# Final Submittal
(Blue Paper)

# FINAL SRO

# WRITTEN EXAMINATION

**NORTH ANNA MAY/JUNE 2006-301 EXAM**

**05000338/2006301 & 05000339/2006301**

**WEEKS OF MAY 22 & JUNE 5, 2006 (OP TEST)**

**JUNE 14, 2006 (WRITTEN)**

2/9/06

## Site-Specific SRO Written Examination
## Cover Sheet

**U.S. Nuclear Regulatory Commission**

**Site-Specific SRO Written Examination**

**Applicant Information**

Name:

| | |
|---|---|
| Date: | Facility/Unit: North Anna Units 1 & 2 |
| Region: I☐ II **X** III ☐ IV ☐ | Reactor Type: W **X** CE☐ BW☐ GE☐ |
| 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 overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

**Applicant Certification**

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

______________________________
Applicant's Signature

**Results**

RO/SRO-Only/Total Examination Values _____ / _____ / _____ Points

Applicant's Scores _____ / _____ / _____ Points

Applicant's Grade _____ / _____ / _____ Percent

Name: ______________________________

2006 NAPS Initial License Operator Exam
SRO Exam

1.
The purpose of the Control Rod Drive System is to ____________________.

A. ensure adequate shutdown margin is maintained at all times

B. position control rods by changing direction of rotation of the CRDMs

C.✓ control reactor subcriticality in conjunction with the CVCS

D. accomodate 40% of the design basis load rejection

DISTRACTOR ANALYSIS:

A INCORRECT Shutdown margin is ensured by controlling RCS boron concentration AND positioning control rods, not by positioning control rods alone. Plausible because control rods can affect shutdown margin.

B INCORRECT Control rods are positioned by operation of movable grippers, not by changing rotation of the CRDMs. Plausible because some reactors use a lead screw type arrangement.

C CORRECT IAW Reference 1, "Together with the CVCS, the control rods are positioned to control reactor subcriticality."

D INCORRECT Control rods only accomodate 10% of the design basis load rejection. Plausible because steam dumps accomodate 40% of the DBLR.

REFERENCES:
1. NCRODP-65-NA, "Rod Control System," Page 1 & 2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Control Rod Drive System; Knowledge of system purpose and or function.

K/A MATCH:
- Question was written to the "purpose" part of this K/A.

2.
The following conditions exist prior to starting the 1-RC-P-1A Reactor Coolant Pump (RCP).

1 The associated cold leg and hot let loop stop valves are fully open

2 Panel C-F1, RCP 1A OIL RES HI-LO LEVEL annunciator is LIT

3 The handswitch for the 1-RC-P-1A1, A RCP BEARING LIFT PUMP was placed in START 1 minute and 45 seconds ago and the red indicating light above the switch is lit

4 The white light above the 1-RC-P-1A1 control switch is not LIT

Using the above, which ONE of the following would prevent the 1-RC-P-1A RCP from starting when its handswitch is placed in the START position?

A. 1 ONLY

B. 1 AND 2

C. 3 AND 4

D.✓ 4 ONLY

DISTRACTOR ANALYSIS:

A INCORRECT — This condition satisfies one of the 4 starting interlocks.

B INCORRECT — Neither of these would prevent the RCP from automatically starting. If the RCP 1A OIL RES HI-LO LEVEL annunciator was LIT, then, procedurely, the operator could start the pump after obtaining permission from the Superintendent of Operations or the Operations Manager On Call.

C INCORRECT — This distractor is plausible in that the operator might confuse this time with the 2 minutes required by step 5.1.17 of 1-OP-5.2 to run the Bearing Lift Pump prior to starting the associated RCP. However, there is no Bearing Lift Pump run time from which the RCP will then automatically start.

D CORRECT — Once Bearing Lift Pump discharge pressure has increased to greater than 700 psig, this light will be LIT and is one of the 4 RCP starting interlocks which must be satisfied in order for the RCP to automatically start when its handswitch is positioned to START.

REFERENCES:

1. NCRODP-38-NA, "Reactor Coolant System," pages 18,19,20,43,45,46,65,A-3, fig 38-26, and fig 38-12.
2. 1-OP-5.2, "Reactor Coolant Pump Startup and Shutdown," pages 19,21,24, and 25.

K/A CATALOGUE QUESTION DESCRIPTION:

- Reactor Coolant Pump System (RCPS); Ability to monitor automatic operation of the RCPS, including: RCP lube oil and bearing lift pumps.

K/A MATCH:

- Question was written to test the applicant's ability to monitor plant indications associated with the RCPS to determine if the a RCP would automatically start if an attempt to start the pump was made and based on his/her observations, determine if the plant responded appropriately.

3.
The following plant conditions exist:

- Unit 2 is operating at 100% power.
- Component Cooling Water Pump 2-CC-P-1B is undergoing emergent repairs and will not be available for an estimated 2 hours.
- Component Cooling Water Pump 2-CC-P-1A is in operation.
- Power is lost to the 2J 4160V bus.
- The 2J Emergency Diesel Generator starts automatically but fails to load.

Which ONE of the following statements is correct concerning subsequent operation of the RCPs?

A. The RCPs must be tripped or the motor windings will be damaged.

B.✓ The RCPs can be run indefinitely provided seal injection flow is maintained.

C. Component Cooling Water Pump 2-CC-P-1A will trip and lockout.

D. Component Cooling Water Pump 2-CC-P-1A will trip, then automaticlly re-start 15 seconds after power is restored to the 2J bus.

DISTRACTOR ANALYSIS:

A INCORRECT This would be correct if Component Cooling Water Pump 2-CC-P-1B had been the running pump with the Component Cooling Water Pump 2-CC-P-1A initially out of commission.

B CORRECT The Component Cooling Water Pump 2-CC-P-1A is powered from the 2H Stub bus so its operation will be unaffected by the loss of the 2J bus..

C INCORRECT This answer would have been correct if the Component Cooling Water Pump 2-CC-P-1B had been the running pump.

D INCORRECT If the loss of power had occurred to the 2H bus and the pump had tripped due to an electrical fault (phase or ground overcurrent) this answer would be correct.

REFERENCES:
1. NCRODY-51-NA, "Component Cooling System," pages 24,26,27,30,31,32, Table 51-3, Table 51-4.
2. NCRODP-35-NA, "Vital and Emergency Electrical Distribution System," page 2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Coolant Pump System (RCPS); Knowledge of bus power supplies to the following: CCW pumps.

K/A MATCH:
- The K/A implies that the applicant have a understanding of the bus power supplies to the CCW pumps and the interrelationship between the CCW pumps and the RCPS. With that implication understood, although the running pump is not affected by the loss of the bus in this situation, the applicant must have that knowledge to arrive at the correct answer.

4.
The following plant conditions exist:

- Unit 1 is operating at 100% power.
- Unit 1 VCT pressure is being increased.

Given the above information, which ONE of the following is correct?

If __________, then VCT hydrogen could exceed a maximum of __________.

A.✓ PVC-1118, Hydrogen Pressure Control Valve failed open; 4.0% concentration

B. PVC-1118, Hydrogen Pressure Control Valve failed open; 0.1% by volume

C. PVC-1119, Nitrogen Pressure Control Valve failed open; 4.0% concentration

D. PVC-1119, Nitrogen Pressure Control Valve failed open; 0.1% by volume

DISTRACTOR ANALYSIS:

A CORRECT — At power, PVC-1118 is in service and set to maintain a pressure of 16 psig in the VCT. The explosive limit for hydrogen is 4% concentration or 1% by volume.

B INCORRECT — Hydrogen should be at least 1% by volume.

C INCORRECT — During power operation, this valve is isolated. If it failed shut under this condition, it would have no impact at all. This distractor is plausible in that it would be correct if the plant were shut down and the applicant might confuse the method in which each is operated. The 4% concentration is correct. A possible alternate distractor that might be used in place of this one: SOV-1258, VCT Vent Isolation, fails open. This would vent the nitrogen blanket while PVC-1118 attempted to maintain 16 psig hydrogen pressure.

D INCORRECT — The nitrogen pressure control valve (PCV-1119) is isolated during normal plant operation, but is supplied to the VCT from the LP nitrogen header via PCV-1119 during shutdown conditions to degassify the VCT. It is set to maintain VCT pressure at 16 psig when in service. Since the plant is operating at 100% power, this answer is incorrect. This answer is also incorrect in that hydrogen must be at least 1% by volume.

REFERENCES:
1. NCRODP-41-NA, Chemical and Volume Control System, pages 20, 62, and fig 41-5-NA.
2. NCRODP-20-NA, Compressed Gas System, pages 5, 6, 11, 25, A-4, fig 20-1-NA.

K/A CATALOGUE QUESTION DESCRIPTION:
- Chemical and Volume Control System; Knowledge of the operational implications of the following concepts as they apply to the CVCS: Explosion hazard associated with hydrogen containing systems.

K/A MATCH:
- This question addresses the "hydrogen containing systems" in that the VCT contains hydrogen gas. The question then presents a list of operational failures that, given the correct plant conditions, create implications related to hydrogen reaching explosive limits.

5.
Unit 1 is on RHR in Mode 5.

- RCS level is 14 inches above loop centerline.
- 1-RH-FC-1605, RHR System Flow Controller failed, causing FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW VALVE to fully open.
- All attempts at closing the valve have failed.
- RHR pump amps and flow are fluctuating
- RHR flow is greater than designed flow rate

Given these conditions, which ONE of the following describes the impact on RCS temperature and the actions the crew must take?

RCS temperature will __________. The crew should __________ Heat Exchanger Flow Control Valve, HCV-1758, to stabilize RHR pump conditions.

A. increase open

B. decrease close

C.✓ increase close

D. decrease open

DISTRACTOR ANALYSIS:

A INCORRECT Operators should close the valve.

B INCORRECT Wrong direction on both.

C CORRECT With FCV-1605 in the full open position, more RHR flow bypasses the RHR heat exchanger without being cooled which causes RCS temperature to increase. Step 7 of 1-AP-11, Loss of RHR, directs operators to reduce flow (using FCV-1605 or HCV-1758) if RHR flow is greater than the design flow (which it will be).

D INCORRECT Temperature will increase.

REFERENCES:
1. NCRODP-40-NA, Residual Heat Removal System, pages 2, 9, 11, 17, 18, 25, fig 40-1-NA.
2. 1-AP-11, Loss of RHR, page 1, step 7 on page 7.

K/A CATALOGUE QUESTION DESCRIPTION:
- Residual Heat Removal System (RHRS); Ability to (a) predict the impacts of the folowing malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

K/A MATCH:
- This question addressess the K/A in that in order for the applicant to determine the correct action to mitigate the malfunction, the impact on both the RHR system must be determined/predicted.

6.

The Crew for Unit 1 just commenced the 5 minute warmup period associated with placing the RHR system in service as part of a plant cooldown. Plant conditions are as follows:

- RCS pressure is 400 psig
- RCS Temperature is 320 degrees
- RHR HEAT EXCHANGER 1A is on service
- RHR Pump 1A is running
- FCV-1605, RHR HEAT EXCHANGER BYPASS FLOW VALVE, was just placed in AUTO after completing the 5 minute warmup

About 2 minutes later the crew receives the following alarm:

- RHR HX CC OUTLET HI TEMP / HI FLOW:

The crew observes the following:

- RHR Heat Exchanger CC Outlet Temperature is 200 $^{o}$F
- RHR Heat Exchanger CC Outlet Flow is approximately 1000 gpm

Which ONE of the following describes the actions the crew should take to clear the alarming condition?

A.✓ Throttle open MOV-100A, RHR HEAT EXCHANGER CC RETURN VALVE, by pressing its open pushbutton until the desired valve position is reached, and then releasing the button.

B. Throttle open MOV-100A, RHR HEAT EXCHANGER CC RETURN VALVE, by pressing its open pushbutton until the desired valve position is reached and then simultaneously pressing both its open and close pushbuttons, and then releasing the buttons.

C. Throttle shut MOV-100B, RHR HEAT EXCHANGER CC RETURN VALVE, by pressing its shut pushbutton until the desired valve position is reached, and then releasing the button.

D. Throttle shut MOV-100B, RHR HEAT EXCHANGER CC RETURN VALVE, by pressing its shut pushbutton until the desired valve position is reached and then simultaneously pressing both its open and close pushbuttons, and then releasing the buttons.

DISTRACTOR ANALYSIS:

A CORRECT — MOV-100A/B are unique in the fact that both the open and close pushbuttons do not have to be pressed simultaneously to throttle the valves. Releasing the pushbutton will stop valve travel.

B INCORRECT — MOV-100A/B are unique in the fact that both the open and close pushbuttons do not have to be pressed simultaneously to throttle the valves. This distractor is plausible in that a number of valves operated from the control room require both the open and close pushbuttons be pressed simulatneously to throttle the valves, a nuance that is specifically addressed in the system description describing the operation of these valves.

C INCORRECT — Although this action might act to clear the alarm, it could also adversely impact the operation of the "B" heat exchanger and would not be appropiate or procedurely correct.

D INCORRECT — See the rational given for distractors B and C.

REFERENCES:
1. OP-14.1, "Residual Heat Removal," Section 5.1.
2. NCRODU-40-NA, "Residual Heat Removal System," pages 19, 24, 25, fig 40-1-NA.
3. NCRODP-51-NA, "Component Cooling System," pages 25, 26, fig 51-7-NA.
4. Self Study Guide for Residual Heat Removal System, pages 12, 13, 14, 17, 18, 24.
5. 1-AR-E-A5, RHR HX CC OUTLET HI TEMP / HI FLOW.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Core Cooling System (ECCS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: CCW flow (establish flow to RHR heat exchanger prior to placing in service).

K/A MATCH:
- The question matches the K/A in that it relates to plant conditons shortly after Component Cooling Water (CCW) flow would have been established. By selecting the correct answer the applicant, through inferrence, must have predicted how the plant would respond. It would also demonstrate understanding of the associated controls which operate differently from other valve open/close switches.

7.

A large-break LOCA occurred several hours ago. The Emergency Core Cooling System (ECCS) is to be placed in the cold-leg recirculation mode.

Which ONE of the following describes operator responsibilities with respect to the operation of the ECCS in this mode?

A. Monitoring indications to verify reflux boiling is occuring in the steam generators.

B. Monitoring heat transfer between the Reactor Coolant System and the steam generators during natural circulation flow.

C.✔ Monitoring water injection from the recirculation sump and removal of steam/water from the break.

D. Monitoring heat transfer between the Reactor Coolant System and the steam generators during forced circulation flow.

DISTRACTOR ANALYSIS:

A INCORRECT This is not in accordance with 1-ES-1.3 but is a plausible distractor as it represents one of the alternative modes of ECCS operation.

B INCORRECT This is not in accordance with 1-ES-1.3 but is a plausible distractor as it represents one of the alternative modes of ECCS operation.

C CORRECT This is the method of core decay-heat removal when the plant is aligned for cold-leg recirculation..

D INCORRECT This is not in accordance with 1-ES-1.3 but is a plausible distractor as it represents one of the alternative modes of ECCS operation.

REFERENCES:
1. 1-ES-1.3, "Transfer To Cold Leg Recirculation," step 8.
2. North Anna Emergency Procedures Exam Bank, question 209, page 223, ID 2781.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Core Cooling System (ECCS); Knowledge of operator responsibilities during all modes of plant operation.

K/A MATCH:
- This question matches the K/A in that the operator must have an understanding of all the modes of ECCS operation in order to answer the question correctly.

8.

Unit 1 is at 8% preparing to connect the generator to the grid when a "Pressurizer High Level Reactor Trip" red first-out annunciator alarms and locks in. Operators observe the following:

- Turbine Load is 7% and increasing
- S/G levels are stable
- Pressurizer Level indicates 95% and increasing
- Charging flow is 70 gpm
- Letdown flow is 85 gpm
- Seal Leakoff flow is 1.0 gpm each
- Seal Injection flow is 8 gpm each

Based on the above, which ONE of the following describes how the plant should respond and the actions the crew should take?

The reactor ______________. The crew should ______________.

A. should have tripped;
trip the reactor and take actions for "Reactor Trip or Safety Injection" IAW 1-E-0.

B. should have tripped;
trip the reactor and take actions for "Response to ATWS" IAW 1-FR.S.1.

C. should not have tripped;
take actions for a "Loss of Vital Instrumentation" IAW 1-AP-3.

D.✓ should not have tripped;
take actions in accordance with 1-AR-B-F6, PRZ HI LEVEL CH I-II-III.

DISTRACTOR ANALYSIS:

A INCORRECT — Based on the indications given, the reactor should not have tripped. This distractor is plausible in that if power had been above the P-7 setpoint (10%) then this response would have occurred.

B INCORRECT — The reactor should not have tripped. This distractor is plausible in that the applicant might fail to account for power being below the P-7 setpoint thereby thinking the reactor should have tripped.

C INCORRECT — The first part is correct. This distractor is plausible in that the applicant might ASSUME that the high level is due to an instrument error since no indications are given that the reactor tripped. He/she would thereby mistake the associated action as being correct, which it would be for such an assumption. However, based on NRC guidelines, no assumptions should be made beyond the information given.

D CORRECT — Based on the information given, a high pressurizer level exists. This would be the correct response for the crew to take in this situation.

REFERENCES:
1. 1-AR-B-F6, Annunciator response for PRZ HI LEVEL.
2. NCRODP-77-NA, "Reactor Protection Systems," pages T-12, T-15 thru T-17.
3. 1-ES-0.1, "Reactor Trip Response."
4. 1-F-0, Atttachment 6.
5. 1-FR-I.1, "Response To High Pressurizer Level."

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Trip; Knowledge of the interrelations between a reactor trip and the following: Reactor trip status panel.

K/A MATCH:
- Although a reactor trip did not occur based on information given in the question stem, the applicant must understand the interrelations between the trip status panel (of which "first-out" alarms indicate) and an actual reactor trip.

9.
The PRT level column vent valve was inadvertently left uncapped during recovery from the last refueling outage. Current plant conditions are as follows:

- Unit at 100% power
- The column vent valve has since developed leak-by of approximately 0.1 scfm.
- Containment vacuum pumps became inoperable minutes ago.
- PRT pressure is currently within limits.

Based on the conditions given, which ONE of the following is correct.

A.✓ Containment partial pressure will increase more rapidly.

B. Containment dewpoint indication will increase.

C. PRT pressure indication will be lower than actual pressure.

D. PRT water level indication will be higher than actual water level.

DISTRACTOR ANALYSIS:

A CORRECT — WIth current PRT nitrogen leakage of 0.1 scfm, a net increase in containment partial pressure will result given that no vacuum pumps are in service.

B INCORRECT — Not adding any moisture to air. Leakage is non-compressible gas only. Additionally the containment dewpoint instruments are not operational. These instruments are only used for the Type A testing.

C INCORRECT — Loss of pressure from vent valve wil be the same as that experienced by the tank.

D INCORRECT — There is no effect on level indication sice the tank pressure will effect the high and low sides of the level transmitter equally.

REFERENCES:
1. North Anna Reactor Coolant System Exam Bank question 197, page 209

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurizer Relief Tank/Quench Tank System (PRTS); Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment.

K/A MATCH:
- Although the malfunction does not directly impact containment (i.e. the containment structure), this question matches the K/A in that it affects the containment environment (i.e. parameters that describe its environment) and by inference, have a direct affect on containment.

10.
Unit 2 was operating at 100% power with the following conditions:

- Component Cooling (CC) systems are in a normal lineup.
- CC backpressure regulating valve, 2-CC-PCV-210, is regulating CC system pressure.

An operator in the plant inadvertently isolates instrument air to CC backpressure regulating valve, 2-CC-PCV-210.

Based on the above, which ONE of the following describes the effects with no further operator involvement?

A. Thermal Barrier Trip Valves, TV-116A/B/C, will shut within 10 seconds.

B. The Unit 2 standby CC pump will automatically start.

C.✔ CC system pressure will increase.

D. The Unit 1 standby CC pump will automatically start.

DISTRACTOR ANALYSIS:

A INCORRECT Although it might be possible, with the increase in system pressure that will occur as a result of this event, for flow through RCP thermal barrier heat exchangers to increase to the trip setpoint of 59 gpm, it must remain so for GREATER THAN 10 seconds. This distractor is plausible if the applicant confuses the limit of ">" with "<".

B INCORRECT Although the standby CC pump has auto-start features, system low pressure is not one of them. This distractor is plausible in that many systems the standby pump often starts on system low pressure.

C CORRECT Since CC is in a normal lineup, Units 1 & 2 CC systems will be cross-connected. In this configuration, the two backpressure regulating valves 1-CC-PCV-110 and 2-CC-PCV-210 are set such that one will regulate 5-10 psig higher than the other. The result is that the regulator with the the lower setting (in this case 2-CC-PCV-210) will be the on-service regulator. The valves fail closed on a loss of air signal which results in CC system pressure increasing to the point where the regulator with the higher set pressure begins to maintain pressure.

D INCORRECT Although the standby CC pump has auto-start features, system low pressure is not one of them. This distractor is plausible in that many systems the standby pump often starts on system low pressure.

REFERENCES:
1. 1-OP-51.1, "Component Cooling System," pages 4, 22, and 24.
2. NCRODP-51-NA, "Component Cooling System," pages 2, 3, 10, 20, 24, 27, 28, 30, 31, figures 2, 3, and 8.
3. North Anna Component Cooling Water & Nuetron Shield Cooling Systems Exam Bank question 15, ID 3872.

K/A CATALOGUE QUESTION DESCRIPTION:
- Component Cooling Water System (CCWS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW pressure.

K/A MATCH:
- This question matches the K/A in that it requires the applicant to predict how the CC system will respond to the operation of the CC system controls (although inadvertently) to one of the CC system components causing a change to CC system pressure. It also addresses exceeding design limits in that the applicant must be able to recall the design flow limit associated with CC flow through RCP thermal barrier heat exchangers.

11.
Assume the following plant conditions:

- The operating crew is performing 1-E-3, "Steam Generator Tube Rupture," in response to a tube rupture in the "B" Steam Generator.

- After terminating safety injection, a small-break LOCA occurs.

- Pressurizer level drops to 8% with charging flow at maximum.

The operating crew should _______________.

A. Manually initiate Safety Injection and go to 1-ES-0.0, "Re-diagnosis."

B.✓ Start Charging Pumps and align the boron injection tank as required and then go to 1-ECA-3.1, "SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired."

C. Start Charging Pumps and align the boron injection tank as required and then continue in 1-E-3, "Steam Generator Tube Rupture."

D. Manually initiate Safety Injection and return to 1-E-0, "Reactor Trip or Safety Injection."

<u>DISTRACTOR ANALYSIS</u>:

A INCORRECT The crew should not exit the procedure they are in based on the conditions given. However, this distractor is plausible considering that the crew is responding to one casualty when another casualty occurs.

B CORRECT Given the stated conditions, specifically, PZR level unable to be maintained > 21%, the foldout page of 1-E-3 directs operators to take this action.

C INCORRECT Although the actions stated to be taken are correct, the foldout page then directs an immeciate transition to 1-ECA-3.1.

D INCORRECT The procedure does not direct the performance of this action based on the conditions given. However, this distractor is plausible considering that the crew is responding to one casualty when another casualty occurs.

<u>REFERENCES</u>:
1. North Anna Emergency Procedures Exam Bank, question 302, page 324, ID 2901.
2. 1-E-3, "Steam Generator Tube Rupture," page 22, step 23, and foldout page.

<u>K/A CATALOGUE QUESTION DESCRIPTION</u>:
- Small Break LOCA; Knowledge of operator responsibilities during all modes of plant operation.

<u>K/A MATCH</u>:
- While it may be possible to develop a scenario wherein a small-break LOCA spans all mode of plant operation, it is unlikely that operators would realisticly ever encounter such a scenario. In lieu of substituting another K/A, a scenario was utilized encompasing at least one mode that required operators to derrive answers subsequent to possessing an understanding of their responsibilities.

12.
Which ONE of the following describes either the design or method used to limit thermal stresses on the pressurizer spray valve?

A.✓ The installation of spray bypass valves.

B. The installation of thermal sleeves.

C. The size of the spray line.

D. The installation of an auxiliary spray line.

DISTRACTOR ANALYSIS:

A CORRECT These valves allow a minimum 1 gpm total spray flow into the pressurizer during all operations with the RCPs operations to prevent the spray and surge lines from cooling off due to ambient heat losses.

B INCORRECT The thermal sleeves, while intended to absorb the thermal stresses associated with the temperature gradients, were designed to provide stress protection to the piping welds which serve as the pressure boundary.

C INCORRECT The size of the spray line was designed to allow sufficient spray flow to prevent the PORVs from opening during a 10% step-decrease in power.

D INCORRECT The auxiliary spray line was designed to provide the capability of initiating spray from the CVCS charging pumps when RCPs are not operating.

REFERENCES:
1. NCRODP-38-NA, "Reactor Coolant System," pages 25, 33, 52, figure 38-19.
2. 1-AR-B-G3, PRZ SPRAY LINE LO TEMP.
3. North Anna Reactor Coolant System Exam Bank, question 99, page 109, ID 5698.

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurizer Pressure Control System (PZR PCS); Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Spray valve warm-up.

K/A MATCH:
- None of the plant procedures/documentation speak directly to a warm-up of the spray valve but do speak to design features utilized to reduce/minimize thermal stresses to which the spray line (and hence the spray valve) might be subjected. In this regard, the question addressed the K/A.

13. 012K6.04 001/2/1/REACTOR PROTECTION/C/A/BANK/NA06/RO/MC

Unit 2 is performing a reactor shutdown. Plant conditions are as follows:

- During the shutdown the turbine first stage pressure channel, MS-PT-2446, sticks at 35% power.
- Steam Flows, Feed Flows, and First Stage Pressure are selected to Channel III.
- Reactor power is currently 14%.

Based on the plant conditions given above, which ONE of the following will occur?

A.✓ If 2 reactor coolant pumps are lost a reactor trip will occur.

B. If the generator output breaker is opened a reactor trip will occur.

C. Source range nuclear instruments will have to be manually unblocked.

D. Power range channel low setpoint high neutron flux level will not unblock.

DISTRACTOR ANALYSIS:

A CORRECT — P-8 (power < 30% generated from NIs) is satisfied and will block the 1.3 loss of flow sing loop reactor trip. However P-7 is not satisfied from any input (3.4 PR <10% AND 2.2 first-stage pressures <10%), so the 2.3 loss of flow trip is not blocked.

B INCORRECT — Although the turbine will trip if the generator output breaker is opened since C-5 is not satisfied because of the failed first-stage pressure channel, the reactor will not trip on the turbine trip because P-8 is satisfied.

C INCORRECT — Source range block/unblock input comes from PR, not first-stage pressure. This distractor is plausible in that the applicant may confuse the source from which the blocking signal is generated.

D INCORRECT — See C.

REFERENCES:
1. NCRODP-77NA, "Reactor Protection Systems," pages 31, 32, 42, 68, T-4, T-10, and figure 77-9.
2. NCRODP-24-NA,"Main Turbine System," pages 13, 14, 21, and 31.
3. INPO Exam Bank

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Protection System; Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Bypass-block circuits.

K/A MATCH:
- This question matches the K/A in that it speaks directly to a bypass-block circuit, specifically C-5. The failure occurs at its source of generation, in this case the actual pressure detector that provides an input signal to the bypass-block curcuit (C-5). The applicant must have knowledge of that circuit and how its failure will affect the response of the reactor protection system once additional stimulus (signals) are inserted into the reactor protection system such that a response is elicited.

14.

Unit-1 is stable in Mode 5 with the following conditions:

- 1-RH-P-1A is running, returning flow to the "C" loop
- RCS cold leg temperature is 140ºF
- RCS pressure is 150 psig
- All RCS loops are isolated
- RCS is solid
- "A" charging pump is running
- Instrument Department is conducting solid state protection testing
- A failure during testing causes a single train safety injection to occur. All "B" train equipment actuates.

Assuming no operator action, RCS pressure will rapidly increase to the __________ and stabilize there for the next several hours.

A.✓ PRZR PORV setpoint for approximately 10 minutes, then increase to the RHR suction relief valve setpoint

B. RHR suction relief valve setpoint for approximately 10 minutes, then increase to the PRZR PORV setpoint

C. RHR suction relief valve setpoint

D. PRZR PORV setpoint

DISTRACTOR ANALYSIS:

A. CORRECT PORVs are on the LTOPs setpoints (365 psig and 370 psig). The PORVs would be using nitrogen as the motive force in this mode. These bottles would only be good for approximately 10 minutes. Pressure would then increase to the RHR RV setpoint (467 psig).

B. INCORRECT Backwards as the RHR RV setpoint is > PORV setpoints at this temperature.

C. INCORRECT This would be correct if the PORVs were not on LTOPs setpoints.

D. INCORRECT On LTOPs the PORVs use nitrogen, no air to back it up, so the tanks would deplete in approximately 10 minutes.

REFERENCES:
1. North Anna Integrated Plant Operations Exam Bank, question 93, page 93, ID 4093.
2.
3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Engineered Safety Features Actuation System (ESFAS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ESFAS controls including: RCS pressure and temperature.

K/A MATCH:
- Question matches the K/A since the operator must be able to predict the effect that the event will have on RCS pressure and temperature in order to answer correctly.

15.
Given the following plant conditions:

- Two minutes ago, an MSTV inadvertently closed causing secondary safeties to lift and a reactor trip and safety injection due to high steam flow coincident with low steamline pressure.
- The reactor trip breakers failed to open.
- Operators tripped the reactor locally from 307 switchgear by tripping both M-G Set Motor Supply Breakers per FR-S.1, Attachment 4, "Response to Nuclear Power Generation/ATWS, Remote Reactor Trip."
- It is now desired to reset SI and secure SI equipment.
- RCS pressure is 1950 psig.

Which ONE of the following represents why an automatic SI will not be blocked following a reset from the Main Control Board under the above conditions?

A. RCS pressure is less than the SI setpoint.

B.✓ Permissive P-4 has not actuated.

C. Permissive P-11 has actuated.

D. The SI timing relays.

DISTRACTOR ANALYSIS:

A INCORRECT P-11 (2000 psig) is the set point above which a blocked SI signal will auto unblock.

B CORRECT The block remains in effect until the reactor trip breakers are reset. Resetting the reactor trip breakers automatically defeats the blocking circuit and arms the SI automatic initiating circuits.

C INCORRECT Pressurizer pressure below the SI setpoint will initiate an SI signal but will not prevent resetting SI.

D INCORRECT SI will not reset if the 60 second timer is active. However, the timer timed out 60 seconds ago.

REFERENCES:
1. NCRODP-77-NA, "Reactor Protection Systems," pages 24, 25, 46, A-3, 50, 51, T-12, T-13, figures 77-4 & 77-12.
2. FR-S.1, Attachment 4, "Response to Nuclear Power Generation/ATWS, Remote Reactor Trip," Steps 1 & 3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Engineered Safety Features Actuation System (ESFAS); Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels.

K/A MATCH:
- This question matches the K/A in that the operator must be albe to determine from plant indications (monitor) whether or not ESFAS channels will be able to be reset (ability).

16.
Which ONE of the following indications would be indicative of only a RCP Number 2 seal failure?

A.✓ RCP High Standpipe level.

B. RCP Seal Leakoff High Flow.

C. RCP Thermal Barrier Low Differential Pressure.

D. RCP Thermal Barrier Cooling Water High temperature.

DISTRACTOR ANALYSIS:

A CORRECT A High RCP Standpipe Level is indicative of a #2 RCP Seal failure.

B INCORRECT This would be due to a #1 RCP seal failure.

C INCORRECT This would be due to a #1 RCP seal failure.

D INCORRECT This would be due to a #1 RCP seal failure.

REFERENCES:
1. North Anna Reactor Coolant System Exam Bank, question 68, page 75, ID 5426
2. NCRODP-38-NA, "Reactor Coolant System," pages 17, 42, figures 38-4, 5, 7, 8, 9.
3. 1-AR-C-H3, RCP 1C STANDPIPE LO LEVEL.
4. 1-AR-C-G3, RCP 1C STANDPIPE HI LEVEL.
5. 1-AR-C-H4, RPC 1A-B-C BEARING HI TEMP.

K/A CATALOGUE QUESTION DESCRIPTION:
- Reactor Coolant Pump (RCP) Malfunctions; Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP seal failure/malfunction.

K/A MATCH:
- This question matches the K/A in that it addressed the "or monitor" portion of the RCP seal failure.

17. 015K5.14 001/2/2/NIS/C/A/MODIFIED/NA06/RO/MC

A reduction in feed to the SGs will cause __________ neutron attenuation and __________ indicated reactor power.

(Attenuation - slowed to thermal energy)

(Assume actual reactor power (turbine load) remains constant.)

A. increased decreased

B. decreased decreased

C. increased increased

D.✔ decreased increased

DISTRACTOR ANALYSIS:

A INCORRECT See distractor D analysis.

B INCORRECT See distractor D analysis.

C INCORRECT See distractor D analysis.

D CORRECT Assuming turbine load is not changed, as feedwater to the SGs is reduced, less heat is being removed from the coolant passing through the SG u-tubes. As a result, Tcold increases. This warmer water enters the reactor and fuel area. Since this water is warmer, its density is lower and results in a reduction of the number of neutrons being slowed to thermal energy values. Now more neutrons escape the area of the core to be seen by nuclear instrument detectors. The result is an increase in the number of neutrons being detected by the NIs..

REFERENCES:
1. North Anna Ex-cre Nuclear Instrumentation System Exam Bank, question 75, page 75, ID 2186.
2. SOER-90-3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Nuclear Instrumentation System (NIS); Knowledge of the operational implications of the following concepts as they apply to the NIS: Neutron flux density, definition and relation to reactor power.

K/A MATCH:
- This question matches the K/A in that as the result of an operational change, the operator must determine the implication with respect to reactor power as indicated by the NIS. Although not specifically addressed in the question, the operator must know and understand the definition of flux density in order to answer correctly.

18.
In accordance with 0-AP-48, "Charging Pump Cross-Connect," which ONE of the following statements is correct?

Operators should __________

A. **only** trip the affected unit, enter E-0, and send an operator to the auxiliary building with the appropriate attachment to supply charging to the affected unit.

B.✓ trip both units; if a low head SI pump is supplying the suction of the running charging pump then consider flowing the BIT on the unit to be borated.

C. **only** trip the affected unit, enter E-0, secure letdown, then isolate RCP seal leakoff.

D. trip both units; if a low head SI pump is supplying the suction of the running charging pump then Emergency Borate.

DISTRACTOR ANALYSIS:

| | |
|---|---|
| A INCORRECT | Both units should be tripped. |
| B CORRECT | This is IAW 0-AP-48 step one and the first note on Attachment 4.. |
| C INCORRECT | Both units should be tripped. |
| D INCORRECT | IAW the note on Attachment 4, emergency boration in this configuration will not be effective. |

REFERENCES:
1. 0-AP-48, "Charging Pump Cross-Connect," pages 1 and 2; Attachment 4, page 1.
2. Tech Spec 3.5.2, ECCS-Operating.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Reactor Coolant Makeup; Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Charging Pump problems.

K/A MATCH:
- This question matches the K/A in that the applicant must determine how to respond to the loss of charging (makeup) due to losing all charging pumps specific to one unit. A failure of all charging pumps was chosen due to the impact that loss has on the opposite Unit.

19.
The following conditions exist:

- Both units are operating at 100% power
- All systems are lined up normally
- The backboards operator has inadvertently placed switch 1-SW-TV-101A/B, Service Water Supply and Return to Recirc Air Fans, in the **SW** position.

Assuming all systems have operated as expected, which ONE of the following describes the immediate effects on the Containment Cooling System?

A.✓ Containment temperature and partial pressure will remain the same.

B. Containment temperature will decrease and containment partial pressure will increase.

C. Containment temperature will decrease and containment partial pressure will remain the same.

D. Containment temperature and partial pressure will decrease.

DISTRACTOR ANALYSIS:

A. Correct. Although 1-SW-TV-101A/B will open, the flowpath is isolated by the 110 and 114 MOVs being de-energized.

B. Incorrect. If the examinee does not realize that the MOVs in the flowpath are de-energized this answer could be chosen based on an expected containment temperature response. More cooling flow would cause the partial pressure indicateion to increase due to the postulated decrease in temperature ($P_{TOT}$ - $P_{SAT}$ = Partial pressure).

C. Incorrect. See above for temperature. Pressure response is correct.

D. Incorrect. See above for temperature. If examinee confuses lower temperature with lower pressure, then the second part is plausible.

References:
Objective 7678 for SW.
REFERENCES:
1. NCRODP-47-S, "Primary Ventilation System," pages 20, 21, 33,.
2. 1-F-0, Attachment 5.
3. 1-FR-Z.4, "Response to Containment Positive Pressure," step 3.c, page 3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Containment Cooling System (CCS); Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SEC/remote monitoring systems.

K/A MATCH:
- This question matches the K/A in that it addresses the "or cause-effect relationships . . ." portion of the K/A. Without an understanding of the physical connections between the the pertinent Containment Cooling Systems, the operator could not correctly answer the question. The affected fans draw air through coolers in contrast to fans located within the Containment that exist soley for the purpose of preventing temperature stratification, for example.

20.
Unit 2 is operating at 100% power when the following occur:

- CD TO AIR RECIRC CLRS HI-LO TEMP alarms.
- Containment temperature is noted to have a slow, gradually increasing trend.

Which ONE of the following would account for these conditions?

A. The hot gas bypass valve for the mechanical chiller compressor has failed shut.

B. 1-BC-PCV-135 located in the Bearing Cooling System discharge header to the tower failed shut.

C. 1-SW-TV-101A, Containment Recirc Air Cooler Service Water Supply Valve failed shut.

D.✓ The guide vanes for the mechanical chiller compressor are failed shut.

DISTRACTOR ANALYSIS:

A INCORRECT Should the hot gas bypass valve fail shut, it would have the opposite effect. This distractor is plausible in that if the applicant does not fully understand the operation of the unit, he/she might think that the hot gas ought to be be allowed to bypass the unit under this condition.

B INCORRECT This valve controls backpressure at 20 psig in the discharge header of the Bearing Cooling System which supplies cooling to the air conditioner chillers. If this valve were to experience a fault in the shut direction then backpressure would increase and actually increase cooling flow to the chillers which would in turn result in lower, not higher, containment temperatures. This distractor is plausible in that the valve is located in the discharge header but in parallel path such that it would not restrict cooling flow.

C INCORRECT Although this would result in the conditions given, Chilled Water, not Service Water, is the normal supply to this cooler, a condition also given in the stem of the question. This distractor is plausible in that there are circumstances where the cooler would be supplied by Service Water.

D CORRECT If the guide vanes were being controlled such that they were less open (i.e. control or positioner malfunction) the temperature of the chill water leaving the unit would be higher, thereby resulting in.less cooling to the containment recirc air coolers. Hence, higher containment ambient temperatures.

REFERENCES:
1. 1-AR-G-B2, CD TO AIR RECIRC CLRS HI-LO TEMP.
2. North Anna Chilled Water System Exam Bank, questions 17 on page 17, 19 on page 19.
3. NCRODP-15-NA, "Chilled Water System," pages 29, 34, figure 15-3.
4. NCRODP-47-NA, "Primary Ventilation System," pages 44, 45, figure 47-7.
5. NCRODP-33-NA, "Bearing Cooling System," pages 2, 3, 5, 16, 28, A-2, A-4, figures 33-1 and 33-4.

K/A CATALOGUE QUESTION DESCRIPTION:
- Containment Cooling System (CCS); Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: Chilled water.

K/A MATCH:
- This question matches the K/A in that the applicant must know which systems work to provide cooling (i.e. use of heat-exchangers) to the CCS as well as how the operation of various components in each of the systems affects the final cooling available to the CCS.

21.

Unit 1 has experienced a loss of offsite power coincident with a Containment Depressurization Actuation (CDA).

Which ONE of the following describes the status of the "B" train of the Quench Spray (QS) System?

1-QS-MOV-100B SUCT VLV: ______(1)______

1-QS-MOV-101B DISCH VLV: ______(2)______

1B QS Pump is running: ______(3)______

A. 1 initially shut then opens

   2 initially shut then opens

   3 immediately upon restoration of power to the bus

B. 1 initially open and remains open

   2 initially shut then opens

   3 immediately upon restoration of power to the bus

C. 1 initially shut then opens

   2 initially shut then opens

   3 13.3 seconds after restoration of power to the 1J bus

D.✓ 1 initially open and remains open

   2 initially shut then opens

   3 18 seconds after restoration of power to the 1J bus

DISTRACTOR ANALYSIS:

A INCORRECT — The suction valve is initially open as this is its normal position. The pump starts after at least 15.75 seconds. This distractor is plausible as this is the shutdown configuration of most systems.

B INCORRECT — The pump starts after at least 15.75 seconds. This distractor is plausible since most pumps start upon receiving power.

C INCORRECT — The suction valve is initially open as this is its normal position. The pump starts after at least 15.75 seconds. This distractor is plausible in that the operator might confuse the 13.3 second delay with the 15 second delay.

D CORRECT — On a receipt of a CDA signal, the QS pumps automatically start and the discharge valves open. If a loss of offsite power is experienced coincident to a containment overpressurization, the QS System is actuated with a 29-Second time delay. The undervoltage condition on the QS Pump supply buses causes the breakers to open. After the diesel starts and energizes the supply bus within 13.3 seconds, a timer in the QS pump circuit breaker closes the breaker after an additional $15\pm$ 0.75 seconds..

REFERENCES:
1. NCRODP-53-NA, "Quench Spray System," pages 2, 16, 26, 27, 28, figure 53-1.
2. 1-FR-Z.1, "Response to Hogh Containment Pressure," step 3.
3. North Anna Recirculation Spray System Exam Bank, questions 22 & 26.

K/A CATALOGUE QUESTION DESCRIPTION:
- Containmnet Spray System (CSS); Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning.

K/A MATCH:
- This question matches the K/A in that it specifically asks the operator the conditions of both the spray system valves and pump after an automatic operation.

22.
Assume the following plant conditions:

- Both units are at 100% power
- The Component Cooling Water System is in a normal configuration, except that 1-CC-P-1A is tagged out for bearing replacement
- 1-CC-P-1B and 2-CC-P-1A are running
- 2-CC-P-1B is in Auto
- 1-CC-P-1B has just tripped on overcurrent

Which ONE of the following statements describes the expected plant and crew response to this event in accordance 0-AP-15, Loss of Component Cooling Water?

A.✓ CC flow to RCPs on both Units will decrease; the crew should manually start 2-CC-P-1B and verify flow to the RCPs returns to normal.

B. CC flow to RCPs on both Units will decrease; 2-CC-P-1B will Auto start, the crew should verify flow to the RCPs returns to normal.

C. All CC flow is lost to Unit 1 and Unit 2 RCPs; the crew should trip both Units and enter 1-E-0 and 2-E-0.

D. All CC flow is lost to only the Unit 1 RCPs; the crew should trip the Unit 1 reactor and enter 1-E-0.

DISTRACTOR ANALYSIS:

A. Correct. Since CC is normally cross-tied and there is now only one pump running, the CC flow to both unit's RCPs will decrease. Since the unit 1 "A" CC pump is tagged out no pump will start automatically, but the unit 2 pump can be manually started.

B. Incorrect. There is no auto-start for the opposite unit CC pump when a pump trips.

C. Incorrect. First part is incorrect since flow to the RCPs will only be degraded, not lost. Since a unit 2 CC pump is available it can be manually started to restore adequate flow. Plausible since there are failures that cause a loss of CC to RCPs and eventually require the unit to be tripped.

D. Incorrect. See above.

Bank question 5562D

REFERENCES:

1. NCRODP-13-NA, "Service Water System," pages 5, 6, 10, 11, 12, 34, 35, 46, 47, figures 31-1 & 13-3.
2. NCRODP-77-NA, "Reactor Protection Systems," pages 49, 59, 60, 61, figures 77-12 & 77-14.
3. NCRODP-51-NA, "Component Cooling System," page 31.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Component Cooling Water (CCW); Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW.

K/A MATCH:

- This question matches the K/A in that a lose of CCW will occur by design, automatically, on a CDA signal. The reference to the 2 minutes in the stem was inserted to allow for automatic system response and is not associated with any particular time relay.

23.

Assume that all pressurizer pressure controls are in AUTOMATIC with the plant operating at 100% power when pressurizer pressure transmitter, 1-RC-PT-1444, fails high.

Which ONE of the following describes how the pressurizer PORVs, spray valves, and backup heaters will initially respond?

A.✓ One PORV will open, both spray valves will open, and all backup heaters will de-energize.

B. Both PORVs will open, one spray valve will open, and all backup heaters will remain energized.

C. One PORV will open, one spray valve will open, and all backup heaters will remain energized.

D. Both PORVs will remain closed, both spray valves will open, and all backup heaters will de-energize.

DISTRACTOR ANALYSIS:

A. Correct. PT-1444 feeds the master pressure controller. It will cause PORV 1455C and both spray valves to open, and all heaters to de-energize.

B. Incorrect. Only one PORV and both spray valves are controlled by the master pressure controller. At 100% power there would be one set of backup heaters locked on.

C. Incorrect. See above.

D. Incorrect. See A and B.

Bank question 50379 with slight modifications.

REFERENCES:
1. NCRODP-74-NA, "Pressure Control and Protection System," pages 5, 6, 31, 32, T-8, figure 74-1.
2. 1-AR-B-F2, "PRZ PWR RELIEF VLVS AUTO BLOCK."
3. North Anna Pressure Control and Protection System Exam Bank, question 4, page 4, ID 711.

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurizer Pressure Control System (PZR PCS) Malfunction; Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Operable control channel.

K/A MATCH:
- This question matches the K/A in that the operator must determine the impact of the failure on both affected and unaffected channels be determining the integrated response of the system as a whole.

24.

Containment Purge Supply Fan, 1-HV-F-4A, fails to start when its handswitch is placed in "ON."

Which ONE of the following will prevent the fan from starting?

A. Only Unit 1 Outside Containment Purge Exhaust Isolation, 1-HV-MOV-100D, NOT FULL OPEN.

B. Only Unit 2 Outside Containment Purge Exhaust Isolation, 1-HV-MOV-200D, NOT FULL OPEN.

C.✓ Only Containment Purge Supply Unit air temperature is 30 degrees and reset locally.

D. Only Unit 2 Hi Hi RMS on RM-259 or RM-262.

DISTRACTOR ANALYSIS:

A INCORRECT — Both MOV-HV-100D **AND** MOV-HV-101 (Unit 1 Outside Containment Purge Exhaust Isolation and Bypass) must be fully closed.

B INCORRECT — Both MOV-HV-200D **AND** MOV-HV-201 (Unit 2 Outside Containment Purge Exhaust Isolation and Bypass) must be fully closed.

C CORRECT — Containment Purge Supply Unit air temperature downstream of the heating coils > 35 degrees and temperature switch reset locally.

D INCORRECT — Unit 2 Hi Hi RMS on RM-259 or RM-262 **AND** either MOV-HV-200D (Unit 2 Outside Containment Purge Exhaust Isolation) or MOV-HV-201 (Unit 2 Outside Containment Purge Exhaust Isolation Bypass) not full closed.

REFERENCES:
1. NCRODP-47-NA, "Primary Ventilation System," pages 48 & 49.
2. North Anna Primary Ventilation System Exam Bank, question 38, page 38, ID 2100.

K/A CATALOGUE QUESTION DESCRIPTION:
- Containment Purge System (CPS); Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Containment pressure, temperature, and humidity.

K/A MATCH:
- This question matches the K/A in that it addresses the temperature portion of this K/A and the operator's ability to predict changes in parameters (in this case a pump either running or not) in order to prevent exceeding a design limit (to prevent the heating coils from freezing).

25.
A Large Break LOCA coincident with an ATWS has occurred on Unit 2.

According to the basis documents related to FR-S.1 "Response to Nuclear Power Generation/ATWS," aside from adding negative reactivity to the core "Emergency Boration" is initiated for which ONE of the following reasons?

A. Manual initiation of SI would delay the addition of borated water and complicate recovery.

B. It keeps the core covered without substantially affecting the moderator coefficient.

C. It limits the affect of any overcompensation of intermediate range flux instruments.

D.✓ It is quicker than initiating local actions to trip the reactor.

DISTRACTOR ANALYSIS:

A INCORRECT Although initiation of SI would delay recovery actions as well as affect maintaining a secondary heat sink, it would not delay the addition of borated water.

B INCORRECT The basis document indicates that when CETs approach 1200 degrees, the addition of cooler water may adversely affect criticality due to its affect on the moderator coefficient.

C INCORRECT If intermediate range flux is not decreaing, and undercompensation is ruled out, then emergency boration is initiated to reduce the flux to a point where the source range channels can be manually energized. This distractor is plausible in that an applicant might confuse this information with the compensation that occurs when in the intermediate range using compensating voltage in an intermediate range compensated ion chamber.

D CORRECT After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. Methods of rapid boration include emergency boration.

REFERENCES:
1. Basis document HFRS1 for FR-S.1, pages 70, 79, 98, 100.
2. North Anna Self-Study Guide for Subcriticality Series, page 14.

K/A CATALOGUE QUESTION DESCRIPTION:
- Anticipated Transient Without Scram (ATWS); Knowledge of the reasons for the following responses as they apply to the ATWS: Initiating emergency boration.

K/A MATCH:
- The knowledge of the [primary] reason for initiating emergency boration is considered almost too simplistic to ask. This question matches the K/A in that it addresses secondary reasons for initiating emergency boration that are not so blatently obvious.

26.

Unit 1 was operating at 100% power when the crew entered 1-AP-3, "Loss of Vital Instrumentation," due to the failure of "B" steam generator level channel III. In accordance with 1-AP-3, which ONE of the following describes an immediate action that must be taken?

A.✓ Place the associated MFRV and Bypass Valve in MANUAL.

B. Verify MFRV controlling SG level in AUTO.

C. Place the Control Rod Mode Selector switch in MANUAL.

D. There are no required "immediate" actions.

DISTRACTOR ANALYSIS:

A CORRECT — This is the action called out by step 2 of 1-AP-3. Operation at 100% power was added to the stem to ensure only one correct answer.

B INCORRECT — Since level channel III feeds SGWLCS the MFRV must be controlled manually. Plausible because the other 2 level channels do not feed into SGWLC. If either of these channels fail the MFRV can remain in auto.

C INCORRECT — Although placing rod contol in manual is called out as an immediate action step (if affected), this is not a required action. This distractor is plausible in that it is called out as an immediate action but only if rod control is affected. This distractor also acts to differentiate as to whether the operator is just taking the actions of a related procedure or whether they actually understand which acitons are required to mitigate the failure.

D INCORRECT — Placing both the main feed reg valve and the bypass valves in manual are immediate action steps. This distractor is plausible in that the operator might be aware of the action to be taken but not recall that it is also an immediate action step.

REFERENCES:
1. 1-AP-3, "Loss of Vital Instrumentation,"page 1 and page 2, step 2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Steam Generator System (S/GS); Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: S/G level detector.

K/A MATCH:
- This question matches the K/A in that the operator must understand the effect of a loss or malfunction of the S/G level detector will have on the S/G in order to take the appropriate actions.

27.

Unit 1 was operating at 100% power when the crew manually tripped the reactor due to a steam generator tube rupture which occurred in the "A" steam generator. Currently, plant conditions are as follows:

- RCS subcooling based on CETC is 50 degrees.
- Intact narrow range SG levels indicate 45%.
- SI has actuated.
- The crew is taking actions IAW 1-E-3, "Steam Generator Tube Rupture."
- Condenser Steam Dumps are being used for cooldown.

Which ONE of the following completes the statement in describing operator responsibilities given the above conditions?

Steam flow from ______________________________.

A.✔ each intact SG should be kept less than $1.0x10^6$ LBM/HR

B. both intact SGs together shall not exceed a total of $1.0x10^6$ LBM/HR

C. both intact SGs may be steamed at a maximum rate of $4.0x10^6$ LBM/HR

D. the ruptured SG may be allowed to reach $1.0x10^6$ LBM/HR

DISTRACTOR ANALYSIS:

A CORRECT — If Steam Dumps are used for RCS cooldown, then, to prevent an undesired Main Steamline Isolation, each Main Steamline flow should be kept less than $1.0 \times 10^6$ LBM/HR.

B INCORRECT — The caution is for each separately. Limiting the total to less than $1.0 \times 10^6$ LBM/HR would unnecessarily limit the crew's ability to remove decay heat and is not in accordance with procedure.

C INCORRECT — This distractor is plausible in that the operator might confuse this figure with the design flow rate for one atmospheric dump valve which is sized to pass approximately 10% (425,244 lbm/hr at 1025 psig).

D INCORRECT — The numerical value chosed for this distractor is plausible for the same reason. However, this answer is incorrect in that this steam generator should have no steam flow indicated.

REFERENCES:
1. 1-E-3, "Steam Generator Tube Rupture," Caution and Note before step 12, page 10.
2. NCRODP-23-NA, "Main Steam System," pages 9 & 17.

K/A CATALOGUE QUESTION DESCRIPTION:
- Steam Generator Tube Rupture (SGTR); Ability to operate and monitor the following as they apply to a SGTR: Steam flow indicators.

K/A MATCH:
- This question matches the K/A in that the operator's selection of the correct answer demonstrates the ability to operate (limit steam flow to the appropriate value) by monitoring steam flow (which would be observed on steam flow indicators).

To make the question more operationally oriented, operators are given indications that should enable them to determine (without being told) that RCPs are still operating (i.e. subcooling is not less than 25 degrees), that intact SG levels are greater than 11%, and that SI has not actuated. If the quesiton is considered too difficult, the stem could be modified as follows: "Which ONE of the following completes the statement in describing operator responsibilities with regard to dumping steam" portion of the stem to make the quesiton more challenging.

28.
Plant conditions are as follows:

- The plant is at 68% power.
- A MFP trips.
- Control rods are stepping in.
- All steam dumps fail to open as required.
- A steam dump arming signal is present.
- Tavg is 592 degrees F.

Given the above information, which ONE of the following describes the effect on the main steam system and the action the crew should take IAW 1-AP-2.2, "Fast Load Reduction?"

NOTE: The following refer to S/G conditions one minute after rods begin stepping in but prior to any operator action.

A. Main steam pressure will increase, perform AP-2.1, "Turbine Trip Without Reactor Trip Required."

B. Main steam pressure will decrease, perform E-0, "Reactor Trip or Safety Injection."

C.✓ Main steam pressure will increase, perform E-0, "Reactor Trip or Safety Injection."

D. Main steam pressure will decrease, perform AP-2.1, "Turbine Trip Without Reactor Trip Required."

DISTRACTOR ANALYSIS:

A INCORRECT IAW AP-2.2, Attachment 3, if Tavg is > 591 degrees F, then trip the reactor and go to E-0.

B INCORRECT Steam pressure will increase. This distractor is plausible in that the applicant might incorrectly reason that since rods are moving in that steam pressure will decrease.

C CORRECT Even though rods are moving in, they account for 10% of 50% of a design load reduction (the steam dumps are designed to account for the remaining 40%). Therefore, Tavg, and hence steam pressure, will increase. While AP-2.2 gives the operator the option of transitioning to either E-0 or AP2.1, if certain conditions are met (i.e. > 591 degrees F) the operator is specifically directed to trip the reactor and go to E-0.

D INCORRECT Steam pressure will increase. This distractor is plausible in that the applicant might incorrectly reason that since rods are moving in that steam pressure will decrease. Also, IAW AP-2.2, Attachment 3, if Tavg is > 591 degrees F, then trip the reactor and go to E-0.

REFERENCES:
1. AP-2.2, "Fast Load Reduction," pages 3 - 5, and Attachment 3 page 1.
2. NCRODP-23-NA, "Main Steam System," pages 5, and 30 - 34.
3. NCRODP-77-NA, "Reactor Protection Systems," page T-10 & figure 77-9.

K/A CATALOGUE QUESTION DESCRIPTION:
- Main and Reheat Steam System (MRSS); Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioining steam dump.

K/A MATCH:
- This question matches the K/A in that the applicant is required to predict the impact the malfunctioning steam dumps will have on the main steam system (specifically, main steam pressure). The applicant is then asked to determine the action necessary to mitigate the consequences of the malfunction via selection of procedures.

29.
Unit 1 is operating at power when the following indications are reported:

- Reactor power is 99% and slowly increasing.
- Auctioneered High Tavg is 576 degrees F and slowly decreasing.
- RCS pressure is 2210 psig and slowly decreasing.
- Turbine load is 970 MWe and decreasing.
- Steam Pressure is 815 psig and slowly decreasing.
- Containment Pressure is 9.5 psia and slowly increasing.

Based on these parameters, which ONE of the following is in progress?

A.✔ Steamline break inside containment.

B. Steamline break outside containment.

C. RCS cold leg leak inside containment.

D. RCS hot leg leak inside containment.

DISTRACTOR ANALYSIS:

A CORRECT Rational.

B INCORRECT Rational.

C INCORRECT Rational.

D INCORRECT Rational.

REFERENCES:
1. Transient Accident Analysis
2.
3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Steam Line Rupture; Ability to determine and interpret the following as they apply to the Steam Line Rupture: Difference between steam line rupture and LOCA.

K/A MATCH:
- This question matches the K/A in that the applicant must determine and interpret, from a set of given parameters, if a steam line rupture or a LOCA is occurring. The parameters are given for only one of the two casualties minus a key parameter that would result in analyzing for the other. With both casualty types listed among the available answers, the applicant is forced to check the information given in the stem to distinguish between the two types and, hence, determine the correct answer.

30.

Unit 2 is at 100% rated thermal power with 2A-D1, ROD CONTROL URGENT FAILURE, in alarm. Control Rods are in automatic mode. The turbine governor valves close unexpectedly resulting in a 11% load rejection over a 90 second period.

Which ONE of the following correctly describes how RCS temperature is controlled given the above conditions?

A.✓ RCS temperature will be controlled entirely by steam dumps because the control rods will not move.

B. RCS temperature will be controlled entirely by control rods because the power change is not enough to require steam dump operation.

C. RCS temperature will go down due to higher xenon concentration with no rod motion and no steam dump operation.

D. RCS temperature will be controlled with a combination of control rods and steam dumps because of the rate of load rejection.

DISTRACTOR ANALYSIS:

A CORRECT — Rods will not move due to the urgent failure. Steam dumps will arm with a 10% rejection in 2 minutes as sensed by turbine impluse pressure. Therefore, with an arming signal and a Tave-Tref deviation of 4 F (i.e. a demand signal), the dumps will open.

B INCORRECT — Rods will not move due to the urgent failure. This distractor is plausible if the applicant does not know that the alarm is indication that rods will not move even though the rods and rod control are designed to respond to a load rejection such as this without assistance from the steam dumps.

C INCORRECT — Steam dumps will respond as stated in the analysis for answer "A." This distractor is plausible because in that the applicant might consider that since the rods will not operate that the steam dumps will; except that the steam dumps are armed with a demand signal.

D INCORRECT — Rods will not move due to the urgent failure. This distractor is plausible if the applicant does not know that the alarm is an indication that the rods will not move.

REFERENCES:
1. 1-AR-A-D1, ROD CONTROL URGENT FAILURE
2. NCRODP-23-NA, "Main Steam System," pages 18, 31 - 34.

K/A CATALOGUE QUESTION DESCRIPTION:
- Steam Dump System (SDS) and Turbine Bypass Control; Knowledge of the effect of a loss or malfunction on the folowing will have on the SDS: Controller and positioners, including ICS, S/G, CRDS.

K/A MATCH:
- This question matches the K/A in that it addresses almost every facit of the K/A.

31.

Unit 2 is in Mode 1 at 30% power. Attachment 1, "Turbine Mechanical Trip Test," of OP-15.1, "Operation of the Main Turbine," is in progress.

An operator is currently holding the Overspeed Trip Test Lever in the TO TEST position.

Which ONE of the following would still result in a Main Turbine and Reactor trip?

A. Actual turbine speed increases to 2000 rpm.

B. Loss of power to the Master Trip solenoid valve, SOV-20/AST-2

C.✓ Energizing the Master Trip solenoid valve, SOV-20/AST-2

D. Actual Thrust Bearing wear exceeds its trip setpoint.

DISTRACTOR ANALYSIS:

A INCORRECT — With the overspeed trip test lever in the TO TEST position, the auto-stop supply to one side of the overspeed trip cylinder is blocked. The turbine will not trip on a mechanical overspeed condition.

B INCORRECT — The Low Vacuum Trip is blocked with the overspeed trip test lever in the TO TEST position. SOV-20AST-2 is normally deenergized.

C CORRECT — SOV-20-AST-2 is normally deenergized. It allows testing of the mechanical trips while affording turbine protection by actual signals whenever one of the "customer trips" is activiated. Customer trips are the monitored events in the plant that trip the turbine. When energized, it pulls down on the protection control tripping bar. Sine the solenoid is attached behind the fulcrum point, it causes the dump valve to be lifted to its trip position, initiating a turbine trip.

D INCORRECT — With the overspeed trip test lever in the TO TEST position, the auto-stop supply to one side of the overspeed trip cylinder is blocked. The turbine will not trip on a thrust bearing trip condition.

REFERENCES:
1. OP-15.1, "Operation of the Main Turbine," Attachment 1, page 68.
2. NCRODP-75-NA, "Main Turbine Control and Protection System," pages 13 - 16, 22, and A-2.
3. NCRODP-24-NA, "Main Turbine System," page 30.

K/A CATALOGUE QUESTION DESCRIPTION:
- Main Turbine Generator (MT/G) System; Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: Defeat of reactor trip by overspeed trip test lever.

K/A MATCH:
- This question matches the K/A in that the applicant must understand the design of the turbine trip test lever and how it provides interlock protection to prevent a reactor trip when in use.

32.
Given the following conditions:

- The plant is stable at 90% power
- The main turbine is on the valve position limiter
- $T_{avg}$ is 0.8 degrees greater than Tref
- Bank D rods are at 200 steps with the Rod Control Bank Selector switch in AUTO
- No plant evolutions are in progress.

Which ONE (1) of the following describes the effect on rod control if a loss of condenser vacuum started to occur and why? (Assume **NO** operator action, the turbine trip setpoint is not reached, and feedwater temperature is unaffected.)

A.✓ Bank D rods would remain at 200 steps because a $T_{avg}/T_{ref}$ mismatch would not occur.

B. Bank D rods would remain at 200 steps because the main turbine is on the limiter.

C. Bank D rods would step IN since $T_{ref}$ would be decreasing.

D. Bank D rods would step OUT since $T_{ref}$ would be increasing.

DISTRACTOR ANALYSIS:

A CORRECT Rational.

B INCORRECT Rational.

C INCORRECT Rational.

D INCORRECT Rational.

REFERENCES:
1. 1-AP-14, "Low Condenser Vacuum,"
2.
3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Condenser Vacuum; Ability to operate and/or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position.

K/A MATCH:
- For this K/A, if condenser vacuum were lost, the only (and obvious) result would be either a unit trip or taking actions that would preclude the observance of any affect on rod position other than rods being on the bottom as the result of a trip. This question matches the K/A in that certain conditions are set wherein the operator must demonstrate the ability to monitor the affect of a loss of condenser vacuum.

33.
Unit 1 experienced an event 2 minutes ago and the following conditions now exist:

- Reactor Power is 26% and STABLE.
- RCS Tavg is at normal operating temperature and STABLE.
- 1B and 1C SGs are operating normally.
- Containment pressure is 16.9 psia and rising slowly.

The following conditions now exist on 1A SG:

| | |
|---|---|
| - Steam flow | STABLE |
| - Feed flow | RISING |
| - Pressure | DECREASING |
| - Level | At low level alarm setpoint and DECREASING RAPIDLY the STM GEN 1A LO LEVEL annunciator alarms. |

Which ONE of the following represents the correct actions to take for this event?

A. Increase feed flow to 1A SG then go to 1-AP-31, "Loss of Main Feedwater."

B. Manually initiate SI and go to E-0, "Reactor Trip or Safety Injection."

C. Start a Standby Main Feedwater Pump in accordance with 1-AP-31, "Loss of Main Feedwater."

D.✓ Go to 1-E-0, "Reactor Trip or Safety Injection."

DISTRACTOR ANALYSIS:

A INCORRECT The annunciator response page for this alarm, 1-AR-F-C1, specifically states this as the action to take. This is incorrect for a faulted steam generator.

B INCORRECT This action is plausible because containment pressure is almost at the SI set point. It is incorrect, because E-0 is always entered before initiating a manual SI.

C INCORRECT This action is per AP-31, but only when power is greater than 70%.

D CORRECT This is the correct action for a obviously faulted steam generator .

REFERENCES:

1. 1-AP-31, "Loss of Main Feedwater," pages 1 through 4.
2. Annunciator Page 1-AR-F-C1, STM GEN 1A LO LEVEL.
3. Annunciator Page 1-AR-F-A4, MAIN FD PPS DISCH HDR LO PRESS.
4. NCRODP-26-NA, "Feedwater System," pages 29 and 58.
5. Tech Spec 3.6.4, Containment Pressure.
6. 1-AP-18, "Increasing Containment Pressure," pages 2 through 4.

K/A CATALOGUE QUESTION DESCRIPTION:

- Loss of Main Feedwater (MFW); Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): MFW line break depressurizes the S/G (similar to a steam line break).

K/A MATCH:

- By asking the applicant to determine the correct operator actions when given the indications for this event (i.e. a feed line rupture), the operator must know the operational implications in order to take the appropriate actions. Although it would be more straightforward to ask the operator what event is in progress based on the given indications, without asking what actions are required, the question would not discriminate the operator's understanding with regard to operational implications resulting from a simple loss of MFW vice a MFW line break.

34.
1-ECA-0.0, Loss of All AC Power," directs the operator to:

- "Depressurize ALL INTACT SGs to 145 PSIG:"

The caution that preceeds this step states that:

- "SG pressure should be maintained greater than 120 psig."

Which ONE of the following correctly describes the BASIS for the step AND the caution?

A. To minimize RCS inventory loss.

B. To prevent voiding in the reactor vessel upper head.

C. To prevent losing Pressurizer level.

D.✔ To prevent injection of SI Accumulator nitrogen in the RCS.

DISTRACTOR ANALYSIS:

A INCORRECT This is given as the basis for the RATE at which the SGs should be depressurized. This distractor is plausible in that it is closely related to the FINAL PRESSURE to which the SGs are to be depressurized.

B INCORRECT Although this is almost always an undesireable condition, it is for this very reason that it is plausible as a distractor. For this event, depressurization of SGs should be continued even if this does occur so this answer is incorrect.

C INCORRECT Although this is almost always an undesireable condition, it is for this very reason that it is plausible as a distractor. For this event, depressurization of SGs should be continued even if this does occur so this answer is incorrect.

D CORRECT This is given as the basis.

REFERENCES:
1. 1-ECA-0.0, "Loss of All AC Power," Step 21 and NOTES and CAUTION on page 12.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite and Onsite Power (Station Blackout); Knowledge of the reason for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power.

K/A MATCH:
- This question matches the K/A in that by asking the applicant to describe the "BASIS" for a particular action contained in the EOP for this event, it askes for the reason.

35.
Which ONE of the following describes the effect of having air ejectors in service during normal operations at power?

A. Increase in the condensate pump required NPSH.

B. Increase in the hotwell temperature.

C.✓ Increase in the steam cycle efficiency.

D. Increase in the mass of non-condensable gas in the main condenser.

DISTRACTOR ANALYSIS:

A. Incorrect. No effect on the required condensate pump NPSH.

B. Incorrect. The air ejectors have no effect on hotwell temperature.

C. Correct. This is the purpose of main condenser air ejectors.

D. Incorrect. This is backwards. Actually decreases the mass of non-condensable gases.

Reference:
Based on GFES question P3078.

REFERENCES:
1. NCRODP-25-NA, "Main Condensage System," pages 24, 25, and 27.

K/A CATALOGUE QUESTION DESCRIPTION:
- Condenser Air Removal System (CARS); Knowledge of system purpose and or function.

K/A MATCH:
- This question matches the K/A in that the applicant must understand how the system functions.

36.
Both Units are operating at 100% power. Conditions are as follows:

- Component Cooling Water Pumps 1-CC-P-1A and 2-CC-P-1A are running.
- CC systems are in a normal lineup.

A loss of offsite power occurs. Emergency diesels are supplying power to their respective busses with the exception of the Unit 1, 1H 4160 VAC bus.

Given the above information, which ONE of the following describes plant conditions 25 seconds after emergency diesels are supplying power with no operator action?

A. Only the 2-CC-P-1A is running supplying cooling water flow to Unit 2 only.

B. 2-CC-P-1A and 1-CC-P-1A are running supplying cooling water flow to both units.

C.✓ 2-CC-P-1A, 2-CC-P-1B and 1-CC-P-1B are running supplying cooling water flow to both units.

D. Only the 2-CC-P-1B and 1-CC-P-1B are running supplying cooling water flow to both units.

DISTRACTOR ANALYSIS:

A INCORRECT — The pump would be supplying both units since Units 1 & 2 CCW systems are normally cross-connected.

B INCORRECT — 2-CC-P-1A would be running but 1-CC-P-1A is powered from the 1H 4160 VAC bus and since power to that bus has not been restored, this pump will not be running.

C CORRECT — Both unit 2 CC pumps would be runningdue to the loss and restoration of power to their respective busses. Unit 1 "B" CC pump would have started when power was restored to the 1J bus.

D INCORRECT — These are not the ONLY pumps running. 2-CC-P-1A would also be running.

REFERENCES:
1. North Anna Self-Study Guide for Component Cooling Water System, pages 8 - 10.
2. NCRODP-51-NA, "Component Cooling System," pages 2, 10, & 31.
3. North Anna Component Cooling Water Exam Bank, questions 47, 61, & 64.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Offsite Power; Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: CCW pump.

K/A MATCH:
- This question matches the K/A in the operator must have the ability to monitor CCW pump response to the plant conditions in order to provide a correct answer.

37.

At full power operation, a loss of power to 1B SG level channel III occurs. Which ONE of the following describes the effect on 1B SG level indication on channels I & II?

Assume no operator action and that all control and/or protection systems operate as designed.

A. They will decrease to approximately 33% and then be maintained at that level with no more than a $\pm$ 5% deviation.

B. They will decrease until a 5% deviation exists between actual level and programmed level and then slowly increase to program level.

C. They will increase to approximately 69.5% and then decrease.

D.✓ They will increase to approximately 75% and then decrease.

DISTRACTOR ANALYSIS:

A INCORRECT Level will increase. This distractor is plausible because the operator might confuse this fault with the program level for 0% turbine load.

B INCORRECT Level would increase. This distractor is plausible because the operator might confuse this with the 5% deviation between actual and programmed level that actuates the annunciator.

C INCORRECT Level would continue to increase to 75%. This distractor is plausible in that the operator might confuse the Steam Generator and Pressurizer hi level set points.

D CORRECT Loss of power to this channel would have the same affect as the level failing low. The bypass FRV would modulate open to increase actual level to programmed level but would stay open due to constant level error caused by the low level input due to the loss of power. SG level would continue to increase to 75% until the Hi-Hi SG level signal closed the bypass FRV. This signal would also trip the main turbine, trip the mainfeedwater pumps, close the main feedwater pump discharge MOVs, close the main feedwater isolation MOVs, and close the MFRVs.

REFERENCES:
1. North Anna Self-Study Guide for Steam Generator Level Control and Protection System, section 2.1.2b.4 on page 12.
2. NCRODP-26-NA, "Feedwater System," pages 32, 33, 34, and 50.
3. NCRODP-77-NA, "Reactor Protection Systems," pages T-4 & T-16.
4. 1-AR-B-G8, PRZ HI LEVEL.
5. 1-AR-R-B1, STM GEN 1A LO-LO LEVEL CH I-II-III.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Vital AC Electrical Instrument Bus; Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: S/G pressure and level meters.

K/A MATCH:
- This question matches the K/A in that it addresses the S/G level portion of the K/A and the oprator's ability to both determine and interpret the effect of a loss of the vital AC instrument bus supplying a particular channel.

38.

Unit 1 was tripped from 100% power following a loss of several electrical busses. The following conditions exist:

- Operating crew has completed 1-ES-0.1, "Reactor Trip Response."
- All electrical busses have been restored with the exception of the 1J emergency bus.
- Repairs to the 1J emergency bus are expected to take 6 hours
- The 1J bus has been dead for one hour.
- All vital busses are being supplied from their respective inverters.

Continued operation in this condition could cause a loss of __________.

A. Vital bus 1-I

B. Air-side seal oil pump, 1-GM-P-1

C.✔ Breaker control power for "A" station service loads

D. DC turbine oil pump, 1-TM-P-5

DISTRACTOR ANALYSIS:

A. Incorrect. DC busses 1-III and 1-IV supply power to the 1-III and 1-IV vital buses, respectively.

B. Incorrect. The air-side seal oil backup pump is powered from 1-IV. The air-side seal oil pump (1-GM-P-1) is powered from MCC 1A2-4.

C. Correct. DC bus 1-III supplies breaker control power to "A" station service 4160V and 480V breakers. (1-IV supplies power to only breaker 15D3.)

D. Incorrect. DC bus 1-II supplies power to the DC turbine oil pump via the H bus.

REFERENCES:
1. NCRODP-35-NA, "Vital and Emergency Electrical Distribution System," pages 3, 10, 36, 37, 45, 49, & A-3.
2. Technical Specification Bases 3.8.4 on page B 3.8.4-2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of DC Power; Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.

K/A MATCH:
- This question matches the K/A in that it specifically addresses the operational implications of a loss of DC power (which would be caused by a station blackout) on battery charger equipment and instrumentation.

39.
The following conditions exist on Unit 2:

- Reactor power is 45% and stable.
- 2-FW-P-1B is supplying feedwater.
- Steam Generator Water Level Controls are in AUTOMATIC.

Which ONE of the following failures will cause RCS Tavg to INITIALLY decrease?

A. Feed Pump differential pressure transmitter, PDT-102, fails high.

B.✓ Feed Pump differential pressure transmitter, PDT-102, fails low.

C. Steam Generator level channel III fails high.

D. Steam Generator level channel II fails low.

DISTRACTOR ANALYSIS:

A INCORRECT — See rational for answer B.

B CORRECT — This failure will cause an auto-start for the standby main feedwater pump which will initially result in an increase in feedwater flow to the steam generators resulting in a decrease in Tavg due to the initial drop in Tcold.

C INCORRECT — This failure will cause the SGWL control system to reduce feedwater to the steam generators causing Tcold to initially increase due to less cooling. Tavg will initially increase as a result.

D INCORRECT — This failure will have no impact since only level channel III has input into SGWL control.

REFERENCES:
1. NCRODP-26-NA, "Feedwater System," pages 28, 29, & 53.

K/A CATALOGUE QUESTION DESCRIPTION:
- Main Feedwater (MFW) System; Knowledge of the effect that a loss or malfunction of the MFW will have on the following: RCS.

K/A MATCH:
- This question matches the K/A in that each of the choices presented to the applicant will have an affect on the RCS and in order to select the correct answer, the applicant must know what the effect will be.

40.
The following Unit conditions exist:

- Both Units are at 100% power, steady state conditions.
- Waste gas decay tank outlet flow control valve 1-GW-FCV-101 is in HAND control.
- Annunciator 2-AR-B-A5, PROCESS VENT VNT STACK A&B LOW RAD MON ALERT/RAD actuates.
- Annunciator 2-AR-B-B5, PROCESS VENT VNT STACK A&B HI HI RADIATION actuates.
- The process radiation alarms associated with these two annunciators have been verified to be valid.
- No other process radiation monitors have exceeded their alarm setpoints.

Which ONE of the following describes the system and/or operator response?

A. Operators should verify that the following valves automatically shut:
   - Unit 1 containment vacuum pump discharge trip valve, 1-GW-TV-102A
   - Unit 2 containment vacuum pump discharge trip valve, 1-GW-TV-102B
   - Waste gas decay tank outlet flow control valve 1-GW-FCV-101.
   Operators must manually trip containment vacuum pumps.

B.✓ Operators should verify that the following valves automatically shut:
   - Unit 1 containment vacuum pump discharge trip valve, 1-GW-TV-102A
   - Unit 2 containment vacuum pump discharge trip valve, 1-GW-TV-102B
   - Waste gas decay tank outlet flow control valve 1-GW-FCV-101.
   Operators should verify that containment vacuum pumps automatically trip.

C. Operators should manually shut the following valves:
   - Unit 1 containment vacuum pump discharge trip valve, 1-GW-TV-102A
   - Unit 2 containment vacuum pump discharge trip valve, 1-GW-TV-102B
   - Waste gas decay tank outlet flow control valve 1-GW-FCV-101.
   Operators must manually trip containment vacuum pumps.

D. Operators should verify that the following valves automatically shut:
   - Unit 1 containment vacuum pump discharge trip valve, 1-GW-TV-102A
   - Unit 2 containment vacuum pump discharge trip valve, 1-GW-TV-102B
   Operators should close waste gas decay tank outlet flow control valve 1-GW-FCV-101.
   Operators should verify that containment vacuum pumps automatically trip.

DISTRACTOR ANALYSIS:

A INCORRECT Although the containment vacuum pumps do not receive a trip signal as a result of the alarms, they trip on interlock when their associated discharge valves shut. This distractor is plausible in that the operators may recall that these pumps do not trip directly from the high radiation signal but may not recall the interlock with the discharge valves.

B CORRECT All three of the valves listed will automatically shut whenever the Westinghouse process radiation monitors (GW-RM-102 and GW-RM-101) or the Kaman Science radiation monitors (1-GW-RM-178-1 and 178-2) detect high-high radiation levels. The two annunciators given in the stem can be actuated by 178-1 and 178-2 respectively. Although the containment vacuum pumps do not receive a trip signal as a result of the alarms, they trip on interlock when their associated discharge valves shut.

C INCORRECT Operators do not need to take any of the actions listed. This distractor is plausible because operators might think that valves might not automatically shut unless the Westinghouse process radiation monitors are also alarming which is why the statement was added to the stem regarding no other radiation monitors were alarming.

D INCORRECT -GW-FCV-101 will automatically shut irregardless if it is in HAND or AUTOMATIC upon receipt of the radiation signals given in the stem.

REFERENCES:
1. 2-AR-B-A5, PROCESS VENT VNT STACK A&B LOW RAD MON ALERT/RAD.
2. 2-AR-B-B5, PROCESS VENT VNT STACK A&B HI HI RADIATION.
3. North Anna Self-Study Guide for Gaseous Waste Disposal System, Section 3.1b.4 on page 22 & Section 4.1b.2 on page 26.
4. NCRODP-45-NA, "Gaseous Waste System," pages 21, 22, A-2, & T-7.
5. 0-AP-5.2, "MGP Radiation Monitoring System," step 5 on Attachment 5, page 2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Accidental Gaseous Radwaste Release; Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste: Ventilation system.

K/A MATCH:
- This question matches the K/A in that it addresses the operator's ability to both monitor and operate the ventilation system as it relates to an accidental gaseous radwaste release.

41.

Unit 1 is in Mode 5 and Unit 2 is operating at 100% power. The following plant conditions exist:

- The Fuel Building Radiation Automatic Interlock (key) switch is in ENABLE.
- Annunciator 1K-D4, RAD MONITOR SYST HI-HI RAD LEVEL, actuated 3 minutes ago due to a valid alarm signal.
- MCR Emergency Ventilation Fans are running.
- The MCR Bottled Air System is dumped.

Assuming all plant systems responded as expected, which ONE of the following caused the conditions given above?

A. Actions taken by the crew in response to Unit 1 Containment, manipulator crane radiation monitor, RMS-RM-162, Hi-Hi alarm.

B. Automatic plant response to Unit 1 Containment, manipulator crane radiation monitor, RMS-RM-162, Hi-Hi alarm.

C.✓ Automatic plant response to Fuel Building, fuel pit bridge radiation monitor, RMS-RM-153, Hi-Hi alarm.

D. Actions taken by the crew in response to Fuel Building, fuel pit bridge radiation monitor, RMS-RM-153, Hi-Hi alarm.

DISTRACTOR ANALYSIS:

A INCORRECT With Unit 1 in Mode 5, RMS-RM-162 is expected to be deenergized. This distractor is plausible since this radiation monitor is the only other monitor that will initiate an automatic plant response and the operator might confuse this with the automatic plant responses generated by RMS-RM-153.

B INCORRECT With Unit 1 in Mode 5, RMS-RM-162 is expected to be deenergized. This distractor is plausible since this radiation monitor is the only other monitor that will initiate an automatic plant response and the operator might confuse this with the automatic plant responses generated by RMS-RM-153.

C CORRECT A Hi-Hi alarm on either RMS-RM-152 or 153 will, following a two-minute time delay, automatically dump the MCR Bottled Air System and start the MCR Emergency Ventilation Fans if the "Fuel Building Radiation Automatic Interlock" (key) switch in the MCR is in the "Enable" position.

D INCORRECT The conditions are a result of automatic plant response two minutes after receipt of the alarm.

REFERENCES:
1. MCRODP-45-NA, "Radiation Monitoring System," pages 27, 28, T-1 & T-11.
2. 1-AR-K-D4, RAD MONITOR SYST HI-HI RAD LEVEL.
3. 1-AP-5, "Unit 1 Radiation Monitoring System," third bullet of NOTE on page 2.
4. 0-AP-5.1, "Common Unit Radiation Monitoring System," steps 1 & 2 on page 2, Attachment 3, step 4 on pages 2 & 3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Area Radiation Monitoring (ARM) System Alarms; Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location.

K/A MATCH:
- This question matches the K/A in that in order to correctly answer the question, the operator must know the alarms generated by different detectors at different locations.

42.
The following unit conditions exist:

- Unit 1 has been operating at 100% power for several months
- A main feedwater piping rupture in the turbine building resulted in a unit trip
- The Auxiliary Feedwater System failed to function.

Which ONE of the following is correct concerning the response of the Reactor Coolant System (RCS), assuming that no operator actions are taken?

A. Steam generator inventory is sufficient to remove decay heat for 30 minutes; RCS temperature stabilizes at Tsat corresponding to the setpoint for the steam generator PORVs.

B. RCS temperature and pressure decrease rapidly; safety injection (SI) actuates; SI flow refills the pressurizer; RCS pressure increases rapidly; pressurizer PORVs cycle to control pressure.

C. RCS temperature and pressure increase; pressurizer PORVs cycle to control RCS pressure; primary relief tank rupture disks blow; safety injection actuates due to increase in containment pressure.

D.✓ Steam generators boil dry; RCS temperature and pressure increase; pressurizer PORVs and/or safety valves cycle to control RCS pressure; core will eventually uncover.

Distractor Analysis:

A. Incorrect. Plausible since the candidate may believe that the remaining steam generator inventory may last for an extended period of time. However, due to the loss of main feedwater at the onset of the event with no subsequent aux feedwater actuation, post-trip steam generator inventory will be significantly lower than normal. Coupled with significant decay heat and RCP heat loads this will lead to depletion of steam generator inventory early in the event.

B. Incorrect. Plausible since the RCS temperature and pressure response is the normal initial response to post-trip conditions. However, pressure does not normally decrease to the SI setpoint. The rest of the distractor would be true for a SI when the RCS repressurizes.

C. Incorrect. Plausible since the described RCS and PRT response is correct. However, no SI actuation setpoint will be reached since the magnitude of release from the ruptured PRT will not be sufficient to raise containment pressure above atmospheric pressure.

D. Correct. With a loss of all main and aux feedwater, steam generator inventory will be depleted over time resulting in a total loss of RCS heat removal. Consequently, RCS pressure and temperature will increase resulting in a depletion of RCS inventory from pressurizer PORV and/or safety valve relief leading to core uncovery.

Bank question 2415.

K/A CATALOGUE QUESTION DESCRIPTION:
- Auxiliary/Emergency Feedwater (AFW) System; Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS.

43.
The following conditions exist:

- Reactor Power is 50%
- A total loss of all Service Water has occurred.
- 0-AP-12, "Loss of Service Water," has been entered.
- RCP temperatures are beginning to rise.

Which ONE of the following correctly describes the actions operators must take in response to the given conditions?

A. When RCP Stator Winding temperature reaches 300°F; stop the affected RCP(s), then trip the affected unit.

B. When RCP Stator Winding temperature reaches 225°F; trip the affected unit, then stop the affected RCP(s).

C.✓ When RCP Pump Bearing temperature reaches 225°F; trip the affected unit, then stop the affected RCP(s).

D. When RCP Pump Bearing temperature reaches 195°F; stop the affected RCP(s), then trip the affected unit.

DISTRACTOR ANALYSIS:

| | | |
|---|---|---|
| A | INCORRECT | This is the correct temperature at which to take action. However, the reactor should be tripped before stopping the affected RCP(s). |
| B | INCORRECT | The stator winding temperatue at which to take action is 300°F. |
| C | CORRECT | Correct. |
| D | INCORRECT | The pump bearing temperature at which to take action is 225°F. RCP(s) should not be tripped until after the affected unit is tripped. |

REFERENCES:
1. 0-AP-12, "Loss of Service Water," step 9, page 8.
2. NCRODP-13-NA, "Service Water System," page 35.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Nuclear Service Water; Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A MATCH:
- This question matches the K/A in that is requires the operator to perform specific system actions on other (integrated) affected systems that take the plant through at least two different modes of plant operation.

44.
A loss of off-site power has left the "H" bus without power.

Assuming no operator action, which ONE of the following DC buses would lose power first?

A. 1-III

B. 1-I

C.✓ 1-II

D. 1-IV

DISTRACTOR ANALYSIS:

A INCORRECT This DC bus is powered from the "J" bus which was not affected based on the indications given.

B INCORRECT The amount of load on this DC bus is less than the load on 1-II.

C CORRECT Battery 1-II has no spare capacity as determined by the battery sizing sheets in Calculation EE-0009 relative to the other three and therefore would lose power first.

D INCORRECT This DC bus is powered from the "J" bus which was not affected based on the indications given.

REFERENCES:
1. NCRODP-35-NA, "Vital and Emergency Electrical Distribution System," page 22 and figures 35-1 & 35-2.
2. North Anna Vital and Emergency Electrical Distribution System Exam Bank, question 108, ID 50277 on page 110
3.

K/A CATALOGUE QUESTION DESCRIPTION:
- AC Electrical Distribution; Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: DC system.

K/A MATCH:
- This question matches the K/A in that the applicant must know the effect of the loss of a particular portion of the AC distribution system has on a related portion of the DC system.

45.
Which ONE of the following would prevent the H stub bus breaker (15H12) from reclosing automatically following restoration of power to the emergency bus but would not prevent operators from manually reclosing the breaker?

A. H bus voltage restored for at least 15 seconds

B.✓ 1-RH-P-1A pump breaker closed

C. Ground condition present

D. CDA signal present

DISTRACTOR ANALYSIS:

| | |
|---|---|
| A INCORRECT | If H bus voltage has been restored for at least 15 seconds then neither action would be prevented. |
| B CORRECT | To automatically close the H stub bus breaker the 1-RH-P-1A pump breaker must be open. |
| C INCORRECT | This would prevent a manual or automatic closing of the stub bus breaker. |
| D INCORRECT | This would prevent a manual or automatic closing of the stub bus breaker. |

REFERENCES:
1. NCRODP-35-NA, "Vital and Emergency Electrical Distribution System," pages 34 & 35.

K/A CATALOGUE QUESTION DESCRIPTION:
- AC Electrical Distribution; Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following: Bus lockouts.

K/A MATCH:
- This question matches the K/A in that the applicant must know the lockout that would prevent an automatic reclosing of the stub bus breaker resulting in a BUS lockout.

46.
A failed breaker has caused the loss of a 125-volt battery charger. Fifteen minutes later, as the OATC monitoring the voltage on the bus you will expect bus voltage will drop __________.

A.✔ slowly at first; then later drop faster

B. quickly at first; then later drop more slowly

C. at a constant, but slow rate

D. at a constant, but quick rate

DISTRACTOR ANALYSIS:

A. Correct. After the initial drop the voltage will drop slowly until cell reversal starts having an effect at which time the voltage drop will occur at a faster rate.

B. Incorrect. See A. Could be chosen based on misconception of cell reversal effects.

C. Incorrect. See A. Could be chosen if cell reversal effects are not considered.

D. Incorrect. See A. Could be chosen based on misconception of cell reversal effects.

Based on bank question 4086.

References:
Objective 3280 in Vital and Emergency Electrical.

REFERENCES:
1. 1-AR-K-C2, STA BATTERY VOLTAGE TROUBLE.
2. NCRODP-35-NA, "Vital and Emergency Electircal Distribution System," pages 14 & 30.
3. Technical Specification 3.8.6.2 and Table 3.8.6-1 on pages 3.8.6-3 & 4.
4. 1-ECA-0.0, "Loss of All AC Power," steps 12 & 19.

K/A CATALOGUE QUESTION DESCRIPTION:
- DC Electrical Distribution; Ability to manually operate and/or monitor in the control room: Battery voltage indicator.

K/A MATCH:
- This question matches the K/A in that the operator is specifically asked about battery voltage which is monitored on the control room (MCR).

47.

Both units were operating at 100% power when a loss of off-site power occurs. The following conditions exist on unit 1:

- 1-FW-P-2, Steam-Driven Auxiliary Feedwater Pump, did not start and cannot be started
- 1H EDG is tagged out for major maintenance
- 1J EDG started and loaded
- The crew is currently in 1-ES-0.1, "Reactor Trip Response" and 1-AP-10, "Loss of Electrical Power"
- 15J2, 1J EDG output breaker, has just opened due to a fault.

Which ONE of the following correctly describes the crew's response to the above conditions?

A. Go to 1-E-0, "Reactor Trip and Safety Injection," and then transition to 1-ECA-0.0, "Loss of All AC Power."

B.✓ Go directly to 1-ECA-0.0, "Loss of All AC Power."

C. Remain in 1-ES-0.1, "Reactor Trip Response," and 0-AP-10, "Loss of Electrical Power."

D. Go directly to 1-FR-H.1, "Loss of Secondary Heat Sink."

DISTRACTOR ANALYSIS:

A. Incorrect. There is a transition to ECA-0.0 out of E-0, but there is no reason to go back to E-0 at this time.

B. Correct. ECA-0.0 can be entered directly.

C. Incorrect. A loss of all emergency power has occurred. Although AP-10 will still be applicable, it is not the controlling procedure.

D. Incorrect. Although a loss of all aux feed flow has occurred. ECA-0.0 directs that CSFs only be monitored and FRs should not be entered.

REFERENCES:
1. 2H-F7, 4KV EMER BUS 2H UV
2. 2H-F8, 4KV EMER BUS 2J UV
3. E-0, "Reactor Trip or Safety Injection," step 3 on page 3 and step 16 on page 5 of Attachment 5.
4. ECA-0.0, "Loss of All AC Power," pages 1 & 2.
5. 0-AP-10, "Loss of Electrical Power," pages 1 & 2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Diesel Generator (ED/G) System; Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

K/A MATCH:
- This question matches the K/A in that it requires the applicant to recognize an abnormal indication that meets entry-level conditions for entering emergency and abnormal operating procedures.

48. 064K4.04 001/2/1/ED/G/MEMORY/MODIFIED/NA06/RO/MC

The following conditions exist:

- Both units are at 100% power
- 1-PT-82H, "1H Emergency Diesel Generator Slow Start Test," is being conducted and 1H EDG is currently paralleled to the grid
- A safety injection signal has occurred due to a LOCA

Which ONE of the following signals will trip the 1H EDG?

A. Low crankcase pressure.

B.✓ Generator differential (87X).

C. High jacket cooling temperature.

D. Overexcitation.

DISTRACTOR ANALYSIS:

A INCORRECT High crankcase pressure, not low, will result in a diesel shutdown.

B CORRECT This is one of the 3 conditions which will result in EDG trip when an emergency start condition has occurred.

C INCORRECT High jacket cooling temperature of > 205 degrees F is a trip, but not when an emergency start is present. Could be chosen if the examinee does not realize that even though the EDG was already running it still has an emergency start.

D INCORRECT Overexcitation is not a diesel trip at any time.

Based on bank qustion 5567. Changed inital conditions and some distractors.

REFERENCES:
1. CNRODP-55-NA, "Station Diesel Generator Systems," pages 86, 87, 93, 94, 102, T-12, T-14 thru T-17.
2. 0-OP-6.1, "Operation of the SBO Diesel," step 5.2.27 on page 23.
3. 0-OP-6.4, "Operation of the SBO diesel," step 4.20 on page 13.
4. Technical Specification Bases 3.8.1.3, page B 3.8.1-23.

K/A CATALOGUE QUESTION DESCRIPTION:
- Emergency Diesel Generators (ED/G); Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Overload ratings.

K/A MATCH:
- This question matches the K/A in that the operator must know the interlock that protects the SBO (Emergency Diesel Generator) Diesel from overcurrent.

49.
The following conditions exist:

- Unit 1 is at 30% power.
- 2-IA-C-1 is tagged for PMs.
- A total loss of the switchyard occurs.
- Both Unit 1 EDGs fail to reenergize the emergency busses and the crew implements 1-ECA-0.0, "Loss of All AC Power."
- The crew is directed to "**MANUALLY** dump steam at the maximum rate using S/G PORVs."

In accordance with ECA-0.0, "Loss of All AC Power," which ONE of the following is correct concerning the **INITIAL** crew response?

A. Fail open the S/G PORVs using the keyswitches in the cable vault.

B.✓ Open the S/G PORVs using the controllers on the benchboard.

C. Direct a watchstander to open the S/G PORVs using the handwheels.

D. Direct a watchstander to fail the IA to the S/G PORVs.

DISTRACTOR ANALYSIS:

A INCORRECT The keyswitches only have the ability to isolate, not open. This distractor is plausible if the operator fails to remember that the switches only isolate.

B CORRECT Rational.

C INCORRECT This answer would be correct for long-term operation but not for an initial response.

D INCORRECT Performing this action would have no affect since the valves fail closed. This answer is plausible if the operator thinks that the valve fails open.

REFERENCES:
1. ECA-0.0, "Loss of All AC Power," Step 21.b on page 13.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Instrument Air; Ability to locate and operate components, including local controls.

K/A MATCH:
- This question matches the K/A in that the loss of the switchyard in conjunction with the one air compressor being tagged results in a loss of instrument air which, under the given conditions, requires operation of local benchboard controls.

50.
A liquid radwaste release is in progress.

Which ONE of the following should operators expect to observe upon receipt of a hi-hi radiation signal from LW-RM-111? (Assume no operator action.)

A. Clarifier Holdup Tank Inlet Valve, LW-FCV-100, shuts **and** All running SG Blowdown Pumps stop.

B.✓ Liquid Waste Discharge Valve, LW-PCV-115, shuts **and** Clarifier Holdup Tank Inlet Valve, LW-FCV-100 shuts **and** All running SG Blowdown Pumps stop.

C. Actuation of a radiation alarm **and** Liquid Waste Disch Valve, LW-PCV-115, shuts.

D. **Only** actuation of a radiation alarm.

DISTRACTOR ANALYSIS:

A INCORRECT A high radiation signal from LW-RM-111 also causes LW-PCV-115 to shut.

B CORRECT A high radiation signal from LW-RM-111 shuts LW-PCV-115 and FCV-LW-100. The blowdown pumps trip due to FCV-LW-100 closure.

C INCORRECT A high radiation signal from LW-RM-111 also causes FCV-LW-100 to shut which, in turn, causes the blowdown pumps to trip.

D INCORRECT This would be the expected response upon a receipt of a hi radiation signal from LW-RM-100, Liquid Waste Evaporator radiation monitor.

REFERENCES:
1. NCRODP-43-NA, "Liquid Waste System," pages 4, 16, 17, 39, and figure 43-6.
2. NCRODP-46-NA, "Radiation Monitoring System," pages 6, 15, 30, and T-7.

K/A CATALOGUE QUESTION DESCRIPTION:
- Liquid Radwaste System (LRS); Ability to manually operate and/or monitor in the control room: Automatic isolation.

K/A MATCH:
- This question matches the K/A in that the applicant is asked to determine the automatic liquid radwaste system responses that could be monitored in the control room as a result of an isolation signal.

51.
The N-16 main steam radiation monitor's indication is invalid below 20% reactor power for which ONE of the following reasons?

A. Steam flow is so low that all of the N-16 decays prior to reaching the detector.

B.✔ The neutron flux is insufficient to produce representative amounts of N-16.

C. Microprossessor neutron flux peaks cannot be identified at this power.

D. At this power level, the 0.5-inch slot in the detector shield limits the collimation of the radiation field.

DISTRACTOR ANALYSIS:

| | |
|---|---|
| A INCORRECT | Not **ALL** of the N-16 decays. this distractor is plausible considering the short half-life of N-16. |
| B CORRECT | At low powers, neutron flox is insufficient to produce representative amounts of N-16. |
| C INCORRECT | This detector measures gamma, not neutron, flux. This distractor is plausible in that the applicant might confuse this detector with those designed to detect neutron flux, in which case this rational could apply. |
| D INCORRECT | Although this detector does incorporate such a design, its ability to detect would not be diminished as a result of low power. |

REFERENCES:
1. North Anna Radiation Monitoring System Exam Bank, question 59, ID 1339, page 61.
2. NCRODP-46-NA, "Radiation Monitoring System," pages 3, 4, 19, & 20.

K/A CATALOGUE QUESTION DESCRIPTION:
- Process Radiation Monitoring (PRM) System; Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation theory, including sources, types, units, and effects.

K/A MATCH:
- This question matches the K/A in the following respects: the low power aspect addresses operational implications applicable to the PRM system; distractors "C" and "D" require the applicant to have knowledge of radiation theory, sources, and types.

52.
The crew has entered 1-FR-C.1, "Response to Inadequate Core Cooling."

You have been directed to open 1-SI-MOV-1869A, SI Hot Leg Injection Alternate Header Isolation IAW 1-FR-C.1.

Which ONE of the following describes your actions?

A. Turn on control power to the valve by shutting its breaker locally then operate the valve from the MCR; a key is required.

B. Turn on control power to the valve by shutting its breaker locally then operate the valve from the MCR; a key is NOT required.

C.✓ Turn on control power to the valve from the MCR and then operate the valve from the MCR; a key is required.

D. Turn on control power to the valve from the MCR and then operate the valve from the MCR; a key is NOT required.

DISTRACTOR ANALYSIS:

A INCORRECT Control power to the valve is turned on from the MCR. However, this distractor is credible in that the operator might confuse this action with a similar action required for SI Accumulator isolation valves.

B INCORRECT Control power to the valve is turned on from the MCR. However, this distractor is credible in that the operator might confuse this action with a similar action required for SI Accumulator isolation valves. A key is required to operate the valves.

C CORRECT MOV-1869A is controlled by a pushbutton handswitch and a keyswitch on MCR vertical board Safeguards Panels. The handswitch has tow positions, ON and OFF, that energize or deenergize the valve control circuits. Its keyswitch has two positions, OPEN and CLOSE, to allow operation of the valve.

D INCORRECT A key is required to open the valve.

REFERENCES:
1. 1-FR-C.1, "Response to Inadquate Core Cooling," step 2 RNO a)2).
2. NCRODP-52-NA, "Safety Injection System," pages 10, 25, T-1 & T-4.
3. NCRODP-35-NA, "Vital and Emergency Electrical Distribution System," page T-1.

K/A CATALOGUE QUESTION DESCRIPTION:
- Inadequate Core Cooling; Ability to locate and operate components, including local controls.

K/A MATCH:
- This question matches the K/A in that in order to answer the question correctly the operator must know the location of and how to operate controls associated with inadequate core cooling.

53.
Plant conditions are as follows:

- Unit 2 is stable at 5% reactor power waiting to clear a Chemistry hold.
- A failure of "B" reserve station service transformer is caused by internal arcing.
- 15G1 automatically opens.
- All equipment is available and operated as expected.

Which ONE of the following describes the affect on the plant?

15G10 will automatically __________.

A. open and unit 2 circulating water pumps will continue to run

B. open and unit 2 circulating water pumps trip due to undervoltage

C.✔ close and unit 1 circulating water pumps will continue to run

D. close and unit 1 circulating water pumps trip due to undervoltage

DISTRACTOR ANALYSIS:

A INCORRECT 15G10 will automatically shut.

B INCORRECT 15G10 will automatically shut.

C CORRECT

D INCORRECT Unit 1 circualting water pumps will not trip.

REFERENCES:
1. North Anna Basic Electrical Distribution System Exam Bank, question 55, ID 2952, page 56.
2. NCRODP-18-NA, "Basic Electrical Distribution," pages 36, 50, T-8, and fig 18-8.

K/A CATALOGUE QUESTION DESCRIPTION:
- Circulating Water System; Knowledge of bus power supplies to the following: Emergency/essential SWS pumps.

K/A MATCH:
- This question matches the K/A in that the applicant must know which circulating water pumps are powered from which buses in order to answer the question correctly.

54.
Unit 1 is at 100% reactor power. Letdown flow is currently 88 gpm.

- 1-CH-RI-128, Reactor Coolant Letdown Radiation Monitor, alarms.
- The crew has verified that the alarm is not due to an instrument malfunction.

Which ONE of the following describes the required actions for a high radiation alarm on 1-CH-RI-128?

A.✓ Notify Health Physics and Chemistry Department to obtain and analyze a sample from the RCS letdown.

B. Reduce power as required to maintain less than 80% of the allowable Technical Specifications 3.4.16 limit.

C. Reduce letdown flow to 45 gpm and monitor 1-CH-RI-128 response.

D. Increase RCS cleanup flow by opening another letdown orifice and increasing charging pump flow.

DISTRACTOR ANALYSIS:

A CORRECT — The applicant must know that the alarm does not in itself confirm the existence of high RCS activity due to the detector location and features as the alarming condition could be attributable to an increase in background radiation levels. A sample must be taken to confirm high RCS activity.

B INCORRECT — This action is performed if the high RCS activity is confirmed from the specific activity measurement from an RCS sample.

C INCORRECT — The NOTE following step 3 of Attachment 8 of 1-AP-5 states, "Maintaining letdown in service through the IXs is preferred to minimize RCS activity," indicating that flow should not be reduced.

D INCORRECT — This action would be performed after consulting with Health Physics and Chemistry IAW step 4 of Attachment 8 of 1-AP-5.

REFERENCES:
1. 1-AP-5, "Unit 1 Radiation Monitoring System," pages 2 & 3, Attachment 8 page 1.
2. Technical Specification 3.4.16, page 3.4.16-1 & Figure 3.4.16-1 on page 3.4.16-3.
3. NCRODP-41-NA, "Chemical and Volume Control System," page 16.

K/A CATALOGUE QUESTION DESCRIPTION:
- High Reactor Coolant Activity; Ability to determine and interpret the following as they apply to the High Reactor Colant Activity: Location or process point that is causing an alarm.

K/A MATCH:
- This question matches the K/A in that the operator must determine that coolant activity is actually high (which is confirmed by Health Physics/Chemistry analysis). The operator must also interpret the alarm in that it could be caused by an increase in area radiation.

55.

Unit 1 and Unit 2 are both operating at 100% power with all systems configured normally.

Which ONE of the following is correct regarding the statement:

Service water pump 1-SW-P-1A will receive an automatic start signal upon receipt of a __________.

A. train B safety injection signal on Unit 2 only

B. train B safety injection signal on either Unit 1 or Unit 2

C.✓ train A safety injection signal on either Unit 1 or Unit 2

D. train A safety injection signal on Unit 1 only

DISTRACTOR ANALYSIS:

A INCORRECT Service Water Pump 1B starts automatically on receipt of a Train B SI signal from either unit.

B INCORRECT Service Water Pump 1B starts automatically on receipt of a Train B SI signal.

C CORRECT Service Water Pump 1A starts automatically on receipt of a Train A SI signal from either unit or a loss of reserve station service power, on the assiciated unit, provided certain conditions are met.

D INCORRECT Service Water Pump 1A starts automatically on receipt of a Train A SI signal from either unit.

REFERENCES:
1. North Anna Service Water System Exam Bank, question 17, ID 1595 on page 17.
2. NCRODP-13-NA, "Service Water System," pages 10, 34, 35, 36, T-6 & fig 13-7.
3. NCRODP-18-NA, "Basic Electrical Distribution System," figure 18-2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Service Water System (SWS); Knowledge of bus power supplies to the following: Service water.

K/A MATCH:
- This question matches the K/A in that the applicant must know which bus the service water pump receives power from in order to provide a correct answer.

56.

Unit 1 was operating at 100% power, with both Unit 2 Service Water Pumps in operation when a total loss of off-site power occurred in conjunction with a Containment Depressurization Actuation (CDA). The following conditions exist:

- Moments prior to the above events, the 1H bus lost power.
- Operators have been unable to restore power to the 1H bus .

Give the above conditions, which ONE of the following describes the flow of service water to the Recirculation Spray (RS) Heat Exchangers? (Assume no **local** valve operations were performed.)

Service water supplied to the Unit 1 "C" RS Heat Exchanger will be via ___________.

A. RSHX Supply Valve, SW-MOV-101C ONLY.

B. RSHX Supply Valve, SW-MOV-101D ONLY.

C. RSHX Supply Valve, SW-MOV-101A **AND**
RSHX supply header cross-connect valves, SW-MOV-102A and -102B

D.✓ RSHX Supply Valve, SW-MOV-101B **AND**
RSHX supply header cross-connect valves, SW-MOV-102A and -102B

DISTRACTOR ANALYSIS:

A INCORRECT — This valve did not open automatically due to the loss bus 1H, its power supply.

B INCORRECT — There will be a flow path from SW supply header #2 through MOV-SW-101D, which opened automatically as a result of the CDA, and then through MOV-SW-103C which also opened automatically upon receiving the CDA signal. Both of these valves receive power from bus 1J. However this is not the ONLY flowpath as 101B through 102A and 102B is also an option.

C INCORRECT — MOV-SW-101A would not open due to the loss of its power supply, bus 1H.

D CORRECT — MOV-SW-101B would have opened automatically upon receipt of the CDA and SW-MOV-102A and 102B are always open.

REFERENCES:

1. NCRODP-13-NA, "Service Water System," page 37, figures 13-2, 13-313-6, & 13-7

K/A CATALOGUE QUESTION DESCRIPTION:

- Service Water System (SWS); Knowledge of bus power supplies to the following: ESF-actuated MOVs.

K/A MATCH:

- This question matches the K/A because the flow of water to the "C" RS Heat Exchanger is determined by the position of valve in each of the supply lines. In order to correctly determine the supply flow path, the applicant must know which valves experienced a loss of power due the the loss of the 1H bus.

REFERENCES:

1. NCRODP-13-NA, "Service Water System," page 37, figures 13-2, 13-313-6, & 13-7

K/A CATALOGUE QUESTION DESCRIPTION:

- Service Water System (SWS); Knowledge of bus power supplies to the following: ESF-actuated MOVs.

K/A MATCH:

- This question matches the K/A because the flow of water to the "C" RS Heat Exchanger is determined by the position of valve in each of the supply lines. In order to correctly determine the supply flow path, the applicant must know which valves experienced a loss of power due the the loss of the 1H bus.

57. 078A3.01 001/2/1/IAS/MEMORY/MODIFIED/NA06/RO/MC

The following conditions exist:

- 1-IA-C-1 Instrument Air Compressor was running in HAND.
- Instrument Air header pressure is 110 psig.
- The "AUTO" pushbutton for the 2-IA-C-1 Instrument Compressor was depressed.
- 2-IA-C-1 was in the "Hand" mode

Which ONE of the following describes the Instrument Air system 12 minutes later?

Instrument air pressure will _____ .

A. be higher and both compressors will be running

B. be higher and only one compressor will be running

C.✓ remain unchanged and both compressors will be running

D. remain unchanged and only one compressor will be running

DISTRACTOR ANALYSIS:

A INCORRECT When the "Auto" pushbutton is depressed, the compressor will start. Since instrument air header pressure is 110 psig and the compressor loads at 108 psig decreasing and unloads at 113 psig increasing, header pressure will remain unchanged.

B INCORRECT Botn instrument air compressors will be running.

C CORRECT When the "Auto" pushbutton is depressed, the compressor will start. Since instrument air header pressure is 110 psig and the compressor loads at 108 psig decreasing and unloads at 113 psig increasing, header pressure will remain unchanged. If the compressor is in AUTO and runs for 15 minutes unloaded, it will automatically shut down.

D INCORRECT Both instrument air compressors will be running.

REFERENCES:
1. North Anna Compressed Air System Exam Bank, question 27, ID 5062, on page 27.
2. NCRODP-17-NA, "Compressed Air System," page 30.
3. 1-OP-46.1, "Operation of 1-IA-C-1, Instrument Air Compressor," pages 4 thru 6.

K/A CATALOGUE QUESTION DESCRIPTION:
- Instrument Air System (IAS); Ability to monitor automatic operation of the IAS, including: Air pressure.

K/A MATCH:
- This question matches the K/A in that in order to answer correctly, the applicant must understand how system air pressure will be affected by automatic operation of the instrument air system, specifically with regard to the automatic (loading/unloading/starting/stopping) of one of the compressors.

58.
The crew is responding to a loss of secondary heat sink IAW FR-H.1, "Response to Loss of Secondary Heat Sink," and must align a low pressure water source to feed SGs.

Which ONE of the following describes the **preferred** source and lineup IAW FR-H.1?

A. Service Water System using the RSHX supply header.

B. Service Water System using the Diesel-Driven Fire Pump, 1-FP-P-2.

C.✓ Fire Protection System using the Motor-Driven Fire Pump, 1-FP-P-1.

D. Fire Protection System using the Diesel-Driven Fire Pump, 1-FP-P-2.

DISTRACTOR ANALYSIS:

A INCORRECT While this lineup is correct, it is considered the source of last resort because of the chemical contamination that would result in the S/Gs.

B INCORRECT The Service Water System would be used only if the Fire Protection System was not available. Also, IAW FR-H.1, the Diesel-Driven Fire Pump would only be used if the Motor-Driven Fire Pump was not available.

C CORRECT This is the preferred source and lineup.

D INCORRECT IAW FR-H.1, the Diesel-Driven Fire Pump would only be used if the Motor-Driven Fire Pump was not available.

REFERENCES:
1. 1-F-0, Attachment 3, Heat Sink.
2. FR-H.1, "Response To Loss Of Secondary Heat Sink;" Step 24, page 24; Attachment 3, page 2; Attachment 4, pages 1 & 2.
3. AP-22.5, "Loss Of Emergency Condensate Storage Tank 1-CN-TK-1," page 3, page 3 of Attachment 2, and page 2 of Attachment 3.
4. NCRODP-13-NA, "Service Water System," pages 4, 9, 10, and 16.
5. North Anna Auxiliary Feedwater System Exam Bank, question 66, ID 6098, page 66.

K/A CATALOGUE QUESTION DESCRIPTION:
- Fire Protection System (FPS); Knowledge of the physical connections and/or cause-effect relationships between the Fire Protection System and the following Systems: AFW System.

K/A MATCH:
- This question matches the K/A in that the applicant must know how and in what way the two systems are physically connected in order to render a correct answer.

59.

Unit 1 was operating at 100% power when a large break loss of coolant accident occurs. The operating crew is performing 1-E-0, REACTOR TRIP OR SAFETY INJECTION. Only one train of safeguards equipment actuated. The following plant conditions exist:

- All train "A" containment isolation phase A and phase B valves are CLOSED.
- All train "B" containment isolation phase A and phase B valves are OPEN
- Containment pressure is 21 psia.

Based on the present plant status, which ONE of the following is correct?

Single train containment isolation is __________ for satisfying post-accident containment isolation criteria. The train B containment isolation valves should be verified closed by using __________.

A. insufficient; 1-PT-91 Containment Penetrations.

B.✔ sufficient; 1-E-0.

C. sufficient; 1-PT-91 Containment Penetrations.

D. insufficient; 1-E-0.

DISTRACTOR ANALYSIS:

A INCORRECT Isolation is sufficient.

B CORRECT Containment isolation is sufficient with one train closed. However, isolation valves should be verified closed using the attachments.

C INCORRECT Isolation valves should be verified closed using the attachments.

D INCORRECT Isolation is sufficient.

REFERENCES:
1. North Anna Containment Isolation System Exam Bank, question 2, ID 5710, on page 2.
2. 1-E-0, "Reactor Trip or Safety Injection," step 12 on page 8 & Continuous Action Page.
3. 1-FR-Z.1, "Response to High Containment Pressure," step 2 on page 2.
4. FSAR Chapter 3, "Design Criteria," Sections 3.1.47.2 and 3.1.48.1 on pages 3.1-40 & 41.

K/A CATALOGUE QUESTION DESCRIPTION:
- Containment System; Ability to (a) predict the impacts of the following malfunction or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.

K/A MATCH:
- This question matches the K/A in that the applicant must determine the impact of a malfunction of one train of Phase A/B on a containment isolation signal. The applicant is then required to determine the actions and procedures to mitigate the consequences of the malfunction.

60.

Unit 1 is operating at 100% power when Engineering discovers a common mode failure that renders all three AFW pumps inoperable on Unit 1.

Which ONE of the following correctly describes the actions required by Technical Specifications?

A. Within one hour initiate action to place Unit 1 in Mode 4 on RHR.

B.✓ Immediately initiate action to restore <u>one</u> AFW train to operable. LCO 3.0.3 requirements are suspended.

C. Immediately initiate action to place Unit 1 in Mode 4 on RHR.

D. Immediately initiate action to restore <u>one</u> AFW train to operable. LCO 3.0.3 requirements are <u>not</u> suspended.

<u>DISTRACTOR ANALYSIS</u>:

| | |
|---|---|
| A INCORRECT | Although the TS would not apply in Mode 4, this is an incorrect action as the unit should not be moved without an operable AFW train. There is no one hour to take action. |
| B CORRECT | The crew needs to immediately take action to restore one AFW train to operable. |
| C INCORRECT | Although action must be taken immediately, the unit should not be moved with no operable AFW trains. |
| D INCORRECT | Although two AFW trains are required in this mode there is a separate action for having two inoperable trains of AFW. |

References
Tech Spec 3.7.5

61.

During the Unit 1 refueling outage, you are tasked with performing the initial valve lineup for 1-OP-31.2A, "Valve Checkoff - Auxiliary Feedwater." You are required to check 1-FW-149, Turb Drvn AFW Pump to S/G MOV Hdr Disch Isol Valve, in the "Locked Closed" position.

Which ONE of the following statements describes the correct method for checking the position of 1-FW-149 in accordance with OPAP-0012, Valve Operations?

A. Visually verify that the stem is in the down position, then check the lock and chain are installed properly.

B.✓ Unlock the valve, using reasonable force, turn the valve in the clockwise direction. Re-install the lock and chain.

C. Check the lock and chain are installed properly, using reasonable force, turn the valve in the clockwise direction.

D. Unlock the valve, using reasonable force, turn the valve slightly in the counter clockwise direction and then reclose the valve. Reinstall the lock and chain.

Distractor Analysis:

A. Incorrect. This would be correct for the independent verifier.

B. Correct. This is how a locked closed valve would be initially checked.

C. Incorrect. Correct actions, but incorrect order.

D. Incorrect. Some of the steps are correct, but not all of them. Valve would NOT be turned in the open direction.

Modified from bank question 5712. Now intial performer, not Independent Verifier. Changes correct answer from A to B.

Reference:
OPAP 0012, pages11-14.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of how to conduct and verify valve lineups.

K/A MATCH:
- Question was written to both parts of this K/A in accordance with the guidance provided in NUREG 1021 Rev.9 Section 401.

62.
The following Unit 2 conditions exist:

- The reactor is at 100% Rated Thermal Power.
- Loop "A" Tavg instrument failed hi.
- Control rods have been placed in manual.
- PRZR Level has been placed in manual.
- Steam Dumps have been placed in Steam Pressure Mode.
- The DELTA T DEFEAT SWITCH has been selected to defeat the failed channel.
- The Tavg DEFEAT SWITCH has been selected to defeat the failed channel.

If no other actions were taken, which ONE of the following correctly states the plant status and/or required operator actions?

A. The reactor should have already tripped on OTDT. The RO is required to immediately trip the reactor.

B.✔ If another loop OTDT trip setpoint is subsequently exceeded, then the operators are to ensure the reactor automatically trips.

C. The turbine should have already undergone a runback. The BOP is required to manually runback the turbine.

D. The reactor should have already tripped on OPDT. The RO is required to immediately trip the reactor.

DISTRACTOR ANALYSIS:

A INCORRECT — Not in a 2/3 logic (I.E. loop is not bypassed). Plausible because loop can be removed from control, but not protective, circuits.

B CORRECT — OTDT is essentially in 1/3 logic because the protective circuit is not bypassed by the actions stated in the stem.

C INCORRECT — Runback occurs on 2/4 within 3% of trip setpoint. Plausible because applicant may not understand or correctly remember runback logic.

D INCORRECT — See A.

REFERENCES:
1. 2-AR-B-A8, LOOP 2A-B-C TAVG DEVIATION.
2. NCRODP-77-NA, "Reactor Protection Systems," pages 12, 13, 14, 18, 19, T-10, and figures 77- & 77-2.

K/A CATALOGUE QUESTION DESCRIPTION:
- Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A MATCH:
- This question matches the K/A in that the applicant must make operational judgements based on plant configuration and instrument interpretation to determine the required operator/plant response.

63.
Which ONE of the following describes part of the procedure change process that is unique to temporary procedures?

A.✓ The requester must enter an expiration date and it should not exceed 120 days.

B. A 10CFR50.59 review must be performed for non-intent changes.

C. If the procedure will be used to satisfy Technical Specification requirements then the validation review must include actual performance of the procedure.

D. They require a Writers Guide Review.

DISTRACTOR ANALYSIS:

A CORRECT — As per VPAP-0502, Procedure Process Control, para 4.25 on page 16 and Attachment 3, page 1, block 6.

B INCORRECT — For temporary procedures, this is only requred when involving compensatory measures implemented to support operability.

C INCORRECT — Section 4.26 of VPAP-0502 secifically states that actual performance of the procedure should not be used on procedures which satisfy Technical Specification requirements.

D INCORRECT — They require an editorial review not a writers guide review.

REFERENCES:
1. VPAP-0502, "Procedure Process Control," pages 16, 27, and 100.
2. DNAP-1408, "Operability Determination Program," page 14, para 3.3.3.
3. VPAP-1101, "Test Control," page 15, para d.
4. DNAP-0502, "Dominion Nuclear Procedure Process Control," page 28.
5. VPAP-1403, "Temporary Modifications," pages 20 - 23.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of the process for controlling temporary changes.

K/A MATCH:
- This question matches the K/A in that the operator must know the process for controlling temporary changes in order to provide a correct answer.

64.
Your crew had been on shift for about an hour when you were asked about a Technical Specification limiting condition for operation for planned maintenance that was entered during the last shift.

Which ONE of the following was designed to provide you the information requested with little or no delay?

A. Plan Of The Day

B. Abnormal Status Log.

C.✓ Action Statement Status log.

D. Record of Plant Issues submitted during the last shift.

DISTRACTOR ANALYSIS:

| | |
|---|---|
| A INCORRECT | An LCO would not appear on the POD after so short a period of time. |
| B INCORRECT | The Abnormal Status log does not have Action Statements in it. |
| C CORRECT | The Action Statement Status log has a copy of all the current action statements. |
| D INCORRECT | A Plant Issue (PI) is not required to be submitted if the entry into an LCO was for planned maintenance. |

REFERENCES:
1. OPAP-0005, "Shift Relief & Turnover," Section 6.1.4 on page 6.
2. VPAP-1501, "," Attachments 1 & 2.
3. North Anna Administrative Procedures Exam Bank, question 56, ID 3231, page 57.
4. DNAP-2000, "Dominion Work Management Process," Attachment 9, page 1.

K/A CATALOGUE QUESTION DESCRIPTION:
- Ability to track limiting conditions for operations.

K/A MATCH:
- This question matches the K/A in that it specifically asks the applicant about one tool that is used to track LCOs.

65.
Unit 2 is refueling with a core on-load in progress when the following events occur:

- Annunciator E-C6, SPENT FUEL PIT LO LEVEL actuates.
- A report is received that the refueling cavity level is rapidly decreasing.
- There is one fuel assembly in the cavity.
- Power is lost to the Manipulator Crane.
- There is one fuel assembly in the fuel building transfer canal.

IAW 2-AP-52, "Loss of Refueling Cavity Level During Refueling," which ONE of the following represents the correct sequence of actions in response to the above conditions?

A. Verify lower internals removed from the reactor vessel and adequately shielded, then return the fuel assembly in the cavity to the reactor vessel.

B. Verify lower internals installed in the reactor vessel, then return the fuel assembly in the cavity to the reactor vessel.

C.✓ Return the fuel assembly in the cavity to the reactor vessel manually then evacuate all unnecessary personnel from containment.

D. Evacuate all unnecessary personnel from containment. When power is restored to the manipulator crane then return the fuel assembly in the cavity to the reactor vessel.

<u>DISTRACTOR ANALYSIS</u>:

A INCORRECT — If lower internals are removed from the reactor vessel, cannot place a fuel assembly in the vessel.

B INCORRECT — The order is reversed.

C CORRECT — This is the correct order. The first part is an immediate action step which has a note preceeding is stating that if electrical power is unavailable to the manipulator crane the manual operation may be required.

D INCORRECT — The steps are reversed.

<u>REFERENCES</u>:
1. 2-AP-52, "Loss of Refueling Cavity Level During Refueling," pages 1 & 2.
2. E-C6, SPENT FUEL PIT LO LEVEL.
3. 0-AP-27, "Malfunction of Spent Fuel Pit System," pages 1, & 4.

<u>K/A CATALOGUE QUESTION DESCRIPTION</u>:
- Knowledge of new and spent fuel movement procedures.

<u>K/A MATCH</u>:
- This question matches the K/A in that it applies to a procedure used to respond to an abnormal condition when moving new and spent fuel. The applicant must have knowledge of the procedure in order to answer correctly.

66.
The Auxiliary Building operator must enter radiation areas to gather routine data and perform daily equipment checks. Consequently, this operator has logged onto a RWP designated only for Operations personnel entering the RCA. This RWP includes the following information:

- General Areas 1- 20 mR/hr
- Entry to contaminated areas require coveralls, shoe covers, gloves, hood, etc.
- Digital Alarming Dosimeter setpoints: 150 mR/hr and 50 mR

DETERMINE what type of RWP this is and the HIGHEST dose rate posting that the Auxiliary Building Operator is allowed to enter WITHOUT Rad Protection personnel coverage?

A. Special RWP
CAUTION RADIATION AREA

B. Special RWP
CAUTION HIGH RADIATION AREA

C. Standing RWP
CAUTION RADIATION AREA

D.✓ Standing RWP
CAUTION HIGH RADIATION AREA

DISTRACTOR ANALYSIS:

A INCORRECT This is a Standing RWP and the DAD alarm setpoint is indicative of areas greater than 5 mr/hr. Plausible if candidate interprets the information as a "special" RWP for only designated personnel.

B INCORRECT This is a Standing RWP. Plausible if candidate interprets the information as a "special" RWP for only designated personnel.

C INCORRECT The DAD alarm setpoint is indicative of areas greater than 5 mr/hr. Plausible if the candidate does not understand that the 50 mr setpoint is cummulative, i.e., assumes that it is a dose rate.

D CORRECT

REFERENCES:
1. VPAP-2101N, Radiation Protection Program (North Anna) Section 6.8.5
2.
3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A MATCH:
- Question was written to the second part of this K/A in accordance with the guidance provided in NUREG 1021 Rev.9 Section 401.

67.
A plant worker is unconscious and presumed badly injured in a Very High Radiation Area.

A 50-year-old volunteer member of the rescue team is attempting to reach the man. The rescuer has already received a whole body exposure of 4 REM.

Which ONE of the following identifies the additional allowable exposure the rescue worker may receive in this rescue attempt?

A. 1 REM

B. 6 REM

C. 21 REM

D.✔ No Limit

DISTRACTOR ANALYSIS:

A INCORRECT There is no dose limit to save a life. Plausible because 5 REM is listed in Enclosure limit associated with Emergency Worker Exposure Limits.

B INCORRECT There is no dose limit to save a life. Plausible because 10 REM is listed in Enclosure limit associated with Emergency Worker Exposure Limits.

C INCORRECT There is no dose limit to save a life. Plausible because 25 REM is listed in Enclosure limit associated with Emergency Worker Exposure Limits.

D CORRECT There is no limit to save a life. Note that the injured person is in a Very High Radiation Area which is defined as an area with dose rate in excess of 500 RADS/hr making this a life saving rescue.

REFERENCES:
1. 0-EPIP-20111, Enclosure 1

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A MATCH:
- This question matches the K/A in that the applicant must know both exposure limits as well as permissible levels in excess of those authorized which includes voluntary live-saving limits of which the key is to know that there is no limit.

68.

Within the EOPs at North Anna, action steps that are considered "Immediate Actions" are identified by which ONE of the following?

A. Letters

B. Bullets

C. Bold Circles

D.✓ Brackets

DISTRACTOR ANALYSIS:

| | |
|---|---|
| A INCORRECT | Letters are used to designate sequential steps. |
| B INCORRECT | Bullets are used to designate subtasks taht do not have to be performed in a sequence. |
| C INCORRECT | As per OPAP-0013, a decision making step is an EOP step designated by a bold circle around the step number. |
| D CORRECT | As per DNAP-0509 and OPAP-0002. |

REFERENCES:

1. DNAP-0509, "Dominion Nuclear Procedure Adherence and Usage;" page 15, para 3.6.2.b; page 18, para 3.8.1.a; page 35, para 4.e.1; page 36, para I.
2. OPAP-0013, "Emergency Operating Procedure Decision Steps Pilot Program," page 7, para 6.0.
3. OPAP-0002, "Operations Department Procedures," page 23, para 6.4.4.e.1.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of EOP layout, symbols, and icons.

K/A MATCH:
- This question matches the K/A in that the applicant must have knowledge of the symbols used within the EOPS and their specific meaning.

69.

Unit 2 operating crew entered FR-H.1, "Response to Loss of Secondary Heat Sink," in response to a Red Heat Sink condition. The crew is still progressing through FR-H.1 when critical safety function status tree conditions are reported as follows:

- Subcriticality: Orange path to FR-S.1, "Response to Nuclear Power Generation/ATWS."
- Core Cooling: Green
- Heat Sink: Green
- Integrity: Green
- Containment: Red path to FR-Z.1, "Response to High Containment Pressure."
- Inventory: Green

Which ONE of the following describes the actions the crew should take in response to the conditions given above?

A. Immediately exit FR-H.1 and transition to FR-Z.1.

B. Immediately exit FR-H.1 and transition to FR-S.1.

C.✓ Complete FR-H.1, then transition to FR-Z.1.

D. Complete FR-H.1, then transition to FR-S.1.

DISTRACTOR ANALYSIS:

A INCORRECT — SInce FR-H.1 was entered on a Red path it must be completed unless a higher priority Red path condition is diagnosed.

B INCORRECT — Although the procedure should be immediately exited, the transition to FR-S.1 would be incorrect. As per F-0 Bases, first para on page 15, a red path takes priority over an orange path even if the orange path exists on a higher priority Critical Safety Function.

C CORRECT — Once H.1 has been entered on a Red path it must be completed unless a higher red path condition is diagnosed. See rational for answer "A."

D INCORRECT — The crew should immediatly exit FR-H.1. The transition is also incorrect. See rational for answer "B."

REFERENCES:

1. F-0BG, "Background Information for F-0 Critical Safety Function Status Trees," pages 11, 12, 15, & 16.
2. HE-0BG, "Background Information for E-0 Reactor Trip or Safety Injection," pages 1, 43, & 64.
3. HFR-H1BG, "Background Information for FR-H.1 Response to Loss of Secondary Heat Sink," page 2.
4. North Anna Self-Study Guide for Heat Sink Series, page 6.
5. North Anna Self-Study Guide for Critical Safety Function Trees, page 11.

K/A CATALOGUE QUESTION DESCRIPTION:
- Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

K/A MATCH:
- This question matches the K/A in that the applicant must have knowledge of the bases for prioritizing implementation of emergency procedures in order to prioritize his/her responses which is what the quesiton asks.

70.
The unit supervisor announces to the crew that 1-ES-0.0, "Rediagnosis," must be entered.

Which ONE of the following conditions would require the RO to agree with this course of action (i.e. allow entry into 1-ES-0.0)?

A. 1-E-0, "Reactor Trip or Safety Injection," has not been exited and Safety Injection has not actuated but is required.

B. 1-E-0, "Reactor Trip or Safety Injection," has not been exited and Safety Injection has actuated but is not required.

C. 1-E-0, "Reactor Trip or Safety Injection," has been exited and Safety Injection has not actuated and is not required.

D.✓ 1-E-0, "Reactor Trip or Safety Injection," has been exited and Safety Injection has actuated and is required.

DISTRACTOR ANALYSIS:

A INCORRECT E-0 must have been exited.

B INCORRECT E-0 must have been exited.

C INCORRECT SI must be either required on in service.

D CORRECT This is per the note at the beginning of 1-ES-0.0.

REFERENCES:
1. 1-ES-0, "Re-Diagnosis." Note at beginning.

K/A CATALOGUE QUESTION DESCRIPTION:
- Rediagnosis; Knowledge of the interrelations between the (Reactor Trip or Safety Injection/Rediagnosis) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A MATCH:
- This question matches the K/A in that in order to make the correct decision regarding the procedure to transition to, the operator must have knowledge of the interrelations between the rediagnosis and the various conponents, functions, and control of safety systems, instrumentation, and interlocks. It also tests the direct correlation between the reactor trip or safety injection and rediagnosis. ROs are required to know procedure intry conditions.

71.
Which ONE of the following describes the sequence of actions required to establish bleed and feed heat removal, in accordance with, 1-FR-H.1, "Response to Loss of Secondary Heat Sink?"

A. "Feed" is established first by initiating Main Feed to a S/G, then "Bleed" is established by depressurizing a S/G with an atmospheric steam dump.

B. "Bleed is established first by opening Both Pressurizer PORVs, then "Feed" is established by initiating Safety Injection.

C.✓ "Feed" is established first by initiating Safety Injection, then "Bleed" is established by opening All Pressurizer PORVs.

D. "Bleed is established first by depressurizing a S/G with an atmospheric steam dump, then "Feed" is established by initiating Main Feed to a S/G.

DISTRACTOR ANALYSIS:

A INCORRECT — Feed is established first, however it is with SI flow, and Bleed is with pressurizer PORVs.

B INCORRECT — Beed is established after Feed is established.

C CORRECT — Feed by initiating an SI is established first, and then the Pressurizer PORVs are opened to establish a bleed path.

D INCORRECT — Bleed is established after Feed is established, and not from these sources.

REFERENCES:
1. 1-FR-H.1, "Response To Loss Of Secondary Heat Sink," pages 18 - 24 beginning with step 14 and caution that follows.

K/A CATALOGUE QUESTION DESCRIPTION:
- Loss of Secondary Heat Sink; Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink): Normal, abnormal and emergency operating procedures associated with (Loss of Secondary heat Sink).

K/A MATCH:
- Even though this question does not specifically ask "what are the implications of actions taken" (i.e. emergency operating procedures used), it matches the K/A in that in order for the operator to determine the correct sequence of actions to be taken associated with a loss of secondary heat sink, he/she must understand that the implication to the plant as a result of those actions will be the re-establishment of a means of heat removal in spite of the loss of the secondary heat sink.

72.

Plant conditions are as follows:

- The crew has entered 1-FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition."
- Letdown is in service.
- Normal spray is <u>not</u> available.
- The crew has commenced an RCS temperature soak.
- Shortly after commencing the soak you notice RCS pressure increasing.

Given the above conditions, and IAW 1-FR-P.1, which ONE of the following describes the appropriate response?

A. Open Reactor vessel head vent valves.

B. Use ONE PRZR PORV to reduce pressure.

C. Increase feed to SGs to reduce pressure.

D.✔ Use Auxiliary Spray to reduce pressure.

DISTRACTOR ANALYSIS:

A INCORRECT Although opening of head vents is correct in some cases (C.1, H.1, it is not correct in this situation.

B INCORRECT This would be correct only if letdown was not in service.

C INCORRECT IAW Step 27b)3), perform actions of other guidelines in effect that do not cool down or raise RCS pressure until the RCS temperature soak has been completed.

D CORRECT IAW Step 25 RNO, if normal spray in not available and letdown is in service, then auxiliary spray would be used.

REFERENCES:
1. 1-FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition," Steps 25 & 27 on pages 18 & 20, Attachment 1.
2. North Anna Self-Study Guide for Pressurized Thermal Shock Series, pages 5 - 7.
3. OPAP-0006, "Shift Operating Practices," para 6.13.3 on page 19.

K/A CATALOGUE QUESTION DESCRIPTION:
- Pressurized Thermal Shock; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

K/A MATCH:
- While this procedure contains no actions that could be defined "Immediate Actions" as understood by industry standards (such as North Anna procedures, E-0 & FR-S.1) the K/A was retained since it does resemble the immediate action requirements as called out in OPAP-0006. The applicant would reasonably be expected to respond to the condition presented in the question without reference to procedures and do so immediately, to reduce risk and protect the plant from this RAD risk level.

73.
The crew is performing a natural circulation cooldown in accordance with ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (Without RVLIS)," when the SRO states that his intention is to transition to FR-I.3, "Response to Voids in Reactor Vessel," in order to perform a head venting operation.

As RO, which ONE of the following describes why you would, or would not, support the SRO's intention and why or why not (IAW the bases)?

You recommend to the SRO that the crew __________.

A. should not take this action as it will result in a loss of pressure control.

B.✓ should not take this action as pressure decreases (from venting) will result in a loss of RCS inventory.

C. should take this actions as pressure decreases (from venting) will result in increased core cooling and partial elimination of the steam void.

D. should take this action as it will restore RVLIS indication.

DISTRACTOR ANALYSIS:

A INCORRECT Although this is the correct recommendation, the rational is not IAW the bases.

B CORRECT According to the bases, at no time is it appropriate to make a transition to FR-I.3 and perform a head venting operation as the steam void would not be elimited but would result in more water flashing to steam in the head region, replacing the steam that was vented.

C INCORRECT This action should not be taken for any reason.

D INCORRECT This action should not be taken for any reason.

REFERENCES:
1. HESO4BG, "Natural Circulation Cooldown with Steam Void in Vessel (Without RVLIS) Background Information," pages 2 & 3.

K/A CATALOGUE QUESTION DESCRIPTION:
- Natural Circulation Operations; Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Operations): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

K/A MATCH:
- This question matches the K/A in that asks about RO actions and recommendations related to the ES-0.3 procedure and related to his function as a member of the control room team such that the procedure is not violated and limitations are not exceeded.

74.

Unit 1 was operating at 100% power when a loss of offsite power occurred. The following plant conditions exist:

The crew is currently in 1-ES-0.3, "Natural Circulation Cooldown With Steam Void in Vessel"
The cooldown rate is 50° F/hr and initiation of Reactor Coolant System (RCS) depressurization has begun
To keep pressurizer (PRZR) level within the desired band, an operator establishes charging significantly greater than letdown
The operator now observes PRZR pressure increasing and PRZR level decreasing

The reason this is occurring is because vessel voids are __________.

A.✓ decreasing due to RCS pressure increasing

B. decreasing due to the compression of any non-condensables

C. increasing due to the RCS cooldown

D. increasing due to condensation of steam in the PRZR

Distactor Analysis:

A. Correct. With a void present, the pressurizer will not respond in the normal manner. If letdown < charging, pressurizer pressure will increase, the vessel void will shrink and pressurizer level will decrease.

B. Incorrect. There will be no significant amount of non-condensables. Plausible because there will be some, but the percentage will be insignificant.

C. Incorrect. Plausible if the applicant confuses the effects of cooldown on vessel voids.

D. Incorrect. Plausible if the applicant believes the pressurizer bubble is controlling the void volume in the vessel.

Bank question 60349.

REFERENCES:
1. ES-0.3, "Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS)," steps 3 - 10 on pages 5 - 8.
2. HES03BG, "Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS) Background Information," pages 2, 27, & 30.

K/A CATALOGUE QUESTION DESCRIPTION:
- Natural Circulation with Steam Void in Vessel with/without RVLIS; Knowledge of the reasons for the following responses as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): 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.

K/A MATCH:
- This question matches the K/A in that the proposed actions will result in changes to temperature/pressure conditions the affect of which the applicant must know and undersatnd in order to prevent exceeding operating limitations.

75.
Given the following plant conditions:

- 1-ECA-2.1, "Uncontrolled Depressurization Of All Steam Generators" is in progress.
- The crew is at step: Check if SI flow should be reduced.

Which ONE of the following sets of parameters will be used to determine if SI flow should be reduced in accordance with ECA-2.1?

A. pressurizer level, RCS pressure, AFW flow.

B. RCS pressure, AFW flow, RCS subcooling.

C.✔ RCS subcooling, RCS pressure, pressurizer level.

D. pressurizer level, AFW flow adequate, RCS subcooling.

DISTRACTOR ANALYSIS:

A INCORRECT The EOP directs the operator to check RCS subcooling, RCS pressure, and Pressurizer level.

B INCORRECT The EOP directs the operator to check RCS subcooling, RCS pressure, and Pressurizer level.

C CORRECT The EOP directs the operator to check RCS subcooling, RCS pressure, and Pressurizer level.

D INCORRECT The EOP directs the operator to check RCS subcooling, RCS pressure, and Pressurizer level.

REFERENCES:
1. 1-ECA-2.1, "Uncontrolled Depressurization of All Steam Generators," Step 14.

K/A CATALOGUE QUESTION DESCRIPTION:
- Uncontrolled Depressurization of all Steam Generators; Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A MATCH:
- The question matches the K/A in that the operator must have a thorough integrated knowledge of plant components, functions, and safety system operation and how they will all function together during this event. The question also matches the K/A in that it requires the oerator to have an understanding of the pertinent plant parameters and use them effectively to utilize manual features following an automatic plant response.

76.
Unit 2 initial conditions are as follows:

- 40% power
- Control Bank A and B Group Step Counters at 230 steps
- Both Control Bank C Group Step Counters at 223 steps
- Both Control Bank D Group Step Counters at 95 steps
- All IRPIs are within 6 steps of their associated group step counters
- Rod control in MANUAL
- Bank overlap unit thumbwheel setpoints are:

S1 = 128 S2 = 230 S3 = 256 S4 = 358 S5 = 384 S6 = 486

You observed the following immediately after the reactor operator took a total of four (4) rod steps out (no other operator action has been taken):

- 40% power
- Control Bank A and B Group Step Counters at 230 steps
- Both Control Bank C Group Step Counters at 224 steps
- Both Control Bank D Group Step Counters at 98 steps
- All IRPIs are within 8 steps of their associated group step counters
- 2A-F1, CMPTR ALARM ROD DEV/SEQ, is LIT

Based on the plant conditions listed above, which of the following correctly states the failure that has occurred and the Limiting Condition for Operation (LCO) that must be entered?

A. Control Bank D P/A Converter has failed.
3.1.6, Control Banks shall be limited within the insertion, sequence, and overlap limits specified in the COLR.

B.✓ Bank Overlap Unit has failed.
3.1.6, Control Banks shall be limited within the insertion, sequence, and overlap limits specified in the COLR.

C. Control Bank D P/A Converter has failed.
3 .1.7, The Rod Position Indication system and the Demand Position Indication System shall be operable.

D. Bank Overlap Unit has failed.
3 .1.7, The Rod Position Indication system and the Demand Position Indication System shall be operable.

DISTRACTOR ANALYSIS:

From the given plant conditions, control rod bank overlap following rod withdrawal is 100 steps rather than the previous, COLR required, 102 steps. Failure of the Bank Overlap Unit (BOU) is indicated by the presence of alarm 1A-F1 with the BOU thumbwheels set properly for 102 steps overlap. Failure of Control Bank D P/A converter would not cause control bank rods to step out of sequence, resulting in improper overlap, because its only control function is to send a signal to the logic cabinet to prevent the automatic withdrawal of control bank rods at the C-11 setpoint.

A INCORRECT. Incorrect failure, correct Tech Spec.

The function of the four P/A converters (one for each control bank) are to convert the pulsed output signals from the rod control system logic cabinet (demand position indication) to an analog signal that is used in the rod insertion limit monitoring circuits. The Control Bank D P/A converter also sends a signal to the logic cabinet to automatically stops the withdrawal of Control Bank D rods at the C-11 (near the top of the core). The output of the four P/A converters feeds a bank display unit that contains a digital meter reading of bank demand position for each of the control banks (T-1409A-D located on Vertical Panel 1-1).

This failure is plausible if the applicant fails to recognize improper control bank C and D overlap, believes the P/A converter provides the overlap logic function, or believes that control bank C or D group step counters are inoperable and that the P/A converter inputs to these rather than the vertical display units.

Since the deviation between any IRPI and it's associated Bank Demand Position Indication (group step counters) is less than 12 steps, Bank Demand Position Indication has remained operable per TS 3.1.7. This action is plausible if the applicant believes that the group step counters for control banks C and/or D are reading incorrectly and are inoperable.

B CORRECT Correct failure and correct Tech Spec.

The function of the BOU is to determine, through the master cycler, which rod bank is to be moved by selecting the appropriate slave cycler. Whith the bank selector switch in MANUAL, the BOU will send a signal to the master cycler to overlap the control banks as determined by thumbwheels S1 through S6.

With Control Bank C rods at 224 steps and Control Bank D rods at 124 steps, bank overlap is 100 steps and less than the COLR limit. TS 3.1.6, Condition A, Control bank sequence or overlap limits not met, is applicable and has associated Required Actions with Completion Times of 1 hour and 2 hours.

C INCORRECT Incorrect failure and incorrect Tech Spec.

See A. and B.

D INCORRECT Correct failure, incorrect Tech Spec.

See A. and B.

<u>REFERENCES</u>:

1. NCRODP-65-NA, Rod Control System, 03/02/04
2. North Anna Power Station Self Study Guide for Rod Control System (65), 04/12/05
3. Technical Specification 3.1.6, Reactivity Control Systems, Amendments 231/212
4. Technical Speification 3.1.7, Rod Position Indication, Amendments 231/212
5. 1A-F1, CMPTR ALARM ROD DEV/SEQ, Rev. 1
6. N1C18 Core Operating Limits Report, Rev. 1

<u>K/A CATALOG QUESTION DESCRIPTION</u>:

- Control Rod Drive System; Knowledge of the purpose and function of major system components and controls.

<u>K/A MATCH</u>:

- Ability to answer first part of question (diagnosis of failure) requires an understanding of the purpose and function of a major system component (the BOU).
- Second part of question (ability to select appropriate Technical Specification) was written in order to make the question consistent with SRO-required level of knowledge: 10 CFR 55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

77.

Which one of the following correctly describes the basis for the lower limit (40° F) on RWST water temperature in North Anna Technical Specification 3.5.4, Refueling Water Storage Tank (RWST)?

A. To minimize the amount of boron precipitation in the core until the ECCS is switched from cold leg to hot leg injection following a LOCA.

B. To minimize the occurrence of chloride and caustic stress corrosion on mechanical components following actuation of quench spray.

C. To ensure adequate boron solubility following a MSLB inside containment in order to ensure subcriticality.

D.✓ To ensure post-LOCA containment pressure reduction is NOT excessive in order to ensure adequate energy removal from the break.

DISTRACTOR ANALYSIS:

A Incorrect. Plausible because this is similar to the basis for the maximum RWST boron concentration.

B Incorrect. Plausible because this is one basis for maintaining chemical addition tank NaOH concentration within upper and lower limits.

C Incorrect. Plausible because the upper limit on RWST water temperature and the lower limit on RWST boron concentration are factors in the MSLB analysis.

D Correct. In the ECCS analysis, the quench spray temperature is bounded by the RWST lower temperature limit. If the lower limit on RWST temperature is violated and a LOCA occurs, quench spray further reduces containment pressure, decreasing the rate at which steam can be vented out the break, which decreases the rate of energy removal from the RCS.

REFERENCES:
1. Tech Spec 3.5.4, Refueling Water Storage Tank (RWST)
2. Tech spec 3.5.4 Basis, Refueling Water Storage Tank (RWST)
3. Tech Spec 3.6.6 Basis, Quench Spray (QS) System
4. Tech Spec 3.6.8 Basis, Chemical Addition System

K/A CATALOG QUESTION DESCRIPTION:
- Emergency Core Cooling; Ability to explain and apply all system limits and precautions.

K/A MATCH:
-Question requires knowledge of basis for RWST lower temperature limit.
10CFR55.43(b)(2) Facility operating limitations in technical specifications and their bases.

78.
At 09:58 Unit 1 conditions were as follows:

- Power was 85%, being decreased at a turbine ramp rate of 5% per minute
- Pressurizer pressure 2200 psig and slowly decreasing
- Charging flow 80 gpm
- Letdown flow 85 gpm
- Seal Injection flow 25 gpm
- Total Seal Leakoff flow 5 gpm
- PRZR PORV 1-RC-PCV-1455C indicated mid position and would not close
- PRT level, temperature, and pressure were increasing

Unit 1 was manually tripped at 10:01 and all systems responded normally.

At 10:04 Unit 1 conditions were as follows:

- PRZR PORV 1-RC-PCV-1455C indicated mid position and would not close
- PRZR PORV block valve 1-RC-MOV-1536 was manually closed
- Pressurizer pressure 2000 psig and slowly increasing
- Pressurizer level is slowly trending back to program level
- PRT level, temperature, and pressure were stable

Which of the following correctly describes the Tech Spec LCOs (listed below) that were applicable and NOT met at 09:58 and at 10:04?

3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
3.4.11, Pressurizer Power Operated Relief Valves (PORVs)
3.4.13, Operational Leakage

**References Provided**

| | At 09:58 | At 10:04 |
|---|---|---|
| A.✓ | 3.4.1, 3.4.11, and 3.4.13 | 3.4.11 ONLY |
| B. | 3.4.1, 3.4.11, and 3.4.13 | 3.4.11 and 3.4.13 |
| C. | 3.4.11 and 3.4.13 ONLY | 3.4.11 ONLY |
| D. | 3.4.11 and 3.4.13 ONLY | 3.4.11 and 3.4.13 |

DISTRACTOR ANALYSIS:

TS 3.4.1

At 09:58 Unit 1 is in Mode 1. Pressurizer pressure is less than the COLR limit of 2205 psig. TS 3.4.1 is applicable in Mode 1 (and per TS Note, the Thermal Power Ramp is NOT greater than 5% RTP per minute).

At 10:04 Unit 1 is in Mode 3. TS 3.4.1 is only applicable in Mode 1.

TS 3.4.11

At 09:58, Unit 1 is in Mode 1. PORV 1455C is inoperable (will not close). TS 3.4.11 is applicable in Modes 1-3 and Condition C applies.

At 10:04, Unit 1 is in Mode 3. PORV 1455C is still inoperable and TS 3.4.11 still applies.

TS 3.4.13

At 09:58, Unit 1 is in Mode 1. PORV 1455C is leaking into the PRT at approximately 15 gpm (105-60-30 gpm). Leakage into the PRTis identified leakage (captured and conducted to a collecting tank). TS 3.4.13 is applicable in Mode 1 and the limitation of item c. "10 gpm identified LEAKAGE" applies, as does Condition A.

At 10:04, Unit 1 is in Mode 3. Although TS 3.4.13 is still applicable, the identified leakage from PORV 1455C has been reduced by closing its block valve. RCS LEAKAGE no longer exceeds TS 3.54.13 limits.

A Correct. Correct for both 09:58 and 10:04

B Incorrect. Correct for 09:58, incorrect for 10:04.

C Incorrect. Incorrect for both 09:58 and 10:04.

D Incorrect. Incorrect for both 09:58 and 10:04.

REFERENCES:
1. Technical Specification 3.4.1, Amendments 231/212 **(provide without bases to applicant)**
2. Technical Specification 3.4.11, Amendments 231/212 **(provide without bases to applicant)**
3. Technical Specification 3.4.13, Amendments 231/212 **(provide without bases to applicant)**

K/A CATALOG QUESTION DESCRIPTION:
- Small Break LOCA; Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

K/A MATCH:

Stem provides indications of SBLOCA. Question requires applicant to apply Tech Specs for given plant conditions.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

79.
Unit 1 conditions are as follows:

- 80% power, ramping to 100% at 1%/min.
- Pressurizer pressure is 2210 psig.
- The control switches for all groups of pressurizer backup heaters are red-flagged
- 1B-H8, PRZ BACKUP GROUP HTRS OL TRIP, is LIT

Which ONE of the following indications would require the crew to initiate actions as required by Technical Specification 3.4.9, Pressurizer, AND what is the basis for these actions?

A.✓ Amber indicating light above the Group 1 heater control switch is LIT
Ensure RCS loop subcooling can be maintained during an extended Loss of Offsite Power

B. Amber indicating light above the Group 1 heater control switch is LIT
Prevent rapid pressure rises caused by normal operational perturbations

C. Amber indicating light above the Group 2 heater control switch is LIT
Ensure RCS loop subcooling can be maintained during an extended Loss of Offsite Power

D. Amber indicating light above the Group 2 heater control switch is LIT
Prevent rapid pressure rises caused by normal operational perturbations

DISTRACTOR ANALYSIS:

A Correct. Correct indication, correct basis.

Group 1 heaters are powered by the 1J emergency bus, and therefore fall under Tech Spec 3.4.9 (LCO 3.4.9.b). The amber light indicates that an automatic feeder breaker trip has occurred due to overload as indicated by 1B-H8.

The Tech Spec bases for requiring two operable groups of pressurizer heaters, capable of being powered from an emergency bus, is as stated in the bases document (Applicable Safety Analyses, paragraph 3).

B Incorrect. Incorrect indication, incorrect basis.

The red indicating lights lit are normal indications for the group 1 and 4 heater feeder breakers being closed. These indications, together with the 1B-H8 alarm, indicate that there is an OL trip present on Group 1 pressurizer heaters. Neither group 2 or 5 heaters are powered from an emergency bus, so tech spec 3.4.9 is NOT applicable to these heaters.

The bases statement is also incorrect for any group of pressurizer heaters. This statement is plausible because it is the correct bases for maintaining pressurizer level less than or equal to 93% per tech spec 3.4.9 (LCO 3.4.9.a.)

C Incorrect. Incorrect indication, correct basis. See A and B.

D Incorrect. Correct indication, incorrect basis. See A and B.

REFERENCES:
1. Technical Specification 3.4.9, Amendments 231/212
2. Technical Specification 3.4.9 Bases, Rev. 0
3. NCRODP-74-NA, Pressure Control and Protection System, 9/30/05
4. Self Study Guide for Pressurizer Control and Protection System (74), 04/07/05
5. 1B-H8, PRZ BACKUP GROUP HTRS OL TRIP, Rev. 1

K/A CATALOG QUESTION DESCRIPTION:
- Pressurizer Pressure Control System (PZR PCS); Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

K/A MATCH:
- Question stem and distractors present pressurizer heater control system operating parameters (control switch positions, indicating lights, alarms) and requires examinee to determine which set of parameters is an entry level condition for Technical Specifications.

Second part of (SRO) question requires examinee to correctly determine basis for Technical Specification. 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

80.

Unit 1 conditions are as follows:

- The reactor is at 100% power at End of Life
- Control rods are in Manual with Control Bank D at 218 steps
- The crew must reduce power to 50% within the next six hours

Which ONE of the following correctly describes why the power reduction should be performed using control rods and boration (rather than only boration) and the basis for maintaining Axial Flux Difference (AFD) within Technical Specification limits?

A. During power reduction, the MTC will add more positive reactivity to the top of the core than at the bottom of the core, causing AFD to shift positive.
The limits on AFD ensure that the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_H$) is not exceeded.

B.✓ During power reduction, the MTC will add more positive reactivity to the top of the core than at the bottom of the core, causing AFD to shift positive.
The limits on AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded.

C. During power reduction, the MTC will add more positive reactivity to the bottom of the core than at the top of the core, causing AFD to shift negative.
The limits on AFD ensure that the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_H$) is not exceeded.

D. During power reduction, the MTC will add more positive reactivity to the bottom of the core than at the top of the core, causing AFD to shift negative.
The limits on AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded.

DISTRACTOR ANALYSIS:

A. Incorrect. Correct effect, incorrect basis.

Plausible bases because ($F^N_H$) is a Tech Spec power distribution limit (FQZ is the other).

B Correct. Correct effect, correct basis.

C Incorrect. Incorrect effect, incorrect basis.

D Incorrect. Incorrect effect, correct basis.

REFERENCES:
1. North Anna Technical Specification Bases, 3.2.3, Axial Flux Difference (AFD)

K/A CATALOG QUESTION DESCRIPTION:
- Nuclear Instrumentation System (NIS); Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects on axial flux density of control rod alignment and sequencing, xenon production and decay, and boron vs. control rod reactivity changes.

K/A MATCH:
- Question requires knowledge of effects on axial flux density (AFD) of boron vs. control rod reactivity change and basis for maintaining AFD within tech spec limits. 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

81.
Unit 1 conditions are as follows:

- Unit recovery is in progress following refueling
- RCS pressure is 340 psig
- All loop stop valves are open
- The A RHR train is in operation
- The crew has just completed tagging out B RHR pump to repair a seal leak
- There are no operations in progress that would cause a dilution of the RCS
- RCS loop parameters are as follows:

| | Loop A | Loop B | Loop C |
|---|---|---|---|
| $T_C$ | 153° F | 152° F | 152° F |
| SG NR Level | 12% | 15% | 10% |
| SG water temp | 78° F | 80° F | 85° F |

- The crew is preparing to start the first RCP following vacuum fill of the RCS loops

If the A RH pump trips on ground overcurrent, which one of the following correctly describes the minimum action(s) and the associated completion time(s) necessary to place Unit 1 in compliance with Technical Specifications?

A.✔ Immediately initiate action to restore one RHR loop to operable and place it in operation.

B. Immediately initiate action to fill any SG to at least 17% N/R, then monitor natural circulation for removal of decay heat.

C. Immediately initiate action to fill any SG to at least 17% N/R, and start the associated RCP.

D. Immediately initiate action to restore one RHR loop to operable status, and immediately initiate action to fill any SG to at least 17% N/R, then monitor natural circulation for removal of decay heat.

DISTRACTOR ANALYSIS:

In mode 5 with loops not filled, LCO 3.4.8 requires two RH loops to be Operable and one loop to be Operating. For the stated conditions, the SG tubes have been drained, and because an RCP has not yet been started, air has not been swept from the tops of the tube bundles. So, although vacuum fill of the RCS has been complete, the loops cannot be considered to be filled until an RCP has been started to sweep air from the SG tubes.

A Correct. Correct actions and completion time.

B Incorrect. Incorrect actions, correct completion time. Plausible because these actions would restore a loop to operable per TS-3.4.7, but incorrect because the SG tubes are still air-bound and cannot support natural circulation heat removal.

C Incorrect. Incorrect actions, correct completion time. See B. Also, immediately starting an RCP is not required for the stated plant conditions.

D Incorrect. Incorrect actions, correct completion time. See B and C.

REFERENCES:
1. Tech Spec 3.4.6, RCS Loops - Mode 4
2. Tech Spec 3.4.7, RCS Loops - Mode 5, Loops Filled
3. Tech Spec 3.4.8, RCS Loops - Mode 5, Loops Not Filled
4. 1-OP-5.2, Reactor Coolant Pump Startup and Shutdown, Rev. 36

K/A CATALOG QUESTION DESCRIPTION:
- Loss of RHR System; Ability to apply technical specifications for a system.

K/A MATCH:
- Question presents examinee with a loss of both trains of RH and requires them to select a set of actions to meet technical specification action requirements. Selection of appropriate answer requires knowledge of basis for tech spec note that provides limited exception to an action requirement.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

82.
Unit 1 plant conditions are as follows:

- 80% power.
- Block valve 1-RC-MOV-1535 is closed, with power available, due to seat leakage on PORV 1-RC-PCV-1456 estimated at 10.5 gpm (PORV 1456 is in automatic control).
- 1-RC-PCV-1455C has just spuriously opened and cannot be closed.
- RCS pressure is 2200 psig and slowly decreasing with all pressurizer heaters on.

Which one of the following lists actions that are required to be taken per 1-AP-44, "Loss of Reactor Coolant System Pressure," and Technical Specifications?

**References Provided**

A. GO TO 1-E-0, Reactor Trip or Safety Injection, while continuing with 1-AP-44. Be in Mode 3 within 6 hours and Mode 4 within 12 hours .

B. Close and maintain power to block valve 1-RC-MOV-1536 within 1 hour. Restore pressure to within limits within 2 hours.

C.✔ Close block valve 1-RC-MOV-1536 and remove power within 1 hour. Restore 1-RC-PCV-1455C OPERABLE within 72 hours.

D. Close and maintain power to block valve 1-RC-MOV-1536 within 1 hour. Be in Mode 3 within 6 hours and Mode 4 within 12 hours .

DISTRACTOR ANALYSIS:

With seat leakage in excess of the LCO 3.4.13 limit for identified leakage, PORV 1456 is inoperable and capable of being manually cycled. LCO 3.4.11 condition B applies and action B.1, close and maintain power to associated block valve (1535), has been taken as indicated in the stem. Although block valve 1535 is closed, it remains operable because power is available.

A. Incorrect. Incorrect first action and incorrect second action.

A reactor trip, which would place Unit 1 in Mode 3, is not required by either 1-AP-44 or tech specs (and is not warranted by plant conditions, with RCS pressure significantly above the low pressure reactor trip setpoint and decreasing slowly). The SBLOCA caused by the open PORV can be stopped by closing the associated block valve. 1-AP-44 only directs entry into 1-E-0 if the PORV and its associated block valve cannot be closed.

Unit 1 would not be required to enter Mode 4 unless block valve 1536 could not be closed and deenergized, 2 PORVs are inoperable and NOT capable of being manually cycled, or both block valves were inoperable for more than 2 hours.

B. Incorrect. Correct first and third actions. Incorrect second action and fourth action.

1-AP-44, step 1 RNO (an immediate action) directs closing the associated block valve for any open PORV that cannot be manually closed. Because PORV 1455C is open and cannot be closed, it is inoperable and not capable of being manually cycled. LCO 3.4.11 condition C applies and the required actions are 1) close associated block valve (1536) within 1 hour (1-AP-44 directs this as an immediate action), 2) remove power from associated block valve within 1 hour, and 3) restore PORV (1455C) to operable status within 72 hours.

Neither 1-AP-44 or LCO 3.4.11 direct or imply that PORV 1456 should be reopened. Doing so would reinitiate an RCS leak greater than LCO 3.4.13 allowable identified leakage (10 gpm). This action is plausible if the examinee is concerned about the mitigation of RCS pressure transients with both PORVs isolated or that unisolating PORV 1456 is necessary in order to avoid a forced shutdown due to LCO 3.0.3.

Since one PORV is capable of being manually cycled, entry into Mode 3 is not required.

C. Correct. Correct first action and correct second action. See B.

D. Incorrect. Correct first and second actions (See B.) Incorrect third action.

Since one PORV is capable of being manually cycled, entry into Mode 3 is not required.

References:

1. 1-AP-44, Loss of Reactor Coolant System Pressure, Rev. 17
2. Tech Spec 3.4.11, Pressurizer Power Operated Relief Valves **(provide without bases to applicant).**
3. Tech Spec 3.4.13, RCS Operational Leakage,
4. Tech Spec 3.4.11 Bases, Pressurizer Power Operated Relief Valves (PORVs)

K/A CATALOG QUESTION DESCRIPTION:

027 Pressurizer Pressure Control Malfunction
G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls

K/A MATCH:

Question requires examinee to choose between a reactor trip and closing the block valve to isolate an open pressurizer PORV with one PORV already isolated. Additionally, question requires examinee to correctly determine appropriate tech spec actions based on operability definitions contained in tech spec bases. 10CFR55.43(b)(2).

83.
Unit 1 conditions are as follows:

- A SGTR occurred in the C SG
- Containment pressure is 10 psia
- All containment radiation monitors have remained less than the high alarm setpoint
- Both the A and B trains of RVLIS are inoperable
- The crew has completed 1-E-3, Steam Generator Tube Rupture
- The crew is performing 1-ES-3.1, Post-SGTR Cooldown Using Backfill
- The crew commenced depressurizing the RCS using normal pressurizer spray

- The following plant parameters were observed as the depressurization progressed:

| RCS Pressure | PRZR Level | RCP 1- A/B/C #1 seal d/P |
|---|---|---|
| 460 psig | 40% | 430 psid |
| 420 psig | 42% | 390 psid |
| 380 psig | 41% | 350 psid |
| 340 psig | 41% | 310 psid |
| 300 psig | 41% | 270 psid |

- The average of the 5 highest CETCs remained constant at 395° F throughout the depressurization.

Which ONE of the following correctly describes a disadvantage of the backfill method of post-SGTR cooldown,

AND

Given the conditions above, what is the minimum RCS pressure to avoid entering 1-ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired?"

**Reference Provided**

A. Potential adverse effects on primary system chemistry.
RCS Pressure 420 psig.

B. Potential contamination of intact steam generator feedwater supply.
RCS Pressure 420 psig.

C.✔ Potential adverse effects on primary system chemistry.
RCS Pressure 300 psig.

D. Potential contamination of intact steam generator feedwater supply.
RCS Pressure 300 psig.

DISTRACTOR ANALYSIS

Time 20:00 establishes initial conditions for the depressurization. From conditions in the stem, containment conditions are normal. Saturation pressure for CETC temperature plus the 25 oF required subcooling margin (Psat for 420 °F) is 308.8 psia (323.5 psig).

A. Incorrect. Correct disadvantage. Incorrect time.

RCS backfill involves depressurizing the RCS to drain ruptured SG water back into the RCS for processing. Secondary plant chemistry parameters are different than primary plant chemistry parameters. This may may result in dilution of RCS boron concentration to less than required for adequate shutdown margin in addition to other adverse affects.

At time 20:02, pressurizer level has increased 2%. This is not a rapid increase (and RCS upper head voiding will not occur with RCPs running), is not approaching the 69% upper limit of continuous action step 9, Control Charging and Letdown Flow to Maintain Przr Level (greater than 36% and less than 69%) and is under the control of the operators via control of charging, letdown, and the depressurization rate. This is an expected response for lowering RCS pressure less than ruptured SG pressure. Plausible if examinee believes that a 2% increase in pressurizer level is unexpected and/or unstable.

B. Incorrect. Incorrect reason, incorrect time.

Plausible disadvantage because it is a potential disadvantage of the steam generator blowdown method of post-SGTR cooldown (1-ES-3.2).

Plausible time if examinee believes (based on Step 10 Note) that RCS depressurization must be stopped when RCS pressure decreases to less than 400 psig (note that RCP #1 seal d/P remains above the 200 psid trip limit for all times listed) or incorrectly calculates Psat (psig) for CETC temperature plus 25 °F margin.

C. Correct.

Correct disadvantage (see A.)

RCS pressure remains above Psat for CETC temperature plus 25 °F subcooling. Pressurizer level remains stable and less than 69%. RCP #1 seal d/Ps remain greater than 200 psid.

D. Incorrect.

Incorrect reason (see B). Incorrect time. RCS subcooling for the given CETC temperature is less than 25°F.

<u>REFERENCES</u>:

1. 1-ES-3.1, Post-SGTR Cooldown Using Backfill, Rev. 12, Step 10 (page 8 of 10) (**Provide to applicant**).
2. Steam Tables (**Provide to applicant)**
3. 1-OP-5.2, Reactor Coolant Pump Startup and Shutdown, Rev. 36\
4. NCRODP-38-NA, Reactor Coolant System, 9/30/05
5. Self Study Guide for Procedure 1-E-3 Series, 8/18/05
6. Facility exam bank Emergency Procedures question # 309 (ID: 2742)
7. Background Information for WOG ERG ES-3.1, HP-Rev. 1

<u>K/A CATALOG QUESTION DESCRIPTION</u>:

- Steam Generator Tube Rupture (SGTR); Ability to determine or interpret the following as they apply to a SGTR: Pressure at which to maintain RCS during S/G cooldown.

<u>K/A MATCH</u>:

- Question requires applicant to determine when to stop depressurizing RCS during SG cooldown based on multiple plant parameters. Additionally, question requires knowledge of reason for selecting/not selecting given procedure. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

84.
Unit 1 conditions are as follows:

- RCS temperature is 300° F during a plant heatup following refueling.
- A scaffold worker inadvertently placed a piece of tube-lock across the terminals of battery 1-III.
- The resulting arc melted the cables from the battery to the DC bus, which tripped the 1-III battery charger input breaker to 1-EP-CB-12C, 125VDC bus 1-III.

Which one of the following correctly describes the plant response and subsequent recovery actions in accordance with 0-AP-10, Loss of Electrical Power?

A. Loss of normal letdown capability ONLY.
Place excess letdown in service.
Aligning swing charger 1C-2 to DC bus 1-III WILL restore the bus to OPERABLE in accordance with Tech Spec requirements.

B. Loss of normal letdown capability ONLY.
Place excess letdown in service.
Aligning swing charger 1C-2 to DC bus 1-III 1C-2 will NOT restore the bus to OPERABLE in accordance with Tech Spec requirements.

C. Loss of BOTH normal AND excess letdown capability.
Reduce charging and seal injection flow, and increase RCS sample system flow.
Aligning swing charger 1C-2 to DC bus 1-III WILL restore the bus to OPERABLE in accordance with Tech Spec requirements.

D.✔ Loss of BOTH normal AND excess letdown capability.
Reduce charging and seal injection flow, and increase RCS sample system flow.
Aligning swing charger 1C-2 to DC bus 1-III will NOT restore the bus to OPERABLE in accordance with Tech Spec requirements.

DISTRACTOR ANALYSIS:

A Incorrect. Incorrect effect. Incorrect recovery actions. Incorrect assessment of DC bus operability.

Incorrect effect. Per 0-AP-10, Attachment ~~17~~ **15**, Loss of DC Bus 1-III, loss of DC b us 1-III results in loss of both normal and excess letdown. Plausible because loss of DC bus 1-I results in loss of normal letdown ONLY.

Incorrect recovery actions. Per 0-AP-10, Attachment 15, Loss of DC Bus 1-III, reduce charging/seal injection and increase RCS sample system flow due to loss of all letdown capability. Plausible because loss of DC bus 1-I requires excess letdown be placed in service.

Incorrect assessment of DC bus operability. Per bases of TS-3.8.4, DC Sources - Operating, the battery must be connected to the bus for the bus to be considered operable. Plausible if applicant believes aligning the swing charger to the DC bus restores the bus to fully operable.

B Incorrect. Incorrect effect. Incorrect recovery actions. Correct assessment of DC bus operability.

Incorrect effect and recovery actions. See A.

Correct assessment of DC bus operability. Per bases of TS-3.8.4, DC Sources - Operating, the battery must be connected to the bus for the bus to be considered operable.

C Incorrect. Correct effect. Correct recovery actions. Incorrect assessment of DC bus operability.

Correct effect. Per 0-AP-10, Attachment 15, Loss of DC Bus 1-III, loss of DC bus 1-III results in loss of both normal and excess letdown.

Correct recovery actions. Per 0-AP-10, Attachment 15, Loss of DC Bus 1-III, reduce charging/seal injection and increase RCS sample system flow due to loss of all letdown capability.

Incorrect assessment of DC bus operability. Per bases of TS-3.8.4, DC Sources - Operating, the battery must be connected to the bus for the bus to be considered operable. Plausible if applicant believes aligning the swing charger to the DC bus restores the bus to fully operable.

D. Correct. Correct effect. Correct recovery actions. Correct assessment of DC bus operability.

Correct effect and recovery actions. See C.

Correct assessment of DC bus operability. See B.

REFERENCES:

1. Tech Spec 3.8.4, DC Sources - Operating, and Bases
2. 0-AP-10, Attachment 17, Loss of DC Bus 1-I, Rev. 47
3. 0-AP-10, Attachment 15, Loss of DC Bus 1-III, Rev. 47

K/A CATALOG QUESTION DESCRIPTION:
- Loss of DC Power; Ability to determine or 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.

K/A MATCH:
First part of question requires examinee to determine effects of a loss of DC bus using given conditions. Second part of question requires examinee to determine required DC bus operability per tech specs using given conditions and knowledge of bases. 10CFR55.43(b)(2).

85.

Unit 1 conditions are as follows:

- At 10:00, Unit 1 tripped from 100% power due to a loss of offsite power.
- During performance of 1-ES-0.1, Reactor Trip Response, the crew was unable to open 1-MS-TV-111B, Turbine-Driven AFW Pump Steam Supply Valve.

- At 10:05 a fire occurred in the Motor-Driven Auxiliary Feedwater Pump House.
- The fire is still burning and is confined to the area immediately surrounding 1-FW-P-3B with a potential threat to the 1-FW-P-3A pump.
- The crew completed actions as directed by 1-FCA-6, Motor-Driven Auxiliary Feedwater Pump Room Fire, with the following exception:

  - 1-MS-TV-111B, Turbine-Driven AFW Pump Steam Supply Valve, would not open.

- The crew has not yet been able to enter the MDAFW pump house.

Assuming no other malfunctions, which one of the following correctly describes the current status of the AFW system?

A.✓ 1-FW-P-2 is running and is capable of providing the design flow rate to the A SG. Both motor driven AFW pumps are in PTL.

B. 1-FW-P-2 is running and supplying AFW to ONLY the A SG. 1-FW-P-3A is running and supplying AFW to ONLY the C SG.

C. 1-FW-P-2 is running and is NOT capable of providing the design flow rate to the A SG. Both motor driven AFW pumps are in PTL.

D. 1-FW-P-2 is running and supplying AFW to all SGs. 1-FW-P-3A is running and supplying AFW to all SGs.

DISTRACTOR ANALYSIS:

A Correct. 1-FCA-6 Step 2 RNO directs operators to place both Motor-Driven AFW Pump selector switches to Pull-To-Lock after placing 1-FW-P-2 in operation, regardless of the extent of the fire. Tech Spec 3.7.5 bases states that opening of either trip valve will provide sufficient steam to the steam driven pump to produce the design flow rate.

B Incorrect. Plausible because this is the normal lineup of the AFW pumps.

C Incorrect. Although 1-FW-P-2 is running, per Tech Spec Bases 3.7.5, it is capable of producing the design flow rate with only one of two parallel trip valves open.

D Incorrect. Per 1-FCA-6, 1-FW-P-2 is not aligned to all SGs using 1-AP-22.4, Loss of Both Motor Driven AFW Pumps, until after the fire is extinguished. Further into 1-AP-22.4, 1-FW-P-3A would eventually be realigned to a "normal" alignment, i.e. supplying the 1C SG.

REFERENCES:

1. NCRODP-26-NA, Feedwater System, 08/16/02
2. Tech Spec Bases 3.7.5, Auxiliary Feedwater (AFW) System
3. 1-FCA-6, Motor-Driven Auxiliary Feedwater Pump Room Fire, Rev. 3
4. 1-AP-22.4, Loss of Both Motor-Driven AFW Pumps, Rev. 5

K/A CATALOG QUESTION DESCRIPTION:

- Auxiliary/Emergency Feedwater (AFW) System; Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pump failure or improper operation.

K/A MATCH:

- Question requires knowledge of major actions within procedure addressing potential AFW pump failures due to fire. Question also requires assessment of facility conditions (AFW status with respect to ability to meet design requirements) based on system knowledge and Tech Spec bases. 10CFR55.43(B)(5).

86.
Unit 1 conditions are as follows:

- 100% power
- The following annunciators alarmed 3 minutes ago:

  -1H-B4, Battery Chgr 1-IV Trouble
  -1K-C2, Sta Battery Voltage Trouble
  -1D-C8, Smoke Det Sys Smoke Indication Trouble

- An Operator just reported a fire in Station Battery I-IV
- DC Bus 1-IV voltage is 0 Volts
- The crew just entered 0-FCA-0, Fire Protection - Operations Response

In accordance with 0-FCA-0, which ONE of the following correctly describes the next procedures the Unit 1 crew must implement in response to the above plant conditions?

A. 1-E-0, Reactor Trip or Safety Injection then 1-FCA-2, Emergency Switchgear Room Fire

B.✓ 1-FCA-2, Emergency Switchgear Room Fire then 1-E-0, Reactor Trip or Safety Injection

C. 0-AP-10, Loss of Electrical Power then 1-FCA-2, Emergency Switchgear Room Fire

D. 1-FCA-2, Emergency Switchgear Room Fire then 1-OP-2.2, Unit Power Operation From Mode 1 to Mode 2

DISTRACTOR ANALYSIS:

A Incorrect. Plausible because 1-FCA-2, Step 1 directs a manual trip of Unit 1. 0-FCA-0 does not direct a unit trip and a unit trip would not normally (in the absence of the fire) be required for the given plant conditions.

B Correct. A component (DC bus 1-IV) has been lost in the affected area and the fire is still burning, indicating a loss or imminent loss of the emergency power Safe Shutdown Function function. 0-FCA-0 (Attachment 2) directs entry into 1-FCA-2, Emergency Switchgear Room Fire, under these conditions.

C Incorrect. Plausible because loss of an electrical bus is an entry condition for 0-AP-10. If a fire was not present in the Emergency Switchgear Room, this would be the correct answer for the given plant conditions.

D Incorrect. Plausible because Tech Spec 3.8.9, Distribution Systems - Operating, requires that the Unit be placed in Mode 3 within 6 hours if certain subsystems (located in the Emergency Switchgear Room) cannot be restored to operable status within 2 hours.

REFERENCES:
1. 0-AP-10, Loss of Electrical Power, Rev. 47
2. 1-E-0, Reactor Trip or Safety Injection, Rev. 33
3. 0-FCA-0, Fire Protection, Operations Response, Rev. 9
4. 1-FCA-2, Emergency Switchgear Room Fire, Rev. 23
5. Self Study Guide for Fire Contingency Action Procedures, 12/15/05
6. NCRODP-35-NA, Vital and Emergency Electrical Distribution System, 10/13/03

K/A CATALOG QUESTION DESCRIPTION:
- DC Electrical Distribution; Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

K/A MATCH:
- Question stem presents examinee with symptoms indicating a loss of a portion of the DC electrical distribution system as well as a fire. Examinee must apply knowledge of procedure use and entry conditions in order to select the correct procedure to implement. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

87.
Which one of the following is correct concerning operation of the charging pumps during the response to a fire in the main control room in accordance with 0-FCA-1, Control Room Fire?

A.✓ Only the H bus-powered charging pumps are assumed to be operable, and at least one will be verified running on each unit while transferring charging pump control to the Auxiliary Shutdown Panel.

B. Only the H bus-powered charging pumps are assumed to be operable, but they are placed in PTL prior to evacuating the MCR. A charging pump will only be started if required to emergency borate due to stuck control rods.

C. Only the J bus-powered charging pumps are assumed to be operable, but they are placed in PTL prior to evacuating the MCR. A charging pump will only be started if required to emergency borate due to stuck control rods.

D. Only the J bus-powered charging pumps are assumed to be operable, and at least one will be verified running on each unit while transferring charging pump control to the Auxiliary Shutdown Panel.

DISTRACTOR ANALYSIS:

A. Correct. Correct power supply. Correct charging pump operation.

Power supply: Following a control room fire, only the "H" bus-powered equipment is assumed to be operable, since the "H" EDG/emergency bus have provisions for control room isolation and operation from outside the control room.

Charging pump operation: Correct per 0-FCA-1.

B. Incorrect. Correct power supply (see A). Incorrect charging pump operation. Plausible because this action is correct for post-MCR evacuation per 0-AP-20.

C. Incorrect. Incorrect power supply (see A). Incorrect charging pump operation (see B).

D. Incorrect. Incorrect power supply (see A). Correct charging pump operation (see A).

REFERENCES:
1. Reactor Operator Program Self Study Guide For Fire Contingency Action Procedures (97), 12/15/05
2. 0-FCA-1, Control Room Fire.
3. 1-AP-20, Operation from the Auxiliary Shutdown Panel.

K/A CATALOG QUESTION DESCRIPTION:
- Plant Fire on Site; Ability to determine and interpret the following as they apply to the Plant Fire on Site: Vital equipment and control systems to be maintained and operated during a fire.

K/A MATCH:
- Question requires knowledge of normal power supply to, and requirements for, charging pump operation during the response to a fire in the control room.

88.
Unit 1 conditions are as follows:

- Motor Driven AFW Pump 1-FW-P-3A is out of service for motor replacement
- The following busses are the ONLY 4160 V busses currently energized:
    - 4160 V Emergency Bus 1H
    - 4160 V Station Service Bus 1A
- No RCPs are running
- 1-MS-TV-115, Terry Turbine Trip valve, has tripped and is mechanically bound shut
- The crew transitioned to 1-FR-C.1, Inadequate Core Cooling, due to a Red Path
- ALL SG Narrow Range levels are 0%
- ALL SGs are Intact
- RVLIS full range indication is currently 40% and slowly decreasing
- CETCs are currently 710° F and slowly increasing
- Containment pressure is 15 psia and slowly increasing
- The crew has NOT been able to establish SI flow

Which ONE of the following describes the next set of actions the crew will take to provide core cooling in accordance with 1-FR-C.1 based on the above conditions?

A. Start Motor Driven AFW Pump 1-FW-P-3B AND depressurize all SGs to 120 psig.

B. Start the 1A RCP AND open both PRZR PORVs and block valves.

C. Start Motor Driven AFW Pump 1-FW-P-3B AND maintain total AFW flow greater than 340 gpm until narrow range level is greater than 11% in at least one SG.

D.✔ Depressurize all SGs to atmospheric pressure AND check if SI accumulators should be isolated.

DISTRACTOR ANALYSIS:

A. Incorrect. The power supply to the B AFW pump is bus 1J, which is deenergized. Plausible because procedure flowpath is to depressurize all (intact) SGs to 120 psig (Step 11) after checking intact SG levels in Step 9.

B. Incorrect. RCPs are only started if core exit TCs are greater than 1200 degF AND the loop is available, i.e. NR SG level greater than 11% [22%]. Although TCs will eventually exceed 1200 degF, without AFW, SG levels will not permit RCP start. Plausible if applicant recognizes that no AFW pumps are available but fails to recognize requirement for SG level for forced circulation to remove decay heat.

C. Incorrect. The power supply to the B AFW pump is bus 1J, which is deenergized. Plausible because maintaining AFW greater than 340 gpm until above 11% NR level in one SG is the RNO for Step 9 a) Narrow range level - greater than 11% [22%]

D. Correct. With no AFW flow (and SG inventory insufficient to permit RCP start), core cooling will be by depressurizing SGs to atmospheric conditions, then checking if SI accumulators should be isolated based on hot leg temperatures.

REFERENCES:
1. 1-FR-C.1, Response to Inadequate Core Cooling, Rev. 12

K/A CATALOG QUESTION DESCRIPTION:
- Inadequate Core Cooling; Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Availability of main or auxiliary feedwater.

K/A MATCH:
- Question requires applicant to determine whether or not AFW is available and, based on plant conditions, determine alternate methods of heat removal per Inadequate Core Cooling Procedure.

89.
Plant conditions at 10:00 were as follows:

- Units 1 and 2 are at 100% power
- The Service Water System is in NORMAL alignment for Winter Service Operations
    - 1-SW-P-1A in AUTO
    - 1-SW-P-1B running with 40 psig discharge pressure
    - 2-SW-P-1A in AUTO
    - 2-SW-P-1B running with 42 psig discharge pressure
    - Both Service Water headers are aligned for full bypass flow
    - All CCHX are in service

At 10:01, 2-SW-P-1B tripped due to ground overcurrent. The crew entered 0-AP-12, Loss of Service Water, and started 1-SW-P-1A to restore flow to Supply Header 1.

Service Water pump status at 10:06 was as follows:

- 1-SW-P-1A running with 35 psig discharge pressure
- 1-SW-P-1B running with 40 psig discharge pressure
- 2-SW-P-1A in AUTO
- 2-SW-P-1B in Pull-to-Lock

Which ONE of the following correctly describes the action(s) to take to achieve the required pump discharge pressure in accordance with 0-OP-49.6, Service Water System Throttling Alignment, and basis for the minimum pressure?

Maintain at least 54 psig at the discharge of ________________.

A. 1-SW-P-1A and -1B, by throttling SW Bypass MOVs 1-SW-MOV-223A and -223B, to ensure that design flows to the RSHXs are achieved following an accident.

B.✓ 1-SW-P-1A and -1B by throttling Unit-1 and Unit-2 CCHX SW Discharge Valves in order to ensure that design flows to the RSHXs are achieved following an accident.

C. 1-SW-P-1A ONLY by throttling Unit-1 CCHX SW Discharge Valves ONLY in order to prevent pump run-out. 1-SW-P-1B already meets minimum discharge pressure requirements.

D. 1-SW-P-1A ONLY by throttling Unit-1 CCHX SW Discharge Valves ONLY in order to ensure that design flows to the RSHXs are achieved following an accident. Supply Header 2 already meets minimum system pressure requirements.

DISTRACTOR ANALYSIS:

A Correct basis, incorrect actions.

TS 3.7.8 A.1 action statement specifically requires throttling SW to the CCHXs. Additionally, 0-OP-49.6, Precaution and Limitation 4.3, prohibits using SW Bypass Valves for satisfying the 54 psig throttling requirement. Plausible because P&L 4.3 also states that these valves can be throttled and describes a normal method of throttling these valves.

B Correct.

With both units in Mode 1 and one SW pump inoperable, Tech Spec 3.7.8 Required Action A.1 requires throttling SW System flow to CC heat exchangers (within 72 hours). Bases for 3.7.8 A.1 state that this is necessary to ensure that design flows to the RS HXs are achieved following an accident in the event of a single failure disabling a(nother) SW pump. 0-OP-49.6 Precaution and Limitation 4.11.2 states that, in order to satisfy 3.7.8 with only three operable SW pumps and both SW headers in service, SW flow must be throttled at the CCHXs so that each SW pump discharge pressure is at least 54 psig.

C Incorrect. Partially correct actions. The discharge pressure on both operating SW pumps must be at least 54 psig to meet design basis safety analyses. See B.

D. Correct actions, and basis for 1-SW-P-1A, but not for Supply Header 2. See B. Plausible if applicant believes that, since Supply Header 2 contains 2 operable pumps, the 54 psig minimum pressure does not apply to header 2.

REFERENCES:
1. 0-OP-49.6, Rev. 15, Service Water System Throttling Alignment
2. Tech Spec 3.7.8, Service Water (SW) System and Bases
3. NCRODP-13-NA, Service Water System, 04/21/05
4. Reactor Operator Self Study Guide for Service Water System, 04/07/05

K/A CATALOG QUESTION DESCRIPTION:
- Service Water System (SWS); Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS.

K/A MATCH:
- Question requires applicant to select correct procedure action to restore SW header paramaters to meet accident analysis assumptions following a partial loss of SW (pump trip). 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

90.
Plant conditions are as follows:

- Operators were performing a Fire Protection surveillance when a large leak developed directly underneath post indicating valve 1-FP-73, loop isolation valve to Unit 2 yard loop.

- Operators took the following actions to isolate the leak:

  - Closed 1-FP-81, Construction Side Fire Protection Supply Header Isolation Valve
  - Closed 1-FP-53, Loop Isolation Valve to Unit 1 Yard Loop
  - Placed Local Test Control Switch for 1-FP-P-2, Diesel Driven Fire Pump to OFF

- All other portions of the Fire Protection system remain unaffected and are operable.

Which ONE of the following correctly describes the required actions and associated minimum completion times required to satisfy TRM 7.1, Fire Suppression Systems, for the plant conditions list above?

**References Provided**

A. Immediately open 1-FP-246, warehouse loop cross-tie.
Within 1 hour, establish an hourly fire watch at hose houses B & C and the SWPH.
Within 12 hours, establish a once per shift fire watch at the AGFOST and the SW Chem. Add. Bldg.

B. Within 1 hour, establish an hourly fire watch at the SWPH.
Within 12 hours, route an equivalent capacity fire hose from either hose house A or D to hose houses B & C.
Within 12 hours, establish a once per shift fire watch at the AGFOST and the SW Chem. Add. Bldg.

C. Immediately open 1-FP-246, warehouse loop cross-tie.
Within 1 hour, establish an hourly fire watch at hose houses B & C, and route an equivalent capacity fire hose from either hose house A or D to hose houses B & C.
Within 12 hours, establish a once per shift fire watch at the AGFOST and the SWPH.

D.✓ Within 1 hour, establish an hourly fire watch at the SWPH.
Within 1 hour, route an equivalent capacity fire hose from either hose house A or D to hose houses B & C.
Within 12 hours, establish a once per shift fire watch at the AGFOST and the SW Chem. Add. Bldg.

DISTRACTOR ANALYSIS:

Conditions in the stem affect the FP system as follows:

- Diesel Fire Pump is inoperable
- Service Water Pump House North wall hose cabinet is inoperable
- Foam system for the above ground fuel oil tank is inoperable
- Fire protection to the Oil Pump House and Oil Storage Tanks is isolated
- Fire protection to Hose Station B is isolated
- Fire protection to Hose Station C is isolated
- Fire protection to the SW Chem. Add. Bldg. is isolated

A Incorrect.

Opening 1-FP-246 is not required. No total system is inoperable and no pressure maintenance system is inoperable. Additionally, opening 1-FP-246 does nothing to restore FP to the isolated portion of the system. Plausible if applicant does not understand FP pressure maintenance.

Incorrect for Hose House C. Plausible if applicant believes Hose House C is covered by 7.1.5.

Correct for Service Water Pump House and Oil Pump House/Storage Tanks only.

B Incorrect. Correct for Service Water Pump House and Oil Pump House/Storage Tanks. Incorrect for Hose Station C. Plausible if applicant believes Hose Station C falls under 7.1.8.

C Incorrect. Correct for Hose Station C and Oil Pump House/Storage Tanks. Incorrect for Service Water Pump House. Plausible if applicant believes the SWPH falls under 7.1.8 (See D for correct answer)

D Correct.

The Hose Rack at the Service Water Pump House is a Primary station per Table 7.1.5-1. Condition A is applicable and Required Action A.1.2.1 has a completion time of 1 hour.

Hose Station C is a Primary yard hydrant/hose house per Table 7.1.6-1. Condition A is applicable and Required Action A.1 has a completion time of 1 hour.

The fixed foam suppression system for the above ground fuel oil tank is item 4.a in Table 7.1.8-1 and is inoperable without FP water (facility verify). Condition B is applicable and Required Action B.1.1 has a completion time of 12 hours. (Facility verify any hose stations within oil pump house are not more limiting).

REFERENCES:
1. TRM 7.1, Fire Suppressioin Systems, subsections 7.1.1, 7.1.5, 7.1.6, 7.1.8, Table 7-1 (**Provide to applicants**)
2. NCRODP-06-NA, Fire Protection System, 1/11/06
3. Technical Drawings 11715-FB-101A and 41B (**Provide to applicants**)

K/A CATALOG QUESTION DESCRIPTION:
- Fire Protection System (FPS); Ability to (a) predict the impacts of the following malfunctions or oprations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Manual shutdown of the FPS.

K/A MATCH:
- Stem provides conditions such that a portion of the FP system was manually shutdown and requires examinees to use TRM to mitigate consequences of this operation. 10CFR55.43.(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

91. Plant conditions are as follows:

- A fuel handling accident occurred that resulted in severe fuel clad damage.
- An Alert was declared 90 minutes ago and the TSC was activated.
- Two individuals responding to the fuel handling accident were just contaminated in the Unit 2 containment.
- One of the contaminated individuals is injured and must be transported offsite for treatment.
- The control room is performing 0-AP-51, Personnel Injury - Operations Response.

Which ONE of the following correctly describes the actions the Shift Manager must take for the contaminated injured person in accordance with 0-AP-51?

A.✔ Have the TSC initiate EPIP-5.01,Transportation of Contaminated Injured Personnel.

B. Coordinate offsite transportation with Sheriff's Dispatcher.

C. Implement EPIP-5.01, Transportation of Contaminated Injured Personnel, from the Main Control Room.

D. Coordinate the offsite transportation with the receiving hospital.

DISTRACTOR ANALYSIS:

A Correct. With the TSC activated (normally 60 minutes maximum), the TSC assumes responsibility for transportation of injured contaminated personnel using EPIP-5.01.

B Incorrect. Plausible because this the correct answer if the TSC is not activated is to "Coordinate offsite transportation with security." Responsibility for coordinating offsite transportation rests with the TSC. Additionally, per EPIP-5.01 Step 3, security would be responsible for coordinating with local law enforcement.

C Incorrect. Although entry conditions for this procedure are met, per 0-AP-51, responsibility for implementing it rests with the TSC, once activated.

D Incorrect. Plausible because HP procedures direct a Health Physics Technician to accompany the ambulance, but the RP Supervisor is not responsible for coordinating offsite transportation.

REFERENCES:
1. 0-AP-51, Personnel Injury - Operations Response, Rev. 6
2. EPIP-5.01, Transportation of Contaminated Injured Personnel, Rev. 11

K/A CATALOG QUESTION DESCRIPTION:
- Conduct of Operations; Ability to supervise and assume a management role during plant transients and upset conditions.

K/A MATCH:
- Question requires applicant, in role of Shift Manager, to determine proper course of action to obtain aid for a contaminated injured person following activation of TSC. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

92.
Plant conditions are as follows:

- Power Range NI channel N-42 became inoperable 2 days ago.
- The In-Core Detection Subsystem was inoperable.
- 40 hours ago, Unit 1 power was reduced to approximately 74%.
- N-42 is in TRIP and Operations has just completed 1-PT-23, Attachment 2, QPTR Hand Calculation. **(Provided)**

- The In-Core Detection subsystem has been returned to service and a flux map was performed to verify the results of the QPTR hand calculation.

Which one of the following is correct concerning the actions required/allowed by Technical Specifications?

**References Provided**

A. Reduce power. Maximum power may be > 50% but must be < 70% RTP.

B. Adjust operable PRNIs to equal calorimetric power.

C. Reduce power. Maximum power must be < 50% RTP.

D.✔ Power can be increased to a maximum value of 88.6% RTP.

DISTRACTOR ANALYSIS:

A Incorrect. Not required unless PRNIs must be adjusted (see B) per 1-PT-24.1. Plausible because this action would be taken before adjusting PRNIs, if adjustment was required.

B Incorrect. Adjustment of PRNIs is not required unless Feed Flow Calorimetric Power exceeds the PRNI channels by more than 2%. Per the conditions in the stem, all operable PRNI channels exceed the Feed Flow Calorimetric. Per 1-PT-24.1, Step 6.2, adjusting the PRNIs is optional as determined by the SRO.

C Incorrect. Plausible if applicant incorrectly determines that AFD is outside of allowable band.

D Correct. QPTR using operable PRNIs is 1.03. Per Tech Spec 3.2.4 Required Action A.1, thermal power must be reduced greater than or equal to 3% for each 1% of QPTR above 1.00. (1.03-1.02) = 1%, so a 3% power reduction from 74.1% is required.

REFERENCES:
1. 1-PT-24.1, Calorimetric Heat Balance (Computer Calculation) Attachment 2, Rev. 25 **(provide completed copy to applicants)**
2. North Anna Technical Specification 3.2.3 Axial Flux Difference (AFD)
3. Core Operating Limits Report N1C18/LM COLR, Figure 3.2-2, Axial Flux Difference Limits
4. North Anna Technical Specification 3.2.4 Quadrant Power Tilt Ratio (QPTR) **(provide without bases to applicants)**
5. 1-PT-23, Quadrant Power Tilt Ratio Determination, Rev. 27
6. Reactor Data Book - Normalized PRNI Currents

K/A CATALOG QUESTION DESCRIPTION:
- Conduct of Operations; Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A MATCH:
- Question provides reactor heat balance and power range instrumentation readings. Applicant must evaluate data and choose a course of action. 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

93.
Plant conditions are as follows:

- Unit 1 is at 100% power.
- The low temperature alarm for 1-HT-HTT-ET-106N, BIT outlet piping heat trace, locked in.
- Temperature was verified to be low.
- The redundant circuit was placed in service and IV'd per 1-AR-33.
- A troubleshooting plan was developed to determine the extent of the problem with the normal circuit.
- The troubleshooting plan WILL require the normal circuit to be re-energized.
- The troubleshooting plan was evaluated in accordance with VPAP-2802, Notifications and Reports, and there is NO potential for causing a reportable event.

In accordance with DNAP-2000, Dominion Work Management Process, this troubleshooting activity is considered to be _________.

**Reference Provided**

A. risk level 1

B.✔ risk level 2

C. risk level 3

D. risk level 4

DISTRACTOR ANALYSIS:

A Incorrect. Re-energizing HT circuit means it is no longer removed from service, so is therefore at least risk level 2.

B Correct. HT circuit is placed in service, but will not result in unit transient, reactivity change, or reportable event.

C Incorrect. See A & B.

D Incorrect. See A & B.

REFERENCES:
1. DNAP-2000, Work Management Process, Rev. 4 **(Att. 4, pg. 5 of 8 provided to applicants)**

K/A CATALOGUE QUESTION DESCRIPTION:
- Equipment Control; Knowledge of the process for managing troubleshooting activities.

K/A MATCH:
- Question requires knowledge of level of approval necessary to expand the boundaries of troubleshooting activities, based on risk level.

94.

TS-3.1.9, Physics Test Exceptions - Mode 2, allows the number of required channels for LCO 3.3.1, Reactor Trip System Instrumentation, Functions 2 (PR Neutron Flux), 3 (PR Neutron Flux Rate), and 18d (PR Neutron Flux, P-10), to be reduced to "3" required channels.

Which one of the following correctly describes the basis for allowing "3" required channels for the functions described above?

A. To permit setpoint adjustment of the channel following Physics Testing.

B. To permit a PRNI to be connected to the reactivity computer for QPTR determination.

C.✓ To permit a PRNI to be connected to the reactivity computer for the Differential Boron Worth Test.

D. To permit calibration of the channel during the Bank Worth Test - Rod Swap Method.

DISTRACTOR ANALYSIS:

A Incorrect. Plausible because Tech Spec 3.3.1 allows bypassing a channel, under certain conditions, to permit setpoint adjustment.

B Incorrect. See C. Plausible because PRNI's are used for QPTR determination.

C Correct. The Differential Boron Worth Test is conducted by determining the change in equilibrium boron concentration at different rod bank positions. The reactivity change is measured using a reactivity computer, which requires bypassing a PRNI.

D Incorrect. PRNI channels are not calibrated during low power physics testing. Plausible because the Bank Worth Test is a physics test (and does NOT use the reactivity computer).

REFERENCES:
1. North Anna Technical Specification Bases 3.1.9, Physics Tests Exceptions - Mode 2

K/A CATALOG QUESTION DESCRIPTION:
- Equipment Control; Knowledge of the process for determining the internal and external effects on core reactivity.

K/A MATCH:
- Question requires knowledge of process for conducting Tech Spec Physics Testing, including basis for allowing bypass of Tech Spec instrumentation. 10CFR55.43(b)(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

95.
Unit 1 has just entered Mode 5 for a mid-cycle maintenance outage.

Which ONE of the following correctly describes how the containment must be equalized to atmospheric pressure in accordance with 1-OP-21.2, Containment Purge, and a condition that would result in automatic isolation of containment purge once established?

A. Equalize using 1-HV-MOV-100B, Cont Purge Supply Valve.
Containment Gaseous Radiation Monitor high-high alarm.

B. Equalize using 1-HV-MOV-100B, Cont Purge Supply Valve.
Ventilation vent B Radiation Monitor high-high alarm.

C.✔ Equalize using 1-HV-MOV-102, Cont Purge Relief Valve.
Containment Gaseous Radiation Monitor high-high alarm.

D. Equalize using 1-HV-MOV-102, Cont Purge Relief Valve.
Ventilation vent B Radiation Monitor high-high alarm.

<u>DISTRACTOR ANALYSIS</u>:

A Incorrect. Incorrect valve, correct radiation monitor.

Per 1-OP-21.2, opening 1-HV-MOV-100B to break containment vacuum is forbidden in order to prevent possible collapse of purge system duct work. Plausible because 1-HV-MOV-100B is subsequently opened for a containment purge.

Correct radiation monitor. See C.

B Incorrect. Incorrect valve, incorrect monitor. See A and D.

C Correct. Correct valve, correct radiation monitor.

Per 1-OP-21.2, if the containment is not at atmospheric pressure, then operators are to open 1-HV-MOV-102 to raise containment to atmospheric pressure (Step 5.1.8).

Per 1-OP-21.2, Containment Gaseous Radiation Monitor high-high alarm automatically isolates purge.

D Incorrect. Correct valve, incorrect radiation monitor. See C. Plausible because containment purge discharges to Ventilation vent "B" stack.

<u>REFERENCES</u>:
1. 1-OP-21.2, Containment Purge, Rev. 28

<u>K/A CATALOG QUESTION DESCRIPTION</u>:
- Radiation Control; Knowledge of the process for performing a containment purge.

<u>K/A MATCH</u>:
- Question requires knowledge of process (procedure) for equalizing containment pressure with atmospheric pressure when initiating a containment purge. One answer choice (distractor) may result in system damage. Question also requires knowledge of radiological conditions that invalidate the release permit required to conduct the purge. 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

96.

Unit 1 conditions are as follows:

- A large-break LOCA occurred
- The crew is performing 1-ES-1.3, Transfer to Cold Leg Recirculation, Step 12, Verify Containment Sump Boron Concentration
  - RWST level is 3%
  - Recirculation was established using ONLY the B Train LHSI Pump
  - The A Train LHSI Pump is in P-T-L because 1-SI-MOV-1860A, LHSI Pump Suction From Containment Sump, can NOT be opened
- All running SI/RS pumps have normal amps, flows, and discharge pressures
- The STA informs the SRO that the core cooling CSF status tree is ORANGE, and that all other trees are either YELLOW or GREEN

- While the crew was conducting a transition brief, the 1J 4160V Emergency Bus deenergized.

Which ONE of the following procedures is the crew is required to implement NEXT in accordance with DNAP-0509, Dominion Nuclear Procedure Adherence and Usage?

A. 1-ES-1.3, Attachment 3, Containment Sump Screen Blockage

B.✔ 1-FR-C.2, Response to Degraded Core Cooling

C. 1-ECA-1.1, Loss of Emergency Coolant Recirculation

D. 0-AP-10, Loss of Electrical Power

DISTRACTOR ANALYSIS:

A Incorrect. Per Continuous Action Page for 1-ES-1.3, indications of sump blockage are not present. Plausible because Step 2 of Attachment 3 directs transition to 1-ECA-1.1 if both Low-Head SI Pumps are NOT available.

B Correct. Neither condition of 1-ES-1.3 Step 1 Note is applicable (i.e. the crew has completed Step 8 and sump blockage has not occurred) so FRs should be implemented per DNAP-0509, which states, "the Recovery Procedure in progress shall be suspended and the FR required by the ORANGE path shall be performed."

C Incorrect. There is no provision in 1-ES-1.3 to transfer to 1-ECA-1.1 following completion of step 8 unless Attachment 3, Containment Sump Screen Blockage, is performed. Per the Continuous Action Page, containment sump screen blockage does not exist. Plausible because 1-ECA-1.1 is designed to restore emergency coolant recirculation (Step 1 RNO directs restoration of electrical power) and is directed per Step 8 RNO (the crew is performing Step 12 and Step 8 is NOT a continuous action step).

D Incorrect. 1-ES-1.3 does not direct transition to 0-AP-10 ( or restoration of electrical power in general) at any step. Plausible because this action would address the cause of the loss of emergency coolant recirculation.

REFERENCES:
1. 1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 17
2. DNAP-0509, Dominion Nuclear Procedure Adherence and Usage, Attachment 5 (EOP/AP Usage)

K/A CATALOG QUESTION DESCRIPTION:
- Emergency Procedures/Plan; Knowledge of EOP implementation hierarchy and coordination with other support procedures.

K/A MATCH:
- Question requires applicant to choose from a number of potentially viable procedures to address plant conditions and apply implementation hierarcy to select the correct procedure IAW plant guidelines. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

97.
Unit 1 conditions are as follows:

- Unit 1 has been operating at 100% steady state power for 29 days.
- Following a lamp check of the MCB annunciators, 1-A-H7, AFD Monitor, remained lit and would not clear.
- Benchboard and computer point values for PRNI Delta Flux, and Delta Flux Limits from the Reactor Data Book (RDB) are as follows:

| | Benchboard | Computer | Limits |
|---|---|---|---|
| N41 AFD | N41C: -3.1 | U0976: -3.10 | -12.0/+6.0 |
| N42 AFD | N42C: -3.2 | U0977: -3.15 | -12.0/+6.0 |
| N43 AFD | N43C: -3.2 | U0978: -3.25 | -12.0/+6.0 |
| N44 AFD | N44C: -3.1 | U0979: -3.15 | -12.0/+6.0 |

- All other Main Control Room annunciators are NOT LIT.
- There are no error messages displayed on PCS and there are no Controller or Interface failures on the System Status display.

Which ONE of the following describes the correct required actions for the given plant conditions?

A.✓ Declare 1-A-H7 INOPERABLE.
Submit a priority one Work Request for 1-A-H7.

B. Declare the Unit 1 Plant Computer INOPERABLE.
Enter 1-AP-42.1, Loss of Unit 1 Plant Computer System (PCS).

C. Declare 1-A-H7 INOPERABLE.
Perform 1-PT-21.1, Reactor Core Flux Mapping.

D. Continuously monitor Unit 1 PCS points U0976, U0977, U0978, and U0979.
Submit a priority one Work Request for 1-A-H7.

DISTRACTOR ANALYSIS:

A Correct.

Per Tech Spec 3.2.3, Axial Flux Difference, Bases, the AFD Monitor Alarm (1-A-H7) is required per SR 3.2.3.1. Since the alarm is lit due to an equipment malfunction, it will no longer function to indicate AFD is outside of COLR limits (see 1-A-H7, AFD Monitor).

Per OPAP-0006, Shift Operating Practices, Rev. 7, step 6.7.3, a Priority 1 Work Request is required.

B Incorrect. Plausible because the AFD Monitor program runs on the Plant Computer. The Plant Computer is not inoperable as indicated by the absence of error messages.

C Incorrect. Althought the AFD Monitor alarm is inoperable, neither Tech Spec 3.2.3 or 1-A-H7 reference or require 1-PT-21.1 or flux maps.

D Incorrect. A priority 1 work request is required. Additionally, continuous monitoring is not required. 1-A-H7 references 1-PT-20.1, Axial Flux Difference (AFD) - Weekly and Special. Rev. 17 of 1-PT-20.1 specifies hourly monitoring and logging of AFD using benchboard readings. Plausible because inoperative alarms may require some form of compensatory monitoring per OPAP-0006.

REFERENCES:
1. 1-A-H7, Rev. 3, AFD Monitor
2. OPAP-0006, Rev. 7, Shift Operating Practices
3. North Anna Technical Specification 3.2.3, Axial Flux Difference (AFD) Bases
4. 1-PT-20.1, Rev. 17, Axial Flux Difference (AFD) - Weekly and Special
5. 1-AP-42.1, Rev. 13, Loss of Unit 1 Plant Computer System (PCS)

K/A CATALOG QUESTION DESCRIPTION:
- Emergency Procedures/Plan; Knowledge of the process used to track inoperable alarms.

K/A MATCH:
- Question requires applicant to select correct course of action for an inoperable alarm required by technical specifications using knowledge of tech spec and bases, Abnormal Procedure entry conditions, and Operating department administrative procedures regarding work priority level determination.
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

98. Which ONE of the following correctly describes the most probable location of a Design Basis LOCA outside of containment,

AND

What is the Emergency Operating Procedure mitigating strategy assuming the leak cannot be isolated in accordance with 1-ECA-1.2, LOCA Outside of Containment?

A. LHSI Pump Cold Leg Injection piping upstream of isolation valves 1-SI-MOV-1890C, D. Sequentially reduce SI flow and re-align the plant to the pre-SI configuration.

B. LHSI Pump Hot Leg Injection piping upstream of isolation valves 1-SI-MOV-1890A, B. Sequentially reduce SI flow and re-align the plant to the pre-SI configuration.

C.✔ LHSI Pump Cold Leg Injection piping upstream of isolation valves 1-SI-MOV-1890C, D. Increase/conserve RWST level, cooldown/depressurize the RCS, and establish RHR cooling.

D. LHSI Pump Hot Leg Injection piping upstream of isolation valves 1-SI-MOV-1890A, B. Increase/conserve RWST level, cooldown/depressurize the RCS, and establish RHR cooling.

DISTRACTOR ANALYSIS:

A Incorrect. Correct location, incorrect strategy.

Components between the RCS and the low pressure LHSI Pump cold leg injection piping include two check valves and normally open isolation valve MOV-1890C(D).

Incorrect strategy. Plausible because if the LOCA was isolable, transition would be from 1-ECA-1.2 to 1-E-1, Loss of Reactor or Secondary Coolant, and then 1-ES-1.1, SI Termination. The major actions listed in the distractor are those for 1-ES-1.1.

B Incorrect. Incorrect location, incorrect strategy.

Location - Components between the RCS and the low pressure LHSI Pump hold leg injection piping include three check valves and normally closed isolation valve MOV-1890A(B). A LOCA through the upstream hot leg injection piping is less likely than through the cold leg piping due to the addition of an additional in-series check valve and because the upstream isolation valves are normally closed.

Strategy - See A.

C Correct. Correct location, correct strategy.

Location - See A and B.

If the LOCA is not isolable, trasition will be from 1-ECA-1.2, LOCA Outside of Containment, to 1-ECA-1.1, Loss of Emergency Coolant Recirculation. Without emergency coolant recirculation capability, core cooling will be prolonged by delaying RWST depletion and will be achieved by establishing RHR cooling. The objectives listed are for 1-ES-1.2, Post LOCA Cooldown and Depressurization.

D Incorrect. Incorrect location, correct strategy.

See A, C.

REFERENCES:
1. Self-Study Guide for ECA-1 Series, 9/23/05
2. 1-ECA-1.2, LOCA Outside Containment, Rev. 5
3. Self-Study Guide for Procedure 1-E-1 Series, 8/18/05

K/A CATALOG QUESTION DESCRIPTION:
- LOCA Outside Containment; Knowledge of symptom based EOP mitigation strategies.

K/A MATCH:
- Question requires applicant knowledge of bases for LOCA Outside Containment EOP and subsequent mitigation strategy based on ability to isolate leak. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

99.

Unit 1 conditions are as follows:

- All three PRZR safety valves were declared inoperable while the unit was in Mode 1.
- Cooling down to Mode 4 in accordance with Technical Specifications.
- Shortly after entering Mode 2, power was lost to all three 4160 Volt Station Service busses.
- Trip valve 1-CC-TV-101B, RCPs thermal barrier return inside isolation, has failed closed.
- Both trains of the RVLIS are inoperable.
- 1-ES-0.2A, Natural Circulation Cooldown with CRDM Fans, is in progress at Step 15, Check for full reactor vessel and the crew observes the following:
    - RCS hot leg temperatures are 520° F and slowly decreasing
    - RCS cold leg temperatures are 500° F and slowly decreasing
    - Pressurizer pressure is 1700 psig and stable
    - Pressurizer level is 28% and slowly decreasing
    - Charging flow is 94 gpm and stable
    - Letdown flow is 100 gpm and stable
    - Seal injection flow to each RCP is 3 gpm and stable
    - Seal leakoff flow from each RCP is 1 gpm and stable
- Power was just restored to 4160 Volt Station Service Bus 1C

Which ONE of the following correctly describes the NEXT action the crew must perform in accordance with 1-ES-0.2A?

**References Provided**

A. Start the C RCP using 1-OP-5.2, Reactor Coolant Pump Startup and Shutdown, THEN go to 1-OP-3.2, Unit Shutdown From Mode 3 to Mode 4.

B.✓ Remain in 1-ES-0.2A, Natural Circulation Cooldown with CRDM Fans, AND perform Step 16, Check Feedwater Status.

C. Remain in 1-ES-0.2A, Natural Circulation Cooldown with CRDM Fans, AND repressurize RCS within limits of Attachment 1.

D. Go to 1-ES-0.4, Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS).

DISTRACTOR ANALYSIS:

A Incorrect. 1-OP-5.2, Initial Condition 3.3 is "Component Cooling is in service to the RCPs). With 1-CC-TV-101B closed, initial conditions for starting an RCP are not met. Plausible because 1-ES-0.2A, Step 1, Try to Start One RCP, is a continuous action step and power has been restored to the C RCP.

B Correct.

Note before Step 15 states that if RVLIS indication is not available, then PRZR level response is used to detect voids in the RCS. With the combined charging and seal injection flow equal to the combined letdown and seal leakoff flow, the trend in pressurizer level is consistent with the change in RCS temperature, i.e. RCS void growth is not occurring. If a void was forming in the vessel (vessel not full), then pressurizer level would exhibit a large increase.

C Incorrect. Pressurizer level response is as expected (see C). Evaluation of second bullet in Step 15, RVLIS upper range indication - greater than 15%, would, per Step 15 Note, be based on pressurizer level response. Since all expected responses are obtained, only the AER column steps within Step 15 should be performed.

D Incorrect. A steam void is not present in the reactor vessel as shown by pressurizer level and associated plant parameters. Plausible because RVLIS is unavailable and RCS cooldown (and depressurization) must continue to meet a tech spec action statement.

REFERENCES:
1. 1-ES-0.2A, Natural Circulation Cooldown with CRDM Fans, Rev.20 (**Provide Steps 14-16, pages 10-11, and Attachment 1, page 20, to applicant**)
2. Facility Exam Bank, Emergency Procedures Question 83 (ID 3335)
3.Background Information for Westinghouse Owners Group Emergency Response Guideline ES-0.2, Natural Circulation Cooldown, HP-Rev. 1, Step 14.
4. OPAP-0002, Operations Department Procedures, Rev. 8, Step 6.4.4.g

K/A CATALOG QUESTION DESCRIPTION:
- Natural Circulation Operations; Ability to determine and interpret the following as they apply to the (Natural Circulation Operations): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A MATCH:
- Question requires applicant to interpret plant conditions and select appropriate next step within natural circulation cooldown procedure. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

100.

The cooldown and depressurization performed in 1-ES-0.4, Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS), differs from that of the other natural circulation cooldown emergency response guidelines. The 1-ES-0.4 cooldown is accomplished in ______________________.

A.✓ a repeated stair-step fashion: (1) cooldown, then stabilize temperature; (2) depressurize

B. a repeated stair-step fashion: (1) depressurize, then stabilize pressure; (2) cooldown

C. a single, linear fashion: simultaneously cooldown and depressurize

D. two distinct steps: (1) complete the cooldown to cold shutdown; (2) start the depressurization

DISTRACTOR ANALYSIS:

A Correct. Per B/G document, this is the strategy for ES-0.4.

B Incorrect. The order of the actions is inverted. Plausible if applicant misunderstands the strategy.

C Incorrect. See A. Plausible since cooldown/depressurization could occur simultaneously.

D Incorrect. See C and B.

REFERENCES:
1. 1-ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (without RVLIS), Rev. 8
2. Self Study Guide for 1-E-0 Series, 8/15/05
3. Background Information for Westinghouse Owners Group Emergency Response Guideline ES-0.4 Natural Circulation Cooldown with Steam Voids In Vessel (Without RVLIS), HP-Rev. 1

K/A CATALOG QUESTION DESCRIPTION:
- Natural Circualtion with Steam Void in Vessel with/wthout RVLIS; Knowledge of symptom based EOP mitigation strategies.

K/A MATCH:
- Question requires knowledge of basis for performing prerequisite procedures and strategy for controlling pressurizer level during reactor vessel head void growth.