U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit:
Region: I [ ] II [ ] III IV	Reactor Type: W [ ] 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

1. Unit 1 initial conditions:
   Reactor power = 100%
   Condenser vacuum = 25" Hg degrading rapidly

   Current plant conditions:
   1-ES-0.1 (REACTOR TRIP RESPONSE) in progress
   Condenser vacuum = 0" Hg

   Based on the above conditions, which ONE of the following states (1) the temperature at which the RCS would be maintained with no operator action and (2) what temperature the RCS is directed to be maintained by 1-ES-0.1?

   A. (1) 550°F
   (2) 547°F

   B. (1) 550°F
   (2) 535°F

   C. (1) 556°F
   (2) 535°F

   D. (1) 556°F
   (2) 547°F

2. Unit 1 plant conditions:
   Reactor power = 50% power
   Reactor Protection System testing in progress
   'A' Reactor Trip Breaker is CLOSED
   'A' Reactor Trip BYPASS Breaker is CLOSED
   'B' Reactor Trip Breaker is CLOSED
   'B' Reactor Trip BYPASS Breaker is OPEN

   Based on the above conditions, if an operator closes (attempts to close) 'B' Reactor Trip BYPASS Breaker which ONE of the following states (1) the status of the Reactor Trip BYPASS Breakers and (2) the status of the Reactor?

   A. (1) Both Reactor Trip BYPASS Breakers will open
      (2) The reactor will trip

   B. (1) Both Reactor Trip BYPASS Breakers will open
      (2) The reactor will NOT trip

   C. (1) Only 'B' Reactor Trip BYPASS Breaker will open
      (2) The reactor will trip

   D. (1) Only 'B' Reactor Trip BYPASS Breaker will open
      (2) The reactor will NOT trip

3. A loss of which ONE of the following busses will result in an imminent Safety Injection?

A. Vital Bus III

B. Vital Bus I

C. 'A' DC Bus

D. 'B' DC Bus

4. Unit 2 Initial Conditions:

   - 100% Power.
   - Chilled CC is in service to Containment.

   Current conditions:

   - 2-CD-REF-1, Unit 2 Turbine Building Chiller Unit, trips due to a fault.

   Based on the current conditions, which ONE of the following describes (1) the effect on Unit 2 containment indicated partial pressure, AND (2) Unit 2 containment temperature?

A. (1) Indicated partial pressure will INCREASE.
   (2) Containment temperature will INCREASE.

B. (1) Indicated partial pressure will DECREASE.
   (2) Containment temperature will INCREASE.

C. (1) Indicated partial pressure will INCREASE.
   (2) Containment temperature will DECREASE.

D. (1) Indicated partial pressure will DECREASE.
   (2) Containment temperature will DECREASE.

5. Current conditions on Unit 1 are as follows:

   - Reactor power is 100%.
   - All plant systems and components are in a normal configuration.
   - RCS hot and cold leg temperatures are stable.
   - Containment pressure is 12.2 psia and rising slowly.
   - Pressurizer pressure is 2005 psia and slowly lowering.
   - Pressurizer level is 59% and rising.
   - Charging flow is lowering in automatic.

   Which ONE of the following containment leak locations correlates to the above indications?

   A. RCS cold leg

   B. Main steam line

   C. Reactor vessel head

   D. Pressurizer vapor space

6. Unit 1 Initial Conditions:

- A small break loss-of-coolant accident (SBLOCA) occurred from 100% power.
- Operators are performing steps in 1-ES-1.2, "POST LOCA COOLDOWN AND DEPRESSURIZATION."
- A controlled RCS cooldown has been initiated at approximately 90 °F/hr.
- All Steam Generator (S/G) narrow range (NR) levels are approximately 45% and STABLE.
- All S/G pressures are approximately 650 psig and DECREASING.
- Auxiliary Feedwater Flow to EACH steam generator is initially 200 gpm.

Current conditions:

- Operators stopped the CHG pump flowing to the alternate header.
- Operators then "paused" for approximately five (5) minutes after stopping the CHG pump to allow RCS pressure to stabilize or increase before taking further actions to reduce SI flow.
- Following safety injection flow reduction, cooldown rate was calculated to be 77°F/hr.
- No operator actions were performed during the "pause."

Based on the conditions at the end of the "pause" (approximately five (5) minutes after stopping the CHG pump), which ONE of the following predicts:

(1) S/G NR level response
AND
(2) S/G pressure response?

A. (1) levels will be INCREASING.
(2) pressures will be DECREASING.

B. (1) levels will be INCREASING.
(2) pressures will be STABLE at a lower value.

C. (1) levels will be STABLE at the same value.
(2) pressures will be DECREASING.

D. (1) levels will be STABLE at the same value.
(2) pressures will be STABLE at a lower value.

7. Unit 1 initial plant conditions:
    Time = 0800
    Reactor power = 100%

   Current plant conditions:
    Time = 0845
    RCS pressure = 700 psig decreasing
    RCS Subcooling = 20°F decreasing
    Safety Injection Flow = 225 gpm to each loop

   Based on the above conditions after transition to 1-E-1 (Loss of Reactor or Secondary Coolant): (1) which ONE of the following actions are directed by 1-E-1 with regard to RCPs and (2) what is the reason for that action?

   A. (1) Secure RCPs
      (2) To reduce the depletion of RCS water inventory.

   B. (1) Secure RCPs
      (2) To prevent the possibility of flywheel fracture if the pump continues to operate without coolant.

   C. (1) Maintain RCPs operating
      (2) They provide core cooling by pumping a 2 phase mixture through the core and loops.

   D. (1) Maintain RCPs operating
      (2) To prevent phase separation in the core region which could lead to core uncovery.

8. Current Unit 1 plant conditions:
   - Reactor power is at 33%

The latest temperature readings obtained from TR-1-448 are as follows:

Temperatures (°F)	**RCP 'A'**	**RCP 'B'**	**RCP 'C'**
Upper thrust bearing	181	178	163
Lower thrust bearing	173	183	172
Upper radial bearing	143	163	146
Lower radial bearing	172	189	158
Motor stator	285	273	302
Lower bearing seal water	153	183	167
Seal water	195	184	185

Given the above temperature readings, which one of the following correctly states the RCP, if any, that exceeds an ACTION LEVEL limit in Attachment 2, RCP Parameters, of 1-AP-9.00, RCP Abnormal Conditions?

A. RCP 'A'.

B. RCP 'B'.

C. RCP 'C'.

D. No RCPs are exceeding an ACTION LEVEL limit.

9. Initial plant conditions are as follows:

    - Unit 1 is at 100% power.
    - Unit 2 is at 100% power.
    - 1-CH-P-1B, Unit 1 B Charging pump, is out of service with motor removed.
    - 1-CH-P-1C, Unit 1 C Charging pump, is running on its alternate power supply.

    Current plant conditions are as follows:

    - 1-CH-P-1C has tripped.
    - Attempts to start 1-CH-P-1A have been unsuccessful.
    - The crew has entered 1-AP-8.00 "Loss of Normal Charging Flow".
    - No indications of Unit 1 charging system leakage are observed.
    - All Unit 2 Charging pumps are operable.
    - RCP seal injection flow is zero.
    - Component cooling water flow to the thermal barrier is normal.

    Given the above conditions, which ONE of the following would be consistent with the actions required by 1-AP-8.00?

    A. Trip Unit 1 reactor. Do NOT trip Unit 2 reactor.
    Cross-connect charging with Unit 2 per 1-AP-8.00.

    B. Trip Unit 1 reactor. Do NOT trip Unit 2 reactor.
    Cross-connecting charging with Unit 2 is NOT permitted per 1-AP-8.00.

    C. Trip Unit 1 and Unit 2 reactors.
    Cross-connect charging with Unit 2 per 1-AP-8.00.

    D. Trip Unit 1 and Unit 2 reactors.
    Cross-connecting charging with Unit 2 is NOT permitted per 1-AP-8.00.

10. Unit conditions are as follows:

- A unit shutdown is in progress due to increased RCS leakage
- RCS Temperature is 300°F
- RCS Pressure is 300 psig
- The RCS is solid
- 1-RH-P-1A ("A" RHR Pump) is in service with 1-RH-E-1A ("A" RHR Heat Exchanger)

Which ONE of the following would INITIALLY occur if 1-RH-P-1A were to trip on overcurrent?

A. RCS Pressure would increase

B. 1-CH-PCV-1145 (Letdown Pressure Control Valve) would open

C. Charging flow would increase

D. CC head tank level would increase

11. Unit 1 initial conditions:

    Time = 1000
    Reactor power = 100%
    1-RC-PCV-1455B (Pzr Spray Valve) fails open
    RCS pressure = 2100 psig decreasing

    Current conditions:

    Time = 1001
    RCS pressure = 1900 psig decreasing
    1-E-0 REACTOR TRIP OR SAFETY INJECTION in progress

    Based on the above conditions: (1) state which ONE of the following actions is directed by step 7 of 1-E-0 if the Pzr Spray Valve can not be closed and (2) state the reason why?

    A. (1) Secure RCP A
       (2) To stop the RCS depressurization.

    B. (1) Secure RCP A
       (2) To prevent inadvertent Safety Injection.

    C. (1) Secure RCP C
       (2) To stop the RCS depressurization.

    D. (1) Secure RCP C
       (2) To prevent inadvertent Safety Injection.

12. Unit 1 Initial Conditions:

- An Anticipated Transient Without SCRAM (ATWS) occurred at 100% power.
- The reactor remains at power. An operator is inserting control rods in manual.
- Safety Injection is NOT actuated.
- Charging flow was verified to be 77 GPM, and the BATP was placed in FAST.

Current conditions:

- 1-CH-MOV-1350, Emergency Borate MOV, will not open. An operator reports that the valve appears to be mechanically bound.
- Neither Pressurizer PORV automatically opened when RCS pressure rose above 2335 psig. An operator was able to manually open ONLY one Pressurizer PORV to control RCS pressure.

Based on the current conditions, which ONE of the following correctly identifies (1) the next required action to initiate emergency boration, in accordance with 1-FR-S.1, "RESPONSE TO NUCLEAR POWER GENERATION/ATWS," AND (2) the required action to operate the PORV as specified in FR-S.1?

A. (1) Manually actuate SI to provide for maximum flowrate injection into the RCS.
(2) Allow the RCS pressure to lower to 2210 psig before closing the PORV.

B. (1) Place switches for CH-MOV-1115B and -1115D to OPEN and switches for CH-MOV-1115C and -1115E to CLOSE.
(2) Allow the RCS pressure to lower to 2210 psig before closing the PORV.

C. (1) Manually actuate SI to provide for maximum flowrate injection into the RCS.
(2) Close the PORV when RCS pressure equals 2335 psig and lowering.

D. (1) Place switches for CH-MOV-1115B and -1115D to OPEN and switches for CH-MOV-1115C and -1115E to CLOSE.
(2) Close the PORV when RCS pressure equals 2335 psig and lowering.

13. Unit 1 initial plant conditions:

    Reactor power = 100%
    Pzr level = 52 % and stable
    VCT Level = 40% and decreasing slowly
    1-AP-16.00 (Excessive RCS Leakage) initiated

    Current Unit 1 conditions:

    Leak determined to be Steam Generator Tube Leak in the 1A SG = 39 gpm and increasing slowly
    The team has initiated 1-AP-24.00 (Minor Steam Generator Tube Leak)

    Based on the above conditions: (1) how will charging pump amps change as 1-CH-FCV-1122 (Charging Flow Control Valve) opens to maintain pressurizer level and (2) is a reactor trip required at this instant in time per 1-AP-16.00 or 1-AP-24.00?

    A. (1) pump amps will increase
    (2) Yes

    B. (1) pump amps will increase
    (2) No

    C. (1) pump amps will decrease
    (2) Yes

    D. (1) pump amps will decrease
    (2) No

14. Which ONE of the following completes the below statements?

(1) The parameters used for SI Termination criteria in 1-E-2, "FAULTED STEAM GENERATOR ISOLATION," are RCS subcooling AND __________________

AND

(2) The parameters used for SI Termination criteria in 1-ECA-2.1, "UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS?" are RCS subcooling AND ________________?

A. (1) E-2: ONLY RCS pressure and PRZR level.
   (2) ECA-2.1: RVLIS indication greater than values specified in a table (based upon number of running RCPs).

B. (1) E-2: ONLY RCS pressure and PRZR level.
   (2) ECA-2.1: RCS pressure and PRZR level.

C. (1) E-2: Secondary heat sink, RCS pressure, and PRZR level.
   (2) ECA-2.1: RVLIS indication greater than values specified in a table (based upon number of running RCPs).

D. (1) E-2: Secondary heat sink, RCS pressure, and PRZR level.
   (2) ECA-2.1: RCS pressure, and PRZR level.

15. Unit 1 initial conditions:

    Station Blackout occurs
    1-ECA-0.0 (LOSS OF ALL AC POWER) in progress
    SGs are to be depressurized to allow SI accumulators to inject into the RCS.

    Based on the above conditions, which ONE of the following: (1) correctly states the maximum cooldown rate allowed during this depressurization and (2) the basis for that rate ?

    A. (1) 25 °F/Hr
       (2) To prevent a steam bubble from forming in the reactor vessel head.

    B. (1) 25 °F/Hr
       (2) To minimize RCS inventory loss.

    C. (1) 100 °F/Hr
       (2) To prevent a steam bubble from forming in the reactor vessel head.

    D. (1) 100 °F/Hr
       (2) To minimize RCS inventory loss.

16. Initial Conditions:

- Surry Unit 2 is in day 25 of a scheduled refueling outage.
- A Loss of All AC Power occurred on Unit 1, which was operating at 100% power.
- Control room operators implemented 1-ECA-0.0, "LOSS OF ALL AC POWER."
- Safety Injection (SI) initiated on Unit 1.
- Power was restored to one Unit 1 safeguards power train from an Emergency Diesel Generator.

Current conditions:

- Unit 1 control room operators are implementing 1-ECA-0.2, "LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED," and are performing the step that directs restoration of SW to CC HXs IAW 0-AP-12.01, "LOSS OF INTAKE CANAL LEVEL."
- Service water is isolated to all recirculation spray (RS) heat exchangers on both units.
- 2 ESW pumps are running.

Note: In 0-AP-12.01, "time zero" is defined as that time intake canal level reaches 23.5 FT.

Based on the current conditions, which ONE of the following correctly identifies (1) the 0-AP-12.01 restriction, if any, on CC HX SW flow 15 minutes after "time zero" AND (2) the 0-AP-12.01 restriction on CC HX SW flow after nine hours have elapsed from "time zero?"

(Reference provided)

A. (1) There are no restrictions on CC HX SW flow.
   (2) Crosstie CC. 3 CC HXs allowed with SW outlet valves 19 turns open for each HX.

B. (1) There are no restrictions on CC HX SW flow.
   (2) 2 CC HXs allowed with SW outlet valves 19 turns open for each HX.

C. (1) Maximum allowable flow is one CC HX outlet SW valve fully open.
   (2) Crosstie CC. 3 CC HXs allowed with SW outlet valves 19 turns open for each HX.

D. (1) Maximum allowable flow is one CC HX outlet SW valve fully open.
   (2) 2 CC HXs allowed with SW outlet valves 19 turns open for each HX.

17. Unit 2 is performing a plant shutdown with the reactor at 5% power when a loss of Vital Bus 1 occurs.

Which one of the following correctly describes:

(1) the direct effect of the loss of Vital Bus I on the RPS system and

(2) if a trip or shutdown were to occur, the effect on the re-instatement of the Source Range NI's?

A. (1) An RPS trip signal is generated
(2) Re-instatement of SRNIs will occur automatically.

B. (1) An RPS trip signal is generated
(2) Re-instatement of SRNIs will NOT occur automatically.

C. (1) An RPS trip signal is NOT generated
(2) Re-instatement of SRNIs will occur automatically.

D. (1) An RPS trip signal is NOT generated
(2) Re-instatement of SRNIs will NOT occur automatically.

18. Unit 1 Initial Conditions:

- 100% power.

Current conditions:

- A complete loss of DC bus 1B has occurred.

Based on the current conditions, which ONE of the following correctly identifies

(1) the required action in 1-AP-10.06, "LOSS OF DC POWER," for the generator output breakers, AND

(2) the subsequent impact on Reactor Coolant Pump (RCP) operations following completion of the immediate actions of 1-E-0 (Reactor Trip or Safety Injection)?

A. (1) A control room operator is NOT required to manually open the Generator output breakers.
   (2) 'A' RCP will stop, 'B' and 'C' RCPs will remain running.

B. (1) A control room operator is NOT required to manually open the Generator output breakers.
   (2) 'A' RCP will remain running, 'B' and 'C' RCPs will stop.

C. (1) A control room operator is required to manually open the Generator output breakers.
   (2) 'A' RCP will stop, 'B' and 'C' RCPs will remain running.

D. (1) A control room operator is required to manually open the Generator output breakers.
   (2) 'A' RCP will remain running, 'B' and 'C' RCPs will stop.

19. Unit 1 initial conditions;

    Reactor power = 90% and decreasing
    Shutdown in progress for refueling

    Current plant conditions:

    Reactor power = 70% and decreasing
    1G-B5 COMPUTER PRINTOUT ROD CONT SYS in alarm
    QPTR = 1.05% and increasing
    The power decrease is stopped

    (1) Based on the above indications, which ONE of the following conditions has occurred and (2) what is the minimum QPTR that requires operator action when exceeded IAW Tech Spec 3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS?

    A. (1) Stuck control rod
       (2) 2%

    B. (1) Stuck control rod
       (2) 5%

    C. (1) Control bank inserted past insertion limit
       (2) 2%

    D. (1) Control bank inserted past insertion limit
       (2) 5%

20. Following a complete loss of instrument air and subsequent restoration, which ONE of the following components will require manual re-alignment to support the start of 1-RC-P-1C?

A. 1-CC-TV-105C, RCP C Cooler CC Return Trip Valve

B. 1-CC-TV-120C, RCP C Thermal Barrier Component Cooling Water System Trip Valve

C. 1-RC-HCV-1303C, RCP C Seal Leakoff Isolation Valve

D. 1-CH-MOV-1381, RCP Seal Return Valve

21. Unit 1 Initial Conditions:

- 100% Power
- Both Megawatts and Megavars start to oscillate.
- System Operator reports that the regional grid is experiencing dynamic instabilities.

Current conditions:

- 230 KV system voltage is 212 KV
- 500 KV system voltage is 492 KV

Based on the current conditions, which one of the following identifies (1) the required operator action for continued voltage regulator operation, as specified in 0-AP-10.18, "RESPONSE TO GRID INSTABILITY," AND (2) the required entry into Technical Specification (TS) 3.16, "Emergency Power System?"

A. (1) Place the voltage regulator - in MANUAL.
   (2) Entry into TS 3.16 is required ONLY for the 230 KV system. Entry into TS 3.16 for the 500 KV system is NOT required.

B. (1) Place the voltage regulator - in MANUAL.
   (2) Entry into TS 3.16 is required for BOTH the 230 KV system AND the 500 KV system.

C. (1) Verify the voltage regulator - in AUTO, or place the voltage regulator in AUTO if possible.
   (2) Entry into TS 3.16 is required for BOTH the 230 KV system AND the 500 KV system.

D. (1) Verify the voltage regulator - in AUTO, or place the voltage regulator in AUTO if possible.
   (2) Entry into TS 3.16 is required ONLY for the 230 KV system. Entry into TS 3.16 for the 500 KV system is NOT required.

22. Unit 1 Initial Conditions:

- A loss-of-coolant accident (LOCA) occurred from 100% power.
- Operators are implementing 1-ECA-1.2, "LOCA OUTSIDE CONTAINMENT."

Current conditions:

- Operators have completed all the steps in ECA-1.2 that (a) attempt to verify proper valve alignment, and (b) locate and isolate the leak.
- RCS pressure continues to DECREASE.
- Annunciator 1B-F3, SFGS AREA SUMP HI LEVEL, is LIT.
- 1-VG-RM-110, VENT VENT 2 GAS, is below the HIGH setpoint, but is rapidly trending UP.
- RM-GW-130-1, PROCESS VENT STK PART, is below the HIGH setpoint, but is rapidly trending UP.

Based on the current conditions, which ONE of the following:

(1) is the NEXT overall mitigating strategy that should be implemented, as specified by ECA-1.2,

AND

(2) a filtered release to the environment __________________ exist?

A. (1) Depressurize S/Gs to inject accumulators.
   (2) DOES

B. (1) Conserve RWST inventory.
   (2) DOES

C. (1) Depressurize S/Gs to inject accumulators.
   (2) DOES NOT

D. (1) Conserve RWST inventory.
   (2) DOES NOT

23. Unit 1 initial conditions:

Loss of Main and Auxiliary Feedwater
EOP transition from E-0 to 1-FR-H.1(RESPONSE TO LOSS OF SECONDARY HEAT SINK)

Current plant conditions:

The 1A SG is to be depressurized to establish condensate flow to the SG.
RCS Pressure is 2000 psig and slowly increasing
Steam Generator Levels are as follows:

'A' SG Wide Range Level - 42% and slowly decreasing
'B' SG Wide Range Level - 41% and slowly decreasing
'C' SG Wide Range Level - 43% and slowly decreasing

Based on the above conditions: (1) why is steam flow limited to 1.0 x 10^6 Lbm/hr during the depressurization and (2) what actions are directed by 1-FR-H.1 if adequate condensate flow to the 1A SG does not occur with SG pressure = 550 psig?

A. (1) To prevent main steam line isolation
(2) Continue to depressurize 'A' SG until condensate flow established

B. (1) To prevent main steam line isolation
(2) Commence RCS Bleed and Feed

C. (1) To limit cooldown rate to < 100°F/Hr
(2) Continue to depressurize 'A' SG until condensate flow established

D. (1) To limit cooldown rate to < 100°F/Hr
(2) Commence RCS Bleed and Feed

24. Initial Unit 1 Conditions:
    - Power Range NIs are: N-41=36.3%, N-42=36.2%, N-43=34.7%, N-44=36.6%
    - Delta T Power is 35%
    - All three RCPs are operating

    Present Unit 1 Conditions:
    - RCP 1A frame vibrations indicate 10 mils

    - 2 minutes after the frame vibrations increase to 10 mils, the speed sensing panel actuates for RCP 1A.

    Based on the above conditions, which one of the following:

    (1) correctly states the status of the 1C-H4, RCP FRAME DANGER, annunciator and

    (2) the impact on reactor operations from the actuation of the speed sensing panel?

    A. (1) 1C-H4 is illuminated.
       (2) Unit 1 reactor automatically trips.

    B. (1) 1C-H4 is illuminated.
       (2) Unit 1 reactor does NOT automatically trip.

    C. (1) 1C-H4 is NOT illuminated.
       (2) Unit 1 reactor automatically trips.

    D. (1) 1C-H4 is NOT illuminated.
       (2) Unit 1 reactor does NOT automatically trip.

25. A tube leak has developed in the thermal barrier heat exchanger for 1-RC-P-1A.

Once the indicated thermal barrier flow has exceeded 50 gpm, which ONE of the following (1) states the expected automatic plant response and (2) the associated time delay for that response?

A. (1) 1-CC-TV-120A (A RCP Thermal Barrier Isolation Trip Valve) will close
(2) 50 seconds.

B. (1) 1-CC-TV-120A (A RCP Thermal Barrier Isolation Trip Valve) will close
(2) 10 seconds.

C. (1) 1-CC-TV-140A (Thermal Barrier CC Return Trip Valve) and 1-CC-TV-140B (Thermal Barrier CC Return Trip Valve) will close
(2) 50 seconds.

D. (1) 1-CC-TV-140A (Thermal Barrier CC Return Trip Valve) and 1-CC-TV-140B (Thermal Barrier CC Return Trip Valve) will close
(2) 10 seconds.

26. Current plant conditions are as follows on Unit 2:

- Reactor power is 100%
- Pressurizer level is 50% and slowly lowering.
- Letdown flow is 104 gpm

The team initiated 1-AP-16.00 (Excessive RCS Leakage) and completed the immediate actions of 1-AP-16.00. At this time, the reactor operator reports the following:

- Seal Injection Flow to each RCP is 7 gpm
- Seal Leak-off Flow for each RCP is 3 gpm
- Tave is stable at 573 °F
- The leak rate was determined to be 75 gpm

Which ONE of the following correctly describes what value the RO should set charging flow at to maintain pressurizer level stable?

A. 54 GPM

B. 63 GPM

C. 75 GPM

D. 84 GPM

27. Unit 1 Conditions:

Cold Shutdown ~ 50 hours after an extended full power run
1-RC-LI-100A = 12.0 ft.
1A RHR pump operating and pump amps are oscillating
1-RH-FI-1605 = 2000 gpm and oscillating

The team has increased RCS level back into the acceptable region and secured 1-RH-P-1A in accordance with 1-AP-27.00 (LOSS OF DECAY HEAT REMOVAL CAPABILITY).

Which ONE of the following actions are directed by 1-AP-27.00 to restore RHR flow?

A. Vent the 1A RHR pump, restart the 1A RHR pump and verify the RHR heat sink.

B. Vent the 1B RHR pump, start the 1B RHR pump and verify the RHR heat sink.

C. Close 1-RH-HCV-1758 and 1-RH-FCV-1605, then re-start the 1A RHR pump and throttle to the pre-event rate.

D. Close 1-RH-HCV-1758 and 1-RH-FCV-1605, then start the 1B RHR pump and throttle to the pre-event rate.

28. Plant conditions:

Unit 1 = 100%
Unit 2 = shutdown for refueling
MAIN CONTROL ROOM OXYGEN MONITOR alarms
A hazardous substance has been spilled in the main control room
0-AP-20.00, MAIN CONTROL ROOM INACCESSIBILITY, is initiated
Operators proceed to the Auxiliary Shutdown Panel with FCA procedures

Based on the above conditions, which ONE of the following states equipment that is operated from the Auxiliary Shutdown Panel as directed by 0-AP-20.00?

A. CC pumps

B. RHR pumps

C. 1-CH-HCV-1311 (Aux Pressurizer Spray)

D. 1-CH-FCV-1122 (Charging Flow Controller)

29. Initial plant conditions on Unit 1 are as follows:

    - Reactor tripped following an inadvertent SI.

    Current plant conditions on Unit 1 are as follows:

    - 1-ES-1.1, SI Termination, has been initiated.
        - SI signal has been reset.
        - Actions have been completed to re-establish charging and letdown.
    - Seal leakoff flow from each RCP is ~3 gpm.
    - PRT level shows a slow rising trend.
    - VCT level is 43% and decreasing.
    - Reactor Pressure is stable at 2235 psig.
    - Tailpipe temperatures are 105°F and stable.

    Which ONE of the following would be consistent with the above conditions?

    A. 1-CH-MOV-1381 "Seal Return Header Isolation Valve" is closed causing 1-CH-RV-1382B "Seal Return Heat Exchanger Relief Valve" to lift.

    B. A component cooling water leak has developed on the Seal Return Heat Exchanger causing 1-CH-RV-1382B "Seal Return Heat Exchanger Relief Valve" to lift.

    C. 1-CH-MOV-1381 "Seal Return Header Isolation Valve" is closed causing 1-CH-RV-1382A "Seal Return Relief Valve" to lift.

    D. A component cooling water leak has developed on the Seal Return Heat Exchanger causing 1-CH-RV-1382A "Seal Return Relief Valve" to lift.

30. Unit 1 Initial Conditions:

- 100% Power
- Pressurizer Relief Tank (PRT) level is 62% and STABLE
- A leaking pressurizer code safety valve has been identified.

Current conditions:

- A PRT annunciator has just alarmed
- PRT level is 76% and INCREASING
- PRT temperature is 126 °F and INCREASING
- PRT pressure is 3 psig and INCREASING

Based on the current conditions, which one of the following identifies (1) the parameter that caused the PRT annunciator, AND (2) chemistry will sample the PRT gas space when PRT level INCREASES by 5% because large changes in PRT level can result in an undesirable atmosphere in the PRT gas space ________(2)______?

A. (1) High level
   (2) due to PG O_2 off-gassing as PRT temperature changes.

B. (1) High temperature
   (2) due to PG O_2 off-gassing as PRT temperature changes.

C. (1) High level
   (2) due to H_2 concentration potentially exceeding 4% in the process vent.

D. (1) High temperature
   (2) due to H_2 concentration potentially exceeding 4% in the process vent.

31. Unit 1 plant conditions:

Reactor power = 100%
Charging flow = 100 gpm increasing
1-CC-RI-105 (CC Heat Exchanger A/B Outlet Radiation Monitor) alarms HIGH
CC surge tank level = 64% increasing
1-AP-16.00 (EXCESSIVE RCS LEAKAGE) is initiated

Based on the above conditions, which ONE of the following describes where the excess volume in the CC system will go to and (2) what actions is directed first by 1-AP-16.00 to attempt to isolate the leak?

A. (1) The process vent system
(2) Isolate letdown

B. (1) The process vent system
(2) Isolate thermal barrier on suspected RCP

C. (1) The auxiliary building sump
(2) Isolate letdown

D. (1) The auxiliary building sump
(2) Isolate thermal barrier on suspected RCP

32. Unit 1 Initial Conditions:

- 75% Power at Middle-of-Life (MOL) conditions.
- Rod Control is in AUTOMATIC.
- VCT automatic makeup controls are set to the current RCS boron concentration.
- Excess Letdown is in service in preparation for removing Normal Letdown from service.

Current conditions:

- Component Cooling (CC) surge tank level is slowly DECREASING at 1% every 5 minutes.
- Reactor power is slowly INCREASING.

Based on the current conditions and assuming NO other operator actions, which ONE of the following identifies (1) the location of the CC leak that would cause the current conditions, AND (2) the expected impact of the CC leak on rod control?

	LEAKING COMPONENT	ROD CONTROL
A.	(1) Seal Water Heat Exchanger	(2) Rods will step IN.
B.	(1) Seal Water Heat Exchanger	(2) Rods will step OUT.
C.	(1) Excess Letdown Heat Exchanger	(2) Rods will step IN.
D.	(1) Excess Letdown Heat Exchanger	(2) Rods will step OUT.

33. In FR-C.2 (Response to Degraded Core Cooling), if attempts to establish adequate core cooling using the High Head Safety Injection System are ineffective, the intact Steam Generators are depressurized to 200 psig and subsequently to atmospheric pressure.

    Which ONE of the following describes (1) the purpose of the depressurization of the steam generators and (2) why the steam generators are depressurization is stopped at 200 psig?

    A. (1) To depressurize the RCS to increase accumulator and Low Head Safety Injection Flow.
       (2) To isolate the Safety Injection Accumulators and prevent Nitrogen addition into the RCS.

    B. (1) To depressurize the RCS to increase accumulator and Low Head Safety Injection Flow.
       (2) To prevent the loss of RCP support conditions by maintaining adequate RCS pressure.

    C. (1) To maximize Auxiliary Feedwater Flow and enhance RCS cooldown.
       (2) To isolate the Safety Injection Accumulators and prevent Nitrogen addition into the RCS.

    D. (1) To maximize Auxiliary Feedwater Flow and enhance RCS cooldown.
       (2) To prevent the loss of RCP support conditions by maintaining adequate RCS pressure.

34. Concerning a pressurized thermal shock event, a ___(1)____ would cause the most significant Pressurized Thermal Shock (PTS) challenge to the plant, and in accordance with FR-P.1 (Response to Imminent Pressurized Thermal Shock), an appropriate response would be to ____(2)____?

A. (1) Main Steam Line Break
(2) Stop the RCS cooldown and reduce Safety Injection flow

B. (1) Main Steam Line Break
(2) Reduce the RCS cooldown rate to < 100 °F/hr and depressurize the RCS to establish 30 °F subcooling

C. (1) Steam Generator Tube Rupture
(2) Stop the RCS cooldown and reduce Safety Injection flow

D. (1) Steam Generator Tube Rupture
(2) Reduce the RCS cooldown rate to < 100 °F/hr and depressurize the RCS to establish 30 °F subcooling

35. A Large Break LOCA has occurred on Unit 1.

Current conditions:
- All plant systems operated as designed.
- Steam Generator Levels are 14% Narrow Range and stable in all three Steam Generators.
- AFW flow to each Steam Generator is 125 gpm.
- Containment Pressure is currently 15 psia after peaking at 47 psia.
- Containment High Range Radiation Monitors (1-RM-RI-127 and 128) are currently indicating 1×10^4 after peaking at 4.3×10^6 .

Based on the current conditions, which ONE of the states (1) the status of Heat Sink and (2) why?

A. (1) Steam generator Heat Sink requirements are met.
   (2) Adequate Steam generator Inventory.

B. (1) Steam generator Heat Sink requirements are met.
   (2) Adequate AFW flow.

C. (1) Steam generator Heat Sink requirements are NOT met.
   (2) Inadequate Steam generator Inventory and AFW flow is < the limit of 540 gpm
   .

D. (1) Steam generator Heat Sink requirements are NOT met.
   (2) Inadequate Steam generator Inventory and AFW flow is < the limit of 450 gpm.

36. Unit 1 plant conditions:

Plant runback occurs from 100% to 90%
RCS HI PRESSURE ALARM lit
Master Pressure Controller output demand is currently 70%

Based on the above conditions, which ONE of the following states (1) the status of the Pressurizer Spray valves and (2) what the maximum Pressurizer spray flow rate is based on?

A. (1) Full Open
(2) To prevent the PORVs from opening during a 10% step load decrease.

B. (1) Full Open
(2) To prevent exceeding the capacity of the PORVs during a load rejection from 100% power.

C. (1) Modulated Open (< 100%)
(2) To prevent exceeding the capacity of the PORVs during a load rejection from 100% power.

D. (1) Modulated Open (< 100%)
(2) To prevent the PORVs from opening during a 10% step load decrease.

37. Unit 1 initial conditions:
   - Reactor power = 5%
   - Pressurizer Pressure Protection transmitter (1-RC-PT-1456) failed low (I&C investigating)

   Current plant conditions:
   - Pressurizer Pressure Protection transmitter (1-RC-PT-1455) subsequently failed low

   Based on the current plant conditions, which ONE of the following correctly describes (1) the effect on the reactor and (2) the effect these failures on subcooling indication on the Inadequate Core Cooling Monitor (ICCM)?

   A. (1) A Reactor Trip will occur
      (2) No impact on subcooling indication

   B. (1) A Reactor Trip will NOT occur
      (2) Subcooling will indicate -35 °F

   C. (1) A Reactor Trip will occur
      (2) Subcooling will indicate -35 °F

   D. (1) A Reactor Trip will NOT occur
      (2) No impact on subcooling indication

38. Unit 1 plant conditions:
Reactor power = 100%

Current conditions:
LBLOCA occurs
Containment pressure = 25 psia increasing
1-CS-MOV-101A (Containment Spray Pump 'A' Discharge Valve) does not open
1-CS-MOV-101B (Containment Spray Pump 'A' Discharge Valve) does not open

Based on the above conditions, which ONE of the following states:

(1) Which Recirculation Spray (RS) System pump suction(s) is/are being supplied by B Train of Containment Spray?

(2) If sufficient containment spray flow is being supplied to meet the design basis of the CS system?

A. (1) RS Train B ONLY
(2) The CS design basis is being met.

B. (1) RS Train B ONLY
(2) The CS design basis is NOT being met.

C. (1) RS Train A and B
(2) The CS design basis is being met.

D. (1) RS Train A and B
(2) The CS design basis is NOT being met.

39. Unit 1 Initial Conditions:

- Unit is shutdown for refueling operations.
- 1-RM-RM-159/160, Containment Particulate/Gas, both read 875 cps.
- 1-RM-RM-152, New Fuel Storage Area, reads 1.7 mr/hr.

Current conditions:

- Core off-load is in progress, when an event occurs.
- 1-RM-RM-159/160, Containment Particulate/Gas, both read 880 cps.
- 1-RM-RM-152, New Fuel Storage Area, reads 15.3 mr/hr.

Based on the current conditions, which ONE of the following describes REQUIRED operator actions, in accordance with 0-AP-22.00, "FUEL HANDLING ABNORMAL CONDITIONS?"

(1) Fuel handling operations MUST STOP ________________,
AND
(2) after dumping a train of MCR air bottles in accordance with 0-AP-22.00, THEN ________________ ?

A. (1) in the Fuel Building. Fuel Handling operations may continue in Containment.
(2) IMMEDIATELY start one emergency supply fan (1-VS-F-41 or 2-VS-F-41 preferred)

B. (1) in the Fuel Building. Fuel Handling operations may continue in Containment.
(2) Wait 50 minutes before starting one emergency supply fan (1-VS-F-41 or 2-VS-F-41 preferred)

C. (1) in BOTH the Fuel Building AND Containment
(2) Wait 50 minutes before starting one emergency supply fan (1-VS-F-41 or 2-VS-F-41 preferred)

D. (1) in BOTH the Fuel Building AND Containment
(2) IMMEDIATELY start one emergency supply fan (1-VS-F-41 or 2-VS-F-41 preferred)

40. Unit 1 is at 20% power during a power increase following a maintenance shutdown.

Initial conditions:
Time = 1000
An existing 2 gallon per day tube leak exists on the 1A SG
CHG LINE FLOW = 97 gpm and increasing
1-AP-16.00 (EXCESSIVE RCS LEAKAGE) is entered

Current conditions:
Time = 1015
Main Steam Line Rad Monitor level increasing (NOT in alarm)
The NEW Steam Generator Tube leak rate is determined to be 6 gpm

Based on the above conditions, which ONE of the following: (1) correctly states if 1-AP-24.00 (MINOR SG TUBE LEAK) is required to be initiated IAW 1-AP-16.00 and (2) what procedure is required to shut down the unit?

A. (1) Yes
(2) 1-AP-23.00 (RAPID LOAD REDUCTION).

B. (1) Yes
(2) 1-GOP-2.2 (UNIT SHUTDOWN, LESS THAN 30% TO HSD).

C. (1) No
(2) 1-AP-23.00 (RAPID LOAD REDUCTION).

D. (1) No
(2) 1-GOP-2.2 (UNIT SHUTDOWN, LESS THAN 30% TO HSD).

41. Unit 1 Initial Conditions:

- A Steam Generator (S/G) tube rupture caused an automatic reactor trip and SI from 100% power.
- Control room operators are implementing 1-ECA-3.1, "SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY."

Current conditions:

- A maximum-rate cooldown in accordance with ECA-3.1 was commenced at time 1500.
- The following data has been logged over the last hour: (consider that the time is currently 1600)

TIME	RCS COLD LEG TEMP
1500	395 °F
1515	370 °F
1530	346 °F
1545	321 °F
1600	296 °F

Based on the current conditions, which ONE of the following correctly describes the cooldown from 1500 to 1600?

(1) The Technical Specification cooldown rate limit ____________________ ,
AND
(2) The cooldown is ______________________ .

A. WAS exceeded.
required to be temporarily stopped.

B. WAS exceeded.
NOT required to be stopped.

C. was NOT exceeded.
NOT required to be stopped.

D. was NOT exceeded.
required to be temporarily stopped.

42. Unit 1 Initial Conditions:

- A reactor startup is in progress.
- All Steam Generator (S/G) Power Operated Relief Valves (PORV) are 10% open, and the controllers are being operated in automatic.

Current conditions:

- The 1A S/G PORV controller setpoint is at 1000 psig AND the setpoint is continuously DECREASING.

Based on the current conditions, which ONE of the following correctly describes the effect of this failure if no operator action is taken?

A. If uncorrected, the 522 °F minimum temperature for criticality Tech Spec limit may be violated.

B. If uncorrected, the 545 °F minimum temperature for criticality Tech Spec limit may be violated.

C. 1B and 1C S/G PORVs will open to relieve more steam and maintain Tave within acceptable limits.

D. 1B and 1C S/G PORVs will not respond, and Tave will increase above the limit of 577 °F.

43. Unit 2 plant conditions:

    2-GOP 1.5 (UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER) is in progress
    Reactor Power = 12%
    2-FW-P-1A ('A' Main Feed Water Pump) is feeding all SGs at 1500 gpm each

Based on the above plant conditions, which ONE of the following conditions will cause 2-FW-P-1A to trip?

A. MFW pump suction header pressure 65 psig for > 15 sec

B. Bus voltage dips to 65% and returns to normal

C. 'A' Main Feed Pump Recirc Valve closed for > 15 sec

D. 2-FW-P-1A lube oil pressure decreases to 5 psig

44. Unit 1 plant conditions:

Reactor power = 50%

1-GW-RM-130A (Process Vent Particulate Radiation Monitor) high alarm sounds

Based on the above conditions: (1) which ONE of the following valves are interlocked to reposition upon receiving the high alarm and (2) why does that action occur?

A. (1) GW-FCV-101 (Waste Gas Decay Tank bleed FCV)
(2) To isolate a potential release path

B. (1) GW-FCV-101 (Waste Gas Decay Tank bleed FCV)
(2) To redirect flow through the waste gas charcoal filters

C. (1) GW-FCV-100 (Process Vent Flow Control Valve)
(2) To isolate all potential release paths

D. (1) GW-FCV-100 (Process Vent Flow Control Valve)
(2) To redirect flow through the waste gas charcoal filters

45. Unit 1 Initial Conditions:

- The plant operated continuously at 100% power for a period of time before the team manually tripped the reactor due to turbine high vibrations.
- All plant systems and component operated as designed.

Current conditions:

- Offsite power is NORMAL.
- Both motor-driven Auxiliary Feedwater (AFW) pumps 1-FW-P-3A and 1-FW-P-3B are running.
- All Steam Generator (S/G) narrow range levels are LESS THAN 12%.
- Control room operators have transitioned to 1-ES-0.1, "REACTOR TRIP RESPONSE."

Based on the current conditions, which ONE of the following:

(1) identifies the core burnup at time of the trip that will result in the GREATER required AFW system flowrate to maintain S/G levels stable,

AND

(2) is the MINIMUM AFW flowrate required by ES-0.1 for the current plant conditions?

	CORE BURNUP	MINIMUM REQUIRED AFW FLOWRATE
A.	(1) 1,000 MWD/MTU	(2) 350 GPM
B.	(1) 1,000 MWD/MTU	(2) 540 GPM
C.	(1) 10,000 MWD/MTU	(2) 350 GPM
D.	(1) 10,000 MWD/MTU	(2) 540 GPM

46. Unit 2 Initial Conditions:

- 100% Power.
- The MCR Undervoltage (UV) bypass switches for turbine-driven Auxiliary Feedwater (AFW) pump 2-FW-P-2 steam supply valves 2-MS-PCV-202A and 2-MS-PCV-202B are in "BYPASS" position for routine maintenance.
- At the completion of maintenance, the UV bypass switches are returned to the "NORMAL" position, as required.
- Due to an electrical failure, all signals will respond as if the UV bypass switch was still in the "BYPASS" position. Operators are unaware of this condition.

Current conditions:

- A loss of all offsite power has occurred.
- #2 EDG loaded on 2H bus and #3 EDG loaded on 2J bus.
- Safety Injection (SI) is NOT actuated.

Based on the current conditions, which ONE of the following correctly identifies the expected plant response?

A. (1) 2-MS-PCV-202A and -202B will receive automatic open signals.
(2) AFW discharge valves 2-FW-MOV-251A through -251F will receive automatic open signals.

B. (1) 2-MS-PCV-202A and -202B will receive automatic open signals.
(2) AFW discharge valves 2-FW-MOV-251A through -251F will NOT receive automatic open signals.

C. (1) 2-MS-PCV-202A and -202B will NOT receive automatic open signals.
(2) AFW discharge valves 2-FW-MOV-251A through -251F will receive automatic open signals.

D. (1) 2-MS-PCV-202A and -202B will NOT receive automatic open signals.
(2) AFW discharge valves 2-FW-MOV-251A through -251F will NOT receive automatic open signals.

47. Unit 1 Initial Conditions:

- 100% Power.

Current conditions:

- Loss of letdown.
- Steam dump control NOT affected.
- Loss of Component Cooling to ALL RCP thermal barrier heat exchangers.
- Component Cooling to ALL other RCP heat exchangers is NOT affected.

Based on the current conditions, which ONE of the following correctly identifies the vital AC bus or buses that has/have been de-energized?

A. Vital Bus I is de-energized. Vital Buses II, III, and IV are energized.

B. Vital Bus I and Vital Bus III are de-energized. Vital Buses II and IV are energized.

C. Vital Bus II is de-energized. Vital Buses I, III, and IV are energized.

D. Vital Bus II and Vital Bus IV are de-energized. Vital Buses I and III are energized.

48. Unit 1 plant conditions:
Reactor power = 100%
UPS 1A1 Battery charger fails

Based on the above conditions, which ONE of the following actions will automatically occur to power loads on DC bus 1A?

A. UPS 1B1 will power DC bus 1A

B. Battery 1A will pickup loads for the next two hours

C. DC bus 1A will cross connect to DC bus 1B

D. UPS 1A2 will power DC bus 1A

49. A loss of 'A' DC Bus occurs followed by a Safety Injection. Which ONE of the following is correct regarding the operation of 1-SI-P-1A ('A' Low Head Safety Injection Pump)?

A. 1-SI-P-1A is NOT running but can be started from the MCR.

B. 1-SI-P-1A is NOT running and can NOT be started from the MCR.

C. 1-SI-P-1A is running and can be stopped from the MCR.

D. 1-SI-P-1A is running but can NOT be stopped from the MCR.

50. Unit 1 Initial Conditions:

- A spurious safety injection from 100% power occurred four (4) minutes ago.

Current conditions:

- An electrical grid transient has JUST resulted in a Station Blackout.

Based on the current conditions, which ONE of the following correctly identifies the SEQUENCE that ALL equipment will automatically load onto the **"H"** bus after EDG #1 re-energizes the bus? (assume NO operator action)

A. (1) 1-VS-F-58A (Filtered Exhaust Fan), THEN
(2) "E" group pressurizer heaters, THEN
(3) 1-FW-P-3A (Motor Driven Auxiliary Feedwater Pump)

B. (1) 1-VS-F-58A (Filtered Exhaust Fan), THEN
(2) 1-FW-P-3A (Motor Driven Auxiliary Feedwater Pump), THEN
(3) "E" group pressurizer heaters

C. (1) 1-FW-P-3A (Motor Driven Auxiliary Feedwater Pump), THEN
(2) 1-VS-F-58A (Filtered Exhaust Fan), THEN
(3) "E" group pressurizer heaters

D. (1) 1-FW-P-3A (Motor Driven Auxiliary Feedwater Pump), THEN
(2) "E" group pressurizer heaters, THEN
(3) 1-VS-F-58A (Filtered Exhaust Fan)

51. Unit 1 Initial Conditions:

- Radiography operations are in progress on a section of main steam piping.
- The radiographers want to verify the correct position of the camera by using a main steam line radiation monitor located on the same elevation and close to the area where the radiography needs to take place.
- To obtain a baseline reading, the camera source was placed 3.21 feet away from the radiation monitor detector. The radiation monitor read 5.92 R/hr.

Current conditions:

- The camera has been moved into position to image the piping section.
- Engineering calculations show that the camera should be placed 17.46 feet away from the radiation monitor detector.

The distances listed above include the difference in height from the camera to the radiation monitor detector. Consider the radiography camera as a radiation point source. Carry all calculations to three (3) decimal places.

Based on the current conditions, which ONE of the following correctly identifies the expected reading on the radiation monitor, if the camera was positioned correctly?

A. 0.037 R/hr

B. 0.200 R/hr

C. 1.088 R/hr

D. 2.538 R/hr

52. Initial conditions:
    - Unit One at 100% power
    - 1-SW-P-10A ("A" charging pump service water pump) in AUTO
    - 1-SW-P-10B ("B" charging pump service water pump) in HAND

    Current conditions:
    - "A" RSST was lost due to sudden pressure
    - All equipment operated as designed

    Assume sufficient time has elapsed to allow for all automatic actions to occur.

    Which ONE of the following states (1) the current status of the charging pump service water pumps and (2) why?

    A. (1) Both pumps currently in service
       (2) 1-SW-P-10B remained on the bus and 1-SW-P-10A started on low discharge header pressure.

    B. (1) Both pumps currently in service
       (2) 1-SW-P-10B remained on the bus and 1-SW-P-10A started due to opposite emergency bus undervoltage.

    C. (1) Only 1-SW-P-10B in service.
       (2) 1-SW-P-10B remained on the bus and 1-SW-P-10A started on low discharge header pressure but secured upon restoration of 1-SW-P-10B

    D. (1) Only 1-SW-P-10A in service.
       (2) 1-SW-P-10B tripped on undervoltage and 1-SW-P-10A started due to opposite emergency bus undervoltage..

53. Which ONE of the following states the power supplies DIRECTLY to both Unit 1 Rod Drive MG sets and what signal would DIRECTLY open the supply breakers to the MG Sets?

A. 'A' and 'C' 260 Volt Station Service
Reactor Trip

B. 'A' and 'C' 480 Volt Station Service
AMSAC

C. 'A' and 'C' 480 Volt Station Service
Reactor Trip

D. 'A' and 'C' 260 Volt Station Service
AMSAC

54. A loss of coolant accident (LOCA), coincident with a failure of ALL containment spray pumps to start, causes containment pressure to INCREASE.

Which ONE of the following correctly describes the expected equipment status as containment pressure continues to rise?

A. All containment recirculation fans will operate at pressures up to 17.7 psia. At 17.7 psia, containment recirculation fans 1A and 1B will automatically trip. Containment recirculation fan 1C will continue to run at pressures greater than 17.7 psia.

B. All containment recirculation fans will operate at pressures up to 23.0 psia. At 23.0 psia, containment recirculation fans 1A and 1B will automatically trip. Containment recirculation fan 1C will continue to run at pressures greater than 23.0 psia.

C. All containment recirculation fans will operate at pressures up to 17.7 psia. At 17.7 psia, all containment recirculation fans will automatically trip.

D. All containment recirculation fans will operate at pressures up to 23.0 psia. At 23.0 psia, all containment recirculation fans will automatically trip.

55. Unit 1 Initial Conditions:

- 100% Power.

Current conditions:

- Containment pressure transmitter 1-LM-PT-100B failed a calibration surveillance four (4) days ago.
- All Technical Specification 3.7, "Instrumentation Systems," required actions for 1-LM-PT-100B have been completed.

Based on the current conditions, which ONE of the following identifies (1) the MINIMUM containment pressure, AND (2) the MINIMUM number of OPERABLE containment pressure channels that must actuate in order to close 1-RM-TV-100A/B/C (Containment Particulate and Gas Radiation Monitor 1-RM-RI-159/160 trip valves)?

A. (1) 23.0 psia
(2) two (2)

B. (1) 23.0 psia
(2) three (3)

C. (1) 17.7 psia
(2) two (2)

D. (1) 17.7 psia
(2) three (3)

56. Unit 1 Initial Conditions:

- 100% Power.

Current conditions:

- A controller failure causes the Unit 1 Operator to place the Charging Flow Controller to MANUAL.
- The Unit 1 Operator attempts to reduce charging flow to 20 gpm to mitigate a high Pressurizer Level.

Based on the current conditions, which ONE of the following correctly describes the behavior of 1-CH-FCV-1122 when the Operator attempts to reduce charging flow to 20 gpm?

A. The Flow Limit Summator no longer limits flow, and 1-CH-FCV-1122 can be manually closed to allow 20 gpm flow.

B. The Flow Limit Summator no longer limits flow; however, 1-CH-FCV-1122 can only be manually closed to allow 25 gpm flow.

C. The Flow Limit Summator will limit charging flow to a minimum of 25 gpm.

D. The Flow Limit Summator will limit charging flow to a minimum of 30 gpm.

57. Initial Unit 1 Conditions:
- Unit 1 is at 100% power
- All control rods are fully withdrawn

Current Unit 1 Conditions:
- A control rod in the 'D' Control Bank drops to the bottom of the core
- Unit 1 is at 70% power
- Delta flux is within band

Which ONE of the following correctly states (1) the control rod select switch position to recover the rod in accordance with 0-AP-1.01, Control Rod Misalignment AND (2) when the step counters are required to be reset in accordance with 0-AP-1.01?

A. (1) Place the ROD CONT MODE SEL SWITCH to MANUAL for rod recovery.
(2) Step counters are required to be reset prior to rod recovery.

B. (1) Place the ROD CONT MODE SEL SWITCH to the affected bank for rod recovery.
(2) Step counters are required to be reset prior to rod recovery.

C. (1) Place the ROD CONT MODE SEL SWITCH to MANUAL for rod recovery.
(2) Step counters are NOT required to be reset until after the rod is recovered.

D. (1) Place the ROD CONT MODE SEL SWITCH to the affected bank for rod recovery.
(2) Step counters are NOT required to be reset until after the rod is recovered.

58. Unit 1 plant conditions:

Unit Startup in progress following a reactor trip at Middle of Life
Reactor power = 90%
1-GOP-1.5 UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER in progress
Axial Flux Difference = 0
All Control Rods are Fully Withdrawn at 225 steps

Based on the above conditions: (1) which ONE of the following states the maximum rate at which power can be increased to 100% IAW 1-GOP-1.5 (2) how will Axial Flux Difference change as power is increased ?

A. (1) 3% in 1 hour
(2) become positive

B. (1) 3% in 1 hour
(2) become negative

C. (1) No rate limitation exists
(2) become positive

D. (1) No rate limitation exists
(2) become negative

59. Initial Unit 2 conditions:
- Reactor power = 100% steady state

Current Unit 2 conditions:
- A LOCA is in progress
- 2-E-1, Loss of Reactor or Secondary Coolant, is being performed
- All RCPs have been stopped
- Containment pressure = 47 psia and slowly increasing
- Total AFW flow = 485 gpm
- SG WR levels are: "A" = 48%, "B" = 40%, "C" = 39%
- RCS pressure = 920 psig
- IR NIs = 2E-11 amps, with SUR = 0
- CETCs indicate 600°F
- RVLIS full range = 40%

Which ONE of the following correctly states the procedure to which the control room crew is required to transition?

A. FR-C.1, Response to Inadequate Core Cooling

B. FR-C.2, Response to Degraded Core Cooling

C. FR-Z.1, Response to Containment High Pressure

D. FR-H.5, Response to Steam Generator Low Level

60. Unit 1 initial conditions:

    Reactor trip from 40% power
    SI actuated
    1-E-0 REACTOR TRIP OR SAFETY INJECTION initiated

Current plant conditions:

    A NR SG level = 26% decreasing
    B NR SG level = 22% decreasing
    C NR SG level = 29% decreasing

Based on the above plant conditions, which ONE of the following correctly states (1) the MINIMUM SG level at which the first signal to start an AFW pump occurs (assuming no operator action) and (2) the MINIMUM required Steam Generator Narrow Range Level that allows SI flow reduction IAW 1-E-0 REACTOR TRIP OR SAFETY INJECTION?

A. (1) 13%
   (2) Greater than 12%

B. (1) 13%
   (2) Greater than 22%

C. (1) 17%
   (2) Greater than 12%

D. (1) 17%
   (2) Greater than 22%

61. Unit 1 Initial Conditions:

- 100% power.
- Rod control is selected to P-446, Channel III turbine first stage impulse pressure.
- P-447, Channel IV turbine first stage impulse pressure, fails LOW.
- Annunciator 1H-D7, "STM DUMP PERM," is lit.

Current conditions:

- No operator actions have been performed to address the P-447 failure.
- Control rods are INSERTING in automatic.
- Steam flow on all channels is INCREASING.

Based on the current conditions, which ONE of the following correctly identifies the cause?

A. A main steam line safety valve has lifted and will not reseat.

B. Median Tave has failed HIGH.

C. P-446 has failed HIGH.

D. P-464, Steam header pressure, has failed HIGH.

62. Unit 1 Initial Conditions:

- Unit was at 28% power when condenser vacuum began to degrade.

Current conditions:

- Annunciator 1E-E3, "DELTA FLUX DEVIATION," is lit.
- Annunciator 1G-H8, "ROD BANK D EXTRA LO LIMIT," is lit.
- Annunciator 1F-B6, "TURB LO VAC," has been lit for five (5) minutes.
- Control rods are inserting in automatic.
- Condenser vacuum continues to DECREASE with no signs of recovery.
- Turbine is at 14% load and DECREASING.

Based on the current conditions, which ONE of the following identifies the required operator action, in accordance with 1-AP-14.00, "LOSS OF MAIN CONDENSER VACUUM?"

A. Commence an emergency boration using 1-AP-3.00, "EMERGENCY BORATION."

B. IMMEDIATELY trip the Reactor and enter 1-E-0, "REACTOR TRIP OR SAFETY INJECTION."

C. IMMEDIATELY trip the Turbine and stabilize the unit using the steam dumps.

D. IF condenser vacuum is less than 24.5 in-Hg for a five (5) minute period, THEN trip the Turbine and stabilize the unit using the steam dumps.

63. Unit 1 initial conditions:
    Shut down for refueling
    Containment purge in progress

Current plant conditions:
    A High radiation signal on the containment particulate (1-RM-RI-159 CTMT PARTC) radiation monitor occurs

Based on the above conditions which ONE of the following correctly states the status of the containment purge components?

A. Containment purge supply fans (1-VS-F-4A and B) off
   Containment purge supply MOVs (1-VS-MOV-100A and B) closed,
   Containment purge discharge MOVs (1-VS-MOV-100C and D) closed

B. Containment purge supply fans (1-VS-F-4A and B) off
   Containment purge supply MOVs (1-VS-MOV-100A and B) closed,
   Containment purge discharge MOVs (1-VS-MOV-100C and D) open

C. Containment purge supply fans (1-VS-F-4A and B) on
   Containment purge supply MOVs (1-VS-MOV-100A and B) closed,
   Containment purge discharge MOVs (1-VS-MOV-100C and D) open

D. Containment purge supply fans (1-VS-F-4A and B) on
   Containment purge supply MOVs (1-VS-MOV-100A and B) open,
   Containment purge discharge MOVs (1-VS-MOV-100C and D) open

64. Plant initial conditions:

Reactor Power = 100% both units
Instrument Air Systems are in their NORMAL alignments (split out)
Operator reports an air leak on Unit 1 instrument air header
Unit 1 instrument air pressure is currently 85 psig and DECREASING

Based on the above conditions, which ONE of the following correctly states the expected status of (1) Unit 2 Instrument Air pressure and (2) the Unit 1 Instrument Air Compressor?

A. (1) Unit 2 Instrument Air pressure will decrease until air pressure equals 80 psig.
(2) operating.

B. (1) Unit 2 Instrument Air pressure will decrease until air pressure equals 80 psig.
(2) NOT operating.

C. (1) Unit 2 Instrument Air pressure will decrease until 2-IA-C-1 starts.
(2) operating.

D. (1) Unit 2 Instrument Air pressure will decrease until 2-IA-C-1 starts.
(2) NOT operating.

65. Which ONE of the following states (1) the capacity of each Fire Water Tank and (2) if a domestic water leak occurred, the tank level at which the leak would stop?

A. (1) 250,000 gallons per tank
(2) 200,000 gallons per tank

B. (1) 300,000 gallons per tank
(2) 250,000 gallons per tank

C. (1) 300,000 gallons per tank
(2) 200,000 gallons per tank

D. (1) 250,000 gallons per tank
(2) 50,000 gallons per tank

66. Which ONE of the following (1) correctly states the maximum allowable length of a valve wrench used IAW OP-AA-100, Conduct of Operations, AND (2) whether OP-AA-100 allows a valve wrench to be used on manual valves as well as motor operated valves (MOVs)?

A. (1) Valve wrench length is limited to approximately 1.5 times the handwheel diameter.

   (2) Valve wrench is permitted to be used on manual valves but not MOVs.

B. (1) Valve wrench length is limited to approximately 2.0 times the handwheel diameter.

   (2) Valve wrench is permitted to be used on manual valves but not MOVs.

C. (1) Valve wrench length is limited to approximately 1.5 times the handwheel diameter.

   (2) Valve wrench is permitted to be used on both manual valves and MOVs.

D. (1) Valve wrench length is limited to approximately 2.0 times the handwheel diameter.

   (2) Valve wrench is permitted to be used on both manual valves and MOVs.

67. Unit 1 Initial Conditions:

- Core re-fueling operations are in progress.
- Approximately 3/4 of the new core has been loaded without incident.

Current conditions:

- One Source Range count rate is double (2X) the initial reference value.
- The other Source Range count rate is (1.75X) (less than double) the initial reference value.
- The 1/M plot is approaching 0.65.

Based on the current conditions, which ONE of the following identifies the MINIMUM conditions that would require stopping core alterations, in accordance with the Precautions and Limitations of 1-OP-FH-001, "CONTROLLING PROCEDURE FOR REFUELING?"

A. Core alterations are required to be stopped immediately and subcriticality reevaluated.

B. Core alterations may continue, but IF BOTH Source Range count rates reach one doubling from the reference value, then core alterations are required to be stopped immediately and subcriticality reevaluated.

C. Core alterations may continue, but IF the 1/M plot approaches 0.5, then core alterations are required to be stopped immediately and subcriticality reevaluated.

D. Core alterations may continue, but IF BOTH Source Range count rates reach one doubling from the reference value, AND the 1/M plot approaches 0.5, then core alterations are required to be stopped immediately and subcriticality reevaluated.

68. Unit 1 initial conditions:
Core re-load in progress
SR NI background count rate = 10 cps

Current plant conditions:
1G-C1, NIS SOURCE RNG SHUTDN HI FLUX, alarms

Based on the above conditions, which ONE of the following correctly states (1) the minimum count rate that would cause the alarm and (2) what actions are directed by ARP 1G-C1?

A. (1) 42 cps
(2) Direct the refueling SRO to place fuel in a safe condition and evacuate containment.

B. (1) 42 cps
(2) Emergency borate and direct the refueling SRO to stop all refueling activities.

C. (1) 60 cps
(2) Direct the refueling SRO to place fuel in a safe condition and evacuate containment.

D. (1) 60 cps
(2) Emergency borate and direct the refueling SRO to stop all refueling activities.

69. With the unit initially at 100% power, 1-MS-PT-1446 (Ch III Impulse pressure) fails to 0%. Unit conditions are as follows:

- Reactor power 95% - by delta-T
- "D" bank control rods at 212 steps
- Delta Flux is currently -7
- Delta flux target is -1.5
- Tave is 568F
- Tref is 572F
- Steam generator levels are stable at 33% narrow range.
- RCS pressure is currently 2200 psig

Which ONE of the following states the MOST LIMITING LCO (if any), and required actions?

A. No LCO actions are required.

B. 15 minute clock to restore delta-flux in band due to delta flux being outside target band.

C. 30 minute clock to restore pressurizer pressure as pressurizer pressure is outside the allowable band.

D. 1 hour clock to verify permissive status due to pimp failure to P-10 and P-7 interlocks

70. Unit One at 100% power and stable.

Chemistry morning report has been issued with the following parameters given:

- Unit One "A" safety injection accumulator (1-SI-TK-1A) - boron concentration- 2235 ppm.
- Unit One "B" safety injection accumulator (1-SI-TK-1B) - boron concentration- 2500 ppm.
- Unit One "C" safety injection accumulator (1-SI-TK-1C) - boron concentration- 2300 ppm.

- "A" Waste Gas Decay Tank- 1.65% oxygen concentration.
- "A" Waste Gas Decay Tank- 25,720 curies.

Which ONE of the following states **ALL** the above parameters that require entry into a Technical Specification LCO?

A. "A" accumulator boron and "A" Waste Gas Decay Tank O_2 content

B. "B" accumulator boron and "A" Waste Gas Decay Tank O_2 content

C. "A" accumulator boron and "A" Waste Gas Decay Tank curie content.

D. "B" accumulator boron and "A" Waste Gas Decay Tank curie content.

71. You are assigned to oversee work being performed in a Radiation area.

Which ONE of the following describes: (1) the types of radiation that are measured by the "DAD" and (2) the requirements for DAD placement if work is to be performed in a contaminated area?

A. (1) Gamma & X-Ray ONLY
   (2) Inside protective clothing with TLD

B. (1) Gamma & X-Ray ONLY
   (2) Outside protective clothing in a whirlpack

C. (1) Gamma, Beta and Neutron
   (2) Inside protective clothing with TLD

D. (1) Gamma, Beta and Neutron
   (2) Outside protective clothing in a whirlpack

72. While taking LOGS in the auxiliary building, a mechanic, who is performing an overhaul on 1-CH-P-1A ("A" charging pump), approaches you and asks for assistance in lifting the auxiliary oil pump. He states that he will only require your assistance for 20-30 minutes.

Which ONE of the following states the proper response to this request?

A. Provide assistance and when logs are complete, ask health physics to assign the dose received while helping the mechanic to the mechanic's RWP.

B. Render the requested assistance on Operations RWP as long as the dose received will not cause you to reach either your DOSE RATE LIMIT or DOSE LIMIT.

C. Inform the mechanic that you are unable to render the requested assistance while on the current Operations RWP.

D. Call health physics shift supervisor and request to be placed on the mechanic's RWP. When complete, contact the health physics shift supervisor again, and get reassigned to the normal operations RWP.

73. Unit 1 initial conditions:

Reactor Trip

Critical safety functions as follows.

SUBCRITICALITY - GREEN
HEAT SINK - ORANGE
CORE COOLING - ORANGE
INVENTORY - YELLOW
CONTAINMENT - RED
INTEGRITY - ORANGE

Based on the above conditions, when addressing Critical Safety Functions (CSFs) which ONE of the following CSFs has the highest priority and should therefore be addressed first?

A. Heat Sink

B. Core Cooling

C. Containment

D. Integrity

74. Unit 1 initial conditions:

Loss of all feedwater has occurred from 100% power
EOPs are progress

Current plant conditions:

Transition to 1-FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK has just been made
1A Wide Range SG level = 4% decreasing
1B Wide Range SG level = 5% decreasing
1C Wide Range SG level = 6% decreasing
RCS pressure = 2300 psig increasing
CETC = 580 ^{o}F increasing
All RCPs are secured

Based on the above conditions: (1) which ONE of the following actions are directed by 1-FR-H.1 and (2) why?

A. (1) Commence bleed and feed
(2) At least Two SGs are considered dry so transition to another form of decay heat removal must be made before conditions degrade further

B. (1) Commence bleed and feed
(2) RCS pressure may reach pressurizer safety valve setpoints, so transition to another form of decay heat removal must be made to prevent water relief through the safety valves.

C. (1) Cross Connect with Unit 2 AFW and feed at the maximum available rate
(2) To reduce RCS temperature to < 550 ^{o}F for establishing a heat sink

D. (1) Cross Connect with Unit 2 AFW and feed at the maximum available rate
(2) To increase SG level to > 7% in any SG for establishing a heat sink

75. Which ONE of the following describes that actions required on a failure of the reactor to trip (ATWS) in accordance with 1-FR-S.1 (Response to Nuclear Power Generation/ATWS)?

A. Place rod control in MANUAL and manually trip the turbine. If turbine will not trip, then close the main steam trip valves. Manually insert control rods.

B. Place rod control in MANUAL and manually trip the turbine. If turbine will not trip, then reduce limiter to zero. Manually insert control rods.

C. Place rod control in AUTOMATIC and manually trip the turbine. If turbine will not trip, then reduce limiter to zero. Verify automatic rod insertion.

D. Place rod control in AUTOMATIC and manually trip the turbine. If turbine will not trip, then close the main steam trip valves. Verify automatic rod insertion.

You have completed the test!

- EAL Tables
- Steam Tables
- Calculator
- This Document

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: The NSS Point of Contact is identified on the Plan of the Day (POD).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. ___	CONSERVE INTAKE CANAL INVENTORY: ☐ • Within 2 hours, start all available ESW pumps IAW 0-OP-SW-002, EMERGENCY SERVICE WATER PUMP OPERATION	☐ Notify NSS Point of Contact to initiate the INTAKE CANAL ALTERNATE MAKEUP GUIDELINE.
18. ___	CHECK CW PUMPS - AT LEAST ONE RUNNING	☐ Start CW pumps IAW OP-48.1.1, STARTING ANY CW PUMP.
19. ___	STOP ALL LIQUID RELEASES: ☐ • LW ☐ • BR ☐ • CP BLDG sumps	
20. ___	CHECK SW TO RS HXS ON EITHER UNIT - IN SERVICE	☐ GO TO Step 23.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Based on heat load, UNIT AT POWER is defined as any Unit which is actually at power or any Unit which has been shutdown for less than 35 days.

21.___ DETERMINE ALLOWABLE CC HX SW OUTLET VALVE POSITIONS FOR NON-ACCIDENT UNIT HXs:

INITIAL UNIT CONDITIONS	Allowable CC HXs and SW outlet valve positions on non-accident Unit
BOTH Units at Power	2 CC HXs with SW outlet valves open 19 turns for each HX
1 Unit at power and 1 Unit shutdown for greater than 35 days	CONDITION 1: With 1 ESW pump operating • 1 CC HX with SW outlet valve open 14 turns
	CONDITION 2: With 2 ESW pumps operating • 1 CC HX with SW outlet valve open 19 turns

22.___ GO TO STEP 24

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Based on heat load, UNIT AT POWER is defined as any Unit which is actually at power or any Unit which has been shutdown for less than 35 days.

23.___ DETERMINE ALLOWABLE CC HX SW OUTLET VALVE POSITION FOR HXs FOR BOTH UNITS

INITIAL UNIT CONDITIONS	Allowable CC HXs and SW outlet valve positions
BOTH Units at Power	Crosstie CC. 3 CC HXs allowed with SW outlet valves 19 turns open for each HX.
• 1 Unit at power AND • 1 Unit shutdown for greater than 35 days	CONDITION 1: 1 ESW pump operating • Unit at Power: 2 CC HXs with SW outlet valves open 19 turns for each HX. • Unit shutdown: 1 CC HX with SW outlet valve open 14 turns.
	CONDITION 2: 2 ESW pumps operating • Unit at Power: 2 CC HXs with SW outlet valves open 19 turns for each HX. • Unit shutdown: 1 CC HX with SW outlet valve open 19 turns.
BOTH Units shutdown for greater than 35 days	1 CC HX for each Unit with SW outlet valve open 19 turns for each HX.

6.3.3 **One-hour Notifications**

NOTE: Some conditions, indicated by "See EPIP-1.01," may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

As soon as practical, but within one hour, the Shift Manager, Station Emergency Manager, or Site Vice President shall notify the NRC Operations Center of:

a. Deviation from Technical Specifications (permitted by 10 CFR 50.54(x)) to protect the health and safety of the public, when no action consistent with license conditions and Technical Specifications can provide adequate or equivalent protection. [10 CFR 50.72(b)(1)]

b. An automatic safety system that does not function as required during operation. See EPIP-1.01. [10 CFR 50.36(d)(1)(ii)(A)]

NOTE: Notifications required by Steps 6.3.3.c., 6.3.3.d., and 6.3.3.e., are exempt from the requirement that Safeguards Information be transmitted only by protected telecommunications circuits approved by NRC.

c. An accidental criticality or loss of SNM. See EPIP-1.01. [10 CFR 70.52 (a), 10 CFR 72.74(a), 10 CFR 74.11a]

NOTE: Step 6.3.3.d. notifications need not duplicate Step 6.3.3.e. notifications. [10 CFR 74.11(c), 10 CFR 72.74(c)]

d. A loss of any [10 CFR 73.71(a)(1), 10 CFR 73.67(e)(3)(vii), 10 CFR 73.67(g)(3)(iii)]:

- SNM shipment
- Spent fuel shipment

or

Availability of supplemental information after initial notification. [10 CFR 73.71(a)(5)] (See also Step 6.15.3.a.3.)

or

Recovery of or accounting for such lost shipment.
See also Step 6.15.3.a.2. [10 CFR 73.71(a)(1), 10 CFR 73.67(e)(3)(vii), 10 CFR 73.67(g)(3)(iii)]

NOTE: Steps 6.3.3.e., 6.3.3.f., 6.3.3.g., 6.3.3.h.notifications need not duplicate Step 6.3.3.d. or 10 CFR 50.72 notifications. [10 CFR 72.74(c), 10 CFR 73.71(e), 10 CFR 74.11(c)]

e. A reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause (See also Step 6.15.3.b.2.).
[10 CFR 73.71(b)(1), 10 CFR 73 App. G.I, 10 CFR 70.52 (a), 10 CFR 72.74(a), 10 CFR 74.11(a)]:

- Theft, loss, or unlawful diversion of SNM
- Significant physical damage to the Station, nuclear fuel, or carrier of nuclear fuel
- Interruption of normal operation through unauthorized use of or tampering with its machinery, components, or controls, including the security system

f. Unauthorized entry into a protected area, material access area, controlled access area, vital area, or transport.

g. Failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, controlled access area, vital area, or transport for which compensatory measures have not been employed.

NOTE: Fitness-for-duty events are reported in accordance with 10 CFR 26 instead of 10 CFR 73.71. See Steps 6.3.6.b. and 6.8.1. [10 CFR 26.73(c)]

h. Actual or attempted introduction of contraband into a protected area, material access area, or transport.

i. Discovery that an undeclared or misclassified event or condition met all the following criteria:

- Exceeded an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure
- The basis for the emergency class no longer exists at the time of discovery
- No other reasons exist for an emergency declaration

In addition, the following shall be notified:

- Department of Emergency Management (at approximately the same time)
- Director Nuclear Protection Services and Emergency Preparedness

6.3.4 **Four-hour Notifications**

NOTE: Some conditions, indicated by "See EPIP-1.01," may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

a. As soon as practical, but within four hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

NOTE: If a unit enters a limiting condition for operation (LCO) and a unit shutdown is started due to the LCO, the event is reportable even if shutdown is not completed. LCOs terminated by a unit shutdown for an unrelated reason are still reportable if the condition would not have been corrected within the LCO time limit for shutdown.

1. Initiation of plant shutdown (reduction of power or temperature) required by Technical Specifications. The initiation of plant shutdown does not include mode changes required by Technical Specifications if initiated after the plant is already in a shutdown condition. See EPIP-1.01. **[10 CFR 50.72(b)(2)(i), 10 CFR 50.36(d)(1)(i)(A), 10 CFR 50.36 (d)(2)(i), NUREG 1022 Item 3.2.1]**
2. Any event that results or should have resulted in ECCS discharge into the RCS as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. **[10 CFR 50.72(b)(2)(iv)(A)]**
3. Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when actuation results from and is part of a pre-planned sequence during testing or reactor operation. **[10 CFR 50.72(b)(2)(iv)(B)]**

NOTE: "Notification to other government agencies has been or will be made" is not necessarily an automatic notification to the NRC. Refer to NUREG – 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, for discussions and examples or contact Station Licensing if clarification is needed. [NUREG-1022, Section 3.2.12]

4. Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned, or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials. **[Commitment 3.2.16]** [10 CFR 50.72(b)(2)(xi)]
5. ISFSI Non-emergency Four-Hour Notifications shall include, if available at time of notification: [10 CFR 72.75(e)(3)]
   - The caller's name and call back telephone number
   - A description of the event, including time and date
   - The exact location of the event
   - The quantities, and chemical and physical forms of the spent fuel, HLW or reactor related Greater than Class C (GTCC) waste involved
   - Any personnel radiation exposure data
6. An action taken in an emergency that departs from a license condition, technical specification, or certificate of compliance when the action is immediately needed to protect the public health and safety and no licensed action that provides adequate or equivalent protection is immediately apparent—see Step 6.14.7.f. [10 CFR 72.75(b)(1)]
7. An event at the ISFSI that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination. [10 CFR 72.75(c)(3)]

8. Groundwater Protection Voluntary Communication Notifications to other government agencies may be reportable under 10 CFR 50.72(b)(2)(xi) requirement for a 4-hour notification to the NRC operations center based upon the following guidance:
   - If a licensee is notifying a local, state, or other federal agency in accordance with an existing law, regulation, or ordinance, then the licensee should make its notification to the NRC under the 50.72 notification requirement.
   - If a licensee is informally communicating with a local, state, or other federal agency (i.e., not under a specific law, regulation or ordinance), then the licensee has discretion as to whether to informally communicate with NRC (e.g., through the site resident inspector and/or regional NRC office) or formally through the 50.72 notification process. If due to the site-specific circumstances or heightened sensitivity to the issue at that site, the issue is likely to produce strong media interest, then the licensee should consider notifying NRC under the 50.72 requirement because this is actually the underlying intent of the regulation.

b. Any person at the Station who observes smoke originating from Station equipment being released into the outdoor atmosphere shall notify the Shift Manager as soon as possible.

1. If the smoke is not from a fire and there are no certified visible emissions evaluators available to determine the opacity of the smoke being released to the outdoor atmosphere, the Shift Manager or other Station personnel shall take the appropriate steps to determine the source, cause, and duration of the smoke being released.
   - Once all of the pertinent information regarding the release of smoke has been obtained, the Electric Environmental Services (ESS) must be notified immediately.
   - The ESS will report the release of smoke into the outdoor atmosphere to the appropriate DEQ regional office as soon as practical, but no later than four daytime business hours of the occurrence, with all of the pertinent information. If the DEQ regional office determines that it is necessary to obtain smoke readings after receiving all of the pertinent information, the ESS will dispatch a certified visible emissions evaluator to the Station to determine the opacity of the smoke being released into the outdoor atmosphere.
2. The ESS will prepare and submit any written reports to the DEQ regional office regarding the release of smoke into the outdoor atmosphere.

6.3.5 **Eight-hour Notifications**

a. As soon as practical, but within eight hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

1. Any condition that results in the condition of the Station, including its principal safety barriers, being seriously degraded. [10 CFR 50.72(b)(3)(ii)(A)]

2. Any event or condition that results in the Station being in an unanalyzed condition that significantly degrades plant safety. [10 CFR 50.72(b)(3)(ii)(B)]

3. Any event or condition that results in valid actuation of any of the following systems, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation: [10 CFR 50.72(b)(3)(iv)(A)]
   - Reactor Protection System (RPS) - (RPS actuation with the reactor critical may be reportable within 4 hours under 10 CFR 50.72(b)(2)(iv)(B), see Step 6.3.4.a.3.)
   - General containment isolation signals affecting containment isolation valves in more than one system or multiple Main Steam Isolation Valves (MSIVs)
   - Emergency Core Cooling Systems (ECCS) including HHSI and LHSI (Actual discharges are reportable within 4 hours under 10 CFR 50.72(b)(2)(iv)(A), see Step 6.3.4.a.2.)
   - Auxiliary Feedwater System
   - Containment heat removal and depressurization systems including Containment spray and fan cooler systems
   - Emergency Diesel Generators (EDGs)

4. Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:
   - Shut down the reactor and maintain it in a safe shutdown condition
   - Remove residual heat
   - Control the release of radioactive material; or
   - Mitigate the consequences of an accident. See EPIP-1.01. [10 CFR 50.72(b)(3)(v)]

5. Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment. See also Step 6.27.2. [10 CFR 50.72 (b)(3)(xii)]
   Could also be a 4 hour report in accordance with 10 CFR 72.75 (b)(5).

6. An event that results in a major loss of emergency assessment capability[1], off-site response capability, or off-site communications capability, e.g., unavailability of any of the following (see Attachment 3, Emergency Response Unavailability, for unavailability criteria)[2]:
   - Safety Parameter Display System[3] (SPDS)
   - Emergency response facilities[4] (see Subsection 4.15)
   - Emergency communications facilities and equipment[5]
   - Prompt Notification System, including sirens
   - Plant monitors necessary for accident monitoring

   See EPIP-1.01. [10 CFR 50.72(b)(3)(xiii)]

7. Any instance of:
   - A defect in any spent fuel storage cask structure, system, or component that is important to safety [10 CFR 72.75(c)1]

     or

   - A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use of the storage cask [10 CFR 72.75(c)2

   See EPIP-1.01.

b. If an Alert, Site Area Emergency, or General Emergency is declared:

1. The Station Coordinator Emergency Preparedness shall prepare a Summary Report from information in completed Emergency Plan Implementing Procedures, Control Room logs, and interviews with persons involved with the declaration and response, as appropriate. See Attachment 8, Example DEM Summary Report.

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1. A major loss of emergency assessment capability includes events that significantly impair fulfillment of the Emergency Plan, including safety assessment capability (e.g., loss of a significant portion of Control Room indications). Loss of on-site meteorological information does not constitute a major loss of assessment capability and should not be reported under this part.
2. Engineering judgment may be needed to assess the significance of losing certain equipment.
3. Unavailability of only the SPDS (one function of the Plant Computer System (PCS)) for less than eight hours is **not** reportable, but unavailability of the SPDS and other assessment capability at the same time **may** be reportable. Scheduled PCS outages or operation of PCS in the Simulator mode are **not** reportable if the SPDS can be made available in less than one hour.
4. EOF loss is reportable only if **both** the LEOF **and** the CEOF are unavailable.
5. A momentary loss of off-site response capability or emergency communications (e.g., the backup power supply fails while security computer and emergency communications are temporarily connected to perform a surveillance test) is **not** reportable.

2. The Site Vice President, Director Nuclear Station Safety and Licensing, or Plant Manager (Nuclear) shall approve the report.

3. Within 8 hours after termination of the event, Nuclear Emergency Preparedness shall ensure the report is delivered to the State Coordinator of the Virginia Department of Emergency Management. [NAEP 4.4; SEP 4.4]

c. If, on Dominion property or at Lake Anna Dam, there is a Dominion employee or contractor fatality (regardless of the time between the injury and death, or the length of an illness) or an event in which three or more Dominion employees or contractors are hospitalized:

1. The Shift Manager shall notify Supervisor Nuclear Site Safety (Station) with the following information:
   - Number of fatalities
   - The employer of those killed
   - The circumstances of the event
   - The extent of injuries

2. Nuclear Site Safety (Station) shall notify OSHA as specified in Step 6.3.5.c.3. See also Step 6.3.4.a.4.

3. Within eight hours after the occurrence, the Supervisor Nuclear Site Safety (Station) (as specified in Step 6.3.5.b.2.) shall notify See Step 6.3.1.a.) the Area Director of OSHA by telephone or facsimile. See Step 6.1.1.a. See also Step 6.3.4.a.4. [29 CFR 1904.8]

d. Whenever fire protection systems, portions of a system, or equipment are impaired or reduced in status for other than scheduled maintenance or scheduled testing activities (meaning an unplanned failure or state of degradation), the Shift Manager shall notify the Supervisor Nuclear Site Safety (Station). **[Commitment 3.2.21] (Surry)**

North Anna notification to the Supervisor Nuclear Site Safety (Station) is within 48 hours per TRM requirements.

6.3.6 **Twenty-four Hour Notifications**

a. As soon as practical, but within 24 hours, the Shift Manager shall notify the NRC Operations Center with the ENS of [10 CFR 20.2202(b)]:

NOTE: The requirements of Step 6.3.6.a.1. do not apply to doses that result from planned special exposures, that are within the limits for planned special exposures, and that are reported in accordance with Step 6.10.11.c. [10 CFR 20.2202(e)]

1. An event that involves licensed material possessed by Dominion that may have caused or threatens to cause:
   - An individual to receive, in a period of 24 hours:
     - A total effective dose equivalent exceeding 5 rems
     - An eye dose equivalent exceeding 15 rems
     - A shallow-dose equivalent to the skin or extremities exceeding 50 rems
   - Release of radioactive material inside or outside a restricted area, so that, if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

   If an event involves radiological overexposure, DEM must be notified as specified in Step 6.27.2. See also Step 6.6.3.c.

2. ISFSI Twenty-Four Hour Notifications shall include, if available at time of notification: [10 CFR 72.75(e)(3)]
   - The caller's name and call back telephone number
   - A description of the event, including time and date
   - The exact location of the event
   - The quantities, and chemical and physical form of the spent fuel or HLW involved
   - Any personnel radiation exposure data

3. An unplanned contamination event that requires access to the contaminated area by workers or the public to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area [10 CFR 72.75(c)(1)]

4. An event in which safety equipment is disabled or fails to function as designed when: [10 CFR 72.75(d)(1)]
   - The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposure to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident,

   and

   - No redundant equipment was available and operable to perform the required safety function

5. An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive material that could exceed regulatory limits or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases)—see Step 6.14.7.f. [10 CFR 72.75(d)(1)(i)]

b. Within 24 hours after discovery of a significant fitness for duty event, a Director shall notify the NRC Operations Center by telephone. See Step 6.1.1. [10 CFR 26.73(b)]

   1. The notifier shall document the notification in Section B of Attachment 4, Significant Fitness for Duty Event NRC 24 Hour Notification.

   2. The notifier shall return the completed original of Attachment 4 to the Fitness for Duty Administrator (Station) for further processing. See Step 6.8.1.

c. Within 24 hours, the Shift Manager shall notify NRC by telephone, telegraph, or facsimile, of any occurrence of an unusual or important event—causally related to Station operation—that indicates or could result in significant environmental impact. See also Step 6.26.2.b. **(North Anna)** [NAPS EPP 4.1 & 5.4.2]

d. Within 24 hours after discovery, Licensing (Station) shall notify (see Step 6.3.1.a.) the NRC Regional Office by telephone of failure to notify NRC of planned removal or significant change in the normal operation of equipment that controls the amount of radioactivity in Station effluents **(North Anna)**.

   [NAPS Unit 1 License, 2.C(3)(b); Unit 2 License, 2.C(3)(a).]

   By the first business day after discovery, Licensing (Station) shall confirm the telephone notification by telegram, mailgram, or facsimile to the NRC Regional Office. See also Step 6.23.6.

e. If any unpermitted, unusual, or extraordinary discharge[1] enters or could be expected to enter State waters, as soon as possible, but not later than 24 hours after discovery, Electric Environmental Services shall notify (see Step 6.3.1.a.) the State Department of Environmental Quality (Water). See also Steps 6.3.4.a.4., 6.3.2.f., and 6.27.3.n. [VPDES Permit]

f. If an unplanned bypass (i.e., intentional diversion of waste streams) occurs from any portion of a treatment works, as soon as possible, but not later than 24 hours after the bypass occurs, Electric Environmental Services shall notify (see Step 6.3.1.a.) the State Department of Environmental Quality (Water). [VPDES Permit]

6.3.7 **Seventy-two Hour Notifications**

If a Notification of Unusual Event is declared:

a. The Station Coordinator Emergency Preparedness shall prepare a Summary Report from information in completed Emergency Plan Implementing Procedures, Control Room logs, and interviews with persons involved with the declaration and response, as appropriate. See Attachment 8, Example DEM Summary Report.

b. The Site Vice President, Director Nuclear Station Safety and Licensing, or Plant Manager (Nuclear) shall approve the report.

c. Nuclear Emergency Preparedness shall ensure the report is delivered to the State Coordinator of DEM within 72 hours after the declaration. [NAEP 4.4; SEP 4.4]

1. Unusual or extraordinary discharge includes, but is not limited to: a) unplanned bypasses, b) upsets, c) spillage of materials resulting directly or indirectly from processing operations or pollutant management activities, d) breakdown of processing or accessory equipment, e) failure of or taking out of service, sewage or industrial waste treatment facilities, auxiliary facilities, or pollutant management activities, or f) flooding or other acts of nature. [VPDES Permit]

ANSWER KEY REPORT
for RO Portion of Exam Test Form: 0

#	ID	Points	Type	Answers 0
1	000W/E13 EK2.2 1	1.00	MCS	A
2	0012 A4.07 1	1.00	MCS	B
3	0013 K2.01 2	1.00	MCS	A
4	0022 K1.04 2	1.00	MCS	B
5	000008AA2.20 2	1.00	MCS	D
6	000009EK2.03 3	1.00	MCS	A
7	000011EK3.12 5	1.00	MCS	A
8	000015AA2.09 2	1.00	MCS	C
9	000022AG2.4.11 2	1.00	MCS	C
10	000025AK1.01 3	1.00	MCS	A
11	000027AK3.03 2	1.00	MCS	C
12	000029EA1.02 2	1.00	MCS	D
13	000038EA1.21 2	1.00	MCS	B
14	000040AA2.05 1	1.00	MCS	D
15	000055EK1.02 1	1.00	MCS	D
16	000056AA1.29 3	1.00	MCS	A
17	000057AA2.20 2	1.00	MCS	B
18	000058AG2.1.23 3	1.00	MCS	D
19	00005AA2.01 2	1.00	MCS	A
20	000065AA1.03 3	1.00	MCS	A
21	000077AA2.07 1	1.00	MCS	C
22	0000W/E04 EA2.1 2	1.00	MCS	B
23	0000W/E05 EA1.1 2	1.00	MCS	A
24	0003A3.04 3	1.00	MCS	A
25	0003A4.08 3	1.00	MCS	B
26	0004A4.08 3	1.00	MCS	B
27	0005 G2.4.9 3	1.00	MCS	D
28	00068 AG2.4.42 2	1.00	MCS	D
29	0006K1.05 2	1.00	MCS	C
30	0007A1.03 1	1.00	MCS	B
31	0008 A2.04 2	1.00	MCS	C
32	0008 G2.1.7 2	1.00	MCS	A
33	000W/E07 EK2.1 2	1.00	MCS	A
34	000W/E08 EK1.3 3	1.00	MCS	A
35	000W/E16 EA1.1 2	1.00	MCS	D
36	0010 A3.02 4	1.00	MCS	A
37	0012 K1.05 2	1.00	MCS	A
38	0026 K3.02 3	1.00	MCS	A
39	0036AA2.01 2	1.00	MCS	C
40	0037AG2.4.4 2	1.00	MCS	A
41	0039 A1.05 1	1.00	MCS	C
42	0039 K4.04 2	1.00	MCS	A
43	0059 K4.16 3	1.00	MCS	B
44	0060AK3.02 2	1.00	MCS	A
45	0061 K5.02 2	1.00	MCS	D
46	0061 K6.01 2	1.00	MCS	A

ANSWER KEY REPORT

for RO Portion of Exam Test Form: 0

#	ID	Points	Type	Answers 0
47	0062 K2.01 1	1.00	MCS	A
48	0062 K3.03 1	1.00	MCS	D
49	0063 K3.02 2	1.00	MCS	B
50	0064 K4.10 2	1.00	MCS	C
51	0073 K5.02 1	1.00	MCS	B
52	0076 K4.02 3	1.00	MCS	A
53	01 K2.05 2	1.00	MCS	B
54	0103 A1.01 1	1.00	MCS	B
55	0103 K4.06 1	1.00	MCS	C
56	011 K6.06 2	1.00	MCS	A
57	014A4.01 2	1.00	MCS	B
58	015K5.10 3	1.00	MCS	D
59	017K5.02 2	1.00	MCS	B
60	035A2.01 2	1.00	MCS	C
61	041 A3.03 2	1.00	MCS	B
62	055 G2.4.45 2	1.00	MCS	B
63	072 K4.01 1	1.00	MCS	A
64	078 K3.03 2	1.00	MCS	C
65	086 A1.05 2	1.00	MCS	B
66	G2.1.29 1	1.00	MCS	A
67	G2.1.40 1	1.00	MCS	A
68	G2.1.44 2	1.00	MCS	A
69	G2.2.40 3	1.00	MCS	B
70	G2.2.42 2	1.00	MCS	C
71	G2.3.5 2	1.00	MCS	B
72	G2.3.7 2	1.00	MCS	C
73	G2.4.14 2	1.00	MCS	C
74	G2.4.18 2	1.00	MCS	A
75	G2.4.49 3	1.00	MCS	C
SECTION 1 (75 items)		**75.00**		