Feedwater Regulating Valve opens because indicated steam flow is greater than indicated feedwater flow.
- B. Feedwater Regulating Valve does not move as a result of the failure.
- C. Feedwater Regulating Valve closes because the pressure transmitter is overcompensating for density.
- D. Feedwater Regulating Valve opens to reduce the level error created by the failure.

Surry

References:

ND-93.3-LP-8, SG Water Level Control System, Rev. 6

Distractor Analysis:

- A. Incorrect because PT-MS-475 does not compensate steam flow for FT-MS-475.
- B. Correct because PT-MS-475 does not compensate steam flow for FT-MS-475.
- C. Incorrect because PT-MS-475 does not compensate steam flow for FT-MS-475.
- D. Incorrect because PT-MS-475 does not compensate steam flow for FT-MS-475.

035 Steam Generator

A3.01: Ability to monitor automatic operation of the S/G including: S/G water level control.

34. 038EK3.09 001/1/1/SAFETY INJECTION/MEM 4.1/4.5/N/SR04301/R/MAB/SDR

Which ONE of the following is correct regarding safety injection termination during a steam generator tube rupture event?

Safety Injection termination ...

- A. may occur with total AFW flow less than 350 gpm as long as 350 gpm is available.
- B. may occur with Pressurizer level less than 22% as long as level is increasing.
- C. may not occur with a void in the reactor head due to presenting RCS pressure control problems.
- D. may not occur with a void in the reactor head due to presenting RCS level control problems.

Sur9

References:

ND-95.3-LP-13, E-3 Steam Generator Tube Rupture, Rev. 11  
1-E-3, Steam Generator Tube Rupture, Rev. 19

Distractor Analysis:

- A. Correct because if no intact SG is available the ruptured SG will be used to cool the RCS. In this instance the AFW flow may be less than 350 gpm, but 350 gpm must still be available to that SG. If sufficient flow is available, then SI termination criteria is considered to be met (ND-95.3-LP-13).
- B. Incorrect because pressurizer level must be greater than 22% to meet the SI termination criteria.
- C. Incorrect because safety injection may be terminated when there is a void in the reactor head. This will present some challenges with RCS pressure and level control, but it is not a large enough concern to prevent SI termination if the specified criteria are met (ND-95.3-LP-13).
- D. Incorrect because safety injection may be terminated when there is a void in the reactor head. This will present some challenges with RCS pressure and level control, but it is not a large enough concern to prevent SI termination if the specified criteria are met (ND-95.3-LP-13).

038 Steam Gen. Tube Rupture

EK3.09: Knowledge of the reasons for the following responses as they apply to the SGTR: Criteria for securing/ throttling ECCS.

35. 039A1.09 001/2/1/MAIN STEAM RADIATION/C/A 2.5/2.7/B/SR04301/R/MAB/SDR

With Unit 1 at 100% power, the Condenser Air Ejector and Main Steam bine Radiation Monitor alarms are recieved. The Condenser Air Ejector Radiation Monitor reads 700 cpm (ALERT and HIGH alarms are in) while local Main Steam NRC Radiation Monitors read "A" .03 mr/hr, and "B" .01 mr/hr, and "C" .01 mr/hr. The Team has implemented 1-AP-16.00, Excessive RCS Leakage, and the RCS leak rate *is* determined to be 60 gpm.

Which ONE of the following describes the actions required?

- A. Verify automatic Condenser Air Ejector divert to Containment, intiate 1-AP-24.00 (Minor SG Tube Leak), manually trip the reactor and go to 1-E-0 (Reactor Trip or Safety injection).
- B. Verify automatic SGBD TV trip isolation and Condenser Air Ejector divert to Containment, manually trip the reactor and initiate ~~SI~~,Go to 1-E-0.
- C. Verify automatic Condenser Air Ejector ~~divert~~ to Containment, initiate 1-AP-24.01 (Large Steam Generator **Tube** Leak), verify letdown isolated, and commence a normal Unit shutdown IAW GOPs.
- D. Verify automatic Condenser Air Ejector divert to Containment, initiate 1-AP-24.01 (Large Steam Generator Tube Leak), and manually trip the reactor and go to 1-E-0.

Surry

References:

- ND-89.3-LP-2, Main Condensate System, Rev. 16
- ND-93.5-LP-3, Post-TMI Radiation Monitoring System, Rev. 6
- 1-AP-16.60, Excessive RCS Leakage, Rev. 11
- 1-AP-24.00, Minor SG Tube leak, Rev. 8
- 1-AP-24.01, Large Steam Generator Tube Leak, Rev. 11

Distractor Analysis:

- A. Incorrect because 60 gpm leakage is more than minor. **AP-24.01** should be entered for a large steam generator tube leak.
- B. Incorrect because **SI** should not **be** initiated.
- C. Incorrect because the reactor must **be** manually tripped with leakage greater than 50 gpm.
- D. Correct because air ejectors will divert to containment on an air ejector high radiation, **AP-24.01** should be entered due to 60 gpm leak rate **with** air ejector high radiation, and **E-0** should be entered following a manual reactor **trip**.

039 Main and Reheat Steam

A1.09: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main steam line radiation monitors.

36. 054G2.4.31 001/1/1/ALARMS RODS/C/A 3.3/3.4/B/SR04301/R/MAB/SDR \_\_\_\_\_

The following Unit 1 conditions exist:

- The Reactor was operating at 78% power when a loss of the "A" Feedwater Pump occurred
- The Team is taking the required immediate actions in accordance with 1-AP-21.00, "Loss of Main Feedwater Flow"
- The Reactor Operator is driving rods in manual to lower Tavg
- Tavg is within 3 °F of Tref
- Annunciator 1G-G8, ROD **BANK D IO** LIMIT, has annunciated

Which ONE of the following is the correct response to the given plant conditions?

- A. Shutdown margin is not sufficient for the given plant conditions and operators should emergency borate to regain the required shutdown margin.
- B.** The operator has driven rods in too far for the existing boron concentration and should borate from **the** Boric Acid Tanks.
- C. **Shutdown** margin is not sufficient for the given plant conditions and operators should trip the Reactor and go to 1-E-0, Reactor Trip or Safety Injection.
- D. The turbine load has decreased too far and the operator should raise turbine load.

Surry

References:

ND-89.3-LP-3, Main Feedwater System, Rev. 12  
ND-95.1-LP-4, Loss of Feedwater, Rev. 3  
1-AP-21.00, Loss of Main Feedwater Flow, Rev. 5  
1G-G8, ROD BANK D LO LIMIT, Rev. 0

Distractor Analysis:

- A. Incorrect because (1) not enough information is given to make the determination that SDM is insufficient, and (2) even if SDM is not above that which is required, emergency boration would not be the preferred method for regaining the required SDM. This is clearly the wrong method for boration because xenon is building *in* and only small borations would be desired to withdraw rods to clear the alarm.
- B. Correct because rods being within 10 steps of its insertion limit would cause the alarm. Boration from the Boric Acid Tanks would be the correct mitigation strategy and as such, is directed by the ARP. Operators would only borate the necessary amount to clear the alarm.
- C. Incorrect because the initial power level was less than 85% and the plant is designed to handle this magnitude of transient. Furthermore, the plant does not need to be tripped with rods approaching or below insertion limits. Rod positions just have to be restored to within limits.
- D. Incorrect because turbine load should not be raised. Immediate actions have the operators reduce turbine load to match steam flow and feed flow. Raising turbine load under these conditions would not be the correct action. It would also be nonconservative to add positive reactivity via the turbine during a transient condition such as described in the stem.

054 Loss of Main Feedwater

G2.4.31: Knowledge of annunciators and indications and use of response instructions.

Bank Question from 2003 Farley Exam (Farley K/A was 054G2.2.20).

37. 055EK1.01 001/1/1/BATTERY/C/A 3.3/3.7/B/SR04301/R/MAB/SDR

The following plant conditions exist:

- A loss of all AC power has occurred.
- Operators have implemented ECA-0.0, Loss of All AC Power.
- Attempts to regain AC power have failed.
- Operators are performing ECA-0.0, Step 28, "Check DC Bus loads"

Which ONE of the following should be performed to lower the Black Battery discharge rate by the largest amount per ECA-0.0?

- A. Secure Air Side Seal Oil Pump only.
- B. Secure Air Side Seal Oil Pump and Emergency Turbine Lube Oil Pump.
- C. Secure Air Side Seal Oil Backup Pump only.
- D. Secure Air Side Seal Oil Backup Pump and Emergency Turbine Lube Oil Pump.

Surry

References:

ND-90.3-LP-6, 125 Vdc Distribution, Rev. 10

ECA-0.0, Loss of All AC Power, Rev. 21

Distractor Analysis:

- A. Incorrect because the Air Side Seal Oil Pump is not a DC load, as is the Air Side Seal Oil Backup Pump. Plausible because the candidate may not know major Black Battery DC Loads, or may not ~~know~~ what actions are permitted by ECA-0.0.
- B. Incorrect because the Air Side Seal Oil Pump is not a DC load, as is the Air Side Seal Oil Backup Pump. Plausible because the candidate may not know major Black Battery DC Loads, or may not know what actions are permitted by ECA-0.0.
- C. Incorrect ECA-0.0 will direct the securing of both Air Side Seal Oil Backup Pump (ASSOBUP) and Emergency Turbine Lube Oil Pump, not just ASSOBUP. Plausible because the applicant may not know that there is more than one pump to secure to conserve Black Batteries.
- B. Correct because per ECA-0.0 step 28 and Basis for this step in ND-95.03-LP-17, the **purpose** is to secure **both** pumps, which **are** large Black Battery DC loads, to conserve the batteries (reducing **battery** discharge rate, thus prolonging battery life).

Surry ILT Bank Question #724

055 Station Blackout

EK1.01: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Effect of battery discharge rate on capacity.

38. 056A2.04 001/2/1/CONDENSATE PUMP/C/A 2.6/2.8/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Two Main Feedwater Pumps **are** operating
- Reactor Power = 85%
- Condensate Pumps 1-CN-P-1A and B are operating
- Condensate Pump 1-CN-P-1C is Tagged Out of Service
- Condensate Pump 1-CN-P-1A trips and cannot be restarted
- Main Feedwater Pump Suction Pressure = 105 psig and slowly lowering
- Steam Generator Levels are slowly lowering
- 1H-F8, FW PP SUCT HDR LO PRESS, is in alarm

Which ONE of **the** following is the correct operator action?

- A. Enter 1-AP-21.00, **Loss** of Main Feedwater Flow, and reduce turbine load to match steam flow and feedwater flow.
- B. Manually trip the Reactor and enter E-0, Reactor Trip or Safety injection.
- C. Secure one of **the** operating Main Feedwater Pumps and monitor the operating Main Feedwater Pump Suction Pressure.
- D. Enter 1-AP-21.00, Loss of Main Feedwater Flow, **and** start a second HP Drain Pump.

Surry

References:

ND-89.3-LP-2, Main Condensate System, Rev. 16  
ND-89.3-LP-3, Main Feedwater System, Rev. 12  
ND-95.1-LP-4, Loss of Feedwater, Rev. 3  
1-AP-21.00, Loss of Main Feedwater Flow, Rev. 5  
1H-F8, FW PP SUCT HDR LO PRESS, Rev. 0  
1H-G8, FW PP DISCH HDR LO PRESS, Rev. 0  
1J-G4, CN PPS DISCH HDR LO PRESS, Rev. 0

Distractor Analysis:

- A. Correct because **MFW** Pump Low Suction Pressure and Discharge Pressure Alarms are entry conditions into **AP-21.00**. Furthermore, with power at **65%**, the direction is to reduce turbine load to match steam **and** feed flows. This **will** also help to recover **MFW** Pump suction and discharge pressure.
- B. Incorrect because no trip criteria are **met** and **AP-21.00** directs power reduction.
- C. Incorrect because tripping a **MFW** Pump will not alleviate **the** issue and there is no procedural guidance to trip a **MFW Pump**. Typically a **MFW** Pump will be secured at about 40% power.
- D. Incorrect because there is no guidance to start a second heater drain pump. The correct response is to lower turbine load.

056 Condensate

A2.04: Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:  
Loss of condensate pumps.

39. 056AA1.26 001/1/1/DIESEL BREAKER/C/A 2.5/2.6/B/SR04301/R/MAB/SDR

The following conditions exist:

- A Loss of Off-Site Power has occurred
- #1 Emergency Diesel Generator has started but failed to auto load
- It has been determined that the auto-closure circuit for 15H3, #1 EDG Output Breaker, is inoperable and that 15H3 can be manually closed
- When the operator places the sync switch for 15H3 to "ON" he observes 120volts on the "incoming" meter, 0 volts on the "running" meter, and the synchroscope is stationary at "3-o'clock"

Which ONE of the following actions is necessary prior to closing 15H3?

- A. Raise EDG speed until the synchroscope is turning slowly in the fast direction, then close 15H3 at "11 o'clock".
- B. Momentarily press the "field flash" pushbutton, then sync and close 15H3.
- C. Raise EDG voltage until the running meter indicates 120 volts, then sync and close 15H3.
- D. No additional action is necessary. Close 15H3.

Surry

References:

ND-90.3-LP-1, Emergency Diesel Generator, Rev. 14

ND-90.3-LP-7, Station Service and Emergency Distribution Protection and Control,  
Rev. 17

Distractor Analysis:

- A. incorrect because the bus is dead. Raising EDG speed will not synchronize the phases.
- B. incorrect because it will not be possible to synchronize (nor is it necessary because the bus is dead). Also, field flash PB does not need to be pushed.
- C. Incorrect because raising the EDG voltage will not raise running voltage. Incoming voltage is the EDG voltage (not running voltage).
- D. Correct because the synchroscope has been turned on, there is no over-current or differential and the aux trip relay does not need to be reset (ND-90.3-LP-7 pg. 18). Therefore, all criteria for manually closing the breaker are met.

056 Loss of Off-Site Power

AA1.26: Ability to operate and / or monitor the following as they apply to the Loss of Off-Site Power: Circuit Breakers

40. 053AK3.01 001/1/1/VITAL AC/CA 4.1/4.4/B/SR04301/R/MAB/SDR

Which ONE of the following reasons correctly states why the reactor would be tripped for a sustained loss of Vital Bus II?

- A. Power to the Reactor Protection System is lost.
- B. Pressurizer pressure control is lost.
- C. Control of Steam Generator Feed Regulating Valve is lost.
- D. "B" Reactor Coolant Pump must be stopped.

Surry

References:

1-AP-10.02, Loss of Vital Bus II, Rev. 9

ND-90.3-LP-5, Vital and Semi-Vital Bus Distribution, Rev. 11

Bistractor Analysis:

- A. Incorrect because RPS is de-energize to trip. If due to other channel failures, etc., the loss of VB II will not preclude a trip if one is needed.
- B. Incorrect because Pzr P Controller will transfer to AUTO-HOLD, but **MANUAL** control is still possible, thus precluding the need for rx trip.
- C. Incorrect because FW-FCV-1488 Flow Controller will transfer to AUTO-HOLD, but **MANUAL** control is still possible, thus precluding the need for rx trip.
- D. Correct because Component Cooling is lost to the "B" RCP Lube Oil Cooler. RCP Parameters will eventually exceed limits (1-AP-10.02 Att. 1), requiring that the RCP be secured following a manual rx trip.

057 Loss of Vital AC Inst. Bus

AK3.01: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical bus.

Surry ILT Bank Question #223

41. 058AA2.01.001/1/1/LOSS.OEDC POWER/CIA.37141/N/SR04301/R/MAB/SDR

The following Unit 1 conditons exist:

- 1K-A8, UPS SYSTEM TROUBLE, annunciates
- 4K-A7, BATT SYSTEM 1A TROUBLE, annunciates
- An operator reports that Battery Charger DC Output for UPS 1A-1 reads 0 amps

Which ONE of the following correctly describes the power supply to the associated DC and Vital AC buses?

- A. BC Bus 1A-1 will be supplied by only Battery 1A as indicated by DC Bus voltage slowly trending down over time and Vital AC Buses 1 and 1A will be supplied by Bus 1H1-1.
- B. DC Bus 1A-1 will be supplied by only Battery 1A as indicated by DC Bus voltage slowly trending down over time and Vital AC Buses 1 and 1A will be supplied by Bus 1H1-2.
- C. DC Bus 1A-1 will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at 125 VDC and the Vital AC Buses 1 and 1A will be supplied by 1H1-1.
- D. DC Bus 1A-1 will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at 125 VDC and the Vital AC Buses 1 and 1A will be supplied by 1H1-2.

Surly

References:

ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11

ND-90.3-LP-6, 125 VDC Distribution, Rev. 10

1K-A7, BATT **SYSTEM 1A TROUBLE**, Rev. 5

1K-A8, UPS SYSTEM TROUBLE, Rev. 1

11448-FE-1G, Sheet 1 of 1, 125V DC One Line Diagram - Surty Power Station Unit 1,  
Rev. 33

Distractor Analysis:

- A. Incorrect because the battery should not be supplying the DC Bus alone. The DC Bus **is** being supplied by the other **UPS** from 1H1-2. **Also**, vital AC Buses 1 and 1A are being supplied by Bus 1H1-2, which is the alternate AC source.
- B. Incorrect because the battery **should** not be supplying the DC Bus alone. **The DC Bus** is being supplied by the other **UPS** from 1H1-2.
- C. Incorrect because the Vital AC Buses 1 and 1A are being supplied by Bus 1H1-2, which is the alternate AC source.
- D. Correct because the other **UPS will** still be supplying DC Bus 1A-1 and the Alternate **AC** Source 1H1-2 will supply Vital AC Buses 1 and 1A.

058 **loss** of DC Power

AA2.81: 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.

42...059A1.03001/211/MAIN FEEDWATER/MEM 2.7/2.9/N/SR04301/R/MAB/SDR

Which ONE of the following set of practices should be observed by operators for starting the second Main Feedwater Pump per GOP-1.5 (Unit Startup, 2% Reactor Power to Max Allowable Power) and OP-FW-004 (Main Feedwater System Operation)?

- A. The second Main Feedwater Pump should be started prior to exceeding 50% power to preclude problems with main feedwater flow capability. Following pump start, if the Main Feedwater Pump Recirculation Valve is in AUTO, the operator should observe that valve closure will occur as the feed flow rises above 3000 gpm.
- B. The second Main Feedwater Pump should be started between 50% power and 65% power to preclude problems with main feedwater flow capability. Following pump start, if the Main Feedwater Pump Recirculation Valve **is** in AUTO, the operator should **observe** that valve closure will occur as the feed flow rises above 3286 gpm.
- ~~C.~~ The second Main Feedwater Pump **should** be started prior *to* exceeding 50% power to preclude problems with main feedwater flow capability. Operating the second Main Feedwater Pump on recirculation with the discharge MOV closed should be minimized to prevent **overpressurization** of the piping between the discharge check valve and the MOV as the system heats.
- D. The second Main Feedwater Pump should be started between 50% power and 65% power to preclude problems with main feedwater flow capability. Operating the second Main Feedwater Pump on recirculation with the discharge MOV closed should be minimized to prevent overpressurization of the piping **between** the discharge check valve and the MOV as the system heats.

Surry

References:

4-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power, Rev. 32  
1-OP-FW-004, Main Feedwater System Operation, Rev. 8  
ND-89.3-LP-3, Main Feedwater System, Rev. 12

Distractor Analysis:

- A. Incorrect because recirc should modulate closed at 4000 gpm.
- B. Incorrect because recirc **should** modulate closed at 4000 gpm.
- C. Correct because of NOTE on Pg. 34 of 44 of GOP-1.5 and **CAUTION** on Pg 12 of 34 of OP-FW-004.
- D. Incorrect because second feedwater pump should be started prior to 50% power.

059 Main Feedwater

A1.03 Ability to predict and / or monitor changes in parameters (to prevent **exceeding** design limits) associated with operating the **MFW** controls including: Power level restrictions for operation of **MFW** pumps and valves.

43. 059AA1.01 001/1/2/LIQUID RAD RELEASE/C/A 3.5/3.5/M/SR04301/R/MAB/SDR

The following Unit 1 conditions exists:

- A Large Break Loss of Coolant Accident has occurred
- The "B" Train of Recirc Spray (RS), the only available train, is in service
- 1-RM-G7, DISCH TNL ALERT / FAILURE, annunciates
- 1-RM-A8, RS/SW HX B ALERT/FAILURE, annunciates
- Reactor Operator notes the RS/SW HX B Monitor ~~is~~ trending up, but the Discharge Tunnel Rad Monitor **is** indicating all EEEEEEs with Red and Yellow Lights Lit and Green Light out.

Which ONE of the following is the correct operator response?

- A. Ensure no additional releases are in progress and secure RS.
- B. Ensure no additional releases are in progress, and increase radiation monitoring.**
- C. Verify all automatic actions have occurred and reset the Discharge Tunnel Digital Rate Meter and perform a source check.
- D. Verify all automatic actions have occurred and raise the Discharge Tunnel Monitor **set** point.

Surry

References:

ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8  
1-RM-G7, DISCH TNL ALERT / FAILURE, Rev. 4  
1-RM-A8, RS/SW HX B ALERT/FAILURE, Rev. 3

Distractor Analysis:

- A. Incorrect because the last available RS train should not be secured, as stated in RM-G7 and RM-A8 Caution Statements. Plausible because this is the correct course of action if the other train was available.
- B. Correct because the last train of RS should not be secured. Other rad monitors should be checked to see if blowdowns have been diverted, to verify that there is no CCW/SW HX leak, and to verify that no CP Bld Liquid releases are occurring. Additional monitoring is called for by the ARPs due to the fact that the last train of RS should not be secured.
- C. Incorrect because there are no automatic actions to verify. Plausible because the applicant **may not** know that there **are no** auto actions associated **with** these particular monitors. With a failed monitor, ARPs will direct a reset and source check, which adds to the plausibility.
- D. Incorrect because there are no automatic actions to verify. Plausible because the applicant may not know that there are not auto actions associated with these particular monitors and it *is* not uncommon for an alarm setpoint to be raised to alert operators of worsening conditions. The Discharge Tunnel Monitor has the indications of being failed, therefore adjusting the setpoint is not a success path.

059 Accidental Liquid Radwaste Release

AA1.01: Ability to operate and / or monitor the following as they apply to the Accidental Liquid Wadwaste Releases: Radioactive-liquid monitor

Modified Surry ILT Bank Question #1977

44. 061A1.04 001/2/1/AFW CONDENSATE/MEM 3.9/3.9/N/SR04301/R/MAB/SDR

1-CN-TK-1, Emergency Condensate Storage Tank (ECST), is supplying AFW Pumps for Residual Heat Removal via Steam Generators. 1J-F4, CST 110,000 GAL LO LVL, has annunciated. ECST level is 90% and lowering.

Which ONE of the following is correct regarding refilling of the ECST?

- A. Filling shall commence prior to the ECST level reaching 54%. AFW pumps must be secured prior to commencing the fill.
- B. Filling may commence after the ECST level drops below 60,000 gallons as long as refill begins within two hours of securing the AFW pumps.
- C. AFW Pumps must be secured prior to commencing the fill and the ECST must be filled within two hours.
- D. Filling of the ECST shall commence prior to the ECST level reaching 54%. AFW pumps may continue to operate during the refill.

Surry

References:

ND-89.3-LP-4, Auxiliary Feedwater System, Rev. 19  
1J-F4, CST 110,000 GAL LO LVL, Rev. 3  
Tech Spec 3.6-1, Amendment No. 224 and 220

Distractor Analysis:

- A. Incorrect because AFW pumps may continue to run during refill based on ARP 1J-F4 Note.
- B. Incorrect because volume must remain above 60,000 gal (54%).
- C. Incorrect because AFW pumps do not need to be secured for refill.
- D. Correct based on all three of the above references.

061 Auxiliary Feedwater

AI.04: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW source tank level.

45. 061AA2 01001/1/2/ARM RAD MONITOR/MEM.3.5/3.7/B/SR04301/R/MAB/SDR \_\_\_\_\_

If a spent fuel assembly is damaged by being dropped in the spent fuel pool, which ONE of the following pairs of radiation monitors would indicate an increase in radiation level?

- A. Spent Fuel Pit Bridge Crane Radiation Monitor and Auxiliary Building Control Victoreen Area Radiation Monitor
- B. Ventilation Vent Particulate Radiation Monitor and Auxiliary Building Control Victoreen Area Radiation Monitor
- C. Spent Fuel Pit Bridge Crane Radiation Monitor and Ventilation Vent Gaseous Radiation Monitor

Surry (Utility needs to add correct RM equipment numbers.)

References:

ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8  
0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 18  
0-RM-D3, 1-RM-RI-153 HIGH, Rev. 4  
0-RM-B4, 1-RM-RI-152 HIGH, Rev 8

Distractor Analysis: (maybe get Some help to provide a little better distractor analysis?)

- A. Incorrect because **the** Aux Bld Control Victoreen Area Radiation Monitor would not show an increased indication.
- B. Incorrect because the Aux Bld Control Victoreen Area Radiation Monitor would not show an increased indication.
- C. Correct because both monitors would **show** an increased indication.
- D. incorrect because a liquid waste process effluent monitor would not see the **results** of the failed fuel.

061 ARM System Alarms

AA2.01: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (**ARM**) System Alarms: ARM panel displays.

Surry Requal Bank Question #118

46\_062A1.01.001/1/2/EDG DIESEL/MEM 3.4/3 8/N/SR04301/R/MAB/SDR

The following conditions were noted during the performance of 1-OPT-EG-001, Number 1 Emergency Diesel Generator Monthly Start Exercise Test:

- The EDG was loaded at a rate of 550 KW/MIN
- The Maximum load attained was 2650 KW
- The Maximum KVAR was 500 KVAR out
- The output voltage was stable at 4300 VAC

Which ONE of the following was in violation of the EBG Precautions and Limitations per 1-OPT-EG-001?

- A** Load Rate
- B. Maximum Load
- C. Maximum KVAR out
- D. Output voltage

Surry

References:

1-OPT-EG-001, Number 1 Emergency Diesel Generator Monthly Start Exercise Test, Rev. 24

1-OP-EG-001, Number 1 Emergency Diesel Generator, Rev. 17

Distractor Analysis:

- A. Correct because the loading rate should not exceed 500 KW/MIN during normal operations.
- B. Incorrect because max load rating is 2750 KW.
- C. Incorrect because max KVAR out is 500 KVAR.
- D. Incorrect because output voltage should be maintained between 4000 and 4400 VAC.

062 AC Electrical Distribution

A1.01: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significant D/G load limits.

47. 062AA1 06 001/1/1/SERVICE WATER/MEM 2.9/2.9/B/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Power = 100%
- During testing, an Intake Canal Low Level Isolation Signal is inadvertently actuated

Which ONE of the following correctly states the plant response caused by the Low Level Isolation Signal?

- A. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will close and can only be reopened after 5 minutes.
- B. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will go to 25% open and can be fully opened after 5 minutes.
- C. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will close and can be reopened when the low level signal is reset.
- D. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will go 25% open and can be fully opened when the low level signal is reset.

Surry (Utility needs to verify technical accuracy and provide any additional reference material (electrical print?)).

References:

ND-89.5-LP-2, Sewice Water **System**, Rev. 20

Distractor Analysis:

- A. Incorrect because the valves will close, but cannot be re-opened until Canal Low Level Isolation Signal is cleared. If the valves would have been closed due to a CLS, then they could have been re-opened after 5 minutes even without the CLS cleared.
- B. Incorrect because the valves will go fully closed.
- C. Correct because the valves will close, but cannot be re-opened until Canal Low Level Isolation Signal is cleared. If the valves would have been closed due to a CLS, then they could be opened after five minutes without resetting CLS.
- D. Incorrect because, as states above, the valves will close.

062 Loss of Nuclear Svc Water

AA1.06: Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Sewice Water (SWS): Control of flow rates to components cooled by the SWS.

48. 063A4.01 001/21/BREAKERS/C/A 28/3.1/N/SR04301/R/MAB/SDR

Unit 1 was operating at 68% power when the following plant conditions developed:

- 1K-A7, BATT SYSTEM 1A TROUBLE, alarm annunciates
- "A" SG PORV Indicating Lights are not lit
- MSTV Indicating lights are not lit
- POWW 1455C/1456 Indicating Lights are not lit
- "A", "D", and "H" Breaker Indicating Lights are not lit
- There is no indicated letdown flow
- The Turbine Driven AFW Pump is running

Which ONE of the following describes the plant conditions assuming **no** other failures in addition to the *cause* of the above conditions?

- A. The reactor will automatically trip. The turbine will automatically trip when the reactor is manually tripped.
- B. The turbine will automatically trip. The reactor will automatically trip due to the automatic turbine trip.
- C. The reactor must be manually tripped. The turbine must also be manually tripped.
- D. The reactor will automatically trip. The turbine will not automatically trip and must be manually tripped.

Surly

References:

ND-90.3-LP-6, 125 Vdc Distribution, Rev. 10

Distractor Analysis:

- A. **Correct** because the reactor will automatically **trip on** loss of voltage to the "A" RTB UV coil to a loss of the "A" DC Bus (see ND-90.3-LP-6). The turbine will not trip until the reactor is manually tripped in accordance with E-0.
- B. Incorrect because the reactor will automatically trip due to loss of voltage to the "A" RTB UV coil due to the loss of the "A" dc Bus.
- C. Incorrect because the reactor does not need to be manually tripped to trip the reactor and the turbine will automatically trip when the reactor is tripped per E-0.
- D. Incorrect because the turbine does not need to be manually tripped. The turbine will trip when the reactor is manually tripped in E-0 or when the other train of RPS occurs due to low SG levels.

063 DC Electrical Distribution

A4.01: Ability to manually operate and/ or monitor in the control room: Major breakers and control power fuses.

49. 064K2.01 001/2/1/DG AIR COMPRESSOR/C/A 2.7/3.1/N/SR04301/R/MAB/SDR

The following plant conditions exist:

- Bus 1J1 voltage drops to 407 volts and returns to 480 volts seven seconds later and remains stable
- Bus 2J1-1 voltage *is* 441 volts and stable

Which ONE of the following correctly states the source of power for Diesel Generator #3's Air Compressors?

- A. Bus 1J1 remained the power supply throughout the seven second voltage drop.
- B. **Six** seconds after the voltage dropped on Bus 1J1, Bus 2J1-1 became the power supply. Bus 2J1-1 will remain the power supply until manually transferred back to Bus 1J1
- C. **Six seconds** after the voltage dropped on Bus 1J1, Bus 2J1-1 became **the** power supply. Bus 2J1-1 **will** remain **the** power supply for 30 minutes with Bus 1J1 greater than 440 volts, at which time it will automatically return to Bus 1J1.
- 5. Six seconds after the voltage dropped on **Bus 1J1**, Bus 2J1-1 became the power supply. Bus 2J1-1 **will** remain the power supply for six seconds **with** Bus 1J1 greater than **440** volts, at which time it will automatically return to Bus 1J1.

Surry

References:

ND-90.3-LP-1, Emergency Diesel Generator, Rev. 14

P&ID 11448-FE-1AA, Appendix R Evaluation Electrical One Line Diagram Surry Power Station Unit 1, Rev. 23

P&ID 11448-FE-1P1, 480V One Line Diagram MCC 1J1-1A Surry Power Station Unit 1, Rev. 4

Distractor Analysis:

- A. Incorrect because 1J1 voltage was less than 410v for greater than 6 seconds. Therefore, 2J1-1 became the power supply after 6 seconds. The ABT will check for 2J1-1 voltage greater than 440v prior to swapping to the alternate power supply.
- B. Incorrect because this *is* the alternate power supply and the ABT *is* a normal-seeking ABT. Therefore, at the beginning of this sequence, the power supply would have been 1J1.
- C. Correct because 1J1 voltage was less than 410v for greater than 6 seconds. Therefore, 2J1-1 became the power supply after *six* seconds. The ABT will check for 2J1-1 voltage greater than 440v prior to swapping to the alternate power supply. When the normal power supply voltage *is* restored to > 440v, a 30 minute time delay *is* started. If the voltage remains above 440v for 30 minutes, then it transfers **back** to the normal power supply (1J1).
- D. Incorrect because of the 30 minute time delay mentioned above.

064 Emergency Diesel Generator

K2.01: Knowledge of **bus** power supplies to the following: Air Compressors.

50. 065AA2.01 001/1/1/AIR IA RHR/C/A 2.9/3.2/B/SR04301/R/MAB/SDR

The following plant conditions exist:

- Unit 2 is in intermediate shutdown
- Operators are attempting to warm the RHR system
- An instrument air leak has developed, but the location *is* yet to ~~be~~ determined
- An Operator reports the sound of compressed air leaking in the area of the RHR pump platform.
- 1B-E6, IA **LOW HDR PRESS/IA COMPR 1 TRBL**, has annunciated
- Instrument air pressure is approximately stable at 60 psig

Which **ONE** of the following correctly explains the potential effect on warming the RHR system?

- A. If the air leak is a rupture upstream of the isolation valve for the air supply to HCV-1758 (RHR Heat Exchanger Outlet Valve), the valve will fail closed. The line may be crimped if the leak will not **affect** vital control instruments. Operators should use the portable air bottle, via quick disconnect, to operate the valve.
- B. If the air leak **is** a rupture upstream of the isolation valve for the air supply to HCV-1758 (RHR Heat Exchanger Outlet Valve), the valve will fail open. The line may be crimped if the leak will not affect vital control instruments. Operators should use the portable air bottle, via quick disconnect, to operate the valve.
- C. If the air ~~leak~~ is a rupture upstream of the isolation valve for the air supply to HCV-1142 (CVCS Flow Regulator Control Valve), the valve will fail closed. The line may be crimped if the **leak** will not affect vital control instruments.
- D. If the air leak is a rupture upstream of the isolation valve for the air supply to HCV-1142 (CVCS Flow Regulator Control Valve), the valve will fail open. The line may **be** crimped if the leak will not affect vital control instruments.

Surry

References:

ND-88.2-LP-1, Residual Heat Removal System Description, Rev. 8 (Pages 9, 10, 11)  
ND-88.2-LP-2, Operation of Residual Heat Removal System, Rev 15  
P&ID 11448-FM-087A Sh 2 of 2, Residual Heat Removal System, Rev. 26  
P&ID 11448-FM-075E Sh 1 of 2, Compressed Air System, Rev. 43  
1B-E6, IA LOW HDR PRESS/IA COMPR 1 TRBL, Rev. 9

Distractor Analysis:

- A. Incorrect because HCV-1758 fails open and cannot be operated with a portable air bottle. Plausible because the applicant may get confused on which valve in this flowpath has the portable air bottle feature.
- B. Incorrect because HCV-1758 cannot be operated with a portable air bottle. Plausible because the applicant may get confused or which valve in this flowpath has the portable air bottle feature.
- C. Correct because HCV-1142 is fail closed and this is the flow path for system warmup. ARP states that leaks may be stopped via crimping if the leak will not affect vital instrumentation.
- D. Incorrect because HCV-1142 fails closed. Plausible because the applicant may get confused on failure modes of HCV-1142, especially since it does have a backup air bottle feature for App. R purposes.

065 Loss of Instrument Air

AA2.01: Ability to determine and interpret the following as they apply to the loss of instrument air: Cause and effect of low pressure instrument air alarm.

51. 067G2.4.18 001/1/2/FIRE KWST CC/MEM 2 7/3.6/B/SR04301/R/MAB/SDR

In FCA-8.00, Limiting Auxiliary Building Fire, if Charging Pump CC Pumps are not running, the operator is directed to shift charging pump suction to the WWST. Which ONE of the following describes the basis for this step?

- A. Suction is shifted to the WWST to maximize boron injection before the charging pumps overheat and are lost due to a time-overcurrent breaker trip.
- B. Suction is shifted to the RWST to maximize boron injection before the charging pumps overheat and are lost due to a instantaneous-overcurrent breaker trip.
- C. The loss of Charging Pump CC will eventually result in a loss of VCT level due to a loss of makeup; therefore suction is shifted to the RWST.
- D. The RWST supplies cooler water to the Charging Pumps; thereby minimizing the cooling requirements for the Charging Pumps.

Surry (Utility needs to verify technical accuracy and supply additional supporting material if any is available.)

References:

ND-95.6-LP-3, Safe Shutdown Fire FCAs, Rev. 5  
0-FCA-8.00, Limiting Auxiliary Building Fire, Rev. 13

Distractor Analysis:

- A. Incorrect because the concern is with overheating the pump, not maximizing boron injection prior to the pump overheating. Supplying cooler RWST water will reduce the pump temperatures.
- B. Incorrect because the concern is with overheating the pump, not maximizing boron injection prior to the pump overheating. Supplying cooler RWST water will reduce the pump temperatures.
- C. Incorrect because VCT level will not be reduced as a result of no CC.
- D. Correct because cooler RWST water will help reduce pump temps when CC is lost.

067 Plant Fire On-Site

G2.4.18: Knowledge of specific bases for EOPs.

52. 068K4 01 001/2/2/RADIATION MONITOR/C/A 3.4/4.1/B/SR04301/R/MAB/SDR

**The following Conditions exist:**

- Both Units are at 180% Power
- Unit 1 Operators have discovered indication of a small tube leak in the "A" Steam Generator ~~for~~ their Unit
- Spent Fuel is being moved in the Spent Fuel Storage Pool to facilitate rack inspections
- 0-RM-M4, 1-VG-RI-104 HIGH, alarms
- All Radiation Monitors appear to be operating satisfactorily
- Ventilation and Radiation Monitors are in their normal alignment

Which ONE of the following could cause RM-VG-104 (#1 Vent Stack RM) to detect higher than *normal* activity?

- A. A Steam Generator Tube Leak on Unit 1.
- B. A spill of high activity coolant in the Chemistry Hot Lab.
- C. A spill of high activity coolant in the High Rad Sample System Room.
- D. A dropped fuel assembly in the fuel building.

Sur9

References:

0-RM-M4, 1-VG-RI-104 HIGH, Rev. 2

0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 18

ND-95.3-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8

Distractor Analysis:

- A. Incorrect because the normal configuration for the ventilation system would not have Main Condenser Air Ejector aligned to discharge to the Number 1 Vent Stack upstream of Radiation Monitor 1-VG-RM-104.
- B. Correct because 0-RM-M4 alarming could be caused by a coolant spill in the Chem Hot Lab according to the ARP.
- C. Incorrect a spill in the High Radiation Sample System Room would not cause this alarm according to the ARP.
- D. Incorrect because fuel clad damage would not be detected by RM-VG-104 when in its normal configuration. 0-AP-22.00 does not list RM-VG-104 as a potential means of indication for damaged fuel clad.

068 Liquid Wadwaste

K4.01: Knowledge of design feature(s) and / or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids.

Sur9 Requal Exam Bank Question #462 (ID:ARP0076)

53\_071K4.06001/2/2/WASTE GAS GASEOUS/MEM2.7/3.5/NSR04301/R/MAB/SDR

A discharge of a waste gas decay tank is in progress when RM-GW-101 reaches the high alarm setpoint and alarm 0-RM-K3, 1-GW-RI-101 HIGH, annunciates. Which ONE of the following is NOT an automatic action initiated by the high radiation levels from the waste gas decay tank release?

- A. 1-GW-FCV-101, Decay Tank Bleed Isolation Valve, closes.
- B. 1-GW-FCV-160, CTMT Vacuum Pump Discharge Isolation Valve closes.
- C. 1-GW-FCV-260, CTMT Vacuum Pump Discharge Isolation Valve, closes.
- D. Associated vacuum pumps trip.

Surry (Utility needs to verify technical accuracy)

References:

ND-92.4-LP-1, Gaseous and Liquid Waste Processing Systems, Rev. 8

ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8

0-RM-K3, 1-GW-RI-101 HIGH, Rev. 0

Distractor Analysis:

- A. Incorrect because according to ARP, this valve will close on reaching the high alarm setpoint.
- B. Incorrect because according to ARP, this valve will close on reaching the high alarm setpoint.
- C. Incorrect because according to ARP, this valve **will** close on reaching the high alarm setpoint.
- D. Correct because the pumps must be manually secured if GW-160 or GW-260 are closed. This info is in a CAUTION in the AWP and a step is provided in the ARP to secure the pumps following the closure of GW-160 / 260.

071 Gaseous and Liquid Waste Processing Systems

K4.86: Knowledge of design(s) features and / or interlocks which **provide** for the following: Sampling and monitoring of waste gas release tanks.

54. 073G2.1.23 001/2/1/PROCESS RADIATION/C/A 3.9/4.0/N/SR04301/R/MAB/SDR

Which **ONE** of the following is sufficient conclusive indication to warrant a correct entry into AP-16.00, Excessive RCS Leakage?

- A. Rising containment humidity, rising containment temperature, and rising containment pressure.
- B. Rising steam generator water level, rising charging flow, and rising Condenser Air Ejector Radiation Monitor reading.
- C. Rising Condenser Air Ejector Radiation Monitor reading, rising steam generator blowdown radiation monitor reading, and stable containment pressure.
- D. Rising containment sump level, lowering pressurizer pressure, and rising containment pressure.

Surry

References:

1-AP-16.00, Excessive RCS Leakage, Rev. 11

1-AP-24.00, Minor SG Tube Leak, Rev. 8

1-E-0, Reactor Trip or Safety Injection, Rev. 46

ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8

Distractor Analysis:

- A. Incorrect because a steam line break can cause containment humidity, temperature, and pressure to rise. Distractor is plausible because these are all possible for an RCS leak.
- B. Correct because SG water level is indication that there may be a tube leak or steam/feed mismatch. The charging flow and Air Ejector Wad monitor corroborates that the problem is tube leakage.
- C. Incorrect because these parameters **are** indications that there may be a tube leak; however, these **same** indications may present themselves with a fuel failure or crud burst. Distractor **is** plausible because these parameters may indicate as stated if RCS leakage actually exists. Distractor is incorrect because these parameter trends may be caused by increased RCS activity.
- D. Incorrect because the Combination **of** these parameters may be caused by a steam leak/break. Distractor is plausible because these parameters may actually change as indicated during an RCS leak.

073 Process Radiation Monitoring

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

55. 076AK2.01 001/1/2/RCS ACTIVITY/C/A 2.6/3.0/B/SR04301/R/MAB/SDR

Hydrogen peroxide has just been added to Unit 2 RCS resulting in an increase in the primary coolant activity. The first indication that the activity level has increased will be seen on the \_\_\_\_\_ and the team should \_\_\_\_\_

- A. Containment particulate radiation monitor; increase flow through the letdown cation bed.
- B. Letdown radiation monitor; monitor letdown filter differential pressure.
- C. Letdown radiation monitor; monitor seal return filter differential pressure.
- D. Containment particulate radiation monitor; decrease flow through the letdown cation bed.

Surry

References:

ND-93.05-LP-1, Pre-TMI Radiation Monitoring System  
ND-88.3-LP-3, Seal Injection, Rev. 6

Distractor Analysis:

- A. Incorrect because containment particulate radiation monitor would not change significantly.
- B. Correct because letdown radiation monitors would indicate quickly due to hydrogen peroxide increasing reactor coolant activity and letdown filter dP would also rise.
- C. Incorrect because the hydrogen peroxide should not affect the seal return dP, at least not as readily or as soon as the letdown filter dP. There is 8 gal of CVCS water that goes to each RCP for seal injection. Five of these gallons flows down the shaft past the thermal barrier and ends up in the RCS. The other three gallons eventually passes through the seal return filter. The CVCS water that enters the RCP seal area has already been filtered prior to getting to the RCP seals. This prefiltering is designed to protect the seals. The water coming from the RCP seal area should be relatively clean CVCS water, not RCS water; therefore making the seal return filter a relatively poor indicator of a crud burst.
- D. Incorrect because containment particulate radiation monitor Would not change significantly.

Surry Bank ILT Exam Question #1606

076 High Reactor Coolant

AK2.01: Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors.

56. 076K2.04 001/2/L/SERVICE WATER/C/A 2 5/2.6/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- A Large Break LOCA occurred 45 minutes ago
- Recirculation Spray is operating
- MCC 11-11-2 de-energizes

Which ONE of the following correctly describes the impact on Sewice Water to and from the Recirc Spray Heat Exchangers?

- A. Recirc Spray Heat Exchanger 1-RS-E-1A Sewice Water Inlet (MOV-SW-104A) and Outlet (MOV-SW-105A) Valves de-energize.
- B. Recirc Spray Heat Exchanger 1-RS-E-1B Service Water Inlet (MOV-SW-104B) and Outlet (MOV-SW-105B) Valves de-energize.
- C. Recirc Spray Heat Exchanger Service Water Inlet (MOV-SW-103A) and Recirc Spray Heat Exchanger 1-RS-E-1A Sewice Water Inlet (MOV-SW-104A) Valves de-energize.
- D. Recirc Spray Heat Exchanger Service Water Inlet (MOV-SW-103B) and Recirc Spray Heat Exchanger 1-RS-E-1B Service **Water** Inlet (MOV-SW-104B) Valves de-energize.

Surry

References:

ND-91-LP-6, Recirculation Spray System, Rev. 9

ND-89.5-LP-2, Service Water System, Rev. 20

P&ID 11448-FE-1M, Sh 1 of 1, 480V One Line Diagram Surry Power Station - Unit 1, Rev. 59

P&ID 11448-FE-1L, Sh 1 of 1, 480V One Line Diagram Surry Power Station - Unit 1, Rev. 52

Bistractor Analysis:

- A. Correct because MOV-SW-104A and 105A are both powered from 1H1-2.
- B. Incorrect because MOV-SW-104B and 105B are both powered from 1J1-2.
- C. Incorrect because MOV-SW-103A is powered from 1H1-1.
- D. Incorrect because MOV-SW-103B is powered from 1J1-1 and MOV-SW-104B is powered from 1J1-2.

076 Service Water

K2.04: Knowledge of bus power supplies to the following: Reactor building closed cooling water.

57\_078A4.01.001/2/1/INSTRUMENT AIR/C/A.3.1/3.1/B/SR04301/R/MAB/SDR

Unit 1 is at 50% power and the team is experiencing problems controlling feedwater flow. An Instrument Air Low Pressure Alarm is received in the Control Room. While monitoring Instrument Air pressure, the RO notes pressure is 50 psig and slowly lowering.

Which ONE of the following actions should be taken?

- A. Commence a slow power reduction to Hot Shutdown.
- B. Commence a fast power reduction to Cold Shutdown.
- C. Trip the Reactor and go to 1-E-0, Reactor Trip or Safety Injection.
- D. Isolate Sewice Air frsm instrument Air and start the Sullair Diesel.

Surry

References:

ND-92.1-LP-1, Station Air **Systems**, Rev. 13

1B-E6, IA LOW HDR PRESS/ IA COMPR 1 TRBL, Rev. 9

0-AP-40.00, Non-recoverable Loss of instrument Air, Rev. 17

Distractor Analysis:

- A. Incorrect because 1B-E6 and AP-40.00 directs rx trip, not power reduction.
- B. Incorrect because 1B-E6 and AP-40.00 directs rx trip, **not** power reduction. (Initial distractor from exam bank was changed because it may **have** been a second correct answer).
- C. Correct because this *is* the guidance provided by 1B-E6 and AP-40.00.
- D. Incorrect because 1B-E6 and AP-40.00 directs rx trip, not power reduction when pressure reaches 50 psig.

078 Instrument Air

A4.01: Ability to manually operate and / or monitor in the control room: Pressure gauges.

Surry Requal Exam Bank Question 428

58. 078K4.02 001/2/1/INSTRUMENT AIR/MEM 3.2/3.5/B/SR04301/R/MAB/SDR

With ALL air systems aligned in the automatic mode, which ONE of the following describes the operation of the Station Instrument Air (IA) System for Unit 1?  
(Assume no operator action is taken.)

- A. Instrument Air is normally supplied by the Service Air System and the system is backed up by IA when IA pressure reaches 95 psig.
- B. Instrument Air is normally supplied by IA Compressors and the system is manually backed up by the Sullair Diesel.
- C. Instrument Air is normally supplied by the Service Air System and is backed up by the IA System when IA pressure reaches 90 psig.
- D. Instrument Air is normally supplied by the Service Air System and is backed up by the Condensate Polishing Instrument Air System when IA pressure reaches 98 psig.

Surry

References

ND-92.1-LP-1, Station Air Systems, Rev. 13

Distraetsr Analysis:

- A. Incorrect because pressure must drop below 90 psig for IA to backup Service Air.
- B. Incorrect because the IA System is normally supplied by Sewice Air.
- C. Correct because Service Air is the normal supply and IA is the backup when pressure drops below 90 psig.
- D. Incorrect because IA is not backed up by the Condensate Polishing Instrument Air System when pressure drops to 98 psig. It is backed up by the IA System when pressure drops below 90 psig.

078 Instrument Air

K4.02: Knowledge of the IAS design feature(s) and or interlock(s) which provide for the following: Cross-over to other air systems.

Surry Requal Bank Question #512

59. 103A4.04 001/2/1/CLS CONTAINMENT/C/A 3.5/3.5/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exists:

- A steam line rupture in Containment occurred several minutes ago
- Maximum Containment Pressure reached 24 psia
- Containment Pressure Transmitters now read:
  - PT-LM-100A = 17.7 psia
  - PT-LM-100B = 17.8 psia
  - PT-LM-100C = 17.6 psia
  - PT-LM-100D = 17.9 psia

Which ONE of the following correctly describes resetting of Consequence Limiting Safeguards (CLS) given the above conditions?

- A. The CLS TRAIN A(B) RESET PERMISSIVE annunciator is lit. CLS HI and CLS HI-HI may be reset at this time. Upon reset, the multiplying relays will energize.
- B. Neither CLS HI or CLS HI-HI may be reset at **this** time. **The** multiplying relays are de-energized.
- C. The CLS HI-HI RESET PERMISSIVE annunciator is lit. **CLS** HI-HI may be reset at this time. Upon reset the multiplying relays will de-energize.
- D. Neither CLS HI or CLS HI-HI may be reset at this time. The multiplying relays are energized.

Surry

References:

ND-88.4-LP-2, Containment Vessel, Rev. 8

ND-91-LP-5, Containment Spray System, Rev. 13

Distractor Analysis:

- A. Incorrect because pressure must be reduced to less than 14.2 psia on 2/4 channels to reset both Hi and Hi-Hi subsystems.
- B. Correct because pressure must be reduced to less than 14.2 psia on 2/4 channels to reset both Hi and Hi-Hi subsystems. **Also**, when CLS is actuated, the multiplying relays are de-energized.
- C. Incorrect because pressure must be reduced to less than **14.2** psia on 2/4 channels to reset both Hi and Hi-Hi subsystems. Also, when **CLS** is actuated, the multiplying relays are de-energized.
- D. Incorrect because when **CLS** is actuated, the multiplying relays are de-energized.

103 Containment

A4.04: Ability to manually operate and / or monitor in the control room: Phase A and Phase B resets.

68. G2.1 11001/3//TECH SPECS/C/A.3.0/3.8/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Plant is at 74% power after just completing a rapid power reduction due to Heater Brain Pump problems
- Axial Flux Difference was outside of the Target Band on 11/03/2003 from 0800 hours to 0845 hours
- Axial Flux Difference was outside of the Target Band on 11/04/2003 from 0740 hours to 0840 hours
- The Axial Flux Difference has remained within the Technical Specification Limits of Figure 3.12-3, Axial Flux Difference Limits As A Function Of Rated Power: for the entire time

Which ONE of the following actions are required by Technical Specifications?

- A. Reactor power **was** required to be less than 50% by 0825 hours on 11/04/2003.
- B. Reactor power was required to **be** less than 50% **by** 0855 hours on 11/04/2003.
- C. Reactor power was required **to** be less than 50% **by** 0910 hours on 11/04/2003.
- D. No power reduction was required, **but** power should not have been raised above 75% until Axial Flux Difference was within the Target Band.

Surry

Reference:

Technical Specification 3.12.B.4.b.(1), Amendment No. 186

Technical Specification 3.12.B.4.b.(2), Amendment No. 186

Distractor Analysis:

- A. Correct because AFD may deviate from its target band for one hour within a **24** hour period. When this is violated, then power must **be** reduced to less than 50% within 30 minutes. From 11/03 @ 0800 hrs to 11/04 @ 0755 hrs a total of one hour outside of target band **was** accumulated. Therefore, **by** 0825 hrs (30 minutes later) power must be **less than** 50%.
- B. Incorrect because because the correct answer is as described in above analysis. Plausible because 0855 hours is **60** minutes after 0755 hrs, which is when the 30 minute clock starts to have power less than 50%.
- C. Incorrect because the correct answer is as described in above analysis. Plausible because 0910 hrs is 30 minutes after 0840 hrs, which was given as the second time frame where AFD was outside of its target band.
- D. incorrect because because the correct answer is as described in above analysis. Plausible because candidate may confuse 50% and 75% power restrictions.

G2.1.11

Knowledge of less than 1 hour technical specification action statements for systems.

61. G2.1.25 001/1/1/RHR/C/A 3.1/3.8/N/SR04301/R/MAB/SDR

The following conditions exist:

- Unit 1 has been shutdown for 10 days for SG tube plugging
- RCS water level is being maintained at 12.4 feet as indicated on 1-RC-LI-100A
- The "B" and "C" loops are isolated with the primary and secondary SG manways removed for SG tube plugging
- The reactor vessel head is tensioned
- The "A" RHR pump is in operation with oscillating amperage indications
- Flow indication 1-RH-FI-1605 is oscillating between 2500 and 2760 gpm.

Which ONE of the following actions is appropriate for the SRO to direct in accordance with AP-27.00, Loss of Decay Heat Removal Capability?  
(AS-29.00 Attachments 1 and 2 provided)

- A. **Raise** RCS level to 12.5 feet as indicated on 1-RC-LI-100A and stabilize flow at 2600 gpm.
- B. Throttle open 1-RH-HCV-1758 and throttle **close** 1-RH-FCV-1605 to reduce RHR flow to 2200 gpm.
- C. Throttle close 1-RH-HCV-1758 and throttle open 1-RH-FCV-1605 to reduce RHR flow to 1200gpm.
- D. Throttle close 1-RH-FCV-1605 to reduce ~~W-R~~ **Flow** to 2200 gpm and raise level to 12.5 feet as indicated on 1-RC-LI-100A.

Surry

References:

- 1-AP-27.00, Loss of Decay Heat Removal Capability, Rev. 10
- ND-88.2-LP-1, Residual Heat Removal System Description, Rev. 8
- ND-88.2-LP-02, Operation of Residual **Heat** Removal System, Rev. 15
- ND-95.2-LP-12, Loss of RHR Events, Rev. 9

Distractor Analysis:

- A. Incorrect because AP-27 Att. 2 indicates *that* 12.5 feet is in the unacceptable region of operation for 2600 gpm RHR flow rate.
- B. Incorrect because AP-27 Att. 2 indicates that 2200 gpm RHR flow rate is in the unacceptable region of operation for 12.4 feet.
- C. incorrect because AP-27 Att. 1 indicates that 1200 gpm RHR flow rate is **less** than the required flow rate of 2200 gpm.
- D. Correct because these actions place the plant in an acceptable region of AP-27 Att. 1 and 2 for required flow rate for 10 days after shutdown.

AP-27 Att. 1 and 2 **will** need to be provided to the applicant.

G2.1.25: Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

62. G2.2.22 001/3//SAFETY LIMIT/MEM 3.4/4.1/N/SR04301/R/MAB/SDR

Which ONE of the following is correct with respect to Technical Specifications?

- A. The Safety Limit for core thermal power is 109% of Rated Thermal Power and the RCS pressure limit is 2735 psig.
- B. The Safety Limit for core thermal power is 109% of Rated Thermal Power and the single loop loss of flow reactor trip shall be unblocked when power range nuclear flux is greater than or equal to 50% of Rated Thermal Power.
- C. The reactor trip on low pressurizer pressure, *high pressurizer* level, turbine trip, and low reactor coolant flow for two or more loops shall be unblocked when power is greater than or equal to 10% of Rated Thermal Power.
- D. The source range high flux, high setpoint trip shall be unblocked when the intermediate range nuclear flux is less than or equal to  $5 \times 10^{-10}$  amperes.

Surly

References:

Technical Specification 2.1 (Amendments 116); 2.2 (Amendments 203); 2.3 (Amendments 175, 176, 206)

Distractor Analysis:

- A. Incorrect because the safety limit for core thermal power is 118%.
- B. Incorrect because the safety limit for core thermal power is 111%.
- C. Correct because this is the correct statement taken from Tech Specs.
- D. Incorrect because source range high flux, high setpoint trip shall be unblocked when the intermediate range nuclear flux is less than or equal to  $5 \times 10^{-11}$  amperes.

Generic K/A 2.2.22

Knowledge of limiting conditions for operations and safety limits.

~~63.~~ 62.2.27 001/3//REFUELING/MEM.2.6/3.5/N/SR04301/R/MAB/SDR

Which ONE of the following correctly states the level of authorization needed for bypassing the Manipulator Crane Overload Interlock?

- A. Refueling SRO or Fuel Handling Supervisor
- B. Refueling SRO and **Shift** Supervisor
- C. **SNSOC** and Refueling SRO
- D. **SNSOC** only

Surry

References:

VPAP-1401, Conduct of Operations, Rev. 11 (Section 6.5)

Distractor Analysis:

- A. Incorrect because **SNSOC** pre-approval is needed per 1-OP-FH-015 Step 4.12.
- B. incorrect because **SNSOC** pre-approval is needed per 1-OP-FH-015 Step 4.12.
- C. Correct because **SRO** approval is needed per 1-OP-FH-015 Step 4.10 AND **SNSOC** pre-approval is needed per 1-OP-FH-015 Step 4.12.
- D. Incorrect because **SRO** approval is needed per 1-OP-FH-015 Step 4.10.

G2.2.27

Knowledge of the refueling process.

64. G2.3.10 001/3//FUEL HANDLING/MEM 2.9/3.3/N/SR04301/R/MAB/SDR

The following conditions exist:

- Unit 2 is at full power
- Unit 1 is in refueling
- Fuel repair is being performed
- A damaged fuel rod is raised too close to the surface of the water
- Area radiation monitors alarm in the vicinity of the fuel movements
- Operators enter 0-AP-22.00, Fuel Handling Abnormal Conditions
- All components operate as designed

Which ONE of the following are immediate actions of AP-22.00?

- A. Stop fuel handling operations, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C and 1-VS-MOD-103D, Dump Cable Vault Air Bottles by closing 1-VS-MOD-103B.
- B. Stop fuel handling operations, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C and 1-VS-MOD-103D, Bump MER 3 Air Bottles by closing 1-VS-MOD-103A.
- C. Evacuate the affected areas, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C and 1-VS-MOD-103D, Dump MEW 3 Air Bottles by closing 1-VS-MOD-103A.
- D. Stop fuel handling operations, Evacuate the affected areas, Stop Main Control Room Fans 1-VS-F-15 and 1-VS-AC-4.

Surry

References:

0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 18

Distractor Analysis:

- A. Correct these are all listed as immediate actions of AP-22.00.
- B. Incorrect because 1-VS-MOD-103A is in the RNO column to be performed if 103B does not close. However, the stem states that all equipment operates as designed, so the operator would not go to the RNO column.
- C. Incorrect because 1-VS-MOD-103A is in the RNO column to be performed if 103B does not close.
- D. Incorrect because stopping MCR Ventilation Fans is not an immediate action.

G2.3.10: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

65. 6232001/3//RADIATION RESPIR TOR/C/A 2 /2.9/N/SR04301/R/MAB/SDR

**Work** in a radiation area must **be** performed. The following conditions exist:

- A point source is present and emits 50 mrem/hour at 1 foot
- The air has a Derived Air Concentration (DAC) of 10

Which **ONE** of the following methods will result in the lowest amount of accumulated dose?

- A. Two workers using hand tools can perform the work in **one** hour at a distance of two feet wearing no respirator.
- B. Three workers using remote tools perform the work in two hours at a distance of six feet wearing no respirator.
- C. Two workers using hand tools perform the work in four hours at a distance of two feet wearing a respirator with a protection factor of 50.
- D. Three workers using remote tools perform the work in 10 hours at a distance of six feet wearing a respirator with a protection factor of 50.

### Surry

#### References:

Dominion Nuclear Employee Training Manual Volume II BRWT, RPT, CSET, SCAT, FWT, Rev. 11, January, 2003.

#### Distractor Analysis:

- A. Incorrect:  $75 \text{ mrem} > 56.7 \text{ mrem}$ .  $\{[(2 \text{ men})(1 \text{ hr})(50 \text{ mrem/hr})(1/2)^2] + [(10 \text{ DAC})(2 \text{ men})(1 \text{ hr})(2.5 \text{ mrem/DAC-HR})]\} = 75 \text{ mrem}$
- B. Incorrect:  $158.3 \text{ mrem} > 56.7 \text{ mrem}$ .  $\{[(3 \text{ men})(2 \text{ hr})(50 \text{ mrem/hr})(1/6)^2] + [(10 \text{ DAC})(3 \text{ men})(2 \text{ hr})(2.5 \text{ mrem/DAC-HR})]\} = 158.3 \text{ mrem}$
- C. incorrect:  $104 \text{ mrem} > 56.7 \text{ mrem}$ .  $\{[(2 \text{ men})(4 \text{ hr})(50 \text{ mrem/hr})(1/6)^2] + [(10 \text{ DAC})(1/50)(2 \text{ men})(4 \text{ hr})(2.5 \text{ mrem/DAC-HR})]\} = 104 \text{ mrem}$
- D. Correct:  $\{[(3 \text{ men})(10 \text{ hrs})(50 \text{ mrem/hr})(1/6)^2] + [(10 \text{ DAC})(1/50)(3 \text{ men})(10 \text{ hrs})(2.5 \text{ mrem/1 DAC-HR})]\} = 41.7 + 15 = 56.7 \text{ mrem}$ .

#### G2.3.2

Knowledge of facility ALARA program.

66. G2.3.9 001/3//CONTAINMENT PURGE/C/A 2.5/3.4/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- The RCS temperature is 190 °F.
- Operators are performing Section 5.2 of 1-OP-VS-001, Containment Ventilation, to place the Containment Purge System in service using 1-VS-F-58A or 1-VS-F-58B, Filter Exhaust Fans.
- The Containment Purge Form requires 5000 cfm purge flow.

Which ONE of the following correctly states selection criteria, in accordance with 1-OP-VS-001, for choosing which valve to use for obtaining the correct purge flow rate?

- A. 1-VS-MOV-100D (Ctrnt Purge Exh) should be throttled instead of 1-VS-MOV-101 (Ctrnt Purge B/P) due to the high flow rate required **by** the Containment Purge Form.
- B. 1-VS-MOV-101 (Ctrnt Purge B/P) should be throttled instead of 1-VS-MOV-100D (Ctrnt Purge Exh). This is due to **the** need to open the supply breaker **to** 1-VS-MOV-100D in order to throttle it. Opening the breaker will prevent automatic CTMT Purge isolation.
- C. 1-VS-MOV-101 (Ctrnt Purge B/P) should be throttled instead of 1-VS-MOV-100D (Ctrnt Purge Exh) due to the low flow rate required by the Containment Purge Form.
- D. 1-VS-MOV-100D (Ctrnt Purge Exh) should **be** throttled instead **of** 1-VS-MOV-101 (Ctrnt **Purge B/P**). This is due to the need to open the supply breaker **to** 1-VS-MOV-101 in order to throttle it. Opening the breaker will prevent automatic CTMT Purge isolation.

Surry

References:

1-OP-VS-001, Containment Ventilation, Rev. 20

Distractor Analysis:

- A. Correct because 1008 should be throttled due to the Containment Purge Form allowing more than 3000 cfm. The bypass will not have enough capacity at this flow rate.
- B. Incorrect because even though auto containment purge isolation will not occur with the breaker open, the procedure still directs the use of 100D due to the high flow rate. Plausible because applicant may think it logical to not intentionally incapacitate auto containment isolation.
- C. Incorrect because with *the* flow rate greater than 3000 gpm, 100D should **be** used. Plausible because 3000 gpm **is** not a very high flow rate.
- D. Incorrect because the bkr does not need to be opened and at 5000 gpm, the procedure directs 101 to be used for fine tuning **the** flow rate. Plausible because preventing auto ctmt purge isolation is a concern when using 100D.

G2.3.9: Knowledge of the process for performing a containment purge.

- B. Immediate action steps may **be** performed in any order, except for the first four immediate action steps of E-0, Reactor Trip or Safety Injection, which must be performed in the order in which they appear in the procedure.
- C. Immediate action steps may be performed in any order except for the first four immediate action steps of E-0, Reactor Trip or Safety Injection, and the immediate action steps of FR-S.1, Response to Nuclear Generation / ATWS, which must be performed in the order in which they appear in the procedure.
- D. immediate action steps may **be** performed in any order **except for** the immediate

#### Surry

#### References:

ND-95.3-LP-2, Emergency Procedure Writer's Format, Rev. 8

(Have Utility add any additional references that may support answer.)

#### Distractor Analysis:

- A. Incorrect because only immediate actions of E-0 and FR-S.1 must be performed in the order in which they appear in the procedure.
- B. Incorrect because only immediate actions of E-0 and FR-S.1 must be performed in the order in which they appear in the procedure.
- C. Correct because immediate actions of E-0 and FR-S.1 must **be performed in** the order in which they appear in the procedure. This requirement / expectation is stated in ND-95.3-LP-2 Page 12.
- D. Incorrect because ECA-0.0 are not required to be performed in any **specific** order.

G2.4.11: Knowledge of abnormal condition procedures.

68. G2.4.12 001/3//IMMEDIATE ACTIONS/MEM 3.4/3.9/N/SR04301/R/MAB/SDR

A situation presents itself that requires a Reactor Operator (RO) to take quick decisive action to ensure Station Safety. Personnel are not in immediate danger and the action requires no reactivity manipulations.

Which ONE of the following correctly describes the requirements for performing the actions?

- A. The RO may take necessary action without prior approval from another licensed operator.
- B. The RO must immediately request approval from the Unit SRO to perform the action and only take action after approval is granted.
- C. The RO may take action only after another licensed operator has been notified and concurs with the action.
- D. The RO may take action only after obtaining a peer check to concur with the action.

Surly

References:

OPAP-0006, Shift Operating Practices, Rev. 4

Bistractor Analysis:

- A. Correct because OPAP-0006 Step 6.1 0.3 states, "During emergencies, Shift Team members may take necessary immediate actions required to ensure personnel and Station safety without prior approval. The Shift Supervisor shall **be** promptly informed of these actions."
- B. Incorrect because action may be taken prior to obtaining permission.
- C. Incorrect because action may **be** taken prior to notifying or obtaining permission from another Team Member.
- D. Incorrect because immediate action is authorized to protect **the** Station.

G2.4.12: Knowledge of general operating crew responsibilities during emergency operations.

69. G2.4.49.002/3//ROD CONTROL/C/A 4.0/4.0/N/SR04301/R/MAB/SDR

Given the following conditions:

- Reactor Power = 85%
- Control Rods are in automatic
- Control Bank D begins to insert without a turbine runback
- Pave and Tref are matched within 0.5 °F

Which ONE of the following describes the correct immediate operator response to these conditions?

- A. Verify quadrant power tilt **and** axial flux difference within limits.
- B. Place ROD CONT MODE SEL switch in MANUAL.**
- C. Manually trip the reactor.
- D. Verify IRPI operating properly.

Surry

References:

0-AP-1.00, Rod Control System Malfunction, Rev. 9.

Distractor Analysis:

- A. Incorrect because the initial response is **to** place **ROD CONT MODE SEL** switch in **MANUAL**.
- B. Correct per AP-1.00.
- C. Incorrect because this would not be performed until **ROD CQNT MODE SEL** switch was placed to **MANUAL** and rod motion had stopped.
- D. Incorrect because AP-1.00 directs placing **ROD CONT MODE SEL** switch in **MANUAL** as an immediate action.

G2.4.49

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

70. WEO4EK3.2 001/1/1/LOCA OUTSIDE/C/A 3.4/4.0/M/SR04301/R/MAB/SDR

Which ONE of the following correctly states actions contained in 1-ECA-1.2, LOCA Outside Containment, and the reasons for those actions?

- A. Open 1-SI-MOV-1890A (LHSI to Hot Leg) or 1-SI-MOV-1890B (LHSI to Hot Leg) to provide a flow path for Low Head Safety Injection. Then close 1-SI-MOV-1890C (LHSI to Cold Legs) and monitor RCS pressure.
- B. If closing 1-SI-MOV-1890C (LHSI to Cold Legs) does not result in an RCS pressure rise then allow it to remain closed because this will give operators time to check Aux Building alarms while the flow path is isolated.
- C. If the leak is not identified and isolated then transition to 1-E-1, boss of Reactor or Secondary Coolant, because RCS inventory is continued to be lost outside of containment.
- D. If closing 1-SI-MOV-1890C (LHSI to Cold Legs) results in an RCS pressure rise, then place the **LHSI** pumps in **PTL** because their suction valves *from the* RWST will be closed to isolate potential leak paths.

Surry

References:

ND-95.3-LP-21, ECA-1.2 LOCA Outside Containment, Rev. 7

ECA-1.2, LOCA Outside Containment, Rev. 5

Bistractor Analysis:

- A. incorrect because ECA-1.2 does not give any direction to open 1-SI-MOV-1890A & B. These valves should be **left** in the closed position. This distractor is plausible because ECA-1.2 does give guidance to close 1890C.
- B. incorrect because if 1-SI-MOV-1890C is closed and RCS pressure is still decreasing, then the leak was not isolated and the valve needs to be re-opened. This is the normal **SI** flow path and it is important to re-establish this path if closing the valve did not isolate the leak.
- C. Incorrect because *if* the leak is not isolated, then the correct transition would be to go to 1-ECA-1.1, boss of Emergency Coolant Recirculation.
- D. Correct because if WCS pressure rises upon closure of 1-SI-MOV-1890C, then the leak was isolated and 1-ECA-1.2 directs the LHSI **pumps** to be placed in PTL and **the** suction valves from the **RWST** to be closed.

WEO4

EK3.2: Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment): Normal, abnormal, and emergency operating procedures associated with (LOCA Outside Containment).

71. WE06EK3.1 001/1/2/CORE COOLING/MEM 3 4/3.8/B/SE04301/R/MAB/SDR

1-FR-C.1, Response to Inadequate Core Cooling, is being performed. Which ONE of the following is the reason RCPs are stopped prior to depressurizing the SGs to less than 150 psig during an inadequate core cooling event?

- A. RCP operation with the SGs at atmospheric pressure is prohibited due to excessive hydraulic stress on the SG U-tubes.
- B. The SGs will depressurize more quickly if no Forced Circulation RCS flow exists.
- C. To minimize heat input to the RCS.
- D. The SG depressurization will lead to a loss of RCP support conditions.

Surry

References:

ND-95.3-LP-38, Response to Inadequate Core Cooling, Rev. 8  
FR-C.1, Response to Inadequate Core Cooling, Rev. 18

Distractor Analysis:

- A. Incorrect because securing RCPs is necessary because **the** depressurization will result in losing the RCP seal support conditions, which could damage the RCPs.
- B. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS or forced Blow.
- C. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS.
- D. Correct because this is the stated reason in ND-95.3-LP-38. Losing #1 Seal support conditions could result in damage to the RCPs.

074 Inad. Core Cooling

E06EK3.1: Knowledge of the reasons for the following responses as they apply to (Degraded Core Cooling): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Surry Requal Exam Bank Question #467

72. WE08G2.1.7 001/1/2/TUBE RUPTURE STEAM/C/A 3.7/4.4/B/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Reactor power is 58% and rising
- RCS pressure is at 2210 **psig** and slowly lowering
- Tavg is 557 °F and slowly lowering
- Pressurizer level is slowly lowering
- Turbine load is stable at 400 MW
- SG levels are at 46% NR
- SG pressures are at 970 **psig** and slowly lowering
- Containment pressure is 9.5 psia and slowly **rising**
- Condenser Air Ejector RM reads 18 cpm

Which ONE of the following correctly diagnoses the event?

- A. Ruptured and faulted steam line break inside containment.
- B. Steam line break inside containment.
- C. LOCA inside containment.
- D. Steam line break outside containment.

Surry

References:

General operator knowledge.

Distractor Analysis:

- A. Incorrect because although there are parameters to support the steam line break, there are **no** parameters to support a SGTR. Plausible because Condenser Air Ejector RM reading is given, but the value is not representative of a SGTR.
- B. Correct because reactor power and ctmt pressure are rising; RCS pressure, Tavg, and SG pressures are lowering. These **are** all indicative of a steam line break inside ctrnt.
- C. Incorrect because reactor power would **not be** rising during a LOCA as *it* would during a steam line break. Plausible because many of **the** parameters coincide with a LOCA.
- D. Incorrect because ctmt pressure is **rising**. Plausible because of the aforementioned parameters that are indicative of a steam line break.

WE08 RCS Overcooling

G2.1.7: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Surry Requal Bank Question #177 (ID: EOP0076)

7.3 WE11EA12001/1/1/LHSI LOCA RWST HHSI/C/A 3.5/3.8/N/SR04301/R/MAB/SDR

The following conditions exist:

- LOCA has occurred.
- RWST level = 13% and decreasing.
- Recirculation Mode Transfer (RMT) keyswitch is in RMT Mode.
- White **RMT** Status Light is lit.
- Amber RMT Status Light is lit.

1-SI-MOV-1860A (LHSI Suction from Sump) opens fully and 1-SI-MOV-1860B (LHS Suction from Sump) strokes to 50% open where it trips on thermal overload. Which **ONE** of the following gives the correct status of Safety Injection?

- A. LHSI from the RWST is injecting **into** the cold legs **and** HHSI from LHSI **pump** discharge is injecting into the cold legs.
- B. No Safety Injection is injecting water to the cold legs.
- C. HHSI directly from the RWST (not from LHSI discharge) is injecting into **the** cold legs, but no LHSI is injecting into the cold legs.
- D. LHSI from the **RWST** and HHSI directly from the RWST (not from LHSI discharge) is being injected into the cold legs.

Sur9

References:

ND-91.3-LP-3, Safety Injection System Operations, Rev. 15

1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 11

Distractor Analysis:

- A. Correct because 1-SI-MOV-1862A&B will not close until 1-SI-MOV-1860A&B open due to an interlock.
- B. Incorrect because **RWST** is still the suction source to the LHSI pumps.
- C. Incorrect because LHSI Pumps are taking suction from the RWST and injecting into the cold legs and HHSI is not taking suction directly from the RWST.
- D. Incorrect because HHSI is not taking suction directly from the RWST. HHSI is taking suction on the discharge of **the** LHSI Pumps.

WE11

EA1.2: Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation): Operating behavior characteristics of the facility.

74. WE12EK2.2 001/L1/AFW/MEM 3.6/3.9/M/SR04301/R/MAB/SDR

A steam break has occurred and all Steam Generators are faulted.

Which ONE of the following is the basis for maintaining a minimum of 60 gpm AFW flow to each Steam Generator per ECA-2.1, Uncontrolled Depressurization of All Steam Generators?

- A. 60 gpm is needed to meet minimum heat sink flow requirements.
- B. 60 gpm to each Steam Generator will ensure even thermal hydraulic distribution across the core.
- C. 60 gpm is the minimum indicated flow rate to prevent Steam Generator dryout.
- D. 60 gpm is the minimum indicated flow that will ensure the feed lines stay warm to prevent excessive thermal shock to the feed lines during recovery actions.

Surry

References:

ND-95.3-LP-22, ECA-2.1 Uncontrolled Depressurization of All Steam Generators, Rev. 9

1-E-3, ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 16

Distractor Analysis:

- A. Incorrect because this requirement is not based on minimum heat sink flow requirements, it is based on SG dryout.
- B. incorrect because this requirement is not based on thermal hydraulic distribution across the core. It is based on S/G dryout.
- C. Correct because 60 gpm is the minimum verifiable flow rate to a steam generator. This ensures a nominal flow rate of 25 gpm to the S/G, considering detector uncertainties, to prevent dryout and thermal shock to the S/G.
- D. Incorrect because the concern is with thermal shock to the SG if AFW flow rates are raised.

040 (W/E12) Steam Line Rupture - Excessive Heat Transfer

EK2.2: Knowledge of the interrelations between the (Uncontrolled Depressurization of All Steam Generators) 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.

Modified Surry ILT Bank Question #1010

1-E-3, Steam Generator Tube Rupture, has been entered due to a ruptured tube in the "A" Steam Generator. The Team is performing Step 4, which directs "A" Steam Generator Narrow Range SG Level to be greater than 12% prior to stopping feed flow.

### Surry

#### References:

1-E-3, Steam Generator Tube Rupture, Rev. 25  
ND-95.3-LP-13, E-3 Steam Generator Tube Rupture, Rev. 11

#### Distractor Analysis:

- A. Correct because this is the basis as stated in ND-95.3-LP-13.
- B. Incorrect because the concern is not thermal gradients across the tubes. The concern is to cover the tubes for thermal stratification and then stop AFW flow as soon as the tubes are covered to give margin to overfill, while mitigating release to the public.
- C. Incorrect because this SG will not be used for the RCS cooldown.
- D. Incorrect because the dP is still going to induce leakage even at 12% SG level.

#### WE13 Steam Generator Over-pressure

EK2.1: Knowledge of the interrelations between the (Steam Generator Overpressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question is modified from a Braidwood Question.

76. 001G2.4.30-001/2/2/REPORTABILITY/MEM 2.2/3.6/B/SR04301/S/MAB/SDR

Which ONE of the following states an event that is required to be reported to the NRC within 1 hour of discovery?

- A. An inadvertent Safety injection due to an instrument surveillance error.
- B. The Shift Supervisor authorizes the individual insertion of control rods into the core without bank overlap to shutdown the reactor in an emergency.
- C. A hypochlorite spill outside the Polishing Building of which the EPA has been notified.
- D. A radioactive release such that if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

Surry

References:

VPAP-2802, Notifications and Reports, Rev. 17.

Distractor Analysis:

- A. incorrect because this is a 4 hour reportable event. Plausible because the applicant may think that inadvertent safety injection is important enough to require reporting to the NRC within one hour.
- B. Correct per VPAP-2802 Section 6.3.3 for deviation from Tech Specs. (VPAP-2802 Page 77.)
- C. incorrect because this is a 4 hour reportable event. Plausible because the applicant may think that a hypochlorite spill with EPA notification is important enough to require reporting to the NRC within one hour.
- D. incorrect because this is a 24 hour reportable event. Plausible because the applicant may think that a large radioactive release is important enough to require reporting to the NRC within one hour.

001 Control Rod Drive

G2.4.30 Knowledge of which events related to system operations / status should be reported to outside agencies.

~~7Z\_004G2\_132 002/2/1/CVCS/MEM 3 4/3.6/N/SR04301/S/MAB/SDR~~

During Unit 1 REFUELING SHUTDOWN and COLD SHUTDOWN operations, the following valves shall be locked, sealed, or otherwise secured in the closed position except during planned dilution or makeup activities.

- 1-CH-223, or
- 1-CH-212, 1-CH-215, and 1-CH-218

Which ONE of the following correctly describes the time requirement and reason for locking, sealing, or otherwise securing these valves following a planned dilution or makeup activity in accordance with Technical Specifications?

- A. 15 minutes to prevent inadvertent boron dilution of the RCS.
- B. 60 minutes to ensure the proper safety system alignment.
- C. 15 minutes to ensure the proper safety system alignment.
- D. 60 minutes to prevent inadvertent boron dilution of the RCS.

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References:

Technical Specification 3.2.E.3, Amendment 199

Bistractor Analysis:

- A. Correct per Technical Specifications and Basis.
- B. Incorrect because Technical Specifications require within 15 minutes.
- C. Incorrect because Technical Specifications Basis states that these valves shall be closed to provide assurance that an inadvertent boron dilution will not occur.
- D. Incorrect because Technical Specifications require within 15 minutes.

004 Chemical and Volume Control

G2.1.32: Ability to explain and apply all system limits and precautions.

78. 009EA2 39 001/1/1/NATURAL CIRCULATION/C/A 4.3/4 7/M/SR04301/S/MAB/SDR

Given the following Unit 1 conditions:

- A small break LOCA has occurred
- As directed by the EOPs, the RCPs have been tripped
- 1-ES-1.2, Post-LOCA Cooldown and Depressurization, Step 20, "Verify Natural Circulation?" is being performed
- RCS pressure is 1490 psig
- Wide Range T-Cold indications are 505 °F and slowly decreasing
- Wide Range T-Hot indications are 515 °F and slowly decreasing
- CETCs are 581 °F and stable
- Containment Pressure is 10 psia
- Containment Radiation Levels are:  $5.0 \times 10^5$  R/hr
- SG Narrow Range Levels are: A=22%, B=24%, C=22%, and slowly decreasing
- SG Pressures are 715 psig and stable
- RVLIS Full Range = 50%

According to 1-ES-1.2, which ONE of the following correctly states the status of Natural Circulation and the correct operator actions?

- A. Natural Circulation criteria are met. Begin depressurizing when subcooling is  $\geq 85$  °F.
- B. Natural Circulation criteria are not met due to CETCs not decreasing. Depressurize the SGs by raising steam flow rate through the steam dumps. Then depressurize when subcooling is  $> 95$  °F.
- C. Natural Circulation criteria are not met due to SG pressure parameters not satisfied. Depressurize the SGs by raising steam flow rate through the steam dumps. Then depressurize when subcooling is  $\geq 85$  °F.
- D. Natural Circulation criteria are not met due to inadequate subcooling. Cool the RCS by raising steam flow rate through the steam dumps. Then depressurize when subcooling is  $> 95$  °F.

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References:

1-ES-1.2, Post LOCA Cooldown and Depressurization, Rev. 21

Distractor Analysis:

- A. Incorrect because there is not adequate subcooling.
- B. Incorrect because CETCs do not need to be decreasing.
- C. Incorrect because SG parameters are satisfied.
- D. Correct because there is inadequate subcooling ( $16\text{ }^{\circ}\text{F} < 85\text{ }^{\circ}\text{F}$ ). ES-1.2 Step 20 RNO directs dumping of more steam. The basis for Step 21 of dumping steam until subcooling is  $< 95\text{ }^{\circ}\text{F}$  is to ensure that the  $85\text{ }^{\circ}\text{F}$  natural circ criteria is not violated. The Degraded Containment numbers were used due to the  
CETC =  $581\text{ }^{\circ}\text{F}$ ; P = 1490 psig = 1505 psia;  $T_{\text{sat}}(1505\text{ psia}) = 597\text{ }^{\circ}\text{F}$ ;  
Subcooling =  $597 - 581 = 16\text{ }^{\circ}\text{F}$

Surry ILT Bank Exam Question #1069

009 Small Break LQCA

**EA2.39:** Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling.

79. 025G2.4.7 001/1/1/RHR FEED AND BLEED/C/A 3.1/3.8/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- RCS is not pressurized
- RCS level is 16.00 feet as read on 1-RC-LI-100A

Which ONE of the following specifies the minimum mandatory backup cooling method(s) required to be available before entering the above plant conditions, in accordance with OSP-ZZ-004, Unit 1 Safety Systems Status List For **Cold** Shutdown/Refueling Conditions?

- A. Reflux Boiling AND Gravity Feed and Bleed.
- B. Gravity Feed and Bleed ONLY.
- C. Forced Feed and Bleed AND Gravity Feed and Bleed.
- D. Forced Feed and Bleed ONLY.

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References:

- 1-OSP-ZZ-004, Unit 1 Safety Systems Status List For Cold Shutdown / Refueling Conditions, Rev. 27
- 1-AP-27.00, Loss of Decay Heat Removal Capability, Rev. 10
- ND-95.2-LP-12, Loss of RHR Events, Rev. 9

Distractor Analysis:

- A. Incorrect: Per 4-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.
- B. Incorrect: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed **and** Bleed is the only Mandatory Backup method required.
- C. Incorrect: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is **the** only Mandatory Backup method required.
- D. Correct: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.

025 Loss of RHR

G2.4.7: Knowledge of event based EOP mitigation strategy

80. 055EA2.03 001/1/1/BLACKOUT SBO/MEM 3.9/4.7/B/SR04301/S/MAB/SDR

The following conditions exist:

- A loss of all AC power has occurred.
- The STA reports the status of the CSFs are as follows:
  - Subcriticality - RED
  - Core Cooling - RED
  - Heat Sink - RED
  - Integrity - GREEN
  - Containment - GREEN
  - *Inventory* - YELLOW

Which ONE of the following procedures should be used to mitigate these conditions?

- A. 1-FR-S.1, Response to Nuclear Power Generation/ ATWS
- B. 1-ECA-0.0, Loss of All AC Power
- C. 1-FR-H.1, Response to Loss of Secondary Heat Sink
- D. 1-FR-C.1, Response to Inadequate Core Cooling

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References:

1-ECA-0.0, Loss of All AC Power, Rev. 21

Distractor Analysis:

- A. Incorrect because FR's should not be implemented **while** in ECA-0.0. (see NOTE prior to step 1 of ECA-0.0)
- B. Correct because this is the correct procedure to mitigate the loss of ac power.
- C. incorrect because FR's should not be implemented while in ECA-0.0.
- D. Incorrect because FR's should not be implemented while in ECA-0.0.

Surry ILT Exam Bank Question #899

055 Station Blackout

EA2.03: Ability to determine or interpret the following as they apply to Station Blackout:  
Actions necessary to restore power.

81.056G2.4 45 001/2/1/CONDENSATE/C/A 3.3/3.6/M/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Power = 100%
- Condenser vacuum is lowering slowly.
- Steam Generator Bevels are 45% and lowering.
- Several alarms have annunciated, including:
  - 1H-G8, FW PP DISCHHDR LO PRESS
  - 1J-G4, CN PPS DISCH HDR LO PRESS
  - 1C-A1, RCP 1A CC RETURN LO FLOW
  - 1C-B1, RCP 1B CC RETURN LO FLOW
  - 1C-C1, RCP 1C CC RETURN LO FLOW

Which ONE of the following states the SRO's correct prioritization of the above conditions as indicated by the procedures and actions chosen to mitigate or correct the conditions?

- A. Trip the Reactor followed by tripping the Reactor Coolant Pumps. Enter E-0, Reactor Trip or Safety Injection.
- B. Enter AP-10.05, Loss of **Semi-vital** Bus. Verify that the standby condensate pump has started and reduce turbine load.
- C. Enter AP-21.00, Loss of Main Feedwater Flow. Maintain full power operation and manually control Steam Generator levels by placing Feedwater Regulating Valves in MANUAL control.
- D. Enter AP-23.00, Rapid Load Reduction, to bring the unit offline, followed by tripping the Reactor Coolant Pumps.

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References:

ND-90.3-LP-5, Vital and Semi-Vital Bus Bistribution, Rev. 11  
1-AP-10.05, Loss of Semi-Vital Bus, Rev. 16  
1-A\$-21.00, **Loss of Main Feedwater Flow**, Rev. 5  
1-AP-23.00, Rapid Load Reduction, Rev. 15  
1H-G8, FW PP DISCH HDR LO PRESS, Rev. 6  
1J-G4, CN PPS DISCH HDR LO PRESS, Rev. 0  
1C-A1, RCP 1A CC RETURN LO FLOW, Rev. 2  
1C-B1, RCP 1B CC RETURN LO FLOW, Rev. 2  
1C-C1, RCP 1C CC RETURN LO FLOW, Rev. 2

Distractor Analysis:

- A. Incorrect because **loss of SVB** causes indication to be **lost for RCP CC Flow** Indication. RCPs should not be tripped. Plausible because if RCPs actually had no cooling, the Rx should be tripped and RCPs should be secured.
- B. Correct because all indications in the **stern** are caused by a loss of **SVB**. Verifying **S/B Condensate Pump** starts and turbine load reduction are correct per **AP-10.05**.
- C. **Incorrect** because maintaining load at 100% **will** cause **SG levels to continue to go down**. The **FW** and Condensate Recircs have Bailed open on the loss of the **SVB**, thus making a load reduction a necessity. Plausible because **SG levels are** lowering and an Applicant may think that opening a FRV may help to mitigate the condition.
- B. **Incorrect** because the unit should not be taken off line using AP-23.00 and RCPs should not be tripped due to the loss of the SVB. Plausible because rapidly bringing the unit off line and securing RCPs, given the stated conditions, may appear logical to the applicant.

Modified Surry ILT Exam Bank Question #224 (maybe it could be considered a **new** question?)

056 Condensate

G2.4.45: Ability to prioritize and interpret the significance of **each** annunciator or alarm.

82. 057G2.1.6 001/1/1/VITAL AC AO ACTIONS/C/A 2.1/4.3/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Reactor Power = 30%
- Plant is in a Chemistry hold during a power ascension
- A loss of Vital Bus III occurs and operators enter 1-AP-10.03, Loss of Vital Bus III
- Electricians quickly find a fault on Vital Bus 1-III and believe that it will take 10 hours to repair.
- 1-CC-TV-105A, CCW TV for the "A" Reactor Coolant Pump (RCP), has closed and cannot be re-opened.
- RCP temperatures are starting to slowly rise.

Which ONE of the following set of actions should the Senior Reactor Operator (SRO) direct given the above conditions?

- A. The SRO should direct the securing of the "A" RCP. Reactor power may be maintained at 30% for the duration of the 10 hour repair to re-energize Vital Bus 1-III.
- B. The SRO should direct the securing of the "A" RCP. Reactor power may be maintained at 30% for two hours, at which time the SRO should direct preparation to bring the unit to hot shutdown within the following six hours.
- C. The SRO should direct a Reactor Trip, followed by the securing of the "A" RCP. The SRO should then direct performance of 1-E-0, Reactor Trip or Safety Injection, and continue with applicable actions of 1-AP-10.03.
- D. The SRO should direct a controlled plant shutdown. If RCP temperatures exceed action level limits, the pump should be secured and the SRO should direct continuation of the controlled plant shutdown.

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References:

ND-93.3-LP-16, Permissive/Bypass? Trip Status lights, Rev. 8  
ND-93.3-LP-10, Reactor Protection - General, Rev. 5  
ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11  
1-AP-10.03, Loss of Vital Bus III, Rev. 8

Distractor Analysis:

- A. Incorrect because **TS 3.16** and commitments made in GL-91-11 (also located *in* Note prior to Step 17 in AP-10.03). The **VB** must **be** re-powered within 2 hours, or **the** unit must be in HSB within **the** next 6 hours. **Also** incorrect because **AP-10.03** will require a reactor trip. Plausible because *the* loss of **VB** causes a **loss** of cooling to "A" RCP. It **may** appear OK to continue operation because the power is < P-8.
- B. Incorrect because **AQ-10.03** requires a reactor trip and securing of RCP if CCW will not be restored prior to RCP temperatures reaching action level limits. Plausible because of the NOTE mentioned in *the* previous distractor analysis.
- C. Correct because **AP-10.03** directs Rx Trip and securing of RCP if CCW will not be restored prior to getting cooling back **to** that pump. The stem states that the **TV is closed** and cannot be re-opened, thus preventing cooling to be restored to the RCP.
- D. Incorrect because **AP-10.03** directs Rx Trip, not a controlled shutdown. Plausible because power is < P-8, which may allow the applicant to incorrectly believe that a shutdown is acceptable.

057 Loss of Vital AC Inst Bus

G2.1.6: Ability to supervise and assume a management role during plant transients and upset conditions.

83. 058G2.4.32 001/1/1/ANNUNCIATORS/C/A 3.3/3.5/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Unit 1 power is 100%
- No annunciators are lit
- Annunciator 1K-H1 has just extinguished

Which ONE of the following is the correct Abnormal Procedure to enter and correct Event Classification?  
(Reference provided)

- A. Enter 0-AP-10.13, Loss of Main Control Room Annunciators, due to the loss of one of the power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare a Notification of Unusual Event if the loss of annunciators lasts for greater than 15 minutes.
- B. Enter 0-AP-10.13, Loss of Main Control Room Annunciators, due to the loss of both power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare a Notification of Unusual Event if the loss of annunciators lasts for greater than 15 minutes.
- C. Enter 1-AP-10.06, Loss of DC Power, and 0-AP-10.13, Loss of Main Control Room Annunciators, due to a loss of DC power and loss of one of the power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare an Alert if the loss of annunciators lasts for greater than 15 minutes.
- D. Enter 1-AP-10.06, Loss of DC Power, and 0-AP-10.13, Loss of Main Control Room Annunciators, due to a loss of DC power and loss of both power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare an Alert if the loss of annunciators lasts for greater than 15 minutes.

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References:

0-AP-10.13, Loss of Main Control Board Room Annunciators, Rev. 4

EPIP-1.01, Emergency Manager Controlling Procedure, Rev. 43

Distractor Analysis:

- A. Incorrect because 1K-HI not lit is indication of both power supplies to Unit 1 annunciator Panels having been lost.
- B. Correct because 1K-H1 not *lit* is indication of both power supplies to Unit 1 annunciator Panels having been lost. EPIP-1.01 Page 6 states that if safety system annunciators are lost for greater than 15 minutes while above CSD, then a NOUE shall be declared.
- C. Incorrect because ~~1K-HI~~ not lit **is** indication of both power *supplies* to Unit 1 annunciator Panels having been lost. Since the plant is still at 100% power, there is no indication that any DC Bus has been lost; therefore 1-AP-10.06 should not **be** entered. An **Alert** classification based on the loss of DC would be incorrect. **As** stated above, a **NOUE** is the correct classification.
- D. Incorrect because the plant **is** still at 100% power, there is **no** indication that any DC **Bus** has been lost; therefore 1-AP-10.06 should not be entered. An Alert classification based on the loss of DC would be incorrect. **As** stated above, a **NOUE** is the correct classification.

Provide EPIP-1.01 Pages 6 and 27

058 Loss of DC Power

G2.4.32: Knowledge of operator response to a loss of all annunciators

84.062A2.12 001/2/1/VITAL AC BUS/C/A 3.2/3.6/N/SR04301/S/MAB/SDR

Unit 1 is at 100% power. It experiences a loss of Vital Bus I at 1200 hours on Monday. Operators enter 1-AP-10.01, Loss of Vital Bus I, and re-energize the Vital Bus from its alternate source at 1215 hours on Monday.

Which ONE of the following correctly states **the** required actions based on the above condition?

- A. In accordance with 1-AP-10.01, Vital Bus I must be reenergized from its primary **source** by 1400 hours on Monday, or be in Hot Shutdown by 2000 hours on Monday.
- B. In accordance with 1-AP-10.01, Vital Bus I must be re-energized from its primary source by 1415 hours on Monday, or be in Hot Shutdown by 2015 hours on Monday.
- C. In accordance **with** 1-AP-10.01, Vital Bus I must be re-energized from its primary source by 1200 hours on Tuesday, or be in Hot Shutdown by 1800 hours on Tuesday.
- D. No shutdown requirements are in effect as long as **Vital Bus I is** energized.

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References:

1-AP-10.01, Loss of Vital Bus I, Rev. 13

ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11

Distractor Analysis:

- A. Incorrect because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because if the bus is not energized, it must be repowered within 2 hours and 1400 hours is 2 hours after 1200 hours.
- B. Incorrect because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because if the bus is not energized, it must be repowered within 2 hours and 1415 hours is 2 hours after 1215 hours.
- C. Correct because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). The consequences of having VB-I not energized by its primary source are mitigated, or corrected, by ensuring that it is energized from its primary source within the specified time requirement.
- D. Incorrect because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because the Vital Bus is energized and the plant would be operating satisfactorily.

062 AC Electrical Distribution

A2.12: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:  
Restoration of power to a system with a fault on it.

85\_062AA2 04 001/1/1/SERVICE WATER/C/A 2 5/2 9/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Power = 100%
- 1-CH-P-1A Charging Pump is operating
- 1-SW-P-10A Charging Pump Service Water Pump is operating
- 1-SW-P-10B Charging Pump Service Water Pump is in standby
- 1D-G5, SW OW CC PPS DISCH TO CHG PPS LO PRESS, alarms
- 1-CH-P-1A Charging Pump Bearing Temperature = 175 °F
- 1-CH-P-1A Charging Pump Oil Cooler Outlet Temperature = 150 °F
- The Pressure Indication on the discharge of 1-SW-P-10A Charging Pump Sewice Water Pump (SW-PI-26) reached a minimum value of 10 psig where it remains stable.
- The Operator in the field reports back to the Control Room that 1-SW-P-10A Charging Pump Service Water Pump is noisy and has high vibrations.

Which ONE of the following correctly states the appropriate assessment of the above conditions and appropriate operator action based on that assessment?

- A. Bearing Temperature is not within limits. The "A" Charging Bump is INOPERABLE. Direct starting standby Charging Pump Service Water Pump, direct securing the "A" Charging Pump Service Water Pump, and notify the System Engineer.
- B. Bearing Temperature *is* not within limits. 1-CH-P-1A Charging Pump is INOPERABLE. Verify auto start of 1-SW-P-10B Charging Pump Service Water Pump, and notify the System Engineer.
- C. Oil Cooler Outlet Temperature is not within normal operating band. 1-CH-P-1A Charging Pump is OPERABLE. Direct starting standby Charging **Pump** Service Water Pump, direct securing the "A" Charging Pump Service Water Pump, and notify the System Engineer.
- D. Oil Cooler Outlet Temperature **is** not within normal operating band. Performance **of** Charging Pump Operability and Performance Test for 1-CH-P-1A Charging Pump must be directed to determine OPERABILITY.

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References:

- 1D-G5, SW OR CC PPS DISCH TO CHG PPS LO PRESS, Rev. 3  
11448-FM-071B, Sh. 1 of 2, Flow / Valve Operating Numbers Diagram, Circulating and Service Water System, Surry Power Station Unit 1, Virginia Power, Rev. 50.  
ND-89.5-LP-2, Service Water System, Rev. 20  
1-OP-CH-002, Charging Pump A Operations, Rev. 13  
1-OPT-CH-001, Charging Pump Operability and Performance Test For 1-CH-P-1A, Rev. 33

Distractor Analysis:

- A. Incorrect because Bearing Temperature is less than 180 °F. OPT-CH-001 Pg 9 states that the upper admin limit is 180 °F. The Charging Pump is still OPERABLE.
- B. Incorrect because Bearing Temperature is less than 180 °F. OPT-CH-001 Pg 9 states that the upper admin limit is 180 °F. The standby pump will not start until 8 psig.
- C. Correct because Oil Cooler Outlet Temperature is not within the normal operating band (80 - 120 °F) as states in OPT-CH-001. However, the problem is not with the Charging Pump, but with the Service Water flow, so swapping Charging Pump Service Water Pumps is the correct initial action based on the ARP.
- D. Incorrect because there is no indication that the Charging Pump has a problem. Given the above alarm, all indications suggest that **the** problem is with the Service Water flow. Therefore, performance of the Operability and Performance Test for the Charging Pump would serve no purpose.

062 boss of Svc Water

AA2.04: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The normal values and upper limits for the temperatures of the components cooled by SWS.

86. 079G2.4.48 001/2/2/INSTRUMENT AIR/C/A 3.5/3.8/B/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Reactor Power = 100%
- **A** loss of Containment Instrument Air has occurred
- 1B-F6, CTMT INST AIR HDR LO PRESSURE, annunciates
- 1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, annunciates
- Containment Instrument Air Pressure = 75 psig
- Containment instrument Air was crosstied with Instrument Air

Which ONE of the following operator actions is required?

- A. Both Pressurizer PORVs are operable following the crosstie. Verify the operability **by** closing POWV Block Valves, stroking PORVs, then re-opening the PORV Block Valves.
- 5. Both Pressurizer PORVs are operable following the crosstie. No further action associated with the PORVs is required.
- C**. Declare both Pressurizer PORVs inoperable. Close and remove power from both PORV block valves within one hour and be in HSD within 6 hours.
- D. Declare both Pressurizer PORVs inoperable. Close, but leave energized, both POWV block valves within one hour and be in HSD within 6 hours.

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References:

ND-92.1-LP-1, Station Air Systems, Rev. 13  
ND-88.1-LP-3, Pressurizer and Pressure Relief, Rev. 12  
1B-F6, CTMT INST AIR HDR LO PRESS, Rev. 1  
1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, Rev. 4  
Technical Specification 3.1.A.6.c, Reactor Coolant System / Relief Valves

Distractor Analysis:

- A. Incorrect because (per 1D-C6) with CTMT Inst Air P  $\geq$  80 psig, the PORVs are inoperable.
- B. Incorrect because (per 1D-C6) with CTMT Inst Air P  $<$  80 psig, the PORVs are inoperable.
- C. Correct because PORVs are not capable of being manually cycled with CTMT Inst Air P  $<$  80 psig. Therefore, within 1 hour, the CTMT Inst Air pressure must be  $>$  80 psig or the block valves must **be** closed and de-energized. Furthermore, the plant **must** be in HSD within the next 6 hours.
- D. Incorrect because, as stated in "C" above, power must be removed from the block valves.

Surry Requal Bank Question #394 (ARP0001)

079 Station Air

G2.4.48: Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

87. 103A2.01 001/2/1/CONTAINMENT LEAKAGE/C/A 2.0/2.6/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- Plant is in Mode 1
- Personnel Airlock Seal Leakage Testing has just been completed
- The Personnel Airlock Inner Door Seal exceeded Technical Specifications leakage limits
- Earlier in the year the Personnel Airlock Inner Door exceeded Technical Specifications leakage limits and the Personnel Airlock Outer Door was opened for a total of 50 minutes during the inoperability of the Personnel Airlock Inner Door

Which ONE of the following actions would satisfy required Technical Specification Actions for the Personnel Airlock Doors?

- A. The Personnel Airlock Outer Door may not be opened to pursue the repair and retest. The plant must be shutdown and cooled down per Plant General Operating Procedures. The plant must be in Hot Shutdown within 6 hours and Cold Shutdown Within the following 30 hours.
- B. The Personnel Airlock Outer Door may be opened for 10 minutes to pursue the repair and retest of the Personnel Airlock Inner Door Seal. Per VPAP-0106, Subatmospheric Containment Entry, the Shift Supervisor shall supervise the containment entry and exit process.
- C. The Personnel Airlock Outer Door may be opened for 15 minutes to pursue the repair and retest of the Personnel Airlock Inner Door Seal. Per VPAP-0106, Subatmospheric Containment Entry, the Unit SRO shall supervise the containment entry and exit process.
- D. The Personnel Airlock Outer Door may be opened for 1 hour to pursue the repair and retest of the Personnel Airlock Inner Door Seal. Per VPAP-0106, Subatmospheric Containment Entry, the Unit SRO shall supervise the containment entry and exit process.

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References:

VPAP-0106, Subatmospheric Containment Entry, Rev. 5  
Technical Specifications 3.8, Containment (Amendments 172 and 171); 1.0.G,  
Definitions (Amendment 180)

Distractor Analysis:

- A. Incorrect because the Outer Door may be opened for 10 minutes since it has already been opened 50 minutes this year while the inner door was inoperable.
- B. Correct because per Tech Specs, the Outer Door may be opened for 15 minutes or 60 minutes for the year (which leaves 10 more minutes for this instance). Furthermore, the SS must supervise the containment entry and exit processes per VPAP-0106 Section 5.1.
- C. Incorrect because the Outer Door may be opened for 10 minutes and the SS must supervise the containment entry and exit per VPAP-0106 Section 5.1.
- D. Incorrect because the Outer Door may be opened for 10 minutes.

103 Containment

A2.01: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Integrated Leak Rate Tests.

88. G2.1.34 001/3//RCS CHEMISTRY/MEM 2.3/2.9/N/SR04301/S/MAB/SDR

The following Unit 1 conditions exist:

- The Unit has been operating at 100% power for the past two weeks
- Chemistry has just provided the following results from a Reactor Coolant System sample that was **taken** 1 hour ago:
  - RCS Chloride = 0.15 ppm
  - RCS Fluoride = 0.15 ppm
  - RCS Oxygen = 0.15 ppm

Which ONE **of** the following describes **the** above conditions and appropriate operator action?

- A. Oxygen concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the plant to cold shutdown conditions.
- B. Oxygen concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the oxygen concentration within limits. **If** the oxygen concentration is outside of the limit after **24** hours, then **the** plant must be taken to cold shutdown.
- C. Chloride concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the plant to cold shutdown conditions.
- D. Chloride concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the chloride concentration within limits. **If** the chloride concentration is outside of the limit after 24 hours, then the plant must be taken to cold shutdown.

Surry

References:

Technical Specifications 3.1.F.1 and Basis

Distractor Analysis:

- A. Incorrect because, according to the Tech Spec Basis, the plant has **24** hours to see if their corrective actions will bring the parameter within spec. If after **24** hours the parameter is **not** within spec, then the plant **must** be taken to cold shutdown using normal plant procedures.
- B. Correct because, according to the Tech Spec Basis, the plant has 24 hours to see if their corrective actions will bring the parameter within spec. **If** after 24 hours the parameter is not within spec, then the plant **mush** be taken to cold shutdown using normal plant procedures.
- C. incorrect because **Chloride** concentration is within limits.
- D. Incorrect because Chloride concentration is within limits.

G2.1.34: Ability **to** maintain primary and secondary plant chemistry within **allowable** limits.

89. G2.1.4 001/3//TECH SPEC STAFFING/C/A 2.3/3.4/N/SR04301/S/MAB/SDR

The following plant conditions exist:

- Unit 1 is shutdown and subcritical by 5.35% delta k / k
- Unit 1 Tavg is 100 °F
- Unit 2 is shutdown and subcritical by 2.35% delta k / k
- Unit 2 Tavg is 190 °F

Which ONE of the following correctly states the **MINIMUM** shift crew composition per Technical Specifications?

- A. 1 **SS**, 1 Unit **SRO**, 3 **ROs**, 4 **AOs**, and 1 **STA**.
- B. 1 **SS**, 2 Unit **SROs**, 3 **ROs**, 4 **AOs**, and no **STA**.
- C. 1 **SS**, 1 Unit **SRO**, 3 **ROs**, 4 **AOs**, and no **STA**.
- D. 1 **SS**, no Unit **SRO**, 2 **ROs**, 4 **AOs**, and no **STA**.

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References:

Technical Specification Table 6.1-1 (Minimum Shift Crew Composition), Amendment No. 123.

Distractor Analysis:

- A. Incorrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
- B. Incorrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
- C. Incorrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
- D. Correct because it matches the requirement for one unit in Cold Shutdown and one unit in Refueling Shutdown.

G2.1.4

Knowledge of shift staffing requirements.

90. G2.2.23 001/3//AFW TECH SPEC/C/A 2.6/3.8/N/SR04301/S/MAB/SDR

The following conditions exist:

- Unit 1 is at 58% power
- Unit 2 is in startup mode with  $T_{avg} = 410^{\circ}\text{F}$
- Unit 2 Steam Driven AFW Pump **and** Motor Driven AFW Pump are declared to be inoperable at 0800 hours on August 11 (all other **AFW** equipment is operable)
- Unit 2 Motor Driven AFW Pump is restored to operable status at 1100 hours and Unit 2  $T_{avg} = 410^{\circ}\text{F}$

Which **ONE** of the following set of Technical Specification actions is correct?  
(Reference provided)

- A. Initially (with both pumps inoperable) both **AFW** Pumps must be restored or Unit 2 must not enter **Hot** Shutdown. **All** Unit 1 Technical Specification Actions will be **less** restrictive than the Unit 2 Technical Specification Actions.
- B. Unit 2 **AFW** actions do not apply. Initially (with both pumps inoperable) Unit 1 must be in Hot Shutdown by 08/25 at 1400 hours and Cold Shutdown by 08/26 at 2000 hours. After the Motor Driven **AFW** Pump is operable **no** Unit 4 actions would be in effect.
- C. Initially (with both pumps inoperable) Unit 2 must be in Cold Shutdown by 08/12 at 2000 hours **and** restore either **AFW** pump by 08/25 at 0800 hours or Unit 1 must be placed in Hot Shutdown by 08/25 **at** 1400 hours. After the Motor Driven **AFW** Pump is restored, the Steam Driven **AFW** Pump must be restored by 08/14 at 0800 hours or Unit 2 must be in Hot Shutdown by 08/14 **at** 2000 hours.
- D. Initially (with both pumps inoperable) Unit 2 shall not enter Hot Shutdown and must be in Cold Shutdown by 08/12 at 2000 hours and restore either **AFW** pump within 14 days or Unit 1 must be placed in Hot Shutdown by 08/25 at 0800 hours. After the Motor Driven **AFW** Pump is restored, the Steam Driven **AFW** Pump must also be restored by 08/14 at 0800 hours or Unit 1 must be in Hot Shutdown by 08/14 **at** 2000 hours.

Surry

References:

Technical Specifications 3.6.C, 3.6.F, 3.6.G, and 3.01

Distractor Analysis:

- A. Incorrect because Unit 2 does not need to be placed in HSD until 12 hours following 08/14 at 0800 hours.
- B. Incorrect because Unit 2 Technical Specification Actions do apply above 350 °F and 450 psig.
- C. Correct because LCO 3.0.1 is entered with both pumps inoperable because there is not a tech spec condition that covers this situation. Once the MDAFW Pump is operable, LCO 3.0.1 is exited, but 3.6.F and 3.6.G still applies for Unit 2.
- D. Incorrect because Unit 1 does not need to be shutdown with only the Unit 2 Steam Driven AFW Pump inoperable.

G2.2.23

Ability to track limiting conditions for operations.

91 . G2.2.31 001/3//REFUELING/C/A 2.2/2.9/N/SR04301/S/MAB/SDR

Unit 1 has been shut down for 21 days and fuel movement has just commenced. Which ONE of the following is correct with regard to Fuel Building Exhaust and Containment Purge Exhaust?

- A. Fuel Building Exhaust and Containment Purge Exhaust must be manually aligned to continuously pass through CAT1 filters during fuel movements.
- B. Fuel Building Exhaust and Containment Purge Exhaust will automatically align to the CAT1 filters if a fuel handling accident occurred at this time.
- C. There is no need to manually align Fuel Building Exhaust or Containment Purge Exhaust to the CAT1 filters because the fuel has decayed for a sufficient period of time such that radiological consequences from a fuel handling accident would be acceptable without iodine filtration.
- D. Fuel Building Exhaust and Containment Purge Exhaust must be secured during fuel movements to prevent automatically tripping the purge.

Surry

References:

ND-92.5-LP-7, Refueling Abnormal Procedures, Rev. 10

Distractor Analysis:

- A. Correct because the automatic alignment feature is bypassed when fuel has decayed for less than 30 days. Therefore, it must be manually aligned prior to moving fuel.
- B. Incorrect because the automatic alignment feature is bypassed when fuel has decayed for less than 30 days.
- C. Incorrect because 30 days is considered sufficient decay time, not 21 days.
- D. Incorrect because this is only a requirement during movement of the upper internals.

G2.2.31

Knowledge of procedures and limitations involved in initial core loading.

92. G2.2.6.001/3//PROCEDURE CHANGE/MEM 2.3/3.3/N/SR04301/S/MAB/SDR

Which ONE of the following correctly states items that require a Regulatory Screen to be performed in accordance with VPAP-3001, Station and Regulatory Reviews?

- A. Emergency Action Level Change AND Station Curve Changes
- B. Seismic Analyses **AND** Heating-Ventilation and Air Conditioning Analyses
- C. Fire Protection Plan Changes AND Plant Flood Analyses
- D. Offsite Dose Calculation Manual Changes AND Equipment Qualification Analyses

Surry

References:

VPAP-3001, Station and Regulatory Reviews, Rev. 9

Bistractor Analysis:

- A. Incorrect because Emergency Action Level Changes are to be processed IAW VPAP-0502 (see VPAP-3001 Page 2 of Att. 3), a Regulatory Screen is not required. Plausible because both items are listed on VPAP-3001 Att. 3 Page 2.
- B. Correct per VPAP-3001 Page 2 of Att.3.
- C. Incorrect because Fire Protection Plan Changes are to be performed IAW VPAP-2401 (see VPAP-3001 Page 2 of Att. 3), a Regulatory Screen is not required. Plausible because both items are listed on VPAP-3001 Att. 3 Page 2.
- D. Incorrect because ODCM changes are to be performed IAW VPAP-2103N, a Regulatory Screen is not required. Plausible because both items are listed on VPAP-3001 Att. 3 Page 2.

G2.2.6: Knowledge of the process for making changes in procedures as described in the safety analysis report.

93. G2.4.29 001/3//EMERGENCY PLAN/MEM 2.6/4.0/N/SR04301/S/MAB/SDR

Which ONE of the following are all responsibilities that shall NOT be delegated by the Station Emergency Manager?

- A. Ordering Site Evacuation, Authorizing Emergency Exposure Limits.
- B. Authorizing Notifications of NRC, State and Local Agencies of the Emergency Status, Authorizing Emergency Exposure Limits.
- C. Authorizing Notifications of NRC, State and Local Agencies of the Emergency Status, Restricting Access to the Site.
- D. Authorizing Emergency Exposure Limits, Restricting Access to the Site.

Surry

**References:**

ND-95.5-LP-2, Station Emergency Manager, Rev. 8  
Site Emergency Plan, Rev. 46

**Distractor Analysis:**

- A. Incorrect because ordering a site evacuation may be delegated.
- B. Correct because **the** answer is clearly stated in both of **the** references.
- C. Incorrect because restricting access to the site may be delegated.
- D. Incorrect because restricting access to **the** site may **be** delegated.

G2.4.29: Knowledge of the emergency plan.

94. G2.4.38 001/3//SEM/MEM.2.2/4.0/N/SR04301/S/MAB/SDR

Which ONE of the following correctly states the preferred order for assuming the Station Emergency Manager responsibilities from the Shift Supervisor once the Technical Support Center is activated?

- A. Manger Nuclear Operations, Director Nuclear Station Safety and Licensing, Director Nuclear Station Operations and Maintenance, Another Qualified SRO
- B. Site Vice-President, Director Nuclear Station Safety and Licensing, Director Nuclear Station Operations and Maintenance, Manger Nuclear Operations
- C. Site Vice-president, Director Nuclear Station Operations and Maintenance, Director Nuclear Station Safety and Licensing, Manger Nuclear Operations
- D. Site Vim-President, Director Nuclear Station Operations and Maintenance, Manger Nuclear Operations, Director Nuclear Station Safety and Licensing

Surry

References:

ND-95.5-LP-2, Station Emergency Manager, Rev. 8

Distractor Analysis:

- A. Incorrect because this is not the preferred order as specified in ND-95.5-LP-2 Pg 3.
- B. Incorrect because this **is not the** preferred order as specified in ND-95.5-LP-2 Pg 3.
- C. Correct because this is the preferred order as specified in ND-95.5-LP-2 Pg 3.
- D. incorrect because this is not the preferred order as specified in ND-95.5-LP-2 Pg 3.

G2.4.38: Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.

95. WE01G2.1.20 001/1/2/SI TERMINATION/C/A 4.3/4.2/N/SR04301/S/MAB/SDR

Given the following plant conditions following an automatic reactor trip:

- RCS has been verified to be intact per 1-E-0, Reactor Trip of Safety Injection
- AFW Flow to "A" SG = 125 gpm
- AFW Flow to "B" SG = 110 gpm
- AFW Flow to "C" SG = 130 gpm
- NR "A" SG Level = 10%
- NR "B" SG Level = 8%
- NR "C" SG Level = 9%
- RCS Pressure = 1750 psig and slowly rising
- PRZR Level = 24% and slowly rising
- WCS subcooling based on CETCs is 80°F

Operators have reached the point in 1-E-0 where they are to check if SI flow should be reduced.

Which ONE of the following would be the next series of operator actions?

- A. Direct STA to begin monitoring Critical Safety Function Status Trees, Reset **SI** and **CLS**, verify Instrument Air available, then stop all but one Charging Pump, followed by isolating High Head **SI** to the Cold legs.
- B. Transition to 1-ES-1.1, **SI** Termination, establish letdown, followed by raising Pressurizer level to > 35%, then secure all but one Charging Pump.
- C. Establish letdown, followed by raising Pressurizer level to > 35%, transition to 1-ES-1.1, **SI** Termination, then secure all but one Charging Pump.
- D. Direct STA to begin monitoring Critical Safety Function Status Trees, Reset **SI** and **CLS**, verify Instrument Air available, align Charging Pump suction to the VCT, then stop all but one Charging Pump.

Surry

References:

1-E-0, Reactor Trip or Safety Injection: Rev. 46

1-ES-1.1, SI Termination, Rev. 29

ND-95-03-03, E-0, Reactor Trip of Safety Injection, Rev. 14

Distractor Analysis:

- A. Correct because these actions are directed by 1-E-0 Steps 26 through 32.
- B. Incorrect because letdown would not be established prior to Pzr L > 35%. Plausible because transition **to ES-1.1 is** logical and distractor states that **the** goal is to get Pzr L > 35%.
- C. Incorrect because letdown would not be established prior to Pzr L > 35%. Plausible because transition to **ES-1.1** is logical and distractor states that the goal is to **get** Pzr L > 35%.
- D. Incorrect because Charging Pump suction would **not** be aligned to VCT until after all **but one** Charging **Pump is** secured. Plausible because all actions are directed by procedure, except that the order of **the** suction **swap** and pump stopping is reversed.

W/E01 Wediagnosis and SI Termination

G2.1.20: Ability to execute procedures.

96. WE03EA2.1 001/1/2/COOLDOWN LOCA/MEM 3.4/4.2/B/SR04301/S/MAB/SDR

Operators are responding to a **LOCA** outside of containment using 1-ECA-1.2, LOCA Outside Containment. The crew efforts to isolate the break are unsuccessful.

Which ONE of the following identifies the procedure ECA-1.2 will direct the operators to in order to cool and depressurize the reactor coolant system?

- A. 1-E-1, Loss of Reactor or Secondary Coolant
- B. 1-ES-1.2, Post LOCA Cooldown and Depressurization
- C. 1-ES-1.3, Transfer to Cold Leg Recirculation
- D. 1-ECA-1.1, Loss of Emergency Coolant Recirculation

Surry

References:

- 1-E-1, Loss of Reactor or Secondary Coolant, Rev. 21
- 1-ES-1.2, Post **LOCA** Cooldown and Depressurization, Rev. 21
- 1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 12
- 1-ECA-1.1, Loss of Emergency Coolant Recirculation, Rev. 17

Distractor Analysis:

- A. Incorrect as stated in Distractor D Analysis. Plausible because there is a Loss of Reactor Coolant in progress.
- B. Incorrect as stated in Distractor D Analysis. Plausible because the goal is to cool and depressurize the RCS.
- C. Incorrect as stated in Distractor D Analysis. Plausible because this is a normal transition for long term cooling during a LOCA.
- D. Correct because Step 2 RNO of ECA-1.2 directs operators to **ECA-1.1** if efforts to isolate the leak are not successful.

W/E03 LOCA Cooldown - Depress.

EA2.1: 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.

Bank Question TP02301.

97. WE05EA2.1 001/1/1/MAIN STEAM LINE/C/A 3.4/4.4/B/SR04301/S/MAB/SDR

The following conditions exist:

- A manual Safety Injection was initiated due to a Steam Break in Safeguards
- All SG pressures are steadily decreasing
- All SG NR levels are off-scale low and WR levels are steadily decreasing
- Pressurizer level is off-scale low
- Pressurizer pressure is steadily decreasing
- RCS temperature *is* decreasing uncontrollably
- Adequate Auxiliary Feedwater flow exists

Which ONE of the following is the correct procedure transitions for the event in progress?

- A. E-0 to E-2 to ECA-2.1
- B. E-0 to E-1 to E-2 to ECA-2.1
- C. E-0 to E-1 to ECA-2.1
- D. E-0 to E-2 to E-1

Surry

References:

1-E-0, Reactor Trip or Safety Injection, Rev. 46

1-E-2, Faulted Steam Generator Isolation, Rev. 9

1-ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 19

Distractor Analysis:

- A. Incorrect because E-0 would be entered upon Rx Prig. Step 21 of E-0 sends the team to E-2. Step 2 of E-2 sends the team to ECA-2.1.
- B. Incorrect because E-0 Step 21 directs performance of E-2. E-1 is not directed until E-0 Step 23.
- C. Incorrect because E-0 Step 21 directs performance of E-2. E-1 is not directed until E-0 Step 23.
- D. Incorrect because E-2 would not be entered until after E-1.

WE05 Inadequate Heat Transfer - Loss of Secondary Heat Sink

EA2.1: Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry ILT Bank Question #1342

98. WEIOEA2.1001/1/2/NATURAL CIRCULATION/C/A 3.2/3.9/B/SR04301/S/MAB/SDR

During a Natural Circulation Cooldown IAW ES-0.3, Natural Circulation Cooldown with Steam Void in Rx Vessel, a steam bubble forms in the vessel head. The **STA** recommends transition to FR-1.3, Response to Voids in Reactor Vessel, to vent the head.

Which ONE of the following courses of action is appropriate?

- A. Initiate FR-1.3 since ES-0.3 assumes FR-1.3 is in effect to eliminate the steam void.
- B. Initiate SI and go to FR-1.3 to vent the head.
- C. The NC Cooldown should be stopped and a transition to FR-1.3 should be made.
- D. Stay in ES-0.3. Void growth is expected and ES-0.3 provides guidance to control the void growth.

Surry

References:

1-FR-1.3, Response To Voids In Reactor Vessel, Rev. 16

1-ES-0.3, Natural Circulation Cooldown With Steam Void in Rx Vessel, Rev. 12

Distractor Analysis:

- A. Incorrect because ES-0.3 does not assume that FR-1.3 is being used.
- B. Incorrect because SI should not be initiated and there is no need to vent the head.
- C. Incorrect because ES-0.3 does provide guidance for managing void growth.
- D. Correct because ES-0.3 **does** provide guidance for managing void growth.

W/E10 Natural Circ.

EA2.1: Ability to determine and interpret the following as they apply to the (Natural Circulation With Steam Void in Vessel with / without RVLIS): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry Requal Bank Question #247

99. WE13EA2.1 001/1/2/SG STEAM GENERATOR/C/A 2 9/3.4/B/SR04301/S/MAB/SDR

During performance of E-3, Steam Generator Tube Rupture, the operating team is directed to adjust the SG PORV setpoint on the ruptured SG to 1035 psig. The Reactor Operator observes ruptured SG pressure to be 1070 psig and the PORV cycling.

Which ONE of the following is the appropriate course of action and reason for the action?

- A. Transition to FR-H.2, "Response to Steam Generator Overpressure" to prevent an overpressure condition in the ruptured SG.
- B. Increase feed flow to the ruptured SG to stop the release and remain in E-3.
- C. Increase the setpoint above 1070 psig to prevent release to the public and transition to ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery.
- D. Leave the PORV setpoint at 1035 psig to minimize challenges to the SG Code Safeties and remain in E-3.

Surry

References:

1-E-3, Steam Generator Tube Rupture, Rev. 25

ND-95.3-LP-13, E-3 Steam Generator Tube Rupture, Rev. 11

Distractor Analysis:

- A. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, FR-H.2 is not associated with any Red or Orange paths.
- 5. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, feeding a ruptured SG will not limit the exposure to the public.
- C. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, this action may challenge the code safeties, which is not desirable.
- D. Correct because this is the correct direction in the procedure and the correct basis for the step.

W/E13 Steam Generator Overpressure

EA2.1: Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry Requal Exam Bank Question #324

100. WE15EA2+002/1/2/CONTAINMENT FLOOD/MEM 2.7/3.2/B/SR04301/S/MAB/SDR

The Control Room Operators are performing FR-S.2, Response to Loss of Core Shutdown, in response to a yellow path condition shown on the Critical Safety Function (CSF) status tree.

Which ONE of the following is correct with regard to transitions out of this procedure?

- A. The operators must leave this procedure at any step as soon as the Loss of Core Shutdown CSF adverse condition has cleared. (Green path established)
- B. The operators must leave this procedure before completion and go to FR-H.1, Response to Loss of Secondary Heat Sink, if the heat sink CSF status tree indicates a yellow path condition.
- C. The operators must leave this procedure before completion and go to FR-C.3, Response to Saturated Core Cooling, if the Core Cooling status tree indicates a yellow path condition.
- D. The operators must leave this procedure before completion and go to FR-Z.2, Response to Containment Flooding, if the containment CSF status tree indicates an orange path condition.

Surry

References:

ND-95.3-LP-26, Critical Safety Function **Status** Trees, Rev. 5

Distractor Analysis:

- A. Incorrect because the operator does **not** have to immediately leave FR if it is not completed.
- B. Incorrect because yellow path does not warrant this action.
- C. Incorrect because yellow path **does not** warrant this action.
- D. Correct because orange path takes priority.

WE15 Containment Flooding

EA2.1: Ability to determine and interpret the following as they apply to the (Containment Flooding): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry ILT Bank Question # 1350