Final Submittal

SURRY EXAM
50-280, 50-28112004-301

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

1. Reactor Operator Written Examination

WITH
REFERENCES

U.S. Nuclear Regulatory Commission
Site-Specific
RO 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 80.00 percent. Examination papers will be collected six hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value:	________	Points
Applicant's Score:	________	Points
Applicant's Grade:	________	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 Pump 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 Pump supplies the bearing lubrication.

D. The oil lift pump discharge pressure must be greater than *300* psig prior to WCP start. Once the RCP reaches operating speed, the thrust runner circulates oil in the upper and iower bearing assemblies.

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References:
NB-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 fur 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, PRZR 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 a d a 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 RCS 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.

Q04 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: how 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, boss of Decay Heat Removal Capability, has been entered.
- The RHR system has just been made available.

Which ON€ 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 RHR 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 I-A$-27.00, Page 16 of 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.

4. 006K6.03 001/2/1//C/A 3.6/3.9/N/SR04301/R/MAB/SDR

Unit 1 tripped 30 minutes ago. The following plant conditions currently exist:

- *3* CETCs indicate between 700 °F and 750 °F and slowly rising
- No WCPs are operating
- RVLIS Full Range is indicating 46% and slowly lowering
- 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
- FR-C.1, Response to Inadequate Core Cooling, has been implemented
- WCP 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

Which ONE of the following describes the correct strategy for mitigating the consequences of these conditions?

A. Depressurize all intact steam generators at maximum rate, while maintaining steam flow less than 1.0×10^5 PPH to try to establish accumulator and LHSI flow. If CETCs rise above 1200 °F, then check conditions for starting a RCP.

5. Bepressurize all intact steam generators at maximum rate, while maintaining steam flow less than 1.0×10^5 PPH to tty to establish accumulator and LHSI flow. If CETC temperatures rise above 1200 °F, WCPs should not be started due to low RCP seal injection flow.

C. Depressurize all intact steam generators at maximum rate, while maintaining steam flow less than 1.0×10^6 PPH to try to establish accumulator and LHSI flow. If CETC temperatures rise above 1200 °F, RCPs should not be started due to low RCF seal injection flow.

D.✓ Depressurize all intact steam generators at maximum rate, while maintaining steam flow less than 1.0×10^6 FPH to try to establish accumulator and LHSI flow. If CETCs rise above 1200 °F, then check conditions for starting a RCP.

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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

Bistractor Analysis:
A. Incorrect because 1.0 x 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 x 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 x 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 sacked 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 drip 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

Bistractor Anaysis:
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 fiiled 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 RCS 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 GQQ-1.1 Step 5.5.4 directs establishment of normal PRT 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 foilowing establishment of drawing a bubble (GOQ-1.1, Step 5.5.13).
D. Incorrect because RCS should be between 300 and 370 psig on PI-1-403.

007 Pressurizer Relief / QuenchTank
K5.82: 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 = 100%
- 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?

pressure stabilizes around 200Qpsig.

B. POW-1456 opens, pressure decreases to 2000 psig, POW-1456 closes, and pressure stabilizes around 2000 psig.

C.' PORV-1455C opens, at 2000 psig PORV-1455C closes; however, pressure will continue to decrease musing 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

Distractcr 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 POW-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 lo 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, hut 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
K4.01: Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump.

10. 010A1.01 001/2/1/BORON PZR SPRAY/C/A 2.8/2.9/N/SR04301/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 criticaiity?
(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.)
5. 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.

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: PZ!? 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 WWST level reaches 13%, push both RMT pushbuttons for each train if automatic transfer does not occur.

C. With WWST 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 WMT.

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 tu Cold beg 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* gprn 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-I122 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 plausibe 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.

the reactor is at 100%?

A. Rod Insertion Limit Low and Extra Low alarms will be received.

B. Ch 1 OTDT setpoint will decrease.

C.✓ "A" Loop OP and OT Delta T Protection Bistables will trip.

D. The Tavg / Tref Deviation alarm will be received.

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References:
NB-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. 012K1.05 001/2/1/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.

5. 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 becasue 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 4.1/4.2/N/SR04301/R/MAB/SDR

Which ONE of the following correctly states the 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-1 OOA) receives a dose 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 Surry 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 MBAFW 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
A3.02: 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 keg 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-71, ES-1.4, Transfer To Hot Leg Recirculation, Rev. 8
ES-1.4, Transfer Bo Hot Leg Recirculation, Rev. 4

Distractor Analysis:
A. Incorrect because debris in the sump will not block water discharged from the SI 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 20 hours, as stated in the stem.

013 Engineered Safety Features Actuation
K3.01: 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 occured
- 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.l, 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).

014 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-4f 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.

Surry

References:
ND-93.3-LP-16, Permissive/Bypass/Trip Status Lights, Rev. *8*
Surry Simulator Malfunction Cause and Effects, Rev. 6, Malfunction MMS-14
ND-93.2-LP-4, Power Range NIs, Rev. 16
Surry Simulator Malfunction Cause and Effects, Rev. 6, Malfunction MNI-10

Distractor Analysis:
A. Incorrect because 2/4 NIs must be above 10% to enable At-Power Trips.
B. Incorrect because 2/4 NIs must be above 10% to enable At-Power Trips and rods will not move.
C. Correct because rods are in MANUAL at 5% power and will not move (P-2 prevents movement). At-Power trips are enabled when 1 of *2* Pimp channels goes above 10%.
D. incorrect because rods are in MANUAL at 5% and will not move.

015 Nuclear Instrumentation
K4.07: Knowledge of NIS design feature(s) and/or interlock(s) provide far 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-LB-8, SG Water Level Control System, Rev. 6

Distractor Analysis:
A. incorrect because this would occur after 36Q 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. 022AK1.01.001/1/1/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 A$-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.

022 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. 02202.4.22 001/2/1/SAFETY FUNTIONS/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.

Surry

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

Distractor Analysis:
A. Incorrect based on the rules of use for safstv 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-35.3-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 1

Unit 2 is operating at 100% power with Chilled CC in service to containment. 2-CD-REF-1A, Unit 2 Turbine Building Chiller Unit, 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

Bistractor 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. 026A2.07.001/2/1/CONTAINMENT SPRAYICIA 3.6/3.9/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:
- A Large Break LOCA has occurred inside containment
- Safety Injection ha5 actuated
- Containment Pressure peaked at 28 psia
- Current Containment Pressure is 15.8 psia
- "2A" and "2B" Outside Recirculation Spray Pumps are operating
- "1A" Inside Recirculation Spray Pump is operating
- "1B" Inside 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 Inside Recirculation Spray Pump "1A" using the handswitch in the control room.

B.✓ Place the Inside Recirc Spray Pump 1A in PTL, then secure Inside Recirculation Spray Pump "1A" locally at the breaker (14H4).

C. Reset CLS, then place the handswitch for Inside Recirculation Spray Pump "1A" in PTL.

D. Allow Inside Recirculation Spray Pump "18" 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 to 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. Incorrect 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 consequnces 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. 026AK3 02 001/1/1/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 Trip Valve) remain as-is.

C. TV-CC-1098 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-I10A, 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 and 110 only closes on Phase III isolation.
D. Incorrect because lesson plan states CC-109 closes on Phase 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 occurred 45 minutes ago
- Safety injection has actuated
- Containment Pressure peaked at 27 psia
- RCS subcooling is 0 °F
- Steam Generator Levels are 22% and slowly rising
- WWST 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 PRESS CLS CH-1

B.✓ 1B-B1, CS PP 1A LOCKOUT OR Ob TRIP

C. 1A-D7, RS PP 1A LOCKOUT OW OL TRIP

D. 1B-F6, CTMT INST AIR HDR LO PRESS

Sur9

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 PRESS CLS CH-1, Rev. 0
1B-B1, CS PP 1A LOCKOUT OR OL TRIP, Rev. 0
1A-D7, RS PP 1A LOCKOUT OR OL 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 Recirc, CLS and SI must be reset.
B. Correct because 1-E§-1.3 has been completed and the RWST has been emptied; therefore, the CS Pumps would be placed in PTL 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 Inside 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
A4.03: Ability to manually operate and / or monitor in the control room: CIRS fans.

Question Status:
Surry Bank ILT Question #741

27. 027AK3.01 001/1/1/PRESSURIZER SPRAY/C/A 3.5/3.8/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-enerized.
- 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 Pressurizer Sprays, Proportional Heaters, and Backup Heaters, assuming that the Proportional Heaters are now operable?

A.✓ Normal Sprays are OFF (valves closed); Proportional Heaters are ON; Backup Heaters are ON.

B. Normal Sprays are OFF (valves closed); Proportional Heaters are OFF; Backup Heaters are ON.

C. Normal Sprays are OFF (valves closed); Proportional Heaters are ON; Backup Heaters are OFF.

D. Normal Sprays are ***ON*** (valves open); Proportional Heaters are ON; Backup Heaters are ON.

Surry

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

Bidractor Analysis:
A. Correct because backup heaters are always on, proportional heaters will be on below about 2220 psig, and spray valves will be closed below about 2255 psig. (The correct answer was validated on the simulator at 2215 psig.)
5. Incorrect because proportional heaters would be on.
C. Incorrect because backup heaters would be on.
D. Incorrect because spray valves would be closed.

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. 028G2.2.12 001/2/2/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 LQCA.

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

A.✓ Turn Power Adjust potentiometer to obtain 48 KW on the wattmeter and approximately 1225 °F heater output. At no time should the heater temperature be allowed to exceed 1400 °F on the highest thermocouple reading.

B. Turn Power Adjust potentiometer to obtain 48 KW on the wattmeter and approximately 1000 °F heater output. At no time should the heater temperature be allowed to exceed 1200 °F on the highest thermocouple reading.

C. Turn Power Adjust potentiometer to obtain 36 KW on the wattmeter and approximately 1225 °F heater output. At no time should the heater temperature be allowed to exceed 1400 °F on the highest thermocouple reading.

D. Turn Power Adjust potentiometer to obtain 36 KW on the wattmeter and approximately 1000 °F heater output. At no time should the heater temperature be allowed to exceed 1200 °F on the highest thermocouple reading.

Surry

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

Distractor Analysis:
A. Correct because as stated in the procedure temperature must remain less than 1400 °F at all times and the potentiometer shall be adjusted to 48 KW.
B. Incorrect because 48KW does not equate to 1000 °F heater output and the temp limit is 1400 °F.
C. Incorrect because 36 KW does not equate to 1225 °F heater output.
D. Incorrect because as stated in the procedure temperature must remain less than 1400 °F.

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

If the Emergency Boration Flowpath is not available, which ONE of the following describes the reason why charging pump suctions 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 **WWSP** 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.l 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 MFW 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 MFW 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 seondary heat sink because **MFW** 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 thsy 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 exists:
- Present time is 1428 hours
- Reactor tripped at 1405 hours
- All Rod Bottom Lights are lit
- N-35 reading is 2 x 10^{-10} amps
- N-36 reading is 4 x 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 x 10^{-10} amps, verify source range channels energized.

B. Place the source range trip bypass switches in the NORMAL position.

CY 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.

Bistractor Analysis:
A. Incorrect because SR energizes at 2/2 IR < 5 x 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 = 2.5×10^{-8} Amps; PR between 0.2 and 1 %

B.✓ IR = 2.0×10^{-6} Amps; PR between 0.2 and 1 %

C. IR = 1.0×10^{-8} Amps; PR < 0.2 %

D. IR = 1.0×10^{-5} Amps; PR 6 0.2 %

Surry

References:
ND-93.2-LP-4, Power Range Nls, Rev. 16
1-GOQ-1.4, Unit Startup, HSD to 2% Reactor Power, Rev. 29

Distractor Analysis:
A. Incorrect because 2.5×10^{-8} Amps is about where critical data is taken (too low).
E. 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 1.0×10^{-8} Amps is about where critical data is taken (too low).
D. Incorrect because 1.0×10^{-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 I is in a refueling outage when the following events occur.
- Purge Isolation Valves (MOV-VS-100A, 5. C, and D) Close
- Containment Instrument Air Suction Valves (TV-IW-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)

B. RM-161 (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-81-IO1 / 102 do not cause these actions.
B. Incorrect because RM-161 does not cause these actions.
C. Correct per ND-93.5-LP-1.
D. Incorrect because RM-I63 does nut cause these actions.

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

33. 035A3.01 001/2/2/STEAM GENERATOR/C/A 4.0/3.9/N/SR04301/R/MAB/SDR

The following Unit 1 conditions exist:

- Plant is stable at 75% Power
- "A" SG Steam Line FP-MS-475 (CH-IV) is selected for Steam Generator Level control
- "A" SG Steam Line PT-MS-475 (CH-III) 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

Bistractor 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
83.01: 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.

Surry

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 gprn, but 350 gprn must still be available to that SG. If sufficient flow is available, then SI termination criteria is considered to be met (NB-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 fer 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 Radiatic Monitor alarms are recieved. The Condenser Air Ejector Radiation Monitor reads 70(cpm (ALERT and HIGH alarms are in) while local Main Steam NRC Radiation Monitc read "A" .03 mr/hr, and "B" .01 mr/hr, and "C" .01 mr/hr. The Team has implementec 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 I-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, manually trip the reactor and initiate 1-E-0 (Reactor Trip or Safety Injection), and go to 1-AP-24.01 (Large Steam Generator Tube Leak).

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-16.00, 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
81.09: 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 acordance 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 LO 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, boss of Feedwater, Rev. 3
1-AP-21.00, Loss of Main Feedwater Flow, Rev. 5
1G-G8, ROD BANK D LO LIMIT, Rev. 0

Distrador 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 toad to match steam flow and feed flow. Raising turbine load under these conditions would not be the correct action. It would also be nonconservtaive 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"
- Annunciator J-F-6, TURB GEAR ZERO SPEED, *is* lit

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
1J-F6, TURB **GEAR** ZERO SPEED, Rev. 1

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.

D. 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
EKl.O1: 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 = 65%
- Condensate Pumps 1-CN-P-IA 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 bevels 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 Brain Pump.

Surry

References:
ND-89.3-LP-2, Main Condensate System, Rev. **16**
ND-89.3-LP-3, Main Feedwater System, Rev. 12
NB-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 HBR LO PRESS, Rev. 0
1H-G8, FW PP DISCH HDW 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/discharge pressure.
B. Incorrect because no trip criteria are met and AQ-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. Incorret because there is no guidance to start a second heater drain pump. The correct response is to lower turbine load.

056 Condensate
82.04: 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.

FACILITY POST EXAM COMMENT:
Facility Comment: Most trainees applied the Management standard for conservative decision making. If reactor trip is imminent, then manually trip the reactor and perform the immediate actions of E-0. The definition of IMMINENT is within one to two hours and continuing deteriorating conditions exist.

DNOS-0101, Nuclear Safety and Conservative Decision Making, states "Operators faced with unexpected or uncertain conditions will place the plant in a safe condition and will not hesitate, if necessary, to reduce power or trip the reactor."

The conditions provided in this question, one condensate pump running with feed pump suction pressure at 105 psig and decreasing, can only result if the only running condensate pump is also significantly degraded. This places the plant in a condition not considered in the development of **AP-21.00, Loss** of Main Feedwater Flow.

The Supervisor of Shift Operations said that he would not hesitate to trip the reactor given the conditions provided in the stem of the question.

We ran this scenario on the simulator and were unable to keep the unit online. It resulted in a reactor trip 100% of the time.

6 of 10 trainees chose answer **(B).**

Recommendations: Based on the above information, accept (B) as an alternate correct answer.

NRC Resolution: Recommendation accepted; the question has two correct answers **(A** and **B).** The NRC concurs that it is reasonable and conservative for an operator to manually trip the reactor with Main Feedwater pump suction pressure at 105 psig and slowly lowering. The stem Conditions created the sense that the system conditions were continuing to degrade therefore, a natural assumption was that a reactor trip was imminent.

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

The following conditions exist:
- A boss 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 120 volts 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.

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

40. 057AK3.01.001/1/1/VITAL AC/C/A 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

Distractor 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.
5. Incorrect because Pnr 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.
5. 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 boss 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 OF DC POWER/C/A 3 7/4 1/N/SR04301/R/MAB/SDR

The following Unit 1 conditons exist:
- 1K-A8, UPS SYSTEM TROUBLE, annunciates
- 1K-A7, BATT SYSTEM 1A TROUBLE, annunciates
- An operator reports that Battery Charger DC Output for UPS SA-1 reads 0 amps

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

A. DC Bus 1A 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 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 will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at 125 VBC and the Vital AC Buses 1 and 1A will be supplied by 1H1-1.

D.✓ DC Bus 1A will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at 125 VBC and the Vital AC Buses 1 and 1A will be supplied by 1H1-2.

Surry

References:
ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11
NB-90.3-LP-6, 125 VDC Distribution, Rev. 10
1K-A7, BATT SYSTEM 1A TROUBLE, Rev. 5
1K-A8, UPS SYSTEM TROUBLE, Rev. 4
11448-FE-1G, Sheet 1 of 1, 125V DC One Line Diagram - Surry Power Station Unit I, 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 1HI-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 1HI-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 1HI-2 will supply Vital AC Buses 1 and 1A.

058 Loss of DC Power
AA2.01: 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. 059A1.03 001/2/1/MAIN FEEDWATER/MEM 2.7/2.9/N/SR04301/R/MAB/SDR

Which ON€ of the following sets 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 Reciculation Valve is in AUTO, the operator should observe that valve closure will occur as the feed flow rises above 3000 gprn.

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 gprn.

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 overpressurination of the piping between the discharge check valve and the MOV as the system heats.

Surry

References:
1-GO$-1.5, Unit Startup, **2%** Reactor Power to Max Allowable Power, Rev. 32
1-0P-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 4900 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 (WS), 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 EEEEEs 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:
NB-93.5-LP-4, 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

Bistractor 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 knew 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 Radwaste 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% (60,000 gallons). AFW pumps must be secured prior to commencing the fill.

B. Filling may commence after the ECST level drops below 54% (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% (60,000 gallons). 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 3.6-7, 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.
B. Correct based on all three of the above references.

061 Auxiliary Feedwater
A1.04: 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. ~~061~~AA2.01 001/1/2/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

D. Ventilation Vent Gaseous Radiation Monitor and the Liquid Waste Effluent Process 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

Bistractor Analysis: (maybe get some help to provide a little better distractor analysis?)
A. Incorrect because the Aux Bld Control Victoreen Area Wadiation 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.

Sur9 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 KVAW was 500 KVAR out
- The output voltage was stable at 4300 VAC

Which ONE of the following was in violation of the EDG 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, Service Water System, Rev. 20

Bistractor 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 bevel 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 boss of Nuclear Service Water (SWS): Control of flow rates to components cooled by the SWS.

48. 063A4 01 001/2/1/BREAKERS/C/A 2 813 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 bights are not lit
- PORV 1455C/1456 Indicating Lights are not lit
- "A", "D", and "H" 4160 V Bus Breaker Indicating bights 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.?

AY 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.

Sur9

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 2J7-1 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.

D. 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

Bistractor Analysis:

A. Incorrect because 1J1 voltage was less than 41Ov tor 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 tu 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 141 voltage was less than 41Ov for greater than 6 seconds. Therefore, 2J1-1 became the power supply after six seconds. The ABT will check fer 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 Compessors.

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 tu be determined
- An Operator reports the sound of compressed air leaking in the area of the RHR pump platform.
- 1B-F-6, CTMT INST AIR HDR LO PRESS, 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 (RHW 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 (WHW 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 vitat control instruments.

D. If the air leak is a rupture upstream of the isolation valve fur 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)
NB-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-F6, CTMT INST AIR HDR LO PRESS, Rev. 1

Distractor Analysis:
A. Incorrect because HCV-1758 fails open and cannot be operated with a portable air bottle. Plausible because the applicant may get consfused 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 consfused on 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 RWST 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 RWST. Which ONE of the following describes the basis for this step?

A. Suction is shifted to the RWST 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 WWST to maximize boron injection before the charging pumps overheat and are lost due to an 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 acuracy and supply additional supporting material if any is availble.)

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

Bistractor 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 WWST 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* 100% 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** beak 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.

Surry

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

Bistractor 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 Hut 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 Radwaste
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.

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

53. 071K4.06.001/2/2/RADIATION MONITOR/MEM 2.7/3.5/N/SR04301/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 frum 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 Vaive closes.

C. 1-GW-FCV-260, CTMT Vacuum Pump Discharge Isolation Valve, closes.

B/ Associated vacuum pumps trip.

Sur9 (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.
5. Incorrect because according to ARP, this valve will close on reaching the high alarm setpoint.
C. Incorrect because acording *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 ARP 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
K4.06: Knowledge of design(s) features and / or interlocks which provide fur the following: Sampling and monitoring of waste gas release tanks.

54. 073G2 1 23.001/2/1/PROCESS RADIATION/MEM 3 9/4 0/B/SR04301/R/MAB/SDR

Which ONE of the following valves AUTOMATICALLY closes when a HIGH alarm is received on Component Cooling Water Radiation Monitor, RM-CC-105?

A.✓ Surge Tank Vent Valve, HCV-CC-100

B. Excess Letdown Heat Exchanger Outlet, HCV-CC-108.

C. RCP's Thermal Barrier CC Outlet Flow Inside and Outside Trip VLV, TV-CC-140.

D. Thermal Barrier Heat Exchanger Isolation Valve, TV-CC-120A

Surry

References:
ND-88.5-LP-1, Component Cooling, Rev. **19**

Distractor Analysis:
A. Correct as stated in ND-88.5-LP-1 Page 12.
B. incorrect, but plausible due to the outlet valve also being an HCV in the CCW Sys.
C. Incorrect, but plausible due to CCW cooling the RCP Th Barrier HX.
D. Incorrect, but plausible due to CCW cooling the WCP Th Barrier HX.

073 Process Radiation Monitoring
G2.1.23: 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 fitter 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 WCP for seal injection. Five of these gallons flows down the shaft past the thermal barrier and ends up in the WCS. 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 corning from the WCP 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. 076K2.04 001/2/1/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
- 4160V "H" bus de-energizes

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

A. ONLY the Service Water Outlet Valves (MOV-SW-105s) from each Recirc Spray Heat Exchanger de-energize.

B. ONLY the Service Water Inlet Valves (MOV-SW-104s) from each Recirc Spray Heat Exchanger de-energize.

C. The Service Water Inlet Valves (MQV-SW-104s) and Service Water Outlet Valves (MOV-SW-105s) from both Inside Recirc Spray Heat Exchangers de-energize.

BY The Service Water Inlet Valves (MOV-SW-104s) and Service Water Outlet Valves (MOV-SW-105s) from one Inside and one Outside Recirc Spray Heat Exchanger 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, flev. 59
P&ID 11448-E-1L, Sh 1 of 1, 480V One Line Diagram Surry Power Station - Unit 1, Rev. 52

Distractor Analysis:
A. Incorrect because the inlet and outlet valves from one inside and one outside recirc spray H.X. de-energize.
B. Incorrect because the inlet and outlet valves from one inside and one outside recirc spray H.X. de-energize.
C. Incorrect because valves from only one inside recirc spray H.X. de-energine.
D. Correct because valves de-energize for the A (IRS H.X.) and C (ORS H.X.) only.

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 Service Air from 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-AQ-40.00, Non-recoverable **boss** of Instrument Air, Rev. 17

Distractor Analysis:
A. Incorrect because 1B-E6 and AP-40.00 directs rx trip, not power reduction.
5. 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 AQ-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

Distractor 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 Service 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.

Sur9 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 = 19.7 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 HI CLS relays will energize and the HI HI CLS relays will de-energize.

B.✓ Neither CLS HI or CLS HI-HI may be reset at this time. The HI CLS relays are de-energized and the HI HI CLS realys are energized.

C. The CLS HI-HI RESET PERMISSIVE annunciator is lit. CLS HI-HI may be reset at this time. Upon reset the HI HI CLS relays will de-energize.

D. Neither CLS HI or CLS HI-HI may be reset at this time. The HI CLS relays are energized and the HI HI CLS relays are de-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 HI CLS relays are de-energized and the HI HI CLS relays are 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 HI CLS relays are de-energized and the HI HI CLS relays are energized.
D. Incorrect because when CLS is actuated, the HI CLS relays are de-energized and the HI HI CLS relays are energized.

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

60 G2.1.11 001/3//TECH-SPECS/C/A 3.0/3.8/N/SR04301/R/MAB/SDR

The following Unit 1 conditions existed:

- Plant was at 74% power after just completing a rapid power reduction due to High Pressure Heater Drain 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 14/04/2003 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 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 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 "6" 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 2700 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?
(AP-29.00 Attachments 1 and 2 provided)

A. Raise RCS level to 12.5 Beet as indicated on 1-RC-LI-100A and stabilize Blow 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 1200 gprn.

D.✓ Throttle close 1-RH-FCV-1605 to reduce RHR flow to 2200 gpm and raise level to 12.5 feet *as* indicated on 1-RC-LI-100A.

Surry

References:
1-AP-27.00, boss 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 sate of 2200 gpm.
D. Correct because these actions place the plant in an acceptable region of AP-27 Att. I 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

G2.7.25: Ability *to* obtain and interpret station reference materials such **as** graphs, monographs, and tables which contain performance data.

62. G2.2.22001/3//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 $5x10^{-10}$ amperes.

Surry

References:
Technical Specification 2.1 (Amendments 4 16); 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 118%.
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 $5x10^{-11}$ amperes.

Generic K/A 2.2.22
Knowledge of limiting conditions for operations and safety limits.

63. G2 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 SWO

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 SWO 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 exists:
- 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-AQ-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 MCW Ventilation by closing 1-VS-MOD-103C and 1-VS-MOD-103D, Bump 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, Dump 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 MER 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 RNQ column.
C. incorrect because 1-VS-MOD-103A is in the WNO 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. G2.3.2 001/3//RADIATION RESPIRATOR/C/A 2.5/2.9/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 twc 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 ab 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 mrem > 56.7 mrem. {[(2 men)(1 hr)(50mrem/hr)$(1/2)^2$]+[(10 DAC)(2 men)(1 hr)(2.5 mrem/DAC-HR)] = 75 mrem}
B. Incorrect: 158.3 mrem > 56.7 mrem. {[(3 men)(2 hr)(50mrem/hr)$(1/6)^2$]+[(10 BAC)(3 men)(2 hr)(2.5 mrem/DAC-HR)] = 158.3 mrem}
C. Incorrect: 104 mrem > 56.7 mrem. {[(2 men)(4 hr)(50mrem/hr)$(1/6)^2$]+[(10 DAC)(1/50)(2 men)(4 hr)(2.5 mrem/DAC-HR)] = 104 mrem}
5. Correct: [(3 men)(10 hrs)(50 mrem/hr)$(1/6)^2$] + [(10 DAC)(1/50)(3 men)(10 hrs)(2.5 mrem/1 DAC-HR)] = 41.7 + 15 = 56.7 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 1-0P-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 10,000 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 (Ctmt Purge B/P) due to the high flow rate required by the Containment Purge Form.

B. 1-VS-MOV-101 (Ctmt Purge B/P) should be throttled instead of I-VS-MOV-IOOB (Ctmt Purge Exh). This is due to the need to open the supply breaker to I-VS-MOV-1008 in order to throttle it. Opening the breaker will prevent automatic CTMT Purge isolation.

C. I-VS-MOV-101 (Ctrnt Purge B/P) should be throttled instead of 1-VS-MOV-100D (Ctmt Purge Exh) due to the low flow rate required by the Containment Purge Form.

D. 1-VS-MOV-1OQD(Ctmt Purge Exh) should be throttled instead of 1-VS-MBV-101 (Ctmt Purge B/P). This is due to the need to open the supply breaker to I-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 100D should be throttled due to the Containment Purge Form allowing more than 3000 cfm. The bypass will not have enough capacity at this fiow 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 gprn is not a very high flow rate.
D. Incorrect because the bkr does not need to be opened and at 10,000 gprn, 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.

67. G2.4.11 001/3//ABNORMAL PROCEDURE/MEM 3.4/3.6/N/SR04301/R/MAB/SDR

Which ONE of the following correctly states the requirements for performing immediate action steps within emergency procedures?

A. Immediate action steps must be performed in the order in which they appear in any procedure.

B. Immediate action steps may be performed in any order, except for the first four immediate action steps of E-5, 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 action steps of FR-S.1, Response to Nuclear Generation / ATWS, and ECA-0.0, boss of All AC Power, which must be performed in the order in which they appear in the procedure.

Surry

References:
ND-95.3-LP-2, Emergency Procedure Writer's Format, Rev. 8
(Have Utility add any addional references that may support answer.)

Bistractor Analysis:
A. Incorrect because only immediate actions of E-0 and **FR-S.** 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. 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 mus 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-5.0 are not required to be performed in any specific order.

G2.4.11: Knowledge of abnormal condition procedures.

68. ii2.4.12 001/3//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.

Surry

References:
OPAP-0006, Shift Operating Practices, Rev. 4

Distractor Analysis:
A. Correct because OPAP-0006 Step 6.10.3 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
- Tave 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.

Sur9

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 CONP MODE SEL switch was placed to MANUAL and rod motion had stopped.
D. Incorrect because AP-1.00 directs placing ROD CONP 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. WE04EK3.2 001/1/1/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-18908 (LHSI to Hut beg) 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-I, Loss of Reactor or Secondary Coolant, because RCS inventory is continued to be lost outside of containment.

D✓ If closing 1-SI-MOV-1890C (LHSI tu 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

Distractor 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 wuuld be to go tu 1-ECA-1.1, Loss of Emergency Coolant Recirculation.
D. Correct because if RCS pressure rises upon closure of 1-SI-MOV-18906, 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 **WWST** to be closed.

WE04
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/SR04301/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.

5. 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**

Bistractor Analysis:
A. Incorrect because securing RCPs is necessary because the depressurization will result in losing the RCP seal support conditions, which could damage the WCPs.
5. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS or forced flow.
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 NB-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 I 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 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. COCA 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** ctmt.
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.

Sur9 Requal Bank Question #177 (ID: EOP0076)

73. WE11EA1.2 001/1/1/LHSI LOCA RWST HHSI/C/A 3.5/3.8/N/SR04301/R/MAB/SDR

The following conditions exist:
- LOCA has occurred.
- WWST level = 13% and decreasing.
- Recirculation Mode Transfer (RMT) keyswitch is in RMT Mode.
- White RMT Status bight is lit.
- Amber RMT Status Light is lit.

1-SI-MOV-1860A (LHSI Suction from Sump) opens fully and 1-SI-MOV-1860B (LHSI 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.✓ "B" LHSI Pump from the RWST and "A" LHSI Pump from the Containment Sump is injecting into the cold legs and HHSB from LHSI pump discharge is injecting into the cold legs.

B. No Safety Injection is injecting water to the cold legs.

C. HHSB directly from the RWST (not from LHSI discharge) is injecting into the cold legs, but no LHSI is injecting into the cold legs.

D. Both LHSI Pumps from the RWST and HHSB directly from the RWST (not from LHSI discharge) is being injected into the cold legs.

Surry

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-1862B will not close until 1-SI-MOV-1860B opens fully (due to an interlock).
B. Incorrect because RWST and Sump are suction sources for LHSI pumps.
C. Incorrect because LHSI Pumps are taking suction from RWST and Sump 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. HHSB is taking suction on the discharge of the LHSI Pumps.

WE11
EAI.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 2 001/1/1/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 lo 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 rasied.

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

75. WE13EK2.1001/1/2/SGTR OVERPRESSURE/MEM 3.0/3.1/M/SR04301/R/MAB/SDR

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 bevel to be greater than 12% prior to stopping feed flow.

Which ONE of the following correctly states the basis for this step?

A.✓ To ensure that the ruptured steam generator tubes are covered to promote thermal stratification.

B. Bo ensure thermal gradients across the tubes of the ruptured steam generator do not exacerbate existing tube damage.

C. To ensure sufficient heat sink for reactor coolant system cooldown.

D. To prevent excessive primary to secondary leakage.

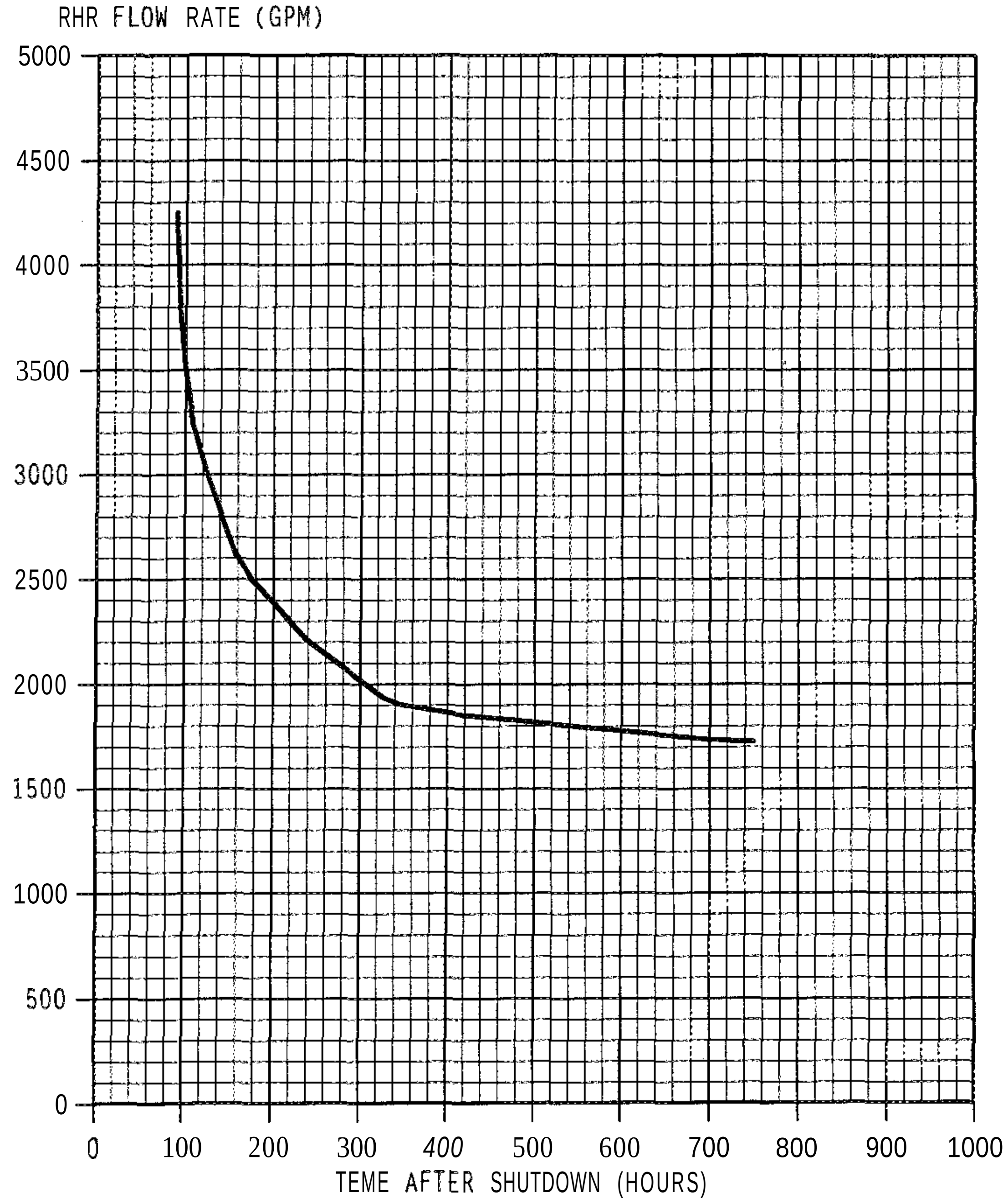
Surry

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

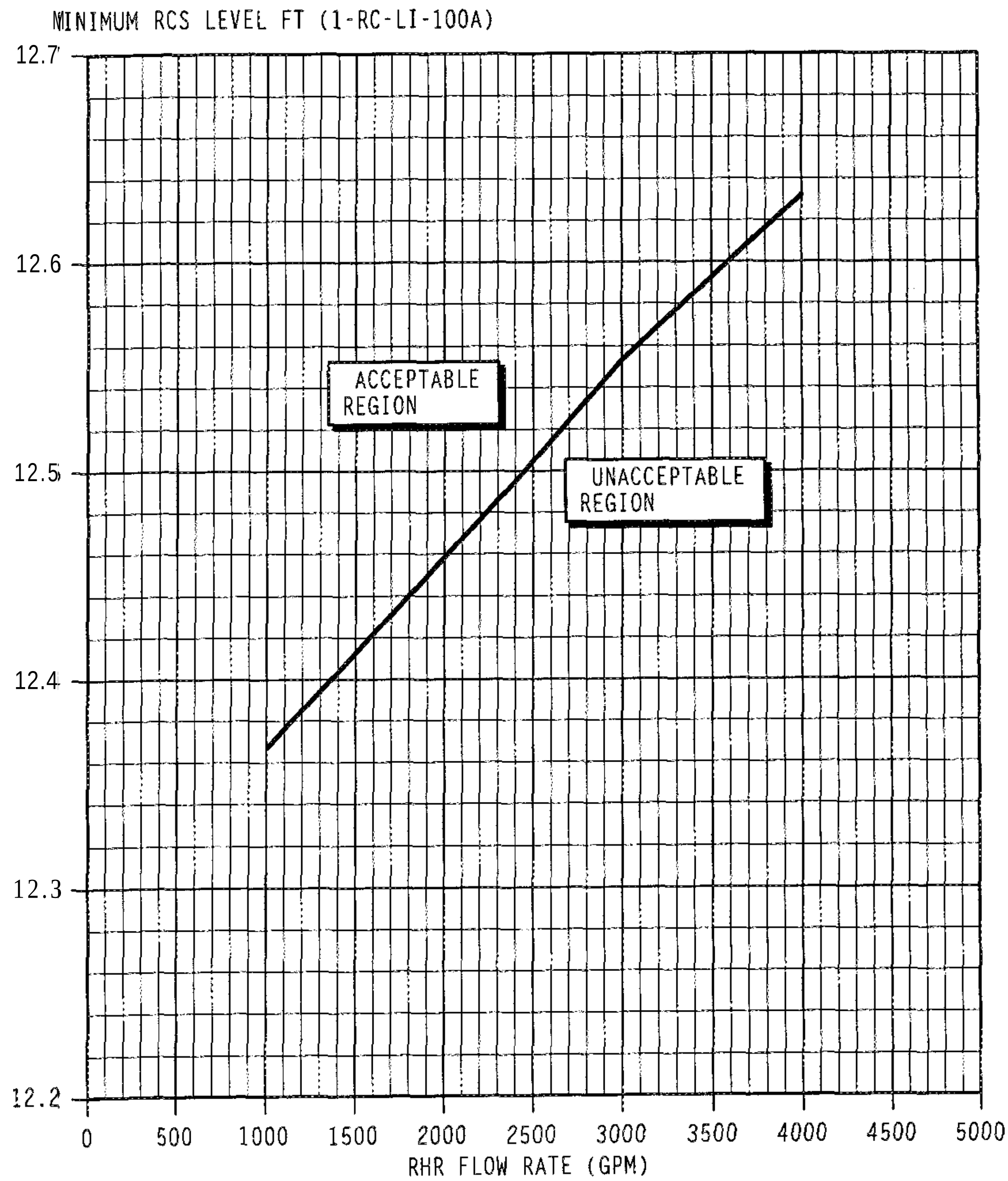
Distractor Analysis:
A. Correct because this is the basis as stated in ND-95.3-LP-13.
B. Incorrect because the concern is not thermal gradients across the tubes. The concern is to cover the tubes for thermal stratification and then stop AFW flow as soon as the tubes are covered to give margin to overfill, 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.



RHR FLOW RATE (GPM)
5000
4500
4000
3500
3000
2500
2000
1500
1000
500
0
0
100
200
300
400
500
600
700
800
900
1000
TEME AFTER SHUTDOWN (HOURS)


MINIMUM RCS LEVEL FT (1-RC-LI-100A)
12.7
12.6
12.5
12.4
12.3
12.2
ACCEPTABLE REGION
UNACCEPTABLE REGION
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
RHR FLOW RATE (GPM)

SURRY LORP EQUATION SHEETS

$Q = \dot{m} c_p \Delta T$

$\dot{Q} = \dot{m} \Delta h$

$Q = UA\Delta T$

$\dot{Q} \propto \dot{m}^3{}_{NatCirc}$

$\Delta T \propto \dot{m}^2{}_{NatCirc}$

$KE = \frac{1}{2} m v^2$

$w = v\Delta P$

$\dot{W}_{pump} = \dot{m} \Delta P v$

$Pwr = W_f \dot{m}$

$Pwr = W_f \Delta h$

$Cycle\ Effeciency = \frac{Net\ Work\ Out}{Energy\ In}$

$s = v_0 t + \frac{1}{2} a t^2$

$v = s/t$

$V_f = V_0 + at$

$a = \frac{(V_f - V_0)}{t}$

$w = \frac{\theta}{t}$

$/ = m\ a$

$w = mg$

$PE = mgn$

$F = PA$

$\dot{m} = v_{av} A p$

$\dot{m} = \rho A v$

$v(P_e - P_1) + \frac{1}{2}(v_e{}^{-2} - v_1{}^{-2}) + g(z_e - z_i) = 0$

$Z_1 + P_1 v_1 + \frac{v_1{}^{-2}}{2g} + h_p = Z_2 + P_2 v_2 + \frac{v_2{}^{-2}}{2g} + h_L$

$g_c = \frac{32.2\, lbm - ft}{lbf - \sec^2}$

$\dot{V} \propto N$

$H_p \propto N^2$

$EHP\ a\ N^3$

$H_L = K \frac{\dot{v}^2}{2}$

$H_L = f \frac{LV^2}{2D}$

$1Mw = 3.41x10^6\, Btu/hr$

$1hp = 2.54x10^3\, Btu/hr$

$1Btu = 778\, ft\ lbf$

$°C = (5/9)(°F - 32)$

$°F = (9/5)(°C) + 32$

$1kg = 2.21\ Ibm$

$1 ft^3 = 7.48\ gal$

SURRY LORP EQUATION SHEETS

$$\Delta E = 931\,\Delta m$$

$$\frac{1}{M} = \frac{CR_1}{CR_x}$$

$$P = P_o e^{\left(\frac{t}{\tau}\right)}$$

$$P = P_o 10^{sur(t)}$$

$$SUR = \frac{26.06}{\tau}$$

$$SUR = \frac{26\rho}{l^* + (\beta - \rho)T}$$

$$SUR = \frac{26.06(\lambda_{eff}\rho)}{(\overline{\beta} - \rho)}$$

$$\tau = \frac{\overline{\beta} - \rho}{\lambda_{eff}\rho}$$

$$\tau = \frac{l^*}{\rho} + \left[\frac{(\overline{\beta} - \rho)}{\lambda_{eff}\rho}\right]$$

$$\lambda_{eff} = 0.1\,\sec^{-1}$$

$$\ell^* = 2x10^{-5}\ \mathbf{sec}$$

$$\tau = \frac{l^*}{(\rho - \overline{\beta})}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\overline{\beta}}{1 + \lambda_{eff}\tau}$$

$$\rho = \frac{(K_{eff} - 1)}{K_{eff}}$$

$$K_{eff} = \frac{1}{(1 - \rho)}$$

$$CR_{S/D} = \frac{S}{(1 - K_{eff})}$$

$$CR_1(1 - K_{eff}1) = CR_2(1 - K_{eff}2)$$

$$DRW \propto \frac{\phi^2{}_{tip}}{\phi^2{}_{avg}}$$

$$SDM = \frac{(1 - {}^{*}\quad)}{K_{eff}}$$

$$A = A_0 e^{-\lambda t}$$

$$\lambda = \frac{\ln 2}{T_{1/2}}$$

$$E = mc^2$$

$$\frac{R}{hr} = \frac{6CE}{d^2(feet)}$$

$$\frac{R}{hr} = \frac{(0.5CE)}{d^2(meters)}$$

$I_1 d_1 = I_2 d_2$ – *Line* **source**

$I_1 d_1^{\,2} = I_2 d_2^{\,2}$ – Point source

$$1Curie = 3.7\,x10^{10}\ dps$$

$$2000\,DAC - hrs = 1\,ALI = 5.0\ Rem$$