76. Unit 1 Initial Conditions:

- The operations team is cooling down the unit in preparation for refueling in accordance with 1-GOP-2.6, "UNIT COOLDOWN, LESS THAN 205 °F TO AMBIENT."
- The pressurizer (PRZR) level is at 70%.
- All pressurizer heaters are available.
- RCS Pressure is approximately 250 psig.
- RCS Temperature is approximately 180 °F.
- 'A' and 'B' steam generators narrow range levels are 50%, while 'C' steam generator is at 98%
- Auxiliary feedwater is available to all SGs.
- All RCS temperature indicators are available.
- All three steam generator PORVs are operable.
- All RCPs are stopped.

Current conditions:

- A large unisolable Component Cooling Water leak caused a complete and sustained loss of Component Cooling Water.
- The operations team entered 1-AP-27.00, "LOSS OF DECAY HEAT REMOVAL CAPABILITY."
- CETC temperatures are approaching saturation.

Based on the current conditions, which ONE of the following is the FIRST method of providing decay heat removal, in accordance with 1-AP-27.00?

A. Forced feed cooling.

B. Reflux boiling heat removal.

C. Natural Circulation.

D. Cooling the RCS with the SFP and RWST coolers.

77. Which ONE of the following correctly describes (1) the definition of the Exclusion Area Boundary for Surry Power Station, and (2) the basis for the Exclusion Area Boundary size?

A. (1) A 1650-foot radius circle centered at the Unit 1 reactor containment building

   (2) An individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not a total radiation dose in excess of 300 Rem to the thyroid from iodine exposure.

B. (1) A 1650-foot radius circle centered at the Unit 1 reactor containment building

   (2) An individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not a total radiation dose in excess of 50 Rem to the thyroid from iodine exposure.

C. (1) A 1650-foot radius circle centered at the Unit 2 reactor containment building

   (2) An individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not a total radiation dose in excess of 300 Rem to the thyroid from iodine exposure.

D. (1) A 1650-foot radius circle centered at the Unit 2 reactor containment building

   (2) An individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not a total radiation dose in excess of 50 Rem to the thyroid from iodine exposure.

78. Unit 1 Initial Conditions:

- 100% Power
- A tube leak in the 'B' Steam Generator (S/G) has been identified.
- Control room operators have transitioned to 1-AP-24.00, "MINOR SG TUBE LEAK."

Current conditions:

- Condenser air ejector radiation monitor, RI-SV-111, alarms but the automatic actions do NOT occur.
- Main Steam (MS) Line B radiation monitor, RI-MS-125, alarms.
- MS Line A and C radiation monitor readings are slightly higher than before.
- The Senior Reactor Operator directs a manual reactor trip and initiation of 1-E-0, "REACTOR TRIP OR SAFETY INJECTION."
- Safety Injection (SI) does NOT automatically actuate.
- At step 4 of 1-E-0, it is determined that SI is NOT REQUIRED.

Based on the current conditions, which ONE of the following is (1) the correct procedural flowpath, AND (2) the correct method to procedurally address the failure of RV-SI-111 automatic actions?

A. (1) Transition to 1-ES-0.1(REACTOR TRIP RESPONSE) and perform 1-AP-24.00 (MINOR STEAM GENERATOR TUBE LEAK) in parallel

   (2) Perform steps in 1-AP-24.00 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-ES-0.1.

B. (1) Transition to 1-ES-0.1(REACTOR TRIP RESPONSE) and perform 1-AP-24.01 (LARGE STEAM GENERATOR TUBE LEAK) in parallel

   (2) Perform steps in 1-AP-24.01 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-ES-0.1.

C. (1) Transition to 1-ES-0.1, (REACTOR TRIP RESPONSE) and then transition to 1-AP-24.01 (LARGE STEAM GENERATOR TUBE LEAK).

   (2) Perform steps in 1-AP-24.01 to correct the failure of RV-SI-111 automatic actions.

D. (1) Transition to 1-ES-0.1, (REACTOR TRIP RESPONSE) and then transition to 1-AP-24.01 (LARGE STEAM GENERATOR TUBE LEAK).

   (2) Perform steps in 1-AP-24.00 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-AP-24.01.

79. Unit 1 Initial Conditions:

- Reactor power = 100%
- Control rod D-6 rod bottom light lit
- 1G-H2, RPI ROD BOTTOM $\leq$ 20 STEPS lit
- Control Rod D6 has been confirmed at the bottom of the core
- 0-AP-1.00 ROD CONTROL SYSTEM MALFUNCTION is entered

Which ONE of the following correctly states (1) which procedure will be required to reduce reactor power and (2) the parameter that is required to be monitored to reduce and stabilize power?

A. (1) 1-GOP-2.1 (Unit Shutdown Power Decrease from Allowable Power to less that 30% Reactor Power)
(2) Loop ΔT

B. (1) 1-GOP-2.1 (Unit Shutdown Power Decrease from Allowable Power to less that 30% Reactor Power)
(2) the highest reading PRNI

C. (1) 1-AP-23.00 (Rapid Load Reduction)
(2) Loop ΔT

D. (1) 1-AP-23.00 (Rapid Load Reduction)
(2) the highest reading PRNI

80. Which ONE of the following correctly identifies two reasons for the Feedwater Line Isolation function in response to a Safety Injection signal, as specified in the bases of Technical Specification 3.7, "INSTRUMENTATION SYSTEMS?"

A. (1) Prevent excessive cooldown of the Reactor Coolant System; AND
(2) Reduces the consequences of a design basis steam generator tube rupture by preventing steam generator overfill.

B. (1) Prevent excessive moisture carry-over that could damage the main turbine blading; AND
(2) Reduces the consequences of a design-basis steam generator tube rupture by preventing steam generator overfill.

C. (1) Prevent excessive cooldown of the Reactor Coolant System; AND
(2) Reduces the consequences of a steam line break inside the containment by stopping the entry of main feedwater.

D. (1) Prevent excessive moisture carry-over that could damage the main turbine blading; AND
(2) Reduces the consequences of a steam line break inside the containment by stopping the entry of main feedwater.

81. Unit 1 Initial Conditions:

- A steam generator tube rupture caused an automatic reactor trip and SI from 100% power.
- Operations personnel are performing actions in 1-E-3, "STEAM GENERATOR TUBE RUPTURE."

Current conditions:

- A maximum-rate cooldown using steam dumps to the condenser has begun.
- SI has just been reset.
- The RO reports that condenser vacuum is 28 " Hg and slowly lowering.
- The TSC informs the operations team that once all actions of E-3 are complete, it is required to implement the post-SGTR procedure that allows the FASTEST means of depressurizing the RCS and ruptured S/G.

Based on the current conditions, which ONE of the following is (1) a required action specified by E-3, AND (2) the correct post-SGTR procedure to implement?

A. (1) Ensure the condenser air ejector is aligned to containment, and then OPEN 1-SV-TV-102A.
(2) GO TO 1-ES-3.2, "POST-SGTR COOLDOWN USING BLOWDOWN."

B. (1) Ensure the condenser air ejector is aligned to containment, and then OPEN 1-SV-TV-102A.
(2) GO TO 1-ES-3.3, "POST-SGTR COOLDOWN USING STEAM DUMP."

C. (1) IF a Hi-CLS signal is NOT actuated, THEN realign the condenser air ejector for normal operations.
(2) GO TO 1-ES-3.2, "POST-SGTR COOLDOWN USING BLOWDOWN."

D. (1) IF a Hi-CLS signal is NOT actuated, THEN realign the condenser air ejector for normal operations.
(2) GO TO 1-ES-3.3, "POST-SGTR COOLDOWN USING STEAM DUMP."

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

- A SBLOCA has occurred.
- Radiation levels in the Auxiliary Building are increasing.
- The crew has transitioned to ECA-1.2 "LOCA Outside Containment".
- The crew closed/verified closed 1-SI-MOV-1890A and -1890B.
- RCS pressure was at 1700 psig and slowly dropping.

Current plant conditions on Unit 1 are as follows:

- The crew has closed 1-SI-MOV-1890C.
- RCS pressure is at 1550 psig and slowly rising.

Which ONE of the following describes (1) the status of the LOCA and (2) the required procedure transition?

A. (1) LOCA has been isolated.
(2) Go to ECA-1.1, "Loss of Emergency Coolant Recirculation".

B. (1) LOCA still exists.
(2) Go to ECA-1.1, "Loss of Emergency Coolant Recirculation".

C. (1) LOCA has been isolated.
(2) Go to 1-E-1, "Loss of Reactor or Secondary Coolant".

D. (1) LOCA still exists.
(2) Go to 1-E-1, "Loss of Reactor or Secondary Coolant".

83. An electrical transient caused the failure of 1-RI-RM-159 (Containment Particulate Radiation Monitor) and 1-RI-RM-160 (Containment Gas Radiation Monitor).

Associated automatic actions have been verified.

Which ONE of the following states additional requirements of Annunciators 1-RM-Q7 (Containment Particulate Alert/Failure) and 1-RM-Q8 (Containment Gas Alert/Failure) for this failure?

A. A Technical Specification 3.0.1 LCO clock exists to return at least one of the radiation monitors to operable status.

B. Samples are required to be taken every 4 hours to ensure compliance with Technical Specification section 3.1.C (RCS Leakage).

C. As long as the manipulator crane, and incore area radiation monitors are operable, no other actions are required.

D. Within 7 days, establish an alternate method of detecting radionuclides in containment with remote indication in the main control room.

84. Unit 1 Initial Conditions:

- At time 0930, unexpected grid fluctuations caused an automatic turbine trip from 100% power.
- Chemistry personnel drew a post-trip RCS sample at time 1005.
- Control room operators have stabilized the unit at 547 °F and normal operating pressure.

Current conditions:

- At time 1045, a Chemistry supervisor reports that the post-trip RCS sample total specific activity reading is greater than the 100/(E bar) limit by 28%.

Based on the current conditions, which ONE of the following (1) is the correct time the LCO for Technical Specification (TS) 3.1.D, Maximum Reactor Coolant Activity, is NOT met; AND (2) the basis of the requirement to cool down the reactor to less than 500 °F, in accordance with TS 3.1.D?

A. (1) LCO not met at 1005;
   (2) In the unlikely event of an assumed 30 minute radioactive release during the design-basis S/G tube rupture, the iodine partitioning factor below this RCS temperature ensures exposure limits are not exceeded at the site boundary.

B. (1) LCO not met at 1045;
   (2) In the unlikely event of a design-basis S/G tube rupture, the saturation pressure corresponding to this RCS temperature is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

C. (1) LCO not met at 1045;
   (2) In the unlikely event of an assumed 30 minute radioactive release during the design-basis S/G tube rupture, the iodine partitioning factor below this RCS temperature ensures exposure limits are not exceeded at the site boundary.

D. (1) LCO not met at 1005;
   (2) In the unlikely event of a design-basis S/G tube rupture, the saturation pressure corresponding to this RCS temperature is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

85. Unit 1 initial conditions:

- Reactor power = 100%
- SGTR = 275 gpm on 1A SG
- Reactor is manually tripped
- 1C RCP trips

Current conditions:

- 1-E-3 (STEAM GENERATOR TUBE RUPTURE) is in progress
- It is determined that Pzr spray is not adequately reducing RCS pressure and the decision is made to use the Pressurizer PORV to reduce RCS pressure.

Based on the above conditions, which ONE of the following states: (1) the reason for minimizing the cycling of the Pressurizer PORV and (2) the procedure that 1-E-3 directs you to perform if, during the depressurization, the Pressurizer PORV and its associated block valve CAN NOT be CLOSED?

A. (1) To reduce the chance of Pressurizer PORV failure.
(2) 1-ECA 3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.

B. (1) To reduce the chance of Pressurizer PORV failure.
(2) 1-ECA 3.1 SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

C. (1) To prevent the Tube rupture from degrading.
(2) 1-ECA 3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.

D. (1) To prevent the Tube rupture from degrading .
(2) 1-ECA 3.1 SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

86. Initial plant conditions on Unit 2 are as follows:

- A power increase is in progress following reactor startup.
- Reactor power is at 8%.
- Pressurizer Spray valve 2-RC-PCV-2455A cannot be opened.
- All three RCPs are operating.
- Normal Charging has been tagged out at 1-CH-MOV-1289A and excess letdown is in-service due to required charging line piping repairs.

Current plant conditions on Unit 2 are as follows:

- RCP ‘C’ trips on ground overcurrent.

Based on the above conditions, which ONE of the following describes whether action statements of the following LCOs are required to be performed:

- Technical Specification Section 3.1.A.4 (Reactor Coolant Loops)
- Technical Specification Section 3.1.A.5 (Pressurizer)

Action statements within...

A. Technical Specification Section 3.1.A.4 is required.
Technical Specification Section 3.1.A.5 is NOT required.

B. Technical Specification Section 3.1.A.4 is NOT required.
Technical Specification Section 3.1.A.5 is required.

C. both Technical Specification Section 3.1.A.4 and 3.1.A.5 are required.

D. neither Technical Specification Section 3.1.A.4 nor 3.1.A.5 are required.

87. Unit 1 initial conditions:

Time = 0800
Plant was on RHR following shutdown for refueling
Containment Closure has been established
SGs are not available
RCS temperature = 190°F stable
RHR flow = 2200 gpm
RCS level = 12.5 feet and decreasing
RVLIS Full Range = 47% and decreasing

Current plant conditions:

Time = 0825
1-AP-27.00 (LOSS OF DECAY HEAT REMOVAL CAPABILITY) has been initiated
RHR pumps have been secured due to vortexing
RCS temperature = 205°F increasing

Based on the above conditions: (1) Classify the event using the Emergency Plan and (2) once RHR level and flow has been restored, state the MAXIMUM cooldown rate allowed per 1-AP-27.00?

(Reference Provided)

A. (1) Alert
(2) 25°F/Hr

B. (1) Alert
(2) 50°F/Hr

C. (1) Site Area Emergency
(2) 50°F/Hr

D. (1) Site Area Emergency
(2) 25°F/Hr

88. Unit 1 initial conditions:

Time = 1000
Reactor power = 100%
1-RC-PORV-1455C (Pressurizer Pressure PORV) indicates open
Both Pzr Spray valves indicate open
RCS Pressure = 2200 psig decreasing
1-AP-31.00 (Increasing or Decreasing RCS Pressure) initiated

Current conditions:

Time = 1001
Reactor Power = 97%
RCS Pressure = 2100 psig increasing
Spray valve in MANUAL and closed
1-RC-PORV-1455C in MANUAL and closed

Based on the above conditions, which ONE of the following correctly states: (1) the component that failed high and (2) the status of 1-RC-PORV-1455C operability per Technical Specifications?

A. (1) 1-RC-PT-1444 (Pressurizer Pressure Control)
(2) PORV is considered OPERABLE

B. (1) 1-RC-PT-1444 (Pressurizer Pressure Control)
(2) PORV is INOPERABLE

C. (1) 1-RC-PT-1445 (Pressurizer Pressure Control)
(2) PORV is considered OPERABLE

D. (1) 1-RC-PT-1445 (Pressurizer Pressure Control)
(2) PORV is INOPERABLE

89. Given the following Plant Conditions:
- RCS Pressure = 1500 psig and slowly decreasing
- RCS Tave = 427 °F and slowly decreasing
- Pressurizer Level = 0%
- Steam Generator Parameters are as follows:

	Pressure	Level (Wide Range)
'A' SG	120 psig - decreasing	23% - decreasing
'B' SG	650 psig - slowly decreasing	56%– slowly increasing
'C' SG	650 psig - slowly decreasing	56%– slowly increasing

- The Air Ejector Radiation Monitor is in High Alarm
- Containment Pressure is 16 psia and slowly increasing
- Containment Particulate and Gas Radiation Monitors (1-RI-RM-159/160) are in High Alarm

Based on the conditions above, which ONE of the following describes the expected procedural sequence for this event?

A. 1-E-3 (Steam Generator Tube Rupture)
1-E-2 (Faulted Steam Generator Isolation)
1-ECA-3.1 (SGTR with Loss of Reactor Coolant - Subcooled Recovery)

B. 1-E-2 (Faulted Steam Generator Isolation)
1-E-3 (Steam Generator Tube Rupture)
1-ECA-3.1 (SGTR with Loss of Reactor Coolant - Subcooled Recovery)

C. 1-E-3 (Steam Generator Tube Rupture)
1-E-2 (Faulted Steam Generator Isolation)
1-E-1 (Loss of Reactor or Secondary Coolant)

D. 1-E-2 (Faulted Steam Generator Isolation)
1-E-3 (Steam Generator Tube Rupture)
1-E-1 (Loss of Reactor or Secondary Coolant)

90. With the unit initially at 100% the following conditions are encountered:

- Annunciator 1A-G1 (TRAVELING SCREENS HI DIFF LVL) was received
- The outside operator reports high delta-P on "A" and "C" high level screens
- Vacuum is 27.6" and degrading
- PCS indications show CW outlet temperatures are 95°F and increasing
- Main generator output is 843 MWe and decreasing

The team has initiated 1-AP-14.00 (Loss of Main Condenser Vacuum).

Which ONE of the following states (1) the cause of the vacuum trend and (2) the procedure directed by 1-AP-14.00 to reduce turbine load?

A. (1) Loss of heat sink to the condenser.

(2) 1-OP-TM-005, Unit Ramping Operations

B. (1) Air in-leakage into the condenser.

(2) 1-AP-23.00, Rapid Load Reduction.

C. (1) Loss of heat sink to the condenser.

(2) 1-AP-23.00, Rapid Load Reduction

D. (1) Air in-leakage into the condenser.

(2) 1-OP-TM-005, Unit Ramping Operations

91. Unit 1 initial conditions:

A reactor trip from 100% power occurs due to a Loss of Offsite Power
Both EDGs start but both output breakers trip on ground over-current
TD AFW pump fails to start
All three Steam Generators indicate 54% Wide Range Level
The STA reports that a RED path on Heat Sink exists and that all other status trees are green.

Based on the above conditions, which ONE of the following correctly states (1) the procedure that is required to be entered and (2) the classification of this event?

A. (1) 1-ECA-0.0 (Loss of All AC Power)
(2) Alert

B. (1) 1-FR-H.1 (Response to Loss of Secondary Heat Sink)
(2) Alert

C. (1) 1-ECA-0.0 (Loss of All AC Power)
(2) Site Area Emergency

D. (1) 1-FR-H.1 (Response to Loss of Secondary Heat Sink)
(2) Site Area Emergency

92. Initial plant conditions:

Unit 2 shutdown with fuel offloaded
Unit 1 = 100% power

Current plant conditions:

Annunciator 1D-G5 (SW OR CC PPS DISCH TO CHRG PPS LO PRESS) is in alarm
1-AP-12.00 (SERVICE WATER SYSTEM ABNORMAL CONDITIONS) has been initiated
Unit 1 operating CHG pump bearing temperatures:
1420 = 170°F
1440 = 180°F
1450 = 195°F

Based on the above conditions: (1) which ONE of the following states the EARLIEST time at which 1-AP-12.00 directs shifting the operating charging pump due to high bearing temperature, and (2) if all charging pumps are incapable of providing flow, correctly states the reason for utilizing a Unit 2 Charging Pump in accordance with Technical Specifications?

A. (1) 1440
(2) To bring the operating unit to cold shutdown

B. (1) 1440
(2) To bring the operating unit to hot shutdown ONLY

C. (1) 1450
(2) To bring the operating unit to cold shutdown

D. (1) 1450
(2) To bring the operating unit to hot shutdown ONLY

93. Given the following plant conditions:

- Unit 1 is at 100%
- A loss of Containment Instrument Air has occurred
- 1B-F6, CTMT INST AIR HDR LO PRESSURE, annunciates
- 1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, annunciates
- Containment Instrument Air was crosstied with Instrument Air
- Containment Instrument Air Pressure = 85 psig and increasing
- All Pressurizer PORV air bottles are properly aligned with air pressures of 1050 psig

Which ONE of the following correctly states (1) the status of LCO 3.1.A.6 (PORV Operability) and (2) the Technical Specification required operator actions, if any?

A. (1) The LCO is met.
(2) No further action associated with the Pressurizer PORVs is required.

B. (1) The LCO is met.
(2) Verify Pressurizer PORV operability by closing Pressurizer PORV Block Valves, manually cycle the Pressurizer PORVs, and then re-open the Pressurizer PORV Block Valves.

C. (1) The LCO is NOT met.
(2) Restore the Pressurizer PORV backup air supply within 14 days OR be in HSD within the next 6 hours.

D. (1) The LCO is NOT met.
(2) Close and remove power from both Pressurizer PORV block valves within one hour AND be in HSD within the next 6 hours

94. Initial plant conditions on Unit 2 are as follows:

- Reactor power is 100%.
- A 20 gpd leak exists on steam generator 'B'.

Current plant conditions on Unit 2 are as follows:

Charging flow has slowly increased.

Auto-makeup to VCT has started.

VCT level is 29% and slowly rising.

Pressurizer level is stable at 54%.

Pressurizer pressure is stable at 2225 psig.

Crew has entered 1-AP-16.00 (Excessive RCS Leakage).

Radiation levels on MSL "B" show a slow increasing trend.

The leak rate has been calculated at 12 gpm.

Transition to 2-AP-24.00, "Minor SG Tube Leak" is __(1)__ AND the correct classification (if any) for the event is a (an) __(2)__?

(Reference provided)

A. (1) required
(2) None Required

B. (1) NOT required
(2) None Required

C. (1) required
(2) NOUE

D. (1) NOT required
(2) NOUE

95. Plant conditions:

RCS cooldown in progress
RCS temperature = 350°F decreasing
RCS pressure = 300 psig

Based on the above conditions, with regards to the Over Pressure Mitigation System (OPMS), prior to decreasing below 350°F ___(1)___, and the TS basis for that configuration is ____(2)___?

(Consider No TS modifications, LCOs...)

A. (1) Pzr level is limited to 33% for the initial 72 hours
   (2) To allow the operator 10 minutes to take action from inadvertent initiation of full (3 pump) charging flow.

B. (1) Two PORVs are required to remain operable
   (2) Based on the Pressurizer PORVs ability to relieve RCS pressure from the start of a RCP with SG temp > RCS temp.

C. (1) Accumulators must be depressurized to less than the Pressurizer PORV setpoint
   (2) To prevent exceeding the PORV capability if an inadvertent OPMS initiation occurs.

D. (1) Verify a maximum of one charging pump is capable of injecting into the RCS.
   (2) To ensure any mass addition can be relieved by one PORV.

96. In accordance with Technical Specification section 3.16 (Emergency Power System), which ONE of the following (1) states the MINIMUM level of fuel oil in the underground fuel oil storage tanks, and (2) the basis for this requirement?

A. (1) 20,000 gallons
(2) To support the full load operation of one EDG for 7 days.

B. (1) 20,000 gallons
(2) To support the full load operation of three EGDs for 36 hours.

C. (1) 35,000 gallons
(2) To support the full load operation of one EDG for 7 days.

D. (1) 35,000 gallons
(2) To support the full load operation of three EGDs for 36 hours.

97. Unit 1 initial conditions:
    Date = 6/24
    Time = 0800
    Reactor power = 100%
    Waste gas storage tank activity level is reported which exceeds TS 3.11, Radioactive Gas Storage, limits

Current conditions:
    Date = 6/26
    Time = 0800
    Reactor power = 100%
    Waste gas storage tank activity level still exceeds Tech Spec 3.11 limits

Based on the above conditions, which ONE of the following correctly states: (1) if Tech Spec 3.0.1 is applicable and (2) the whole body dose that the tank radioactivity limit is designed to prevent exceeding at the exclusion area boundary if the tank were released IAW Tech Spec Basis?

A. (1) Yes
   (2) 50 mrem

B. (1) Yes
   (2) 0.5 rem

C. (1) No
   (2) 50 mrem

D. (1) No
   (2) 0.5 rem

98. Unit 1 initial plant conditions:
Reactor power = 50%
Plant shutdown in progress due to RCS activity greater than TS limits
AFW Pump 1-FW-P-3B OOS

Current plant conditions:
'A' SG tube rupture occurs
'A' SG pressure = 1000 psig
Reactor has been tripped
1-E-3 STEAM GENERATOR TUBE RUPTURE in progress
The TSC has been established
An operator is dispatched to close 1-MS-87 (steam from the A SG to the TD AFW pump) in order to save valuable equipment

Based on the above conditions, which ONE of the following: (1) states the MAXIMUM allowable dose (TEDE) that can be authorized for the operator to receive while isolating steam to the TD AFW pump and
(2) if the valve can not be closed, what procedural actions shall be taken IAW 1-E-3 to mitigate the failure?

A. (1) 10 Rem
(2) Remain in 1-E-3 and trip the TD AFW pump overspeed trip valve.

B. (1) 10 Rem
(2) GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

C. (1) 25 Rem
(2) Remain in 1-E-3 and trip the TD AFW pump overspeed trip valve.

D. (1) 25 Rem
(2) GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

99. Unit 1 Initial Conditions:

- Holding at 50% power for fuel conditioning following a refueling outage.

Current conditions:

- Technicians performing a routine surveillance test on the AMSAC logic system inadvertently cause a half-train Train "A" AMSAC signal to be generated.
- Annunciator F-B-3, AMSAC INITIATED, is lit
- All AMSAC functions occurred.

Based on the current conditions, which ONE of the following correctly describes (1) whether the half-train AMSAC signal should be considered a VALID or INVALID actuation, as defined by VPAP-2802, "Notifications and Reports," AND (2) the most restrictive time requirement to report this event to the NRC, as specified by VPAP-2802?

(Reference provided)

A. (1) VALID actuation
   (2) 4 hour notification

B. (1) INVALID actuation
   (2) 8 hour notification

C. (1) VALID actuation
   (2) 8 hour notification

D. (1) INVALID actuation
   (2) 4 hour notification

100. Unit 1 plant conditions:

Time = 0200
RCS cooldown in progress
RCS temperature = 250 ^{O}F
RCS pressure = 320 psig
1A charging pump is the only running charging pump

Current plant conditions:

Time = 0210
RCS pressure = 280 psig decreasing
The maximum charging flow achieved with the 1A charging pump is 125 gpm

Based on the above conditions, which ONE of the following: (1) states the correct procedure to be entered and (2) what actions are directed by that procedure?

A. (1) 1-AP-16.00 EXCESSIVE RCS LEAKAGE
(2) Align charging pump suction to the RWST

B. (1) 1-AP-16.00 EXCESSIVE RCS LEAKAGE
(2) Align and start 1B and 1C charging pumps

C. (1) 1-AP-16.01 SHUTDOWN LOCA
(2) Align charging pump suction to the RWST

D. (1) 1-AP-16.01 SHUTDOWN LOCA
(2) Align and start 1B and 1C charging pumps

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]**

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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 SRO Portion of Exam Test Form: 0

#	ID	Points	Type	Answers 0
1 76	0026G2.1.7 2	1.00	MCS	C
2 77	0036AA2.03 2	1.00	MCS	A
3 78	0039A2.03 2	1.00	MCS	C
4 79	003AG2.4.31 9	1.00	MCS	C
5 80	0054G2.2.25 1	1.00	MCS	C
6 81	0055G2.4.6 1	1.00	MCS	B
7 82	006A2.12 12	1.00	MCS	C
8 83	0073A2.02 3	1.00	MCS	B
9 84	0076AA2.02 1	1.00	MCS	B
10 85	010G2.4.20 13	1.00	MCS	B
11 86	015/17AG2.2.22 2	1.00	MCS	~~C~~ B
12 87	025AA2.05 4	1.00	MCS	C
13 88	027AA2.15 5	1.00	MCS	B
14 89	035A2.01 19	1.00	MCS	B
15 90	051G2.4.11 12	1.00	MCS	C
16 91	059G2.4.14 15	1.00	MCS	C
17 92	062AA2.06 2	1.00	MCS	A
18 93	079G2.2.22 2	1.00	MCS	~~A~~ C
19 94	G2.1.20 14	1.00	MCS	C
20 95	G2.2.14 22	1.00	MCS	D
21 96	G2.2.22 4	1.00	MCS	C
22 97	G2.3.12 1	1.00	MCS	D
23 98	G2.3.4 22	1.00	MCS	A
24 99	G2.4.30 3	1.00	MCS	D
25 100	G2.4.9 24	1.00	MCS	C
SECTION 1 (25 items)		**25.00**		