Master RO Exam

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER RO EXAMINATION	Region: I / II / III / IV
Date:	Facility/Unit: **Perry**
License Level: RO / SRO	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. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	100.00	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

QUESTION 1

A tagging error resulted in the de-energization of 120 VAC Instrument Panel EB-1-A1. The Control Room Operator reports that the SLC A OUT OF SERVICE alarm has annunciated.

If an ATWS occurs that requires boron injection, how would the Standby Liquid Control 'A' subsystem respond when the Control Room Operator places the SLC PUMP A control switch to ON?

A. Squib Valve 'A' will <u>not</u> fire and SLC Pump 'A' will <u>not</u> start.

B. Squib Valve 'A' will <u>not</u> fire but SLC Pump 'A' will start.

C. Squib Valve 'A' will fire but SLC Pump 'A' will <u>not</u> start.

D. Squib Valve 'A' will fire and SLC Pump 'A' will start.

QUESTION 2

The reactor is critical and plant heatup/pressurization is in progress when the detector voltage for IRM Channel 'D' decreases to 50% of the normal detector voltage. All other IRMs are OPERABLE.

Which one of the following describes the response of IRM Channel 'D'?

A. Only a RPS half scram signal is generated.

B. Only a control rod block signal is generated.

C. A control rod block and RPS half scram signal are generated.

D. A control rod block and RPS half scram signal are not generated.

QUESTION 3

The plant is operating at 100% reactor power when a loss of 4160V Bus EH11 occurs. Subsequently, a reactor scram occurs. The Control Room Operator notes that the Rod Control & Information System (RC&IS) indication is blinking ON and OFF on the full core display.

Which one of the following action(s) can the Control Room Operator perform to verify ALL RODS IN using the RC&IS display?

A. Depress the DATA SOURCE pushbutton to select the operable RC&IS channel.

B. Depress the RAW DATA and SCRAM VALVES pushbuttons to determine control rod positions.

C. Depress the ACKN ACCUM FAULT pushbutton to allow the control rods to settle into the full-in position.

D. Release the DATA MODE pushbutton and select the operable RC&IS channel with the DATA SOURCE pushbutton.

QUESTION 4

Which one of the following describes the interlock associated with the RCIC pump suction valves?

A. The suction source will automatically swap from the Suppression Pool to the CST upon a low level in the Suppression Pool.

B. The suction source will automatically swap from the Suppression Pool to the CST upon a high level in the Suppression Pool.

C. The suction source will automatically swap from the CST to the Suppression Pool upon a low level in the Suppression Pool.

D. The suction source will automatically swap from the CST to the Suppression Pool upon a low level in the CST.

QUESTION 5

The plant has under gone a transient that resulted in a Recirculation Flow Control valve runback.

Which one of the, following describes the allowable operation of the Recirculation Flow Control Valves, prior to resetting the runback?

The Recirculation Flow Control valves can ____________________.

A. be closed using LOOP manual operation, however, they can only be opened to the 12% valve position.

B. be closed using LOOP manual operation, however, they can only be opened to the position that they ran back to.

C. not be closed any further because they are at the full closed stop and cannot be re-opened due to a hydraulic lock on the valves.

D. not be closed any further because they are at the full closed stop, however, they can be opened to the 22% valve position.

QUESTION 6

The plant has scrammed due to a loss of off-site power. HPCS and RCIC did not start. When RPV water level reached Level 1, the Control Room Operator reports that the ADS 105-second time delay logic timer is running.

The Unit Supervisor directs the Control Room Operator to inhibit ADS per PEI-B13, RPV Control (Non-ATWS).

Later the Control Room Operator is directed to arm and depress both Manual Initiation pushbuttons for the ADS 'A' subsystem.

Which one of the following is the response of the ADS 'A' subsystem in this situation?

ADS 'A' subsystem will initiate __________________.

A. immediately, if any Division 1 low pressure ECCS subsystem pressure permissive is satisfied.

B. in 105 seconds, if any Division 1 low pressure ECCS subsystem pressure permissive is satisfied.

C. immediately, regardless of the Division 1 low pressure ECCS subsystem status.

D. in 105 seconds, regardless of the Division 1 low pressure ECCS subsystem status.

QUESTION 7

Which one of the following steam loads is supplied steam from the pressure/flow equalizer manifold upstream of the Main Turbine Stop and Control Valves?

A. Extraction Steam Hot Water Heat Exchanger

B. Reactor Feed Pump Turbines

C. Reactor Core Isolation Cooling (RCIC) Turbine

D. Main Turbine Sealing Steam

QUESTION 8

Which one of the following describes the operation of the Static Transfer Switch associated with the ATWS UPS Inverter?

A. The Static Transfer Switch will shift to the alternate (AC) source on an over-current condition and will automatically shift back to the Inverter when the inverter fault has cleared.

B. The Static Transfer Switch will shift to the alternate (AC) source on an over-voltage condition and will automatically shift back to the Inverter when the inverter fault has cleared.

C. The Static Transfer Switch will shift to the alternate (AC) source on an over-current condition and will <u>not</u> automatically shift back to the Inverter when the inverter fault has cleared.

D. The Static Transfer Switch will shift to the alternate (AC) source on an under-voltage condition and will <u>not</u> automatically shift back to the Inverter when the inverter fault has cleared.

QUESTION 9

A Precaution and Limitation in SOI-N64/62, Off-Gas/Condenser Air Removal System, states, "In the event of a loss of dilution steam from SJAE's, initiate an air purge through the operating preheater/recombiner side for at least one hour".

Which one of the following statements is the bases for this Precaution and Limitation?

A. To prevent damage to the Holdup Line loop seal level indicators due to a rapid change in pressure.

B. To prevent a hydrogen explosion hazard internal to the system.

C. To maintain the recombiner inlet temperature below 300 degrees F.

D. To prevent reverse flow through the recombiners that could result in the introduction of catalyst pellets, fines, or particles into the upstream portion of the system.

QUESTION 10

Control Room HVAC Train 'A' (M25/26) is operating in the Emergency Recirculation mode. A Non-Licensed Operator reports smoke coming from the 'A' Emergency Recirculation Charcoal Filter Plenum and the plenum is glowing red.

Which one of the following describes the method to combat a fire in the 'A' Emergency Recirculation Charcoal Filter Plenum?

A. The Fire Protection System will automatically initiate the charcoal filter deluge system and fill the charcoal filter plenum with water.

B. The Fire Protection System will automatically initiate the charcoal filter preaction system and fill the charcoal filter plenum with water.

C. The Control Room Operator arms and depresses CONT RM EMG RCIRC A CHAR FLTR DELUGE pushbutton (P54-F3180) to manually initiate the charcoal filter deluge system and fill the charcoal filter plenum with water.

D. The Non-Licensed Operator manually valves in the deluge system locally, and then opens the deluge system manual initiation valve to fill the charcoal filter plenum with water.

QUESTION 11

The plant was operating at 45% reactor power when a transient on the First Energy grid caused a fast closure of the Turbine Control Valves for the Perry Main Turbine. The Feedwater Level Control System maintained reactor water level within 10 inches of normal level. Reactor pressure increased slightly but was maintained by the Turbine Bypass Valves.

Which one of the following describes the current plant conditions for this event?

A. The reactor scrammed and the Reactor Recirculation Pumps tripped off.

B. The reactor scrammed and the Reactor Recirculation Pumps downshifted to slow speed.

C. The reactor remained at power with a reduced power due to a Reactor Recirculation flow control valve runback.

D. The reactor remained at power with a reduced power due to a Reactor Recirculation Pump downshift to slow speed.

QUESTION 12

A HIGH alarm is received on the Gaseous channel of the Drywell Atmosphere Airborne Radiation Monitor, D17-K670.

Which one of the following conditions will occur?

A. The Containment Evacuation Alarm will activate.

B. The Plant Emergency Alarm will activate.

C. The COMB GAS DW PURGE INBD (M51-F090) and COMB GAS DW PURGE OTBD (M51-F110) valves will isolate (if they are open).

D. The DW PURGE SUPP TRN A FIRST ISOL DAMPER (M14-F055A) and DW PURGE SUPP TRN A SECOND ISOL DAMPER (M14-F055B) will isolate (if they are open).

QUESTION 13

Hydrogen Igniters are designed to ____________________.

A. burn hydrogen while it is at low concentrations and therefore limit the threat to containment integrity caused by low internal pressure.

B. burn hydrogen while it is at low concentrations and therefore limit the threat to containment integrity caused by high internal pressure.

C. cause a hydrogen deflagration in order to minimize the threat to containment integrity caused by high temperature.

D. cause a hydrogen deflagration in order to minimize the threat to containment integrity due to vacuum breaker failure.

QUESTION 14

Which one of the following inclusive combinations of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning defines the MODE known as 'Cold Shutdown'?

A. The mode switch is in REFUEL, average reactor coolant temperature is 198 °F, and all reactor vessel head closure bolts are fully tensioned.

B. The mode switch is in SHUTDOWN, average reactor coolant temperature is 198 °F, and all reactor vessel head closure bolts are fully tensioned.

C. The mode switch is in SHUTDOWN, average reactor coolant temperature is 198 °F, and some reactor vessel head closure bolts are <u>not</u> fully tensioned.

D. The mode switch is in SHUTDOWN, average reactor coolant temperature is 208 °F, and all reactor vessel head closure bolts are fully tensioned.

QUESTION 15

You are the Licensed Operator 'At the Controls'. You have requested to be temporarily relieved by another licensed operator for a short break. Since assuming the shift, the following items have started, are in progress, or have been completed:

1. The Operations Manager took a ½ PA Day to play golf.
2. Surveillance testing of APRM 'D' is in progress by I&C.
3. Reactor power was decreased 5% due to System Dispatcher request.
4. The ROD DRIFT annunciator is alarming spuriously and I&C has been notified.
5. Preparations for Division 1 Diesel Generator monthly surveillance testing may start next shift.

Which of the above item(s) are REQUIRED to be discussed with the other licensed operator before you can be temporarily relieved?

NOTE: A partial list will be incorrect.

A. 2 and 3

B. 2, 3, and 4

C. 2, 3, 4, and 5

D. 1, 2, 3, 4, and 5

QUESTION 16

A surveillance test that will cause the system to become inoperable is scheduled to be performed. Unless otherwise noted in the surveillance test, the system is considered to be inoperable when the ________________.

A. the Supervising Operator signs the "Authorization to Start The Test" block on the Data Package Cover Sheet.

B. the Unit Supervisor signs the "Authorization to Start the Prerequisites" block on the Data Package Cover Sheet.

C. the Lead Test Performer annotates the start date/time in the Test Tracking Log.

D. the first surveillance step is actually performed which will make the system inoperable, such as installing a jumper or turning a switch.

QUESTION 17

In accordance with IOI-9, Refueling, which one of the following statements best describes the reason that personnel performing Core Alterations shall maintain direct communications with the Control Room?

A. Core Alterations are considered to be a change in core reactivity that requires the knowledge and consent of the Operator at the Controls.

B. Core Alterations are considered to be a special infrequently performed tests or evolutions (IPTEs) that require constant communication with the Control Room.

C. To allow the Operator at the Controls to monitor for inadvertent criticality and inform the Refuel Floor of such event.

D. To allow the on-shift Shift Technical Advisor (STA) to perform a shutdown margin check required during Core Alterations.

QUESTION 18

Which one of the following combinations of reactor power and reactor pressure would indicate that a Safety Limit violation has occurred?

	Reactor Power	**Reactor Pressure**
A.	35%	810 psig
B.	30%	775 psig
C.	20%	770 psig
D.	10%	750 psig

QUESTION 19

Plant startup is in progress at 5% reactor power.

Which one of the following describes the allowable mode of operation and bases for operation of the Containment Vessel and Drywell Purge System (M14)?

A. Refuel mode in order to reduce airborne activity in Containment during RCIC operation.

B. Single Train Drywell Ventilation mode in order to reduce Drywell average air temperature due to steam leaks.

C. Intermittent mode during backwash of RWCU Filter Demineralizer 'A' in order to minimize off-site radiation doses.

D. Intermittent mode during the time that RWCU Filter Demineralizer 'A' is in Hold due to leaks in the RWCU F/D System.

QUESTION 20

Jim Noname, age 37, is a radiation worker at Perry. He has no exposure from any other nuclear facility. Jim's current year-to-date (YTD) radiation exposure (TEDE) is 500 millirem and his lifetime radiation exposure is 15 Rem.

How much more radiation exposure can Jim receive this year before an Increased Dose Control Level Authorization is required?

A. 500 millirem

B. 1500 millirem

C. 3500 millirem

D. 4500 millirem

QUESTION 21

The plant is operating at 100% reactor power when a piping elbow on the common discharge of the Turbine Building Closed Cooling (TBCC) pumps ruptures. The TBCC SURGE TANK LEVEL LOW alarm is locked in. TBCC Pump discharge pressure indications on panel H13-P870 are extremely low.

Which one of the following describes the immediate operator action(s) to be performed?

A. Perform a fast reactor shutdown and transfer the Reactor Recirculation Pumps to slow speed.

B. Perform a fast reactor shutdown and trip the Reactor Recirculation Pumps to OFF.

C. Restore TBCC surge tank level by manually opening Two Bed M/U Wtr Cont Vlv Bypass Vlv, P44-F503 and reduce the TBCC heat load as soon as possible.

D. Open TBCC HX SW TCV BYP, P41-F390 if failure of the TBCC heat exchanger outlet temperature control valve, P41-F003 is suspected.

QUESTION 22

The plant is operating at 100% reactor power when a complete loss of Instrument Air occurs.

Which one of the following describes the expected response for the following Feedwater System air-operated valves?

A.

MFP Full Flow Control Valve (N27-F010)	Fails as-is
MFP Low Flow Control Valve (N27-F110)	Fails closed
MFP and RFP Recirculation Flow Control Valves	Fail open
RFP Seal Injection Valves	Fail open

B.

MFP Full Flow Control Valve (N27-F010)	Fails as-is
MFP Low Flow Control Valve (N27-F110)	Fails as-is
MFP and RFP Recirculation Flow Control Valves	Fail close
RFP Seal Injection Valves	Fail open

C.

MFP Full Flow Control Valve (N27-F010)	Fails as-is
MFP Low Flow Control Valve (N27-F110)	Fails as-is
MFP and RFP Recirculation Flow Control Valves	Fail close
RFP Seal Injection Valves	Fail as-is

D.

MFP Full Flow Control Valve (N27-F010)	Fails as-is
MFP Low Flow Control Valve (N27-F110)	Fails as-is
MFP and RFP Recirculation Flow Control Valves	Fail open
RFP Seal Injection Valves	Fail open

QUESTION 23

During a reactor startup, a single notch control rod withdrawal reduced the reactor period to 45 seconds.

Which one of the following describes the operator action(s) to be taken?

A. Stop control rod withdrawal. Insert control rod(s) in the approved sequence, if necessary, to increase reactor period.

B. Stop control rod withdrawal. Insert cram rod(s), if necessary, to increase reactor period.

C. Control rod withdrawal can continue unless a reactor period of less than 30 seconds is observed.

D. Control rod withdrawal can continue if the reactor period is deemed to be spurious in nature

QUESTION 24

The plant is conducting a refueling outage when the following alarms are received in the Control Room:

- CNTMT VENT EXH RAD HI
- CNTMT VENT EXH RAD A/D HI HI / INOP
- CNTMT VENT EXH RAD B/C HI HI / INOP

In addition to directing the evacuation of the Containment, which one of the following identifies an additional required Immediate Action to be performed in accordance with ONI-D17, High Radiation Levels Within the Plant?

A. Notify Health Physics and Chemistry.

B. Notify state and local authorities if an Emergency Action Level (EAL) has been exceeded.

C. Direct the insertion of any withdrawn control rod.

D. Direct the continued monitoring of the affected area by Control Room personnel to determine the extent of the radiation problem.

QUESTION 25

Which one of the following PEI Alternate Injection methods would require the Turbine Building to be accessible in order for plant operators to perform local actions?

A. CRD Alternate Injection

B. Condensate Transfer Alternate Injection

C. Alternate Injection via the FPCC Header using a Hotwell Pump

D. Alternate Injection via the FPCC Header using the SPCU Pump

QUESTION 26

During RCIC System operation, a RCIC turbine trip occurred.

The Supervising Operator attempted to reset the RCIC turbine by closing the RCIC TURBINE TRIP THRT V LATCH, 1E51-F510, and then taking the control switch to the OPEN position.

The following RCIC System valve position indications exist:

RCIC TRIP THROTTLE VALVE	GREEN light ON	RED light OFF
RCIC TURBINE TRIP THRT V LATCH	GREEN light OFF	RED light ON
RCIC TURBINE GOVERNOR VALVE	GREEN light OFF	RED light ON

Which one of the following currently describes the operation of the RCIC System?

A. The RCIC System should be operating at a speed based on governor demand.

B. The RCIC System is reset awaiting the re-opening of RCIC STEAM SHUTOFF VALVE, 1E51-F045.

C. The RCIC System is still tripped awaiting the reset of the trip device linkage locally at the RCIC turbine,

D. The RCIC System is still tripped awaiting the reset of the RCIC Division 1 and/or Division 2 isolation signals.

QUESTION 27

How long is the Division 1 battery designed to supply the required Loss of Coolant Accident (LOCA) loads without allowing the final discharge voltage to decrease below the minimum design cell voltage?

The Division 1 battery will supply the LOCA loads (with coincident AC power loss) for a MINIMUM of ___________.

NOTE: Selection of a duration less than or greater than the minimum design will be an INCORRECT response.

A. 2 hours

B. 3 hours

C. 4 hours

D. 12 hours

QUESTION 28

The following plant conditions exist:

- An ATWS is in progress
- HPCS injection prevention has been performed per PEI-SPI-5.1
- RPV level is being maintained at + 75 inches

A loss of Bus EH13 occurs.

Which one of the following describes the response of the HPCS Pump breaker?

A. The HPCS Pump breaker remains closed at all times.

B. The HPCS Pump breaker remains open at all times.

C. The HPCS Pump breaker initially opens and, upon re-energization of Bus EH13, re-closes after a 10 second time delay.

D. The HPCS Pump breaker initially opens and, upon re-energization of Bus EH13, re-closes immediately.

QUESTION 29

IMMEDIATELY following a reactor scram from full power, what would be the expected indication observed on the Intermediate Range Monitors?

A. Range 3 due to delayed neutrons dominating from longer-lived delayed neutron precursors.

B. Range 3 due to delayed neutrons dominating from shorter-lived delayed neutron precursors.

C. Range 5 due to delayed neutrons dominating from shorter-lived delayed neutron precursors.

D. Range 5 due to delayed neutrons dominating from longer-lived delayed neutron precursors.

QUESTION 30

The plant is operating at 100% reactor power when all inboard Main Steam Line Isolation Valves inadvertently isolate. The MSIV closure signal to the Reactor Protection System (RPS) failed to scram the reactor

Which one of the following describes the response of the reactor?

Assume **NO** operator action is taken.

A. Reactor power will increase and stabilize at a higher power.
RPV water level will decrease and return to normal level.

B. Reactor power will increase and cause a reactor scram on power.
RPV level will decrease and then stabilize at a higher level.

C. Reactor power will decrease and stabilize at a lower power.
RPV water level will increase and then return to normal level.

D. Reactor power will increase and cause a reactor scram on power.
RPV water level will increase and then return to normal level.

QUESTION 31

The plant was operating at 50% power with both RFPTs on the Master Level Controller when a Feedwater rupture in the Turbine Building caused reactor water level to decrease.

Reactor water level decreased to +80 inches before HPCS and RCIC were able to restore reactor water level to normal.

Which one of the following correctly describes the status of the Recirculation System?

A. Recirculation Pumps are tripped with their flow control valves in their pre-transient positions.

B. Recirculation Pumps are in slow speed with their flow control valves in their pre-transient positions.

C. Recirculation Pumps are tripped with their flow control valves locked up (motion inhibited).

D. Recirculation Pumps are in slow speed with their flow control valves locked up (motion inhibited).

QUESTION 32

The plant is at 100% power. Since the beginning of shift, Control Room Operators have observed the following Drywell parameter trends:

• Drywell Pressure:	Increasing
• Drywell Average Temperature	Increasing
• Drywell Air Cooler Drain Flow Rate	Increasing
• Drywell Floor Drain Sump Fill Rate	Increasing

Which one of the following could be the cause of these indications?

A. There is an accumulator air leak on an inboard MSIV.

B. There is a cooling coil leak on the lower drywell cooler air handling unit.

C. There is an instrument line leak on a water level condensing chamber.

D. There is an outer seal leak on a Reactor Recirculation Pump.

QUESTION 33

An ATWS is in progress. The following plant conditions exist:

- Reactor power is 25%.
- Reactor pressure is at rated pressure.
- Reactor water level is at +180 inches and stable.

Which one of the following describes the effect of reducing reactor pressure?

A. Reactor power will decrease due to the voiding of the core and remain lower than the original power.

B. Reactor power will initially decrease due to the voiding of the core and then increase due to the lowering moderator temperature.

C. Reactor power will increase due to the collapsing of the voids in the core resulting in increased neutron thermalization.

D. Reactor power will decrease due to the concentration of boron in the core absorbing fast neutrons.

QUESTION 34

Concerning the operation of the MSL & MSIV BYP OTBD ISOL, 1B21-F019, local transfer switches located at MCC EF1A07, select the correct statement.

A. Switches are not active until the control transfer to the Division 1 Remote Shutdown Panel is completed.

B. Switches are not active until the control transfer to the Division 2 Remote Shutdown Panel is completed.

C. Switches are always active and if placed in EMERGENCY, will cause 1B21-F019 to close on an MSIV isolation signal.

D. Switches are always active and if placed in EMERGENCY, will cause 1B21-F019 to close (if open).

QUESTION 35

The Feedwater Leakage Control System has been initiated following a Loss of Coolant Accident when the Feedwater System was no longer required for adequate core cooling.

Which one of the following describes the location where the Feedwater Leakage Control System injects seal water?

A simplified diagram of the Feedwater System is attached.

A. Between the inboard (F559A/B) and the outboard (F032A/B) feedwater check valves.

B. Between the outboard feedwater check valves (F032A/B) and the feedwater header shutoff valves (F065A/B).

C. Through the bonnets of the shutdown cooling to feedwater shutoff valves (E12-F053A/B).

D. Through the bonnets of the feedwater header shutoff valves (F065A/B).

QUESTION 36

Fuel element failure is indicated by increasing plant radiation levels.

Upscale alarms are received on all Main Steam Line Radiation Monitors.

Upscale Trip alarms are received on Main Steam Line Radiation Monitors A and B.

Which one of the following action(s) will automatically occur based on these indications only?

A. Off-Gas Discharge Isolation Valve N64-F632 closes.

B. Reactor Water Sample Isolation Valves B33-F019 and B33-F020 close.

C. Main Steam Line Isolation Valves B21-F022A-D and B21-F028A-D close.

D. Mechanical Vacuum Pump Suction Valves N62-F130A and N62-F130B close.

QUESTION 37

Describe the safety function of the Containment Ventilation Exhaust Radiation Monitor (D17-K609A-D) during a refueling outage.

A. Detect a fuel bundle rupture <u>inside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>off-site</u> dose limits are not exceeded.

B. Detect a fuel bundle rupture <u>outside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>off-site</u> dose limits are not exceeded.

C. Detect a fuel bundle rupture <u>inside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>on-site</u> dose limits are not exceeded.

D. Detect a fuel bundle rupture <u>inside</u> Containment which actuates the Containment Evacuation alarm to ensure personnel evacuate Containment.

QUESTION 38

Which one of the following describes the bases for the 'Drywell Pressure-High' function for the Reactor Protection System Instrumentation?

A. To ensure that the Minimum Critical Power Ratio (MCPR) is maintained above the MCPR Safety Limit.

B. To minimize the probability of fuel damage during a break in the Reactor Coolant Pressure Boundary (RCPB).

C. To reduce the amount of energy transferred to the coolant which could challenge the integrity of the Reactor Coolant Pressure Boundary (RCPB).

D. To ensure that sufficient capacity remains in the Scram Discharge Volume to accept the water displaced during control rod insertion from a full scram.

QUESTION 39

Which one of the following describes the bases for Alternate Rod Insertion due to high reactor pressure?

A. ARI reduces the challenge to the integrity of the Reactor Coolant Pressure Boundary.

B. ARI reduces the capability to cool the reactor fuel.

C. ARI reduces unwanted safety relief valve operation resulting in undesired voiding of the core.

D. ARI reduces unwanted safety relief valve operation resulting in undesired heatup of the Suppression Pool.

QUESTION 40

The following plant conditions exist:

- An ATWS is in progress
- MSIVs are isolated
- SRVs are being used to control reactor pressure

As Suppression Pool temperature increases, ECCS pump NPSH ________________.

A. <u>increases</u> resulting in the potential for ECCS pump cavitation.

B. <u>increases</u> resulting in the potential for pump runout.

C. <u>decreases</u> resulting in the potential for ECCS pump cavitation.

D. <u>decreases</u> resulting in the potential for pump runout.

QUESTION 41

The plant is shutting down to perform maintenance. Because of fuel cladding leaks, plant management has decided not to scram the reactor, but rather, to conduct a controlled insertion of control rods to minimize the potential increase in radioactivity release from the fuel.

As rod insertion progresses, the reactor goes subcritical. The Control Room Operator stops insertion of control rods with the intent to slow down the reactor depressurization and cooldown. Practically all of the heat generation at this point is from decay heat.

Thirty (30) minutes later the Control Room Operator notes that IRM flux levels are increasing on a long, stable positive reactor period.

Which one of the following describes the next action the Control Room Operator should take?

A. Insert control rods to a position that causes reactor period to be 60 – 150 seconds.

B. Withdraw the next in-sequence control rod to maintain the power rise to reach the point of adding heat.

C. Manually scram the reactor to terminate the power rise.

D. Monitor IRMs and range them according to the power increase to keep them on-scale.

QUESTION 42

A Loss of Coolant Accident (LOCA) has occurred. From the conditions below, select the set of conditions that would preclude the use of all ranges of RPV Water Level Instrumentation to determine reactor water level.

PEI-SPI Supplement Figure 1a is provided for reference.

A. Reactor Pressure 50 psig
Drywell Temperature 296 °F
Containment Temperature 205 °F

B. Reactor Pressure 25 psig
Drywell Temperature 260 °F
Containment Temperature 251 °F

C. Reactor Pressure 4 psig
Drywell Temperature 212 °F
Containment Temperature 212 °F

D. Reactor Pressure 0 psig
Drywell Temperature 190 °F
Containment Temperature 145 °F

QUESTION 43

Plant Emergency Instruction PEI-B13, RPV Control (ATWS) specifies that, under certain conditions, injection into the RPV be terminated and prevented except for boron and CRD.

The reason that injection into the RPV is terminated and prevented is to __________.

A. decrease the suppression pool heatup rate.

B. decrease the rate and magnitude of power oscillations.

C. increase the thermal driving head.

D. increase the core inlet subcooling.

QUESTION 44

Following a turbine trip at 40% reactor power, reactor pressure spiked to 1095 psig and then immediately decreased to 960 psig. The reactor did NOT scram.

Assuming control rods did NOT insert but all other systems performed as designed, what plant conditions would be observed 10 seconds after the turbine trip?

A. Feedwater flow controllers in MANUAL, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B CLOSED.

B. Feedwater flow controllers in MANUAL, Reactor Recirculation pumps TRIPPED off with pump breakers CB 3A/B and CB 4A/B OPEN.

C. Feedwater flow controllers in AUTO, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B OPEN.

D. Feedwater flow controllers in AUTO, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B CLOSED.

QUESTION 45

A Loss of Coolant Accident (LOCA) occurred and Containment pressure has increased above the Primary Containment Limit (PCL). PEI-T23, Containment Control, directs the Control Room Operators to vent Containment.

Which one of the following is the bases for the requirement to vent Containment?

If Containment pressure exceeds PCL, the _______________.

A. Containment pressure can no longer be determined since all Containment pressure indicators are off-scale high.

B. design pressure limit for the Containment Equipment Hatch has been exceeded.

C. Containment Vent Valves cannot be opened and closed.

D. RPV Vent Valves cannot be opened.

QUESTION 46

During a Main Turbine trip, reactor pressure peaked at 1115 psig. The reactor scrammed and reactor pressure is now 900 psig.

Select the item that describes the operation of the Safety Relief Valves (SRVs) during and following this transient.

Assume **NO** operator action is taken with respect to the SRVs.

A. One SRV opened and remained open until pressure decreased to 936 psig. If pressure increases to 1100 psig, one SRV will re-open.

B. Ten (10) SRVs opened. One SRV remained open until pressure decreased to 926 psig. If pressure increases to 1100 psig, two SRVs will re-open.

C. Ten (10) SRVs opened. Ten (10) SRVs remained open until pressure decreased to 936 psig. If pressure increases to 1100 psig, one SRV will re-open.

D. Nineteen (19) SRVs opened. Ten (10) SRVs remained open until pressure decreased to 936 psig. If pressure increases 1100 psig, two SRVs will re-open

QUESTION 47

Plant Conditions are as follows:

• Reactor is shutdown	all rods are in
• Reactor pressure	600 psig
• Reactor water level	210 inches
• Suppression Pool temperature	115°F
• Suppression Pool level	14.0 feet
• Drywell pressure	1.1 psig
• Containment pressure	0.8 psig
• RHR Loops A and B	Suppression Pool Cooling mode

What action is required to be performed?

A. Reduce reactor pressure to provide a wider operating margin to HCL.

B. Spray Containment.

C. Emergency Depressurize.

D. These conditions require no further actions be initiated.

QUESTION 48

The plant is operating at 100% reactor power when one Reactor Recirculation pump trips. All systems respond as designed to this event. How will RPV water level initially respond and what is the reason for this response?

RPV water level will ________________________.

A. increase due to the displacement of water into the downcomer by increased steam voiding.

B. decrease due to the lack of coolant velocity to sweep voids into the steam separator.

C. increase due to the continuing addition of feedwater at 100% rated feedwater flow.

D. decrease due to the runback of feedwater pumps to minimum speed.

QUESTION 49

The plant is operating at 35% reactor power when the Control Room operators observe Main Condenser vacuum is decreasing (increasing absolute pressure) and Off-Gas System after-filter discharge flowrate is increasing.

Which one of the following could be the cause of these indications?

A. Main Steam to Steam Jet Air Ejector supply pressure is less than 125 psig.

B. Steam Seal header pressure is 1.0 psig.

C. Steam Seal header pressure is 4.0 psig.

D. Steam Seal exhaust vacuum is greater than 12.0 inches water vacuum.

QUESTION 50

The Division 2 Diesel Generator (DG) received an automatic start signal due to a bus undervoltage condition.

Five (5) seconds later the undervoltage condition still exists, starting air pressure has decreased to 150 psig, and DG speed is 100 rpm.

Which one of the following describes the current status of the Division 2 DG?

The Division 2 DG starting air valves are ____________________.

A. open and the Division 2 DG will continue to roll for another 10 seconds unless its speed reaches 441 rpm.

B. open and the Division 2 DG will continue to roll for another 5 seconds unless its speed reaches 200 rpm.

C. closed because starting air pressure has decreased to 150 psig.

D. closed and the Division 2 DG has successfully started.

QUESTION 51

The plant is currently operating at 25% reactor power.

Which one of the following describes the response of the RC&IS System if the Main Turbine were to trip with no reactor scram?

RC&IS will ______________________.

A. implement the constraints of the Rod Pattern Controller, and depending on the control rod pattern, initiate Insert and/or Withdraw blocks.

B. implement the constraints of the Rod Withdrawal Limiter allowing control rods to be withdrawn up to 4 notches.

C. implement no constraints on control rod motion since reactor power is at the Low Power Alarm Point between the Rod Pattern Controller and the Rod Withdrawal Limiter.

D. implement the constraints of the Rod Withdrawal Limiter allowing control rods to be withdrawn up to 2 notches.

QUESTION 52

The plant was operating at 20% reactor power when a malfunction of the Feedwater Level Control System (C34) caused RPV water level to increase to 224 inches before Control Room Operators could restore RPV level back to normal.

Which one of the following is the plant response to this event?

A. There would be no noticeable plant response at this reactor power level.

B. Reactor power initially increased but then reactor power and reactor water level returned to normal after approximately one minute.

C. The Main Turbine tripped but the reactor did not scram.

D. The Main Turbine tripped and the reactor scrammed.

QUESTION 53

A high containment temperature has occurred and the Control Room Operators have entered PEI-T23, Containment Control. The PEI directs the Control Room Operators to "operate all available containment cooling".

Plant conditions are as follows:

- No BOP isolation has occurred.
- CVCW Chiller 'A' is operating.
- CVCW Chill Water Pump 'A' is operating.
- Containment Vessel Cooling Fans 'A', 'C', 'D', and 'F' are operating.

What action can be taken to "operate all available containment cooling"?

A. Start CVCW Chiller 'C'.

B. Start CVCW Chill Water Pump 'C'.

C. Start Containment Vessel Cooling Fans 'B' and 'E'.

D. Manually close the CVCW three-way valve to isolate any chill water bypass flow around the Containment Vessel Cooling Air Handling Unit cooling coils.

QUESTION 54

A cold reactor startup is in progress. Drywell temperature is slowly increasing as reactor heatup and pressurization is being performed.

Which one of the following describes how the Drywell Cooling System (M13) responds during <u>normal</u> plant operation?

Assume the Drywell Cooling System is in normal operation.

A. The standby Lower Drywell Cooling Fan will automatically start when Reactor Vessel Support Skirt area temperature exceeds 120 °F.

B. The Lower Drywell Cooler NCC Bypass Valve, P43-F365, will re-position to cause more cooling water to flow through the in-service Lower Drywell Air Handling Unit cooling coil.

C. The Lower Drywell Cooler NCC Bypass Valve, P43-F365, will re-position to cause less cooling water to flow through the in-service Lower Drywell Air Handling Unit cooling coil.

D. The Lower Drywell Cooler 3-Way NCC Supply Valve, P43-F025, will throttle open to provide additional cooling water flow through the in-service Lower Drywell Air Handling Unit cooling coil.

QUESTION 55

The plant is operating at 90% reactor power when both Nuclear Closed Cooling System Heat Exchanger temperature control valves malfunction. Plant operators are unable to operate the temperature control valves manually. The standby NCC Heat Exchanger is drained for maintenance. As a result, alarm NCC HX OUTLET TEMP HIGH is received and NCC Heat Exchanger outlet temperature is 93 °F and slowly increasing.

Which one of the following operator actions is required to be performed?

A. Start an additional Service Water Pump.

B. Enter PEI-T23, Containment Control.

C. Perform a rapid manual shutdown of the Reactor Water Cleanup System.

D. Close both Recirculation Loop A and B Flow Control Valves until total core flow is 58 Mlbm/hr.

QUESTION 56

The plant is operating at power. The Motor Feed Pump is tagged out due to a motor ground. Reactor Feed Pump 'B' has just been removed from service for corrective maintenance.

What is the current operating guideline for reactor power based on the present status of the Feedwater System?

A. 63%

B. 66%

C. 68%

D. 71%

QUESTION 57

The following plant conditions exist:

- The reactor is in MODE 4
- RHR Loop A is in the Shutdown Cooling mode
- RHR Loop B is in the Suppression Pool Cooling mode

A valid RPV Level 1 reactor water level condition occurs.

Which one of the following describes the automatic response of the RHR system?

A. RHR Pumps A and B trip; RHR Loop B shifts to the LPCI mode; RHR Pump B restarts.

B. RHR Pump A trips; RHR Loop B continues to operate in the Suppression Pool Cooling mode.

C. RHR Pump A trips; RHR Loop B realigns to the LPCI mode.

D. RHR Pump A continues to operate in the Shutdown Cooling mode; RHR Loop B realigns to the LPCI mode.

QUESTION 58

IOI-12, Maintaining Cold Shutdown, specifies that when Reactor Recirculation Pumps are not operating, reactor water level should be maintained greater than 250 inches on the Reactor Shutdown Range Level.

Maintaining reactor water level in a range of 250 to 260 inches on the Reactor Shutdown Range Level will ________________.

A. prevent undetected boiling locally in the core.

B. provide adequate NPSH for the RHR pumps during shutdown cooling operation.

C. provide sufficient water volume to prevent a loss of shutdown cooling due to a low reactor water level isolation.

D. provide sufficient water volume to flood the Main Steam lines.

QUESTION 59

A reactor startup is in progress with reactor pressure at 650 psig when CRD Pump 'A' trips. While preparing to start CRD Pump 'B', four HCU accumulator faults are received; three of which are associated with withdrawn control rods. CRD charging water pressure is 1375 psig, as read on 1C11-R601, CRD PRESSURE CHARGING WATER.

Which one of the following operator actions is required to be performed?

A. The Reactor Mode Switch must be placed in SHUTDOWN immediately.

B. The Reactor Mode Switch must be placed in SHUTDOWN if the conditions above still exist after 20 minutes.

C. A fast reactor shutdown must be commenced immediately.

D. A fast reactor shutdown must be commenced if the conditions above still exist after 20 minutes.

QUESTION 60

PEI-T23, Containment Control, directs the operator to Emergency Depressurize if Suppression Pool level cannot be maintained below 24.5 feet.

Why does the PEI direct emergency depressurization at this point?

Above this Suppression Pool level, ______________________.

A. the operation of SRVs may cause failure of the Containment.

B. the pressure rise in the Containment could cause overflow of the weir wall.

C. boron would be diluted below the Hot Shutdown Boron Weight if boron was being injected.

D. the NPSH for pumps taking suction on the Suppression Pool would be insufficient.

QUESTION 61

The plant is operating at 100% reactor power. AEGTS Train 'A' is in operation with its associated Annulus Differential Pressure Controller in the AUTO mode.

Which one of the following describes the response of the AEGTS System if the absolute pressure in the Annulus <u>decreases below</u> the desired pressure?

A. The standby AEGTS Train 'B' will automatically start and restore Annulus pressure to the desired value.

B. The associated Annulus Differential Pressure Controller setpoint will automatically increase to match the higher Annulus pressure.

C. The AEGTS Train 'A' exhaust damper will throttle open while the recirculation damper will throttle close until the Annulus pressure is restored to the desired value.

D. The AEGTS Train 'A' exhaust damper will throttle close while the recirculation damper will throttle open until Annulus pressure is restored to the desired value.

QUESTION 62

PEI-N11, Containment Leakage Control, has been entered on Area Water Level. It is determined that a primary system is discharging into the affected area. PEI-B13, RPV Control (Non-ATWS) is required to be entered and executed concurrently with PEI-N11.

Which one of the following is the reason for entering PEI-B13, RPV Control (Non-ATWS)?

A. A reduction in RPV water level will effect a decrease in the flow of water into the affected area in order to maintain personnel access in the affected area.

B. A reduction in RPV water level will effect a decrease in the flow of water into the affected area in order to maintain equipment qualifications in the affected area.

C. A reduction in RPV pressure will effect a decrease in the flow of water into the affected area in order to maintain personnel access in the affected area.

D. A reduction in RPV pressure will effect a decrease in the flow of water into the affected area in order to maintain equipment qualifications in the affected area.

QUESTION 63

A fire exists in Reactor Recirculation Pump 'A'. The fire was reported at 1358. The CNTMT CO2 SUPPLY OUTBOARD ISOL VALVE, 1P54-F340, was opened by the Control Room Operators at 1440.

Which one of the following describes the current status of the CO2 system?

CO_2 for the Reactor Recirculation Pump fire was ______________________.

A. automatically released into the Drywell and was discharged for the required amount of time.

B. automatically released into the Drywell and was not discharged for the required amount of time.

C. not automatically released into the Drywell; therefore, the CO_2 System will need to be manually discharged.

D. not automatically released into the Drywell; therefore, a Drywell entry will be required to suppress the fire.

QUESTION 64

When a control rod is selected, the Control Room Operator observes that the control rod has an "Insert Block" and "Insert Inhibit" light.

This means that the control rod **cannot** be INSERTED ________________.

A. since this might allow the LHGR or MCPR limit to be exceeded.

B. since this would indicate a control rod block due to a system fault.

C. since this might allow a control rod to have excessive rod worth.

D. since this would indicate a control rod block due to a bypassed control rod position indicator.

QUESTION 65

Given the following conditions for Reactor Recirculation Hydraulic Power Unit 'B':

- Subloop 1 — READY, LEAD, OPERATIONAL, PRESSURIZED
- Subloop 2 — READY

Alarm 'FCV B HPU NEEDS MAINTENANCE" is received at panel H13-P680.
A Control Room Operator reports that an amber 'OIL WARM' light is illuminated on panel H13-P614 for HPU 'B'.

Which one of the following describes the operational status of HPU 'B'?

A. Subloop 1 and Subloop 2 are in the Maintenance mode.

B. Subloop 1 remains in operation and Subloop 2 remains in Standby.

C. Subloop 1 is in the Maintenance mode and Subloop 2 is in operation but <u>not</u> in LEAD.

D. Subloop 1 is in the Maintenance mode and Subloop 2 is in operation but <u>is</u> in LEAD.

U.S. NUCLEAR REGULATORY COMMISSION
WRITTEN EXAMINATION JANUARY 2001
REACTOR OPERATOR

QUESTION 66

The Control Room has been evacuated and plant control has been established at the Division 1 Remote Shutdown Panel.

Select the correct statement concerning operation of the Residual Heat Removal (RHR) System under these conditions.

A. The RHR PUMP A MIN FLOW VALVE, E12-F064A, will auto open when flow is less than 1650 gpm for 8 seconds when the RHR Pump is running.

B. The RHR A TEST VALVE TO SUPR POOL, E12-F024A, will auto close if a LPCI initiation signal is received.

C. The RHR A Pump, E12-C002A. will auto start if a LPCI initiation signal is received.

D. The RHR A TO RADWASTE ISOLATION VALVE, E12-F049, will auto close if drywell pressure is > 1.68 psig.

QUESTION 67

Assume that all required conditions have been met for an Automatic Depressurization AND that depressurization is in progress. If ALL the Low Pressure ECCS pumps trip off, which one of the following describes how the Automatic Depressurization System is affected?

A. Automatic Depressurization will stop and can be recommenced by depressing the ADS Manual Initiation pushbuttons.

B. Automatic Depressurization will stop and can be recommenced by restarting a Low Pressure ECCS pump.

C. Automatic Depressurization will stop and can only be reestablished by manually opening SRVs.

D. Automatic Depressurization will continue without interruption.

QUESTION 68

The High Pressure Core Spray System (HPCS) automatically initiated due to receipt of both the Low Reactor Water Level and High Drywell Pressure signals.

HPCS initiation may/will be reset _______________________.

A. automatically when both initiation signals clear.

B. manually only after both initiation signals clear.

C. manually after the Low Reactor Water Level initiation signal clears.

D. manually after the High Drywell Pressure initiation signal clears.

QUESTION 69

The plant is operating at 35% reactor power. MSIV B21-F022C has a faulty limit switch which is generating a '< 92% open' signal.

Which one of the following MSIVs, if closed, would cause a ½ scram?

A. MSIV B21-F028A

B. MSIV B21-F028B

C. MSIV B21-F022A

D. MSIV B21-F028C

QUESTION 70

The following plant conditions exist:

- Reactor Mode Switch is in STARTUP/STANDBY
- Intermediate Range Monitors (IRM) A, C, D, E, and G are on Range 3; all other IRMs are on Range 2
- Source Range Monitor (SRM) A is reading 0.5 cps
- SRMs B and C are reading 8.3 x 10E4
- SRM D mode switch is in STANDBY

A rod block signal has been generated.

Which one of the following has caused the rod block?

A. SRM Upscale

B. SRM Downscale

C. SRM Detector Wrong Position

D. SRM Inoperable

QUESTION 71

During reactor power operations, the following plant conditions exist:

- Reactor power 75%
- Core flow 70% (73 Mlbm/hr)
- Total Recirculation drive flow 65% (62 Kgpm)
- Recirculation Loops in operation Both

Which one of the following is the APRM Upscale Thermal Power Trip Setpoint?

A. 84.3%

B. 104.6%

C. 106.9%

D. 107.7%

Delete
AMS 2/13/01

QUESTION 72

The reactor is shutdown with the following plant conditions:

• Reactor water level	255 inches on Shutdown Range Water Level
• Reactor water temperature	120 degrees F
• Reactor pressure	0 psig
• Drywell temperature	110 degrees F

Which one of the following is correct with respect to these plant conditions?

Actual reactor water level will be ______________________.

A. higher than indicated since the reactor water temperature is LOWER than the calibration conditions for the Shutdown Range Water Level.

B. lower than indicated since the reactor water temperature is LOWER than the calibration conditions for the Shutdown Range Water Level.

C. lower than indicated since the drywell temperature is HIGHER than the calibration conditions for the Shutdown Range Water Level.

D. higher than indicated since the drywell temperature is HIGHER than the calibration conditions for the Shutdown Range Water Level.

QUESTION 73

An electrical transient has occurred and Service Water Pump 'D' is lost. Which bus normally powers this pump?

A. Bus H12

B. Bus XH12

C. Bus XH21

D. Bus XH22

QUESTION 74

A Loss of Coolant Accident (LOCA) has occurred and Drywell pressure is 2.4 psig. LPCS is operating on minimum flow. LPCI A, B, and C have been overridden off since they were not required to maintain adequate core cooling.

Ten minutes after the initial Drywell break, RPV water level suddenly decreased below RPV Level 1. One hundred (100) seconds later, RPV water level was restored above RPV Level 3.

It has now been five minutes since RPV water level was restored above RPV Level 3 and the Unit Supervisor has directed the Supervising Operator to verify the current status of the Automatic Depressurization System (ADS).

NO operator actions were taken with respect to ADS other than resetting annunciators that had cleared.

Which one of the following is the correct annunciator status that the Supervising Operator should expect to observe?

	ADS A 105 SEC TIME DELAY LOGIC INITIATED	ADS A TIMER 90 SEC & RUNNING	ADS A INSTANTANEOUS LOGIC INITIATED
A.	ON	ON	ON
B.	OFF	ON	OFF
C.	ON	OFF	OFF
D.	OFF	OFF	ON

QUESTION 75

A Loss of Coolant Accident (LOCA) has occurred and Drywell pressure is 2.2 psig.

Based on these plant conditions, which one of the following describes the operation of the Containment Vacuum Relief Isolation Valves (M17-F015, F025, F035, and F045)?

If a Containment Vacuum Relief Isolation Valve control switch is ______________.

A. placed in OPEN, then the valve will open regardless of Containment pressure.

B. placed in OPEN, then the valve will open only if Containment pressure is negative.

C. placed in CLOSE, then the valve will close regardless of Containment pressure.

D. placed in CLOSE, then the valve will close only if Containment pressure is negative.

QUESTION 76

Given the following plant conditions:

- Drywell pressure is 1.3 psig
- Reactor water level is +105 inches
- Main condenser vacuum is 25 inches Hg A
- Reactor pressure is 75 psig

Which one of the following describes the system components that isolated based on these plant conditions?

A. RWCU isolation valves, MSIVs and MSL Drain isolation valves, RCIC steam supply line isolation valves

B. MSIVs and MSL Drain isolation valves, NCC Containment & Drywell isolation valves, RWCU isolation valves

C. RCIC steam supply line isolation valves, Drywell Floor Drain Sump & Containment Drain Sump isolation valves, Reactor Water Sample isolation valves

D. Reactor Water Sample isolation valves, RWCU isolation valves, MSIVs and MSL Drain isolation valves

QUESTION 77

The plant has experienced a Loss of Coolant Accident (LOCA). All ECCS have operated as designed EXCEPT one Drywell pressure transmitter (B21-N094A) that supplies a high Drywell pressure signal to the Division 1 Containment Spray initiation logic. The transmitter has failed downscale (indicates zero).

Ten (10) minutes after the LOCA initiation signal, plant conditions are as follows:

- Drywell pressure 1.9 psig
- Containment pressure 4.1 psig

Which one of the following describes the status of the Division 1 Containment Spray System?

Division 1 Containment Spray System has ____________________.

A. automatically initiated.

B. not automatically initiated but will initiate if the CNTMT SPRAY A MANUAL INITIATION pushbutton is armed and depressed.

C. not automatically initiated but will initiate if the CNTMT SPRAY A HI DW PRESS BYP keylock switch is placed in BYPASS.

D. not automatically initiated but will initiate if Containment pressure increases to 7.8 psig.

QUESTION 78

Reactor power is 90%.

Which one of the following describes how an SRV, that was stuck open, is verified closed after its control power fuses have been removed in accordance with ONI-B21-1, SRV Inadvertent Opening/Stuck Open?

A. Reactor pressure increases.

B. Main Generator electrical output increases.

C. Indicated steam flow on the effected steam line decreases.

D. Both SOLENOID STATUS A (B) red indicating lights on P601 are off

QUESTION 79

The plant is operating at 80% reactor power. Reactor Feed Pump Turbine (RFPT) 'A' and 'B' controllers are in Automatic when the High Pressure Core Spray (HPCS) System inadvertently initiates and injects into the RPV.

Which one of the following describes the response of the Feedwater Level Control System?

Total feedwater flow will ____________________.

A. decrease; resulting in a reactor scram on low reactor water level.

B. increase; reactor water level will stabilize at some level slightly lower than the tape set value.

C. decrease; reactor water level will stabilize at the same level as the tape set value.

D. decrease; reactor water level will stabilize at some level slightly higher than the tape set value.

QUESTION 80

A secondary flowpath associated with the Annulus Exhaust Gas Treatment System (AEGTS) allows a purge path to be established.

Which one of the following describes the purpose of this secondary flowpath?

A. To control Drywell temperature during plant heatup.

B. To control Drywell pressure during plant heatup.

C. To control Drywell airborne radiation levels to allow Drywell entry during plant heatup.

D. To control Drywell hydrogen concentration during a Loss of Coolant Accident (LOCA).

QUESTION 81

The plant is at 15% reactor power. The Control Room Operator is in the process of synchronizing the Main Generator to the grid and is ready to close Generator Breaker S-610-PY-TIE. A malfunction occurs in the turbine control system and turbine speed increases to just below the Main Turbine overspeed trip setpoint.

Which one of the following describes how the plant will respond to this event?

A. The synchroscope will turn clockwise at a slower rate; Main Generator output voltage will increase.

B. The synchroscope will turn clockwise at a faster rate; Main Generator output voltage will not change.

C. The synchroscope will turn counter-clockwise at a faster rate; Main Generator output voltage will decrease.

D. The synchroscope will turn counter-clockwise at a faster rate; Main Generator output voltage will not change.

QUESTION 82

The Division 1 Diesel Generator is being operated in parallel with the grid. The Diesel Generator Control Transfer Switch is in the LOCAL position.

Which one of the following describes the response of the Division 1 Diesel Generator if a valid Loss of Coolant Accident (LOCA) signal occurs?

The Division1 Diesel Generator output breaker will ______________________:

A. not trip but the diesel generator trips normally bypassed by a LOCA signal will be bypassed.

B. not trip and the diesel generator trips normally bypassed by a LOCA signal will not be bypassed.

C. trip but the diesel generator trips normally bypassed by a LOCA signal will not be bypassed.

D. trip and the diesel generator trips normally bypassed by a LOCA signal will be bypassed

QUESTION 83

The following plant conditions exist:

- The plant is operating at 100% reactor power
- Feedwater Level Control is on the Master Level Controller with Narrow Range Level Channel 'A' selected
- Narrow Range Level Channel 'A' has failed upscale

The reactor will scram on ________________________.

A. low RPV water level; water level will be restored to approximately 200 inches in accordance with ONI-C71-1, Reactor Scram.

B. low RPV water level; water level will be restored to 185 - 215 inches in accordance with PEI-B13, RPV Control (Non-ATWS).

C. high RPV water level; water level will be restored to approximately 200 inches in accordance with ONI-C71-1, Reactor Scram.

D. high RPV water level; water level will be restored to 185 - 215 inches in accordance with PEI-B13, RPV Control (Non-ATWS).

QUESTION 84

The plant is operating at 100 % reactor power when Recirc Pump Seal Flow Regulator, 1C11-D012A, fails closed.

Which one of the following describes the potential consequence of this condition?

If Reactor Recirculation Pump A operation continues, then the _______________.

A. radioactivity discharged to Radwaste will decrease due to the reduced recirc pump seal purge flow.

B. possibility of recirc pump seal damage will decrease due to the reduced recirc pump seal purge flow.

C. possibility of recirc pump seal damage will increase due to the possible ingestion of dirt from an unclean piping system.

D. possibility of recirc pump seal damage will increase unless the alternate recirc pump seal purge supply from the Condensate Transfer and Storage System can be lined up.

QUESTION 85

Reactor Recirculation Pumps 'A' and 'B' tripped off when reactor water level reached Level 2. Preparations are underway to restart Reactor Recirculation Pump 'A' in order to restore forced circulation through the core.

Which one of the following interlocks must be met for Reactor Recirculation Pump 'A' to successfully start and operate in slow speed?

A. RPV water level is greater than RPV Level 3.

B. Flow Control Valve 'A' actuator (D004A) is full open.

C. Differential temperature between reactor steam dome temperature and Reactor Recirculation Pump 'A' suction temperature is greater than 10 degrees F.

D. Differential temperature between reactor steam dome temperature and Reactor Recirculation Pump 'A' suction temperature is less than 50 degrees F.

QUESTION 86

Given the following conditions:

- The Reactor Water Cleanup System (RWCU) is operating in the normal mode
- The RWCU LD ISOLATION BYPASS Switches (E31-S1A,B) on panels H13-P632 and P642 have been placed in "BYPASS"

Select the expected effect on the RWCU System.

The RWCU System isolation signal on ________________________.

A. high non-regenerative heat exchanger outlet temperature is defeated.

B. initiation of the Standby Liquid Control System (SLC) is defeated.

C. high differential flow rate is defeated.

D. low RPV level (Level 2) is defeated.

QUESTION 87

RHR Loop A has just been placed into the Shutdown Cooling mode of operation using the normal return path. The cooldown rate is excessive. The Unit Supervisor directs you to reduce the cooldown rate.

Which one of the following is the correct action to reduce the cooldown rate?

A. Throttle close the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle open the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

B. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

C. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 2000-7100 gpm.

D. Throttle ESW flow through the RHR Heat Exchanger using RHR A HX'S ESW OUTLET VALVE, P45-F068A.

QUESTION 88

The plant is operating at 100% reactor power. RHR A HX'S BYPASS VALVE, E12-F048A, has failed in the fully open position.

Which mode(s) of RHR Loop A are/is OPERABLE?

A. Suppression Pool Cooling only

B. Low Pressure Coolant Injection only

C. Containment Spray and Suppression Pool Cooling

D. Containment Spray and Low Pressure Coolant Injection

QUESTION 89

Plant conditions are as follows:

- Core offload is in progress
- Refuel Bridge is stationed in the 'cattle chute' (Portable Refueling Shield) between the Reactor Pressure Vessel and the Dryer Storage Pool
- Refuel Bridge grapple is loaded with a new fuel bundle
- Reactor Mode Switch is in the SHUTDOWN position
- All control rods are fully inserted

Which one of the following describes the allowable direction(s) that the Refuel Bridge can travel in (i.e., travel direction will **not** be prevented by an interlock)?

A. The Refuel Bridge can move in either direction.

B. The Refuel Bridge can move towards the Dryer Storage Pool only.

C. The Refuel Bridge can move towards the Reactor Pressure Vessel only.

D. The Refuel Bridge cannot move in either direction

QUESTION 90

A plant transient resulted in a loss of extraction steam to FDW Heaters 5A and 5B. The Immediate Actions of ONI-N36, Loss of Feedwater Heating, have been completed. All remaining FDW Heaters are in operation. Current reactor power is 90%.

What is the Main Generator MWe limitation based on the current FDW Heater lineup?

The Main Generator Mwe Limitation Table from ONI-N36 is provided below for reference.

Heater	Number of Trains Lost	Side of Heater Lost	RFP Steam Supply		Basis
			Main	Extraction	
1 & 2	1	Condensate	1125 MWe	1188 MWe	1
5	2	Extraction	938 MWe	1000 Mwe	2
1 & 2, 3, 5, 6	2 Trains of the same heater	Condensate	563 MWe	625 MWe	3

A. 625 MWe

B. 938 MWe

C. 1000 MWe

D. 1188 MWe

QUESTION 91

The plant is operating at 90% reactor power with both Reactor Feed Pump Turbines (RFPTs) in operation. RFPT 'A' Flow Controller is in MANUAL and RFPT 'B' Flow Controller is in AUTOMATIC. The STARUP FDW PUMP SELECT SWITCH is in the MFP position.

Which one of the following describes the response of the Feedwater System if the speed of RFPT 'A' is slowly decreased?

A. RFPT 'B' flow rate will increase, RFPT 'A' flowrate will decrease, and total feedwater flow will remain the same.

B. RFPT 'B' discharge pressure will slightly decrease, RFPT 'A' discharge pressure will slightly increase, and total feedwater flow will remain the same.

C. RFPT 'B' flow rate will remain the same, RFPT 'A' flowrate will decrease, and total feedwater flow will decrease.

D. RFPT 'B' discharge pressure will slightly increase, RFPT 'A' discharge pressure will slightly decrease, and total feedwater flow will increase.

QUESTION 92

The plant is operating at 100% reactor power when Division 1 DC Bus ED-1-A is lost.

Which one of the following conditions will occur?

A. RCIC automatically initiates.

B. Recirculation Pumps 'A' and 'B' trip off.

C. Division 1 Diesel Generator automatically trips, if running.

D. Alarm window "ANN PWR SUPPLY FAIL" on H13-P680 energizes.

QUESTION 93

The plant is in a refueling outage and the M14 Containment Vessel and Drywell Purge System (CVDWP) is operating in the Refuel mode. Containment Ventilation Exhaust Radiation Monitor D17-K609C is in alarm due to a Downscale indication.

An I&C Technician is troubleshooting D17-K609C when the following alarms are received in the Control Room:

- CNTMT & DW PURGE EXHAUST FAN A FLOW LOW
- CNTMT & DW PURGE EXHAUST FAN B FLOW LOW
- CNTMT PURGE SUPPLY FAN A FLOW LOW
- CNTMT PURGE SUPPLY FAN B FLOW LOW
- DW PURGE SUPPLY FAN A FLOW LOW
- DW PURGE SUPPLY FAN B FLOW LOW

Which one of the following conditions is the probable cause for the current status of the CVDWP System?

A. Either Containment Ventilation Exhaust Radiation Monitor D17-K609A or D17-K609D is in an UPSCALE TRIP (HI-HI) condition due to a refueling accident in Containment.

B. Containment Ventilation Exhaust Radiation Monitor D17-K609B is in an UPSCALE TRIP (HI-HI) condition due to a refueling accident in Containment.

C. The I&C Technician inadvertently placed the MODE SWITCH for D17-K609D to the ZERO position.

D. The I&C Technician inadvertently placed the MODE SWITCH for D17-K609A to the TRIP TEST position.

QUESTION 94

Control Room HVAC and Emergency Recirculation (M25/26) Train 'A' has been manually shifted from the NORM mode to the EMERG RECIRC mode by placing CONT RM HVAC TRAIN A MODE SELECT, M25-S7, in the EMERG RECIRC position.

Assume no other operator actions were performed.

Which one of the following describes the current damper lineup for M25/26 Train 'A'?

	HVAC A OTBD SUPP DAMPER F010A	HVAC A INBD SUPP DAMPER F020B	EMG RCIRC DAMPER A F040A	HVAC A RETURN DAMPER F110A	HVAC A EXHAUST DAMPER F130A
A.	Open	Open	Closed	Open	Closed
B.	Open	Open	Closed	Closed	Open
C.	Closed	Closed	Open	Closed	Closed
D.	Closed	Open	Open	Closed	Closed

QUESTION 95

The plant is operating at 75% reactor power.

MSL B INBD MSIV B21-F022B control switch is in the TEST position. The Control Room Operator depresses the MSL B INBD MSIV TEST pushbutton 1B21H-S3B.

Which one of the following describes the response of MSL B INBD MSIV B21-F022B?

A. Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

B. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

C. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

D. Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

QUESTION 96

Which one of the following would be the control rod movement sequence most likely to cause the phenomenon known as the 'reverse power effect'?

A. 1 or 2 notch withdrawal of a deep control rod.

B. 1 or 2 notch withdrawal of a shallow control rod.

C. 1 or 2 notch insertion of a shallow control rod.

D. 10 or 12 notch continuous withdrawal of a shallow control rod.

QUESTION 97

During refueling operations, a FPCC SURGE TANK A LEVEL HI/LO annunciator is received. The Control Room Operator reports that surge tank level is high. The FPCC SURGE TK FILL FROM CST VALVE, G41-F045, is verified closed.

Which one of the following could be the cause of the surge tank high level?

A. Steam Dryer was removed from the Dryer Storage Pool during RPV re-assembly.

B. Emergency makeup valve from the Service Water System (P41) is open or leaking by.

C. Fuel Transfer Tube Drain Tank is pumping down during fuel transfer operations.

D. FPCC flow to the lower fuel pools was increased.

QUESTION 98

Which one of the following load sets will be lost if Bus H11 becomes de-energized?

A. Hotwell Pump A, Hotwell Pump C, and Condensate Booster Pump B

B. Hotwell Pump B, Hotwell Pump C, and Condensate Booster Pump B

C. Hotwell Pump A, Condensate Booster Pump A, and Condensate Booster Pump C

D. Hotwell Pump C, Condensate Booster Pump A, and Condensate Booster Pump C

QUESTION 99

Which one of the following core components acts as a partition to force the majority of coolant and moderator flow into the control rod guide tubes, fuel support pieces, and to the fuel assemblies?

A. Baffle plate

B. Core shroud

C. Core plate

D. Control rod guide tube flow orifices

QUESTION 100

The reactor power 8-hour average limit is 3700 MWt.

What is the basis for this limitation?

A. To prevent exceeding the maximum steady state Main Generator real load.

B. To prevent exceeding the maximum steady state licensed reactor power level.

C. To minimize Recirculation Flow Control Valve (FCV) oscillations.

D. To minimize Main Turbine Control Valve oscillations.

Perry RO Written Examination
Answer Key

Q#	Ans	K/A	Reference
1	B	211000K2.02	SDM-C41, ARI-H13-P601-19D1, ONI-R25-1, PDB-H022
2	C	215003K3.03	SDM-C51
3	D	201005A4.01	SDM-C11, SOI-C11
4	D	217000K1.03	SDM-E51
5	B	202002K1.12	SDM-B33
6	A	218000K4.02	SDM-B21C
7	B	239001K1.22	SDM-B21/N11
8	A	262002K6.03	SDM-R14/15
9	B	271000K6.08	SOI-N64/62
10	D	286000A3.04	SDM-P54, SOI-P54
11	B	202001K5.05	SDM-C71, SMD-B33
12	C	288000K1.05	SDM-D17, SDM-M51
13	B	500000EK1.01	SDM-M56, PEI Basis Document
14	B	G 2.1.22	Technical Specification Table 1.1-1
15	B	G 2.1.2	PAP-0126
16	B	G 2.2.12	PAP-1105
17	A	G2.2.26	IOI-(, ORM 6.2.3, SOI-F11, SOI-F15, PAP-0802
18	B	G2.2.22	Technical Specification - Safety Limits
19	D	G 2.3.9	SOI-M14, SOI-G33
20	A	G 2.3.1	HPI-B0003
21	A	G 2.4.49	ONI-P44
22	D	295019AA2.02	ONI-P52
23	A	G 2.4.10	ARI-H13-P680-6 (B1)
24	A	G 2.4.11	ONI-D17
25	C	G 2.4.35	PEI-SPI 4.3
26	C	217000K5.06	SOI-SDM-E51

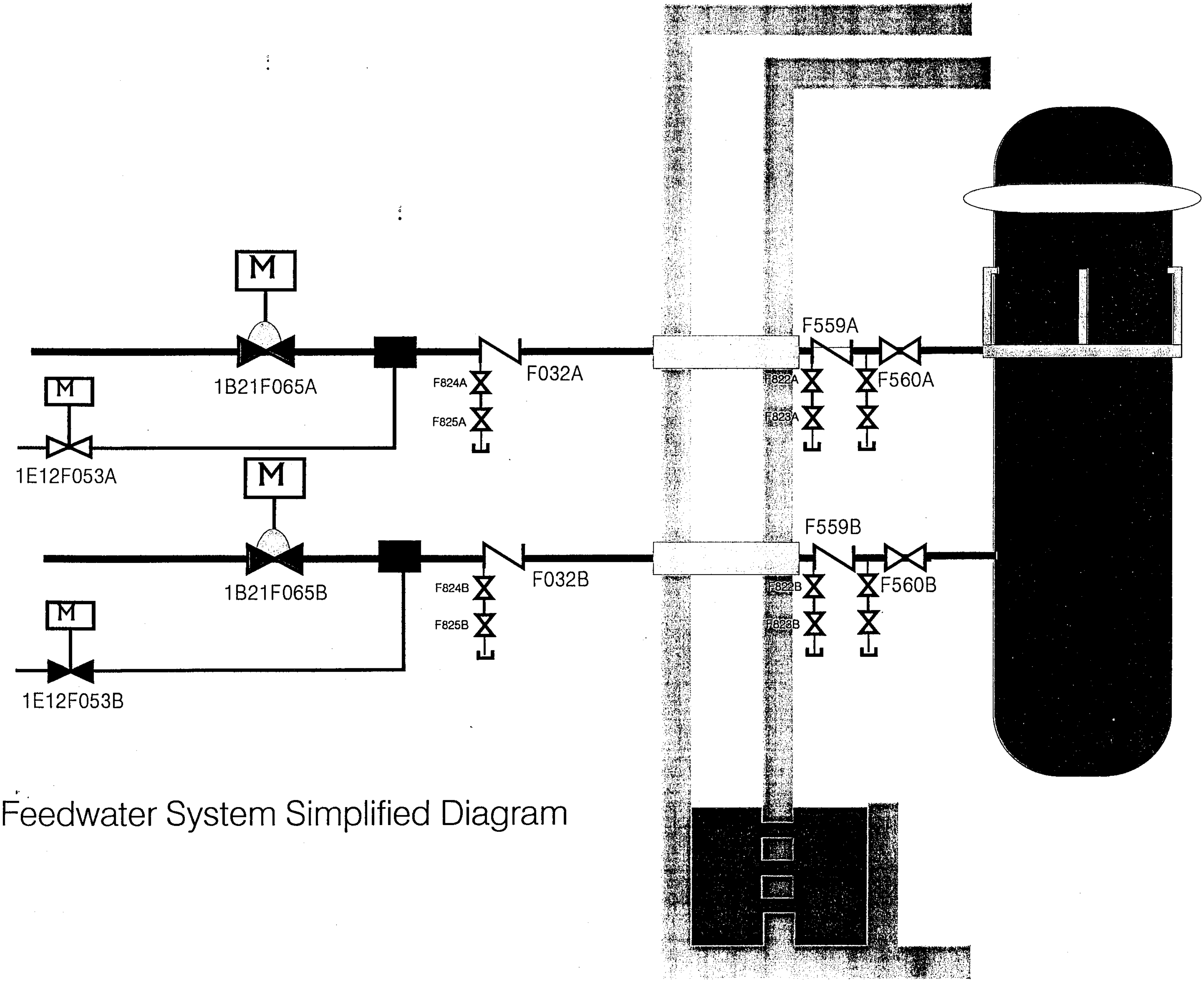
Q#	Ans	K/A	Reference
27	A	295004A2.03	SDM-R42, TS LCO 3.8.4
28	A	295003AK2.04	SDM-E22B
29	C	295006AA1.05	ONI-C71-1, GP Reactor Theory Text
30	B	295007AA2.03	AT&AA Text Chapter 2 (USAR 15B 5.2.2)
31	A	295009AA1.03	SDM-B33, SDM-B21
32	C	295010AA2.05	SDM-E31, Technical Specification 3.4.7
33	B	295015AK1.04	GP Reactor Theory Text, PEI Bases Document
34	D	295016AK2.02	SDM-C61
35	D	G 2.1.28	SDM-N27
36	D	295017AA1.06	SDM-N62, SDM-D17A
37	A	295023AK3.02	SDM-D17A, TS 3.3.6.1, USAR 15.7.6
38	B	295024EK3.06	SDM-C71, LCO 3.3.1.1
39	A	295025EK3.06	SDM-C22, LCO 3.3.1.1, LCO 3.3.4.2
40	C	295026EK1.01	GP Components Text, Suppression pool temperature PEI bases
41	C	295014AK3.01	IOI-4 Caution statement, PAP-0201 section 6.4.5
42	A	295027EK2.02	PEI Bases document
43	A	295031EK1.03	PEI bases document
44	C	295037EK2.02	SDM-C22
45	B	G 2.4.18	PEI-T23
46	B	295025EA1.03	SDM-B21/N11
47	C	295030EA2.02	PEI-T23, PEI-B13
48	A	295001AK3.01	AT&AA Text Chapter 5 (USAR 15.3.1)
49	B	295002AK2.11	ONI-N62
50	B	264000A3.01	SDM-R43
51	A	295005AA1.03	SDM-C11
52	D	295008AK1.01	SMD-C71, SDM-N32/C85, SDM-B21
53	C	295011AK2.01	PEI-T23, SDM-M11, SOI-M11
54	B	295012AK2.02	SDM-M13

Q#	Ans	K/A	Reference
55	D	295018AK2.01	SOI-B33, ONI-P43
56	A	G 2.1.32	SOI-C34, SOI-N27, ONI-N27
57	C	295020AK2.09	SMD-E12
58	A	295021AK1.04	IOI-12
59	B	295022AK1.02	ONI-C11-1, Technical Specification 3.1.5
60	A	295029EK3.01	PRI-T23, PEI bases document
61	D	295035EA1.02	SDM-M15
62	D	295036EK3.02	PEI-N11, PEI bases document
63	C	G 2.4.27	ONI-P54
64	C	201005A2.03	SDM-C11, SOI-C11
65	C,D	202002A3.02	SDM-B33, ARI-H13-P680-4
66	D	203000K4.14	IOI-11, Attachment 1, SDM-C61
67	D	209001K3.02	SDM-B21C
68	C	209002A4.15	SDM-E22A
69	B	212000K6.05	SDM-C71
70	D	215004A3.04	SDM-C51
71	DELETE		
72	C	216000K5.07	SDM-B21
73	C	400000K2.01	DCP 99-5019
74	D	218000A1.05	SDM-B21C
75	B	223001K5.01	SDM-M17
76	D	223002A3.02	SDM-B21(NS4)
77	B	226001K6.08	SDM-E12
78	B	239002A2.03	SDM-B21/N11, ONI-B21-1
79	D	259002A1.02	SDM-C34
80	B	261000K1.02	SDM-M15
81	B	262001A4.02	GP components text - chapter 5, IOI-3
82	B	264000K4.07	SDM-R43
83	B	212000A2.08	ONI-C71-1, PEI-B13, PEI bases document

Q#	Ans	K/A	Reference
84	C	201001K3.01	SDM-B33, SDM-C11
85	D	202001K4.15	SDM-B33
86	C	204000K1.15	SDM-E31, SDM-G33
87	B	205000A1.06	SOI-E12
88	B	G 2.1.33	SDM-E12, Technical Specification 3.5.1, 3.6.1.7, and 3.6.2.3
89	B	234000A3.02	SDM-F11/15
90	C	245000A2.06	ONI-N36
91	A	259001A4.02	SDM-C34, GP components text chapter 2
92	B	263000K3.03	SDM-R42, ONI-R42-1
93	B	272000K4.02	SDM-M14, SDM-D17A
94	D	29003A4.03	SDM-M25/26, SOI-M25/26
95	D	300000K1.05	SDM-B21/N11
96	B	201003K5.05	GP Reactor Theory Text, Chapter 5
97	C	233000A2.03	SDM-G41, ARI-H13-P970-1 (D3)
98	C	256001K2.01	SDM-N21/61
99	C	29002K4.02	SDM-B13
100	D	G 2.1.32	IOI-3

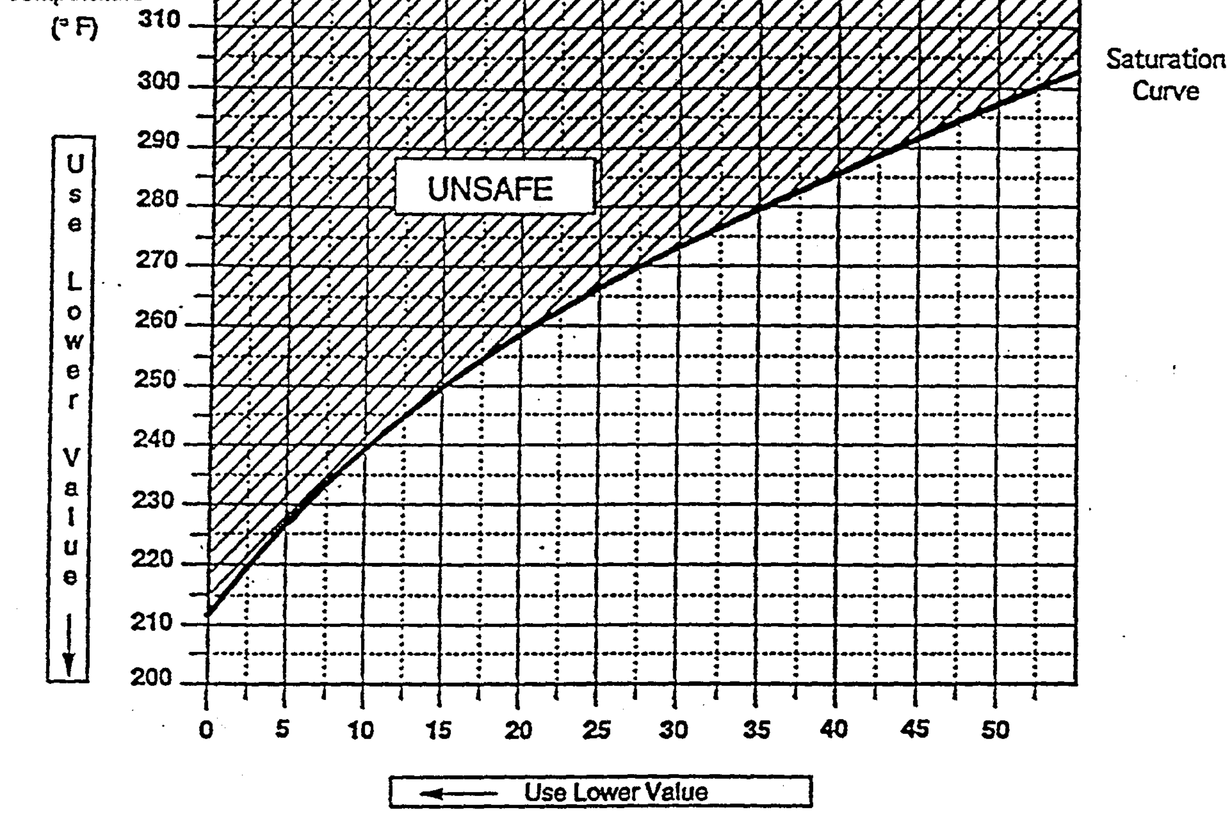
U.S. Nuclear Regulatory Commission
Written Examination January 2001
Reactor Operator
Reference Materials

1. Simplified FDW System Diagram
2. PEI-SPI Supplement Figure 1a



Feedwater System Simplified Diagram

Figure 1a
RPV Low Pressure Saturation Temperature



Directions:

1.0 **IDENTIFY** RPV Pressure on the horizontal axis of the figure.

2.0 <u>**IF**</u> the value falls between marked lines on the figure,
<u>**THEN**</u> **USE** the <u>lower</u> value.

3.0 **IDENTIFY** the point formed by the intersection of the RPV Pressure and the Saturation Curve to determine RPV Saturation Temperature.

4.0 <u>**IF**</u> the value falls between marked lines on the figure,
<u>**THEN**</u> **USE** the <u>lower</u> value.

Master SRO Exam

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information	
Name: MASTER SRO EXAMINATION	Region: I / II / **III** / IV
Date:	Facility/Unit: **Perry**
License Level: RO / **SRO**	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. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	100.00	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

QUESTION 1

In MODES 1, 2, and 3, compliance with LCO 3.6.2.1, Suppression Pool Average Temperature, and LCO 3.6.2.3, RHR Suppression Pool Cooling System, is required to ________________________.

A. ensure the Drywell peak temperature and pressure remain below design limits following a DBA-LOCA.

B. ensure the Primary Containment peak temperature and pressure remain below design limits following a DBA-LOCA.

C. maintain a sufficient amount of cooled water to condense the steam from the SRV quenchers or RCIC turbine exhaust line during all modes of plant operation.

D. maintain an adequate suppression pool heat sink volume to ensure Primary Containment pressure and temperature remain within design limits.

QUESTION 2

Which one of the following events will cause fuel temperature to act first to change the reactivity addition to the core?

A. A control rod drop during reactor power operation.

B. The tripping of the Main Turbine at 45% reactor power.

C. A safety relief valve opening during reactor power operation.

D. The loss of a feedwater heater (extraction steam isolated) during reactor power operation.

QUESTION 3

A CAUTION in the Plant Emergency Instructions states, " Operation of LPCS or RHR with suction from the Suppression Pool and Suppression Pool level less than 5.75 feet may result in equipment damage."

Select the statement below that describes the application of this limit.

A. Equipment damage is expected to occur immediately.
A 10CFR 50.54(x) determination must be made prior to operating a pump at a Suppression Pool level less than 5.75 feet.

B. Equipment damage is expected to occur immediately.
However, a 10CFR 50.54(x) determination is NOT necessary prior to operating a pump at a Suppression Pool level less than 5.75 feet.

C. Equipment damage is NOT expected to occur immediately.
However, a 10CFR 50.54(x) determination must be made prior to operating a pump at a Suppression Pool level less than 5.75 feet.

D. Equipment damage is NOT expected to occur immediately.
A 10CFR 50.54(x) determination is NOT necessary prior to operating a pump at a Suppression Pool level less than 5.75 feet.

QUESTION 4

The plant is operating at 5% power and a test of RCIC has just been completed. In accordance with the power ascension schedule, a test of the Safety Relief Valves (SRVs) is now in progress. Testing is in progress when the Control Room Operators notice that Suppression Pool average water temperature has inadvertently increased to 106 °F.

What are the MINIMUM actions required?

A. Enter PEI-T23, Containment Control and suspend testing of SRVs.

B. Enter LCO 3.6.2.1, Suppression Pool Average Temperature and suspend testing of SRVs.

C. Enter LCO 3.6.2.1, Suppression Pool Average Temperature and place the Reactor Mode Switch in the SHUTDOWN position.

D. Enter PEI-T23, Containment Control and LCO 3.6.2.1, Suppression Pool Average Temperature and suspend testing of SRVs.

QUESTION 5

The Unit Supervisor ordered the Control Room to be abandoned due to toxic fumes. The plant is being operated from the Remote Shutdown Panel (C61-P001). Plant cool down is in progress with preparations being made to place RHR Loop 'A' in the Shutdown Cooling mode of operation. The Unit Supervisor directs the operator to verify that reactor pressure is less than 135 psig prior to placing shutdown cooling into operation.

What is the reason for this direction to ensure reactor pressure is less than 135 psig?

A. The Shutdown Cooling system will isolate if reactor pressure exceeds 135 psig.

B. The RHR pump seals could be damaged if reactor pressure exceeds 135 psig with shutdown cooling in operation.

C. The relief valve on the RHR pump suction line is designed to lift at 135 psig.

D. The relief valve on the Shutdown Cooling suction line is designed to lift at 135 psig.

QUESTION 6

A plant transient has resulted in a reactor scram.

Plant conditions are as follows:

- No systems can be aligned for injection.
- MSIVs are closed.
- Containment pressure is 2.1 psig.
- Drywell pressure is 7.5 psig.
- All control rods are fully inserted.

Given these plant conditions, which one of the following conditions assures adequate core cooling?

A. Reactor water level is unknown, no SRVs are open, and reactor pressure is 265 psig.

B. Reactor water level is unknown, 8 SRVs are open, and reactor pressure is 45 psig.

C. Reactor water level is -40 inches, no SRVs are open, and reactor pressure is 800 psig

D. Reactor water level is -40 inches, 5 SRVs are open, and reactor pressure is 25 psig.

QUESTION 7

A plant event is in progress. PEI-N11, Containment Leakage Control, has been entered.

In the Auxiliary Building, 574', area radiation monitor D21-K112, AB EL 574' East, indicates 4.2 Rem/hr.

The Maximum Safe Operating Condition Value for this area is 4.0 Rem/hr. The Unit Supervisor directs that the reactor be shutdown even though **NO** primary system is discharging into the area.

What is the bases for the Unit Supervisor's decision to shutdown the reactor?

A. Systems required to assure adequate core cooling are required to be isolated.

B. Two or more areas have exceeded their Maximum Safe Operating Conditions Value for Area Radiation; therefore, a direct threat to continued safe operation exists.

C. One area has exceeded its Maximum Safe Operating Conditions Value for Area Radiation; therefore, a direct threat to personnel safety exists.

D. Area radiation levels of this magnitude prohibit personnel access to the Auxiliary Building that may be required to support operation of systems required to maintain the reactor shutdown.

QUESTION 8

A plant startup is in progress with reactor power at 5%. The Unit Supervisor has been directed to raise reactor power to 20%. The on-shift Chemistry Technician reports the following results for the SLC Storage Tank sample:

- Net Tank Volume 4475 gallons
- Solution Concentration-WT % Boron 2.8%

Select the correct response for the indicated conditions.

Technical Specification Section 3.1 is provided for reference.

A. The SLC System is OPERABLE. Reactor startup to 20% power can continue.

B. Restore at least one SLC subsystem to OPERABLE within 8 hours or be in Hot Shutdown within the next 12 hours. Reactor startup to 20% power can continue.

C. Restore at least one SLC subsystem to OPERABLE within 7 days or be in Hot Shutdown within the next 12 hours. Reactor startup to 20% power cannot continue.

D. Restore at least one SLC subsystem to OPERABLE within 8 hours or be in Hot Shutdown within the next 12 hours. Reactor startup to 20% power cannot continue.

QUESTION 9

MCC-EF1A07, which supplies power to some RCIC components, has been lost. Shortly thereafter, a Loss of Coolant Accident (LOCA) occurred and RPV water level decreased below the RCIC initiation setpoint.

A step in PEI-B13, RPV Control (Non-ATWS) states: "Initiate any of the following which should have initiated: RCIC."

RCIC did not initiate.

Given these plant conditions, should the Unit Supervisor direct that the RCIC System be manually initiated from the Control Room?

A. Yes; assuming all RCIC valves are in their normal standby lineup.

B. Yes; all RCIC valves required to be open for RCIC to inject are DC powered.

C. No; RCIC cannot be lined up for injection from the Control Room if the RCIC Turbine Steam Supply Valve, E51-F045, is closed.

D. No; RCIC cannot be lined up for injection from the Control Room under any circumstances with a loss of AC power.

QUESTION 10

The plant is in MODE 1. The Division 1 Diesel Generator is in Secured Status. The Division 2 and 3 Diesel Generators are in Standby Readiness.

The on-shift Fire Protection Technician reports the following information for CO_2 storage tank 0P54-A008 for the Diesel Generator Rooms:

- Tank pressure 300 psig
- Tank volume 2000 lbs

Which one of the following statements is correct concerning 0P54-A008?

PAP-1914, Attachment 4 is provided for reference.

A. 0P54-A008 is OPERABLE because tank pressure and tank volume exceed the minimum requirements.

B. 0P54-A008 is inoperable because tank volume is less than the minimum tank storage volume requirement.

C. 0P54-A008 is inoperable because tank pressure is greater than the minimum tank storage pressure requirement.

D. 0P54-A008 is <u>not</u> required to be OPERABLE because the Division 1 Diesel Generator is in Secured Status.

QUESTION 11

Which one of the following statements describes when Perry Technical Specification LCO 3.0.4 would allow a MODE change from MODE 2 to MODE 1?

A. When the ACTION(S) of the LCO not met permit continued operation of the plant for 30 days.

B. When the ACTION(S) of the LCO not met permit continued operation of the plant for an unlimited period of time.

C. When compliance with the associated ACTION(S) of the LCO not met would place the plant back into MODE 2.

D. When compliance with the associated ACTION(S) of the Operational Requirement (OR) not met would place the plant back into MODE 2.

QUESTION 12

The plant is operating during an emergency. The Operations crew determines that conditions are such that there is no appropriate action to be taken which would be in compliance with the Perry Operating License.

Whose permission, at a minimum, is required to take reasonable action(s) to maintain the plant in a safe condition **AND** when must the NRC be notified of such reasonable action(s)?

A. Operations Shift Supervisor; notify the NRC within one (1) hour.

B. NRC Resident Inspector; notify the NRC within one (1) hour.

C. Operations Manager; notify the NRC within four (4) hours.

D. Licensed Supervising Operator; notify the NRC within four (4) hours.

QUESTION 13

The plant is operating at 100% reactor power when annunciators MAIN TURB & FDW TRIP RCIC/L8 and RPS RX LEVEL HI L8 on panel H13-P680 alarm. All Reactor Narrow Range Level meters indicate that reactor water level is 225 inches.

The plant continues to operate at 100% reactor power.

As the Unit Supervisor, which one of the following directions should be given to the Operator-at-the-Controls?

A. Take manual control of feedwater flow and slowly return reactor water level to its normal band, and then commence a normal plant shutdown.

B. Manually trip the Main Turbine and carryout the Immediate Actions of ONI-N32, Turbine and/or Generator Trip.

C. Manually scram the reactor and carryout the Immediate Actions of ONI-C71-1, Reactor Scram.

D. Enter PEI-B13, RPV Control (Non-ATWS) and manually scram the reactor.

QUESTION 14

Per PAP-1105, Surveillance Test Control, the Unit Supervisor, depending on plant conditions and amount of time available, can permit Contingent SVIs to be performed without the use of a Data Package Cover Sheet (DPCS).

Which one of the following is **NOT** a guideline that the must be adhered to if a Contingent SVI is to be performed without a DPCS?

A. A DPCS is properly completed following performance of the SVI.

B. An updated or current working copy of the SVI is used to perform the SVI.

C. A Plant Narrative Log entry is made annotating the SVI number, the sections of the SVI actually performed, results, and the name(s) of the test performer(s) upon completion of the SVI.

D. The Supervising Operator signs the Plant Narrative Log entry, acknowledging the completion of the SVI and approval of and concurrence with the results.

U.S. NUCLEAR REGULATORY COMMISSION
WRITTEN EXAMINATION JANUARY 2001
SENIOR REACTOR OPERATOR

QUESTION 15

A refueling outage is in progress. Currently, the reactor pressure vessel has been defueled in preparation for in-vessel work.

Select the condition that would require continuous communication between the Control Room and the Refuel Floor.

A. Replacement of a fuel support piece from the Vessel Platform.

B. Replacement of an entire LPRM assembly from the Auxiliary Platform.

C. Replacement of a Feedwater Sparger nozzle while standing on the top guide.

D. Replacement of the RPV vessel head O-rings from the Vessel Platform.

QUESTION 16

During Day 26 of a refueling outage, the plant remains in a shutdown condition with reactor temperature and pressure being maintained at 150 °F and 0 psig respectively.

The RPV has been re-assembled. Refueling personnel are moving spent fuel bundles from the Containment to the FHB Spent Fuel Storage Pool with the Inclined Fuel Transfer System (IFTS).

What are the requirements for Primary Containment and Fuel Handling Building integrity during spent fuel movement?

A. Primary Containment integrity <u>and</u> Fuel Handling Building integrity are required.

B. Primary Containment integrity is required and Fuel Handling Building integrity is NOT required.

C. Primary Containment integrity is NOT required and Fuel Handling Building integrity is required.

D. Primary Containment integrity <u>and</u> Fuel Handling Building integrity are NOT required.

QUESTION 17

In accordance with PAP-0905, Work Order Process, who may initiate the Troubleshooting Log in order to avert an imminent plant shutdown?

A. Responsible System Engineer

B. Shift Supervisor

C. Unit Supervisor

D. Operations Manager

QUESTION 18

The Unit Supervisor has authorized tags to be cleared for a Clearance on the Main Steam System (B21). Numerous tags on manual valves are located in the Aux. Steam Tunnel in a Level 1 Locked High Radiation Area.

Which one of the following describes the tag removal independent verification requirements for the Main Steam System Clearance?

Independent Verification ____________________________.

A. can be waived for both ALARA <u>and</u> personnel safety concerns.

B. can be waived for ALARA concerns but <u>not</u> for personnel safety concerns.

C. can be waived for personnel safety concerns but <u>not</u> for ALARA concerns.

D. <u>cannot</u> be waived for both ALARA <u>and</u> personnel safety concerns.

QUESTION 19

As the Shift Supervisor, you may authorize radiological work activities to occur without an approved Radiation Work Permit (RWP) during an urgent situation provided that ______________________________.

A. the entry is into a High Radiation Area, Level 1 Locked High Radiation Area, or a Level 2 Locked High Radiation Area.

B. the entry is into a Very High Radiation Area (VHRA).

C. authorization is also granted by the Plant Manager.

D. authorization is for access to High Radiation Areas only by the Non-Licensed Operators.

QUESTION 20

Access to the IFTS Valve Room in the Containment is required for surveillance purposes.

The IFTS Valve Room is controlled by a locking device providing for ____________.

A. one uniquely keyed lock whose key is maintained by the Radiation Protection Section.

B. one uniquely keyed lock whose key is maintained by the Control Room Shift Supervisor.

C. two uniquely keyed locks, of which one key is maintained by the Radiation Protection Section and the other key is maintained by the Control Room Shift Supervisor.

D. two uniquely keyed locks, of which one key is maintained by the Radiation Protection Section and the other key is maintained by the Control Room Unit Supervisor.

QUESTION 21

The plant has entered a Site Area Emergency. No emergency facilities are operational. Two PPOs are in the plant performing PEI-SPI actions as directed by the Control Room.

To provide accountability of the two PPOs in the plant, the Shift Supervisor must _____.

A. direct the PPOs in the plant to promptly return to the Unit 2 Control Room and use the designated Accountability Card Reader.

B. provide the names and badge numbers of the PPOs in the plant to the OSC Coordinator, when the OSC becomes operational.

C. provide the names and badge numbers of the PPOs in the plant to the Security Shift Supervisor.

D. complete the Personnel Accountability Checklist for the PPOs in the plant and forward to the Central Alarm Station (CAS).

QUESTION 22

The plant is operating at 100% reactor power when alarm ANN PWR SUPPLY FAIL is received on panel H13-P680.

Additional indications on panel H13-P680 include:

- RECIRC A PUMP DIFF PR, 1B33-R605A indicates downscale
- RECIRC B PUMP DIFF PR, 1B33-R605B indicates downscale
- RFPT 'B' speed has increased to the high-speed stop
- RFPT 'A' speed has decreased after initially increasing

Which one of the following sets of Immediate Operator Actions should the Unit Supervisor verify are performed by the Supervising Operators?

A. Select NARROW RANGE LEVEL CH 'A' and transfer control of RFPT 'A' to the Manual Speed Control Dial.

B. Select NARROW RANGE LEVEL CH 'A' and transfer control of RFPT 'B' to the Manual Speed Control Dial.

C. Select NARROW RANGE LEVEL CH 'B' and transfer control of RFPT 'A' to the Manual Speed Control Dial.

D. Select NARROW RANGE LEVEL CH 'B' and transfer control of RFPT 'B' to the Manual Speed Control Dial.

U.S. NUCLEAR REGULATORY COMMISSION
WRITTEN EXAMINATION JANUARY 2001
SENIOR REACTOR OPERATOR

QUESTION 23

PEI-B13, RPV Control (ATWS) has been entered due to an ATWS with an MSIV isolation. The following plant conditions exist:

- Reactor power 10%
- Reactor pressure 800 psig
- Reactor water level + 115 inches
- Suppression Pool temperature 113 °F
- Drywell pressure 1.1 psig
- Number of SRVs open 2

Based on these plant conditions, which one of the following reactor water level bands is required by PEI-B13, RPV Control (ATWS) in order to lower reactor power?

A. Restore ~~and maintain~~ reactor water level between +185 and +215 inches.

B. Maintain reactor water level between –25 and + 215 inches.

C. Maintain reactor water level between –25 inches and +100 inches.

D. Maintain reactor water level between –25 inches and the level to which it was lowered.

Delete AMS 2/13/01

QUESTION 24

An Override step in PEI-B13, RPV Control (Non-ATWS) asks, "Can Suppression Pool temperature be maintained below HCL?" If the response is NO, then the PEI directs the RPV be rapidly depressurized using the Main Turbine Bypass Valves.

According to this Override step, the RPV must be depressurized _______________.

A. when Suppression Pool average temperature equals HCL.

B. when Suppression Pool average temperature exceeds HCL.

C. when any one Suppression Pool temperature indicator exceeds HCL.

D. before Suppression Pool average temperature exceeds HCL.

QUESTION 25

Following a LOCA, the following plant conditions exist:

• RPV water level	35 inches
• Containment pressure	10 psig
• Containment hydrogen concentration	8.0%
• Drywell hydrogen concentration	7.5%
• Hydrogen igniters	Failed to operate in either division

What is the hydrogen control equipment configuration required under these conditions?

PEI-M51/56, Hydrogen Control, flowchart is provided for reference.

A. Combustible Gas Mixing System OPERATING; Hydrogen Recombiners OPERATING.

B. Combustible Gas Mixing System SECURED; Hydrogen Recombiners OPERATING.

C. Combustible Gas Mixing System OPERATING; Hydrogen Recombiners SECURED.

D. Combustible Gas Mixing System SECURED; Hydrogen Recombiners SECURED.

QUESTION 26

An inadvertent High Pressure Core Spray (HPCS) Pump start occurred.

Before the Control Room Operators could take any action for the inadvertent HPCS Pump start, a loss of DC Bus ED-1-C occurred.

Bus EH13 is being powered from the preferred off-site power source.

Which one of the following methods should be directed by the Unit Supervisor in order to shutdown the HPCS Pump?

A. From the Control Room, take the HPCS PUMP, 1E22-C001, control switch to the STOP position.

B. From the Control Room, take the PREFERRED SOURCE BREAKER, EH1303, control switch to the TRIP (open) position.

C. Locally at HPCS PUMP BRKR EH1304 cubicle, depress the MANUAL TRIP pushbutton.

D. Locally at HPCS PUMP BRKR EH1304 cubicle, take the test control switch to the TRIP (open) position.

QUESTION 27

With the plant at power, Instrument Air and Service Air header pressures are slowly decreasing. The following annunciators are received:

- INSTRUMENT AIR HEADER PRESSURE LOW
- PARALLEL IA AIR HEADER PRESSURE LOW
- SERVICE AIR HEADER PRESSURE LOW
- SA/IA XCONN CLOSE IA RECEIVER PRESSURE LOW

SA/IA XCONN VALVE, 1P52-F050, has automatically closed.

Which one of the following conditions will occur due to the low system air pressure?

A. ADS SRVs will be inoperable.

B. MSIVs will be inoperable.

C. SDV Vent and Drain Valves will fail open.

D. CRD drive water flow will increase.

QUESTION 28

The following plant conditions exist:

- An ATWS is in progress
- HPCS injection prevention has been performed per PEI-SPI-5.1
- RPV level is being maintained at + 75 inches

A loss of Bus EH13 occurs.

Which one of the following describes the response of the HPCS Pump breaker?

A. The HPCS Pump breaker remains closed at all times.

B. The HPCS Pump breaker remains open at all times.

C. The HPCS Pump breaker initially opens and, upon re-energization of Bus EH13, re-closes after a 10 second time delay.

D. The HPCS Pump breaker initially opens and, upon re-energization of Bus EH13, re-closes immediately.

QUESTION 29

IMMEDIATELY following a reactor scram from full power, what would be the expected indication observed on the Intermediate Range Monitors?

A. Range 3 due to delayed neutrons dominating from longer-lived delayed neutron precursors.

B. Range 3 due to delayed neutrons dominating from shorter-lived delayed neutron precursors.

C. Range 5 due to delayed neutrons dominating from shorter-lived delayed neutron precursors.

D. Range 5 due to delayed neutrons dominating from longer-lived delayed neutron precursors.

QUESTION 30

The plant is operating at 100% reactor power when all inboard Main Steam Line Isolation Valves inadvertently isolate. The MSIV closure signal to the Reactor Protection System (RPS) failed to scram the reactor

Which one of the following describes the response of the reactor?

Assume **NO** operator action is taken.

A. Reactor power will increase and stabilize at a higher power.
RPV water level will decrease and return to normal level.

B. Reactor power will increase and cause a reactor scram on power.
RPV level will decrease and then stabilize at a higher level.

C. Reactor power will decrease and stabilize at a lower power.
RPV water level will increase and then return to normal level.

D. Reactor power will increase and cause a reactor scram on power.
RPV water level will increase and then return to normal level.

QUESTION 31

The plant was operating at 50% power with both RFPTs on the Master Level Controller when a Feedwater rupture in the Turbine Building caused reactor water level to decrease.

Reactor water level decreased to +80 inches before HPCS and RCIC were able to restore reactor water level to normal.

Which one of the following correctly describes the status of the Recirculation System?

A. Recirculation Pumps are tripped with their flow control valves in their pre-transient positions.

B. Recirculation Pumps are in slow speed with their flow control valves in their pre-transient positions.

C. Recirculation Pumps are tripped with their flow control valves locked up (motion inhibited).

D. Recirculation Pumps are in slow speed with their flow control valves locked up (motion inhibited).

QUESTION 32

The plant is at 100% power. Since the beginning of shift, Control Room Operators have observed the following Drywell parameter trends:

• Drywell Pressure:	Increasing
• Drywell Average Temperature	Increasing
• Drywell Air Cooler Drain Flow Rate	Increasing
• Drywell Floor Drain Sump Fill Rate	Increasing

Which one of the following could be the cause of these indications?

A. There is an accumulator air leak on an inboard MSIV.

B. There is a cooling coil leak on the lower drywell cooler air handling unit.

C. There is an instrument line leak on a water level condensing chamber.

D. There is an outer seal leak on a Reactor Recirculation Pump.

QUESTION 33

An ATWS is in progress. The following plant conditions exist:

- Reactor power is 25%.
- Reactor pressure is at rated pressure.
- Reactor water level is at +180 inches and stable.

Which one of the following describes the effect of reducing reactor pressure?

A. Reactor power will decrease due to the voiding of the core and remain lower than the original power.

B. Reactor power will initially decrease due to the voiding of the core and then increase due to the lowering moderator temperature.

C. Reactor power will increase due to the collapsing of the voids in the core resulting in increased neutron thermalization.

D. Reactor power will decrease due to the concentration of boron in the core absorbing fast neutrons.

QUESTION 34

Concerning the operation of the MSL & MSIV BYP OTBD ISOL, 1B21-F019, local transfer switches located at MCC EF1A07, select the correct statement.

A. Switches are not active until the control transfer to the Division 1 Remote Shutdown Panel is completed.

B. Switches are not active until the control transfer to the Division 2 Remote Shutdown Panel is completed.

C. Switches are always active and if placed in EMERGENCY, will cause 1B21-F019 to close on an MSIV isolation signal.

D. Switches are always active and if placed in EMERGENCY, will cause 1B21-F019 to close (if open).

QUESTION 35

The Feedwater Leakage Control System has been initiated following a Loss of Coolant Accident when the Feedwater System was no longer required for adequate core cooling.

Which one of the following describes the location where the Feedwater Leakage Control System injects seal water?

A simplified diagram of the Feedwater System is attached.

A. Between the inboard (F559A/B) and the outboard (F032A/B) feedwater check valves.

B. Between the outboard feedwater check valves (F032A/B) and the feedwater header shutoff valves (F065A/B).

C. Through the bonnets of the shutdown cooling to feedwater shutoff valves (E12-F053A/B).

D. Through the bonnets of the feedwater header shutoff valves (F065A/B).

QUESTION 36

Fuel element failure is indicated by increasing plant radiation levels.

Upscale alarms are received on all Main Steam Line Radiation Monitors.

Upscale Trip alarms are received on Main Steam Line Radiation Monitors A and B.

Which one of the following action(s) will automatically occur based on these indications only?

A. Off-Gas Discharge Isolation Valve N64-F632 closes.

B. Reactor Water Sample Isolation Valves B33-F019 and B33-F020 close.

C. Main Steam Line Isolation Valves B21-F022A-D and B21-F028A-D close.

D. Mechanical Vacuum Pump Suction Valves N62-F130A and N62-F130B close.

QUESTION 37

Describe the safety function of the Containment Ventilation Exhaust Radiation Monitor (D17-K609A-D) during a refueling outage.

A. Detect a fuel bundle rupture <u>inside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>off-site</u> dose limits are not exceeded.

B. Detect a fuel bundle rupture <u>outside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>off-site</u> dose limits are not exceeded.

C. Detect a fuel bundle rupture <u>inside</u> Containment which causes the CVDWP (M14) System to isolate to ensure <u>on-site</u> dose limits are not exceeded.

D. Detect a fuel bundle rupture <u>inside</u> Containment which actuates the Containment Evacuation alarm to ensure personnel evacuate Containment.

QUESTION 38

Which one of the following describes the bases for the 'Drywell Pressure-High' function for the Reactor Protection System Instrumentation?

A. To ensure that the Minimum Critical Power Ratio (MCPR) is maintained above the MCPR Safety Limit.

B. To minimize the probability of fuel damage during a break in the Reactor Coolant Pressure Boundary (RCPB).

C. To reduce the amount of energy transferred to the coolant which could challenge the integrity of the Reactor Coolant Pressure Boundary (RCPB).

D. To ensure that sufficient capacity remains in the Scram Discharge Volume to accept the water displaced during control rod insertion from a full scram.

QUESTION 39

Which one of the following describes the bases for Alternate Rod Insertion due to high reactor pressure?

A. ARI reduces the challenge to the integrity of the Reactor Coolant Pressure Boundary.

B. ARI reduces the capability to cool the reactor fuel.

C. ARI reduces unwanted safety relief valve operation resulting in undesired voiding of the core.

D. ARI reduces unwanted safety relief valve operation resulting in undesired heatup of the Suppression Pool.

QUESTION 40

The following plant conditions exist:

- An ATWS is in progress
- MSIVs are isolated
- SRVs are being used to control reactor pressure

As Suppression Pool temperature increases, ECCS pump NPSH ________________.

A. <u>increases</u> resulting in the potential for ECCS pump cavitation.

B. <u>increases</u> resulting in the potential for pump runout.

C. <u>decreases</u> resulting in the potential for ECCS pump cavitation.

D. <u>decreases</u> resulting in the potential for pump runout.

QUESTION 41

The plant is shutting down to perform maintenance. Because of fuel cladding leaks, plant management has decided not to scram the reactor, but rather, to conduct a controlled insertion of control rods to minimize the potential increase in radioactivity release from the fuel.

As rod insertion progresses, the reactor goes subcritical. The Control Room Operator stops insertion of control rods with the intent to slow down the reactor depressurization and cooldown. Practically all of the heat generation at this point is from decay heat.

Thirty (30) minutes later the Control Room Operator notes that IRM flux levels are increasing on a long, stable positive reactor period.

Which one of the following describes the next action the Control Room Operator should take?

A. Insert control rods to a position that causes reactor period to be 60 – 150 seconds.

B. Withdraw the next in-sequence control rod to maintain the power rise to reach the point of adding heat.

C. Manually scram the reactor to terminate the power rise.

D. Monitor IRMs and range them according to the power increase to keep them on-scale.

QUESTION 42

A Loss of Coolant Accident (LOCA) has occurred. From the conditions below, select the set of conditions that would preclude the use of all ranges of RPV Water Level Instrumentation to determine reactor water level.

PEI-SPI Supplement Figure 1a is provided for reference.

A. Reactor Pressure 50 psig
Drywell Temperature 296 °F
Containment Temperature 205 °F

B. Reactor Pressure 25 psig
Drywell Temperature 260 °F
Containment Temperature 251 °F

C. Reactor Pressure 4 psig
Drywell Temperature 212 °F
Containment Temperature 212 °F

D. Reactor Pressure 0 psig
Drywell Temperature 190 °F
Containment Temperature 145 °F

QUESTION 43

Plant Emergency Instruction PEI-B13, RPV Control (ATWS) specifies that, under certain conditions, injection into the RPV be terminated and prevented except for boron and CRD.

The reason that injection into the RPV is terminated and prevented is to __________.

A. decrease the suppression pool heatup rate.

B. decrease the rate and magnitude of power oscillations.

C. increase the thermal driving head.

D. increase the core inlet subcooling.

QUESTION 44

Following a turbine trip at 40% reactor power, reactor pressure spiked to 1095 psig and then immediately decreased to 960 psig. The reactor did NOT scram.

Assuming control rods did NOT insert but all other systems performed as designed, what plant conditions would be observed 10 seconds after the turbine trip?

A. Feedwater flow controllers in MANUAL, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B CLOSED.

B. Feedwater flow controllers in MANUAL, Reactor Recirculation pumps TRIPPED off with pump breakers CB 3A/B and CB 4A/B OPEN.

C. Feedwater flow controllers in AUTO, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B OPEN.

D. Feedwater flow controllers in AUTO, Reactor Recirculation pumps operating in SLOW speed with pump breakers CB 3A/B and CB 4A/B CLOSED.

QUESTION 45

A Loss of Coolant Accident (LOCA) occurred and Containment pressure has increased above the Primary Containment Limit (PCL). PEI-T23, Containment Control, directs the Control Room Operators to vent Containment.

Which one of the'following is the bases for the requirement to vent Containment?

If Containment pressure exceeds PCL, the ______________________.

A. Containment pressure can no longer be determined since all Containment pressure indicators are off-scale high.

B. design pressure limit for the Containment Equipment Hatch has been exceeded.

C. Containment Vent Valves <u>cannot</u> be opened and closed.

D. RPV Vent Valves <u>cannot</u> be opened.

QUESTION 46

During a Main Turbine trip, reactor pressure peaked at 1115 psig. The reactor scrammed and reactor pressure is now 900 psig.

Select the item that describes the operation of the Safety Relief Valves (SRVs) during and following this transient.

Assume **NO** operator action is taken with respect to the SRVs.

A. One SRV opened and remained open until pressure decreased to 936 psig. If pressure increases to 1100 psig, one SRV will re-open.

B. Ten (10) SRVs opened. One SRV remained open until pressure decreased to 926 psig. If pressure increases to 1100 psig, two SRVs will re-open.

C. Ten (10) SRVs opened. Ten (10) SRVs remained open until pressure decreased to 936 psig. If pressure increases to 1100 psig, one SRV will re-open.

D. Nineteen (19) SRVs opened. Ten (10) SRVs remained open until pressure decreased to 936 psig. If pressure increases 1100 psig, two SRVs will re-open

QUESTION 47

Plant Conditions are as follows:

• Reactor is shutdown	all rods are in
• Reactor pressure	600 psig
• Reactor water level	210 inches
• Suppression Pool temperature	115°F
• Suppression Pool level	14.0 feet
• Drywell pressure	1.1 psig
• Containment pressure	0.8 psig
• RHR Loops A and B	Suppression Pool Cooling mode

What action is required to be performed?

A. Reduce reactor pressure to provide a wider operating margin to HCL.

B. Spray Containment.

C. Emergency Depressurize.

D. These conditions require no further actions be initiated.

QUESTION 48

The plant is operating at 100% reactor power when one Reactor Recirculation pump trips. All systems respond as designed to this event. How will RPV water level initially respond and what is the reason for this response?

RPV water level will ________________________.

A. increase due to the displacement of water into the downcomer by increased steam voiding.

B. decrease due to the lack of coolant velocity to sweep voids into the steam separator.

C. increase due to the continuing addition of feedwater at 100% rated feedwater flow.

D. decrease due to the runback of feedwater pumps to minimum speed.

QUESTION 49

The plant is operating at 35% reactor power when the Control Room operators observe Main Condenser vacuum is decreasing (increasing absolute pressure) and Off-Gas System after-filter discharge flowrate is increasing.

Which one of the following could be the cause of these indications?

A. Main Steam to Steam Jet Air Ejector supply pressure is less than 125 psig.

B. Steam Seal header pressure is 1.0 psig.

C. Steam Seal header pressure is 4.0 psig.

D. Steam Seal exhaust vacuum is greater than 12.0 inches water vacuum.

QUESTION 50

The Division 2 Diesel Generator (DG) received an automatic start signal due to a bus undervoltage condition.

Five (5) seconds later the undervoltage condition still exists, starting air pressure has decreased to 150 psig, and DG speed is 100 rpm.

Which one of the following describes the current status of the Division 2 DG?

The Division 2 DG starting air valves are ___________________.

A. open and the Division 2 DG will continue to roll for another 10 seconds unless its speed reaches 441 rpm.

B. open and the Division 2 DG will continue to roll for another 5 seconds unless its speed reaches 200 rpm.

C. closed because starting air pressure has decreased to 150 psig.

D. closed and the Division 2 DG has successfully started.

QUESTION 51

The plant is currently operating at 25% reactor power.

Which one of the following describes the response of the RC&IS System if the Main Turbine were to trip with no reactor scram?

RC&IS will ______________________.

A. implement the constraints of the Rod Pattern Controller, and depending on the control rod pattern, initiate Insert and/or Withdraw blocks.

B. implement the constraints of the Rod Withdrawal Limiter allowing control rods to be withdrawn up to 4 notches.

C. implement no constraints on control rod motion since reactor power is at the Low Power Alarm Point between the Rod Pattern Controller and the Rod Withdrawal Limiter.

D. implement the constraints of the Rod Withdrawal Limiter allowing control rods to be withdrawn up to 2 notches.

QUESTION 52

The plant was operating at 20% reactor power when a malfunction of the Feedwater Level Control System (C34) caused RPV water level to increase to 224 inches before Control Room Operators could restore RPV level back to normal.

Which one of the following is the plant response to this event?

A. There would be no noticeable plant response at this reactor power level.

B. Reactor power initially increased but then reactor power and reactor water level returned to normal after approximately one minute.

C. The Main Turbine tripped but the reactor did not scram.

D. The Main Turbine tripped and the reactor scrammed.

QUESTION 53

A high containment temperature has occurred and the Control Room Operators have entered PEI-T23, Containment Control. The PEI directs the Control Room Operators to "operate all available containment cooling".

Plant conditions are as follows:

- No BOP isolation has occurred.
- CVCW Chiller 'A' is operating.
- CVCW Chill Water Pump 'A' is operating.
- Containment Vessel Cooling Fans 'A', 'C', 'D', and 'F' are operating.

What action can be taken to "operate all available containment cooling"?

A. Start CVCW Chiller 'C'.

B. Start CVCW Chill Water Pump 'C'.

C. Start Containment Vessel Cooling Fans 'B' and 'E'.

D. Manually close the CVCW three-way valve to isolate any chill water bypass flow around the Containment Vessel Cooling Air Handling Unit cooling coils.

QUESTION 54

A cold reactor startup is in progress. Drywell temperature is slowly increasing as reactor heatup and pressurization is being performed.

Which one of the following describes how the Drywell Cooling System (M13) responds during normal plant operation?

Assume the Drywell Cooling System is in normal operation.

A. The standby Lower Drywell Cooling Fan will automatically start when Reactor Vessel Support Skirt area temperature exceeds 120 °F.

B. The Lower Drywell Cooler NCC Bypass Valve, P43-F365, will re-position to cause more cooling water to flow through the in-service Lower Drywell Air Handling Unit cooling coil.

C. The Lower Drywell Cooler NCC Bypass Valve, P43-F365, will re-position to cause less cooling water to flow through the in-service Lower Drywell Air Handling Unit cooling coil.

D. The Lower Drywell Cooler 3-Way NCC Supply Valve, P43-F025, will throttle open to provide additional cooling water flow through the in-service Lower Drywell Air Handling Unit cooling coil.

QUESTION 55

The plant is operating at 90% reactor power when both Nuclear Closed Cooling System Heat Exchanger temperature control valves malfunction. Plant operators are unable to operate the temperature control valves manually. The standby NCC Heat Exchanger is drained for maintenance. As a result, alarm NCC HX OUTLET TEMP HIGH is received and NCC Heat Exchanger outlet temperature is 93 °F and slowly increasing.

Which one of the following operator actions is required to be performed?

A. Start an additional Service Water Pump.

B. Enter PEI-T23, Containment Control.

C. Perform a rapid manual shutdown of the Reactor Water Cleanup System.

D. Close both Recirculation Loop A and B Flow Control Valves until total core flow is 58 Mlbm/hr.

QUESTION 56

The plant is operating at power. The Motor Feed Pump is tagged out due to a motor ground. Reactor Feed Pump 'B' has just been removed from service for corrective maintenance.

What is the current operating guideline for reactor power based on the present status of the Feedwater System?

A. 63%

B. 66%

C. 68%

D. 71%

QUESTION 57

The following plant conditions exist:

- The reactor is in MODE 4
- RHR Loop A is in the Shutdown Cooling mode
- RHR Loop B is in the Suppression Pool Cooling mode

A valid RPV Level 1 reactor water level condition occurs.

Which one of the following describes the automatic response of the RHR system?

A. RHR Pumps A and B trip; RHR Loop B shifts to the LPCI mode; RHR Pump B restarts.

B. RHR Pump A trips; RHR Loop B continues to operate in the Suppression Pool Cooling mode.

C. RHR Pump A trips; RHR Loop B realigns to the LPCI mode.

D. RHR Pump A continues to operate in the Shutdown Cooling mode; RHR Loop B realigns to the LPCI mode.

QUESTION 58

IOI-12, Maintaining Cold Shutdown, specifies that when Reactor Recirculation Pumps are not operating, reactor water level should be maintained greater than 250 inches on the Reactor Shutdown Range Level.

Maintaining reactor water level in a range of 250 to 260 inches on the Reactor Shutdown Range Level will _______________.

A. prevent undetected boiling locally in the core.

B. provide adequate NPSH for the RHR pumps during shutdown cooling operation.

C. provide sufficient water volume to prevent a loss of shutdown cooling due to a low reactor water level isolation.

D. provide sufficient water volume to flood the Main Steam lines.

QUESTION 59

A reactor startup is in progress with reactor pressure at 650 psig when CRD Pump 'A' trips. While preparing to start CRD Pump 'B', four HCU accumulator faults are received; three of which are associated with withdrawn control rods. CRD charging water pressure is 1375 psig, as read on 1C11-R601, CRD PRESSURE CHARGING WATER.

Which one of the following operator actions is required to be performed?

A. The Reactor Mode Switch must be placed in SHUTDOWN immediately.

B. The Reactor Mode Switch must be placed in SHUTDOWN if the conditions above still exist after 20 minutes.

C. A fast reactor shutdown must be commenced immediately.

D. A fast reactor shutdown must be commenced if the conditions above still exist after 20 minutes.

QUESTION 60

PEI-T23, Containment Control, directs the operator to Emergency Depressurize if Suppression Pool level cannot be maintained below 24.5 feet.

Why does the PEI direct emergency depressurization at this point?

Above this Suppression Pool level, ______________________.

A. the operation of SRVs may cause failure of the Containment.

B. the pressure rise in the Containment could cause overflow of the weir wall.

C. boron would be diluted below the Hot Shutdown Boron Weight if boron was being injected.

D. the NPSH for pumps taking suction on the Suppression Pool would be insufficient.

QUESTION 61

The plant is operating at 100% reactor power. AEGTS Train 'A' is in operation with its associated Annulus Differential Pressure Controller in the AUTO mode.

Which one of the following describes the response of the AEGTS System if the absolute pressure in the Annulus decreases below the desired pressure?

A. The standby AEGTS Train 'B' will automatically start and restore Annulus pressure to the desired value.

B. The associated Annulus Differential Pressure Controller setpoint will automatically increase to match the higher Annulus pressure.

C. The AEGTS Train 'A' exhaust damper will throttle open while the recirculation damper will throttle close until the Annulus pressure is restored to the desired value.

D. The AEGTS Train 'A' exhaust damper will throttle close while the recirculation damper will throttle open until Annulus pressure is restored to the desired value.

QUESTION 62

PEI-N11, Containment Leakage Control, has been entered on Area Water Level. It is determined that a primary system is discharging into the affected area. PEI-B13, RPV Control (Non-ATWS) is required to be entered and executed concurrently with PEI-N11.

Which one of the following is the reason for entering PEI-B13, RPV Control (Non-ATWS)?

A. A reduction in RPV water level will effect a decrease in the flow of water into the affected area in order to maintain personnel access in the affected area.

B. A reduction in RPV water level will effect a decrease in the flow of water into the affected area in order to maintain equipment qualifications in the affected area.

C. A reduction in RPV pressure will effect a decrease in the flow of water into the affected area in order to maintain personnel access in the affected area.

D. A reduction in RPV pressure will effect a decrease in the flow of water into the affected area in order to maintain equipment qualifications in the affected area.

QUESTION 63

A fire exists in Reactor Recirculation Pump 'A'. The fire was reported at 1358. The CNTMT CO2 SUPPLY OUTBOARD ISOL VALVE, 1P54-F340, was opened by the Control Room Operators at 1440.

Which one of the following describes the current status of the CO2 system?

CO_2 for the Reactor Recirculation Pump fire was ____________________.

A. automatically released into the Drywell and was discharged for the required amount of time.

B. automatically released into the Drywell and was not discharged for the required amount of time.

C. not automatically released into the Drywell; therefore, the CO_2 System will need to be manually discharged.

D. not automatically released into the Drywell; therefore, a Drywell entry will be required to suppress the fire.

QUESTION 64

When a control rod is selected, the Control Room Operator observes that the control rod has an "Insert Block" and "Insert Inhibit" light.

This means that the control rod **cannot** be INSERTED ________________.

A. since this might allow the LHGR or MCPR limit to be exceeded.

B. since this would indicate a control rod block due to a system fault.

C. since this might allow a control rod to have excessive rod worth.

D. since this would indicate a control rod block due to a bypassed control rod position indicator.

QUESTION 65

Given the following conditions for Reactor Recirculation Hydraulic Power Unit 'B':

- Subloop 1 READY, LEAD, OPERATIONAL, PRESSURIZED
- Subloop 2 READY

Alarm 'FCV B HPU NEEDS MAINTENANCE" is received at panel H13-P680.
A Control Room Operator reports that an amber 'OIL WARM' light is illuminated on panel H13-P614 for HPU 'B'.

Which one of the following describes the operational status of HPU 'B'?

A. Subloop 1 and Subloop 2 are in the Maintenance mode.

B. Subloop 1 remains in operation and Subloop 2 remains in Standby.

C. Subloop 1 is in the Maintenance mode and Subloop 2 is in operation but not in LEAD.

D. Subloop 1 is in the Maintenance mode and Subloop 2 is in operation but is in LEAD.

QUESTION 66

The Control Room has been evacuated and plant control has been established at the Division 1 Remote Shutdown Panel.

Select the correct statement concerning operation of the Residual Heat Removal (RHR) System under these conditions.

A. The RHR PUMP A MIN FLOW VALVE, E12-F064A, will auto open when flow is less than 1650 gpm for 8 seconds when the RHR Pump is running.

B. The RHR A TEST VALVE TO SUPR POOL, E12-F024A, will auto close if a LPCI initiation signal is received.

C. The RHR A Pump, E12-C002A. will auto start if a LPCI initiation signal is received.

D. The RHR A TO RADWASTE ISOLATION VALVE, E12-F049, will auto close if drywell pressure is > 1.68 psig.

QUESTION 67

Assume that all required conditions have been met for an Automatic Depressurization AND that depressurization is in progress. If ALL the Low Pressure ECCS pumps trip off, which one of the following describes how the Automatic Depressurization System is affected?

A. Automatic Depressurization will stop and can be recommenced by depressing the ADS Manual Initiation pushbuttons.

B. Automatic Depressurization will stop and can be recommenced by restarting a Low Pressure ECCS pump.

C. Automatic Depressurization will stop and can only be reestablished by manually opening SRVs.

D. Automatic Depressurization will continue without interruption.

QUESTION 68

The High Pressure Core Spray System (HPCS) automatically initiated due to receipt of both the Low Reactor Water Level and High Drywell Pressure signals.

HPCS initiation may/will be reset ______________________.

A. automatically when both initiation signals clear.

B. manually only after both initiation signals clear.

C. manually after the Low Reactor Water Level initiation signal clears.

D. manually after the High Drywell Pressure initiation signal clears.

QUESTION 69

The plant is operating at 35% reactor power. MSIV B21-F022C has a faulty limit switch which is generating a '< 92% open' signal.

Which one of the following MSIVs, if closed, would cause a ½ scram?

A. MSIV B21-F028A

B. MSIV B21-F028B

C. MSIV B21-F022A

D. MSIV B21-F028C

QUESTION 70

The following plant conditions exist:

- Reactor Mode Switch is in STARTUP/STANDBY
- Intermediate Range Monitors (IRM) A, C, D, E, and G are on Range 3; all other IRMs are on Range 2
- Source Range Monitor (SRM) A is reading 0.5 cps
- SRMs B and C are reading 8.3 x 10E4
- SRM D mode switch is in STANDBY

A rod block signal has been generated.

Which one of the following has caused the rod block?

A. SRM Upscale

B. SRM Downscale

C. SRM Detector Wrong Position

D. SRM Inoperable

U.S. NUCLEAR REGULATORY COMMISSION
WRITTEN EXAMINATION JANUARY 2001
REACTOR OPERATOR

QUESTION 71

During reactor power operations, the following plant conditions exist:

- Reactor power — 75%
- Core flow — 70% (73 Mlbm/hr)
- Total Recirculation drive flow — 65% (62 Kgpm)
- Recirculation Loops in operation — Both

Which one of the following is the APRM Upscale Thermal Power Trip Setpoint?

A. 84.3%

B. 104.6%

C. 106.9%

D. 107.7%

Delete
AMS 2/13/01

QUESTION 72

The reactor is shutdown with the following plant conditions:

• Reactor water level	255 inches on Shutdown Range Water Level
• Reactor water temperature	120 degrees F
• Reactor pressure	0 psig
• Drywell temperature	110 degrees F

Which one of the following is correct with respect to these plant conditions?

Actual reactor water level will be ______________________.

A. higher than indicated since the reactor water temperature is LOWER than the calibration conditions for the Shutdown Range Water Level.

B. lower than indicated since the reactor water temperature is LOWER than the calibration conditions for the Shutdown Range Water Level.

C. lower than indicated since the drywell temperature is HIGHER than the calibration conditions for the Shutdown Range Water Level.

D. higher than indicated since the drywell temperature is HIGHER than the calibration conditions for the Shutdown Range Water Level.

QUESTION 73

An electrical transient has occurred and Service Water Pump 'D' is lost. Which bus normally powers this pump?

A. Bus H12

B. Bus XH12

C. Bus XH21

D. Bus XH22

QUESTION 74

A Loss of Coolant Accident (LOCA) has occurred and Drywell pressure is 2.4 psig. LPCS is operating on minimum flow. LPCI A, B, and C have been overridden off since they were not required to maintain adequate core cooling.

Ten minutes after the initial Drywell break, RPV water level suddenly decreased below RPV Level 1. One hundred (100) seconds later, RPV water level was restored above RPV Level 3.

It has now been five minutes since RPV water level was restored above RPV Level 3 and the Unit Supervisor has directed the Supervising Operator to verify the current status of the Automatic Depressurization System (ADS).

NO operator actions were taken with respect to ADS other than resetting annunciators that had cleared.

Which one of the following is the correct annunciator status that the Supervising Operator should expect to observe?

	ADS A 105 SEC TIME DELAY LOGIC INITIATED	ADS A TIMER 90 SEC & RUNNING	ADS A INSTANTANEOUS LOGIC INITIATED
A.	ON	ON	ON
B.	OFF	ON	OFF
C.	ON	OFF	OFF
D.	OFF	OFF	ON

QUESTION 75

A Loss of Coolant Accident (LOCA) has occurred and Drywell pressure is 2.2 psig.

Based on these plant conditions, which one of the following describes the operation of the Containment Vacuum Relief Isolation Valves (M17-F015, F025, F035, and F045)?

If a Containment Vacuum Relief Isolation Valve control switch is ______________.

A. placed in OPEN, then the valve will open regardless of Containment pressure.

B. placed in OPEN, then the valve will open only if Containment pressure is negative.

C. placed in CLOSE, then the valve will close regardless of Containment pressure.

D. placed in CLOSE, then the valve will close only if Containment pressure is negative.

QUESTION 76

Given the following plant conditions:

- Drywell pressure is 1.3 psig
- Reactor water level is +105 inches
- Main condenser vacuum is 25 inches Hg A
- Reactor pressure is 75 psig

Which one of the following describes the system components that isolated based on these plant conditions?

A. RWCU isolation valves, MSIVs and MSL Drain isolation valves, RCIC steam supply line isolation valves

B. MSIVs and MSL Drain isolation valves, NCC Containment & Drywell isolation valves, RWCU isolation valves

C. RCIC steam supply line isolation valves, Drywell Floor Drain Sump & Containment Drain Sump isolation valves, Reactor Water Sample isolation valves

D. Reactor Water Sample isolation valves, RWCU isolation valves, MSIVs and MSL Drain isolation valves

QUESTION 77

The plant has experienced a Loss of Coolant Accident (LOCA). All ECCS have operated as designed EXCEPT one Drywell pressure transmitter (B21-N094A) that supplies a high Drywell pressure signal to the Division 1 Containment Spray initiation logic. The transmitter has failed downscale (indicates zero).

Ten (10) minutes after the LOCA initiation signal, plant conditions are as follows:

- Drywell pressure 1.9 psig
- Containment pressure 4.1 psig

Which one of the following describes the status of the Division 1 Containment Spray System?

Division 1 Containment Spray System has ___________________.

A. automatically initiated.

B. not automatically initiated but will initiate if the CNTMT SPRAY A MANUAL INITIATION pushbutton is armed and depressed.

C. not automatically initiated but will initiate if the CNTMT SPRAY A HI DW PRESS BYP keylock switch is placed in BYPASS.

D. not automatically initiated but will initiate if Containment pressure increases to 7.8 psig.

QUESTION 78

Reactor power is 90%.

Which one of the following describes how an SRV, that was stuck open, is verified closed after its control power fuses have been removed in accordance with ONI-B21-1, SRV Inadvertent Opening/Stuck Open?

A. Reactor pressure increases.

B. Main Generator electrical output increases.

C. Indicated steam flow on the effected steam line decreases.

D. Both SOLENOID STATUS A (B) red indicating lights on P601 are off

QUESTION 79

The plant is operating at 80% reactor power. Reactor Feed Pump Turbine (RFPT) 'A' and 'B' controllers are in Automatic when the High Pressure Core Spray (HPCS) System inadvertently initiates and injects into the RPV.

Which one of the following describes the response of the Feedwater Level Control System?

Total feedwater flow will ____________________.

A. <u>decrease</u>; resulting in a reactor scram on low reactor water level.

B. <u>increase</u>; reactor water level will stabilize at some level slightly lower than the tape set value.

C. <u>decrease</u>; reactor water level will stabilize at the same level as the tape set value.

D. <u>decrease</u>; reactor water level will stabilize at some level slightly higher than the tape set value.

QUESTION 80

A secondary flowpath associated with the Annulus Exhaust Gas Treatment System (AEGTS) allows a purge path to be established.

Which one of the following describes the purpose of this secondary flowpath?

A. To control Drywell temperature during plant heatup.

B. To control Drywell pressure during plant heatup.

C. To control Drywell airborne radiation levels to allow Drywell entry during plant heatup.

D. To control Drywell hydrogen concentration during a Loss of Coolant Accident (LOCA).

QUESTION 81

The plant is at 15% reactor power. The Control Room Operator is in the process of synchronizing the Main Generator to the grid and is ready to close Generator Breaker S-610-PY-TIE. A malfunction occurs in the turbine control system and turbine speed increases to just below the Main Turbine overspeed trip setpoint.

Which one of the following describes how the plant will respond to this event?

A. The synchroscope will turn clockwise at a slower rate; Main Generator output voltage will increase.

B. The synchroscope will turn clockwise at a faster rate; Main Generator output voltage will <u>not</u> change.

C. The synchroscope will turn counter-clockwise at a faster rate; Main Generator output voltage will decrease.

D. The synchroscope will turn counter-clockwise at a faster rate; Main Generator output voltage will <u>not</u> change.

QUESTION 82

The Division 1 Diesel Generator is being operated in parallel with the grid. The Diesel Generator Control Transfer Switch is in the LOCAL position.

Which one of the following describes the response of the Division 1 Diesel Generator if a valid Loss of Coolant Accident (LOCA) signal occurs?

The Division1 Diesel Generator output breaker will ______________________:

A. not trip but the diesel generator trips normally bypassed by a LOCA signal will be bypassed.

B. not trip and the diesel generator trips normally bypassed by a LOCA signal will not be bypassed.

C. trip but the diesel generator trips normally bypassed by a LOCA signal will not be bypassed.

D. trip and the diesel generator trips normally bypassed by a LOCA signal will be bypassed

QUESTION 83

The following plant conditions exist:

- The plant is operating at 100% reactor power
- Feedwater Level Control is on the Master Level Controller with Narrow Range Level Channel 'A' selected
- Narrow Range Level Channel 'A' has failed upscale

The reactor will scram on ________________________.

A. low RPV water level; water level will be restored to approximately 200 inches in accordance with ONI-C71-1, Reactor Scram.

B. low RPV water level; water level will be restored to 185 - 215 inches in accordance with PEI-B13, RPV Control (Non-ATWS).

C. high RPV water level; water level will be restored to approximately 200 inches in accordance with ONI-C71-1, Reactor Scram.

D. high RPV water level; water level will be restored to 185 - 215 inches in accordance with PEI-B13, RPV Control (Non-ATWS).

QUESTION 84

The plant is operating at 100 % reactor power when Recirc Pump Seal Flow Regulator, 1C11-D012A, fails closed.

Which one of the following describes the potential consequence of this condition?

If Reactor Recirculation Pump A operation continues, then the _______________.

A. radioactivity discharged to Radwaste will decrease due to the reduced recirc pump seal purge flow.

B. possibility of recirc pump seal damage will decrease due to the reduced recirc pump seal purge flow.

C. possibility of recirc pump seal damage will increase due to the possible ingestion of dirt from an unclean piping system.

D. possibility of recirc pump seal damage will increase unless the alternate recirc pump seal purge supply from the Condensate Transfer and Storage System can be lined up.

QUESTION 85

Reactor Recirculation Pumps 'A' and 'B' tripped off when reactor water level reached Level 2. Preparations are underway to restart Reactor Recirculation Pump 'A' in order to restore forced circulation through the core.

Which one of the following interlocks must be met for Reactor Recirculation Pump 'A' to successfully start and operate in slow speed?

A. RPV water level is greater than RPV Level 3.

B. Flow Control Valve 'A' actuator (D004A) is full open.

C. Differential temperature between reactor steam dome temperature and Reactor Recirculation Pump 'A' suction temperature is greater than 10 degrees F.

D. Differential temperature between reactor steam dome temperature and Reactor Recirculation Pump 'A' suction temperature is less than 50 degrees F.

QUESTION 86

Given the following conditions:

- The Reactor Water Cleanup System (RWCU) is operating in the normal mode
- The RWCU LD ISOLATION BYPASS Switches (E31-S1A,B) on panels H13-P632 and P642 have been placed in "BYPASS"

Select the expected effect on the RWCU System.

The RWCU System isolation signal on ________________________.

A. high non-regenerative heat exchanger outlet temperature is defeated.

B. initiation of the Standby Liquid Control System (SLC) is defeated.

C. high differential flow rate is defeated.

D. low RPV level (Level 2) is defeated.

QUESTION 87

RHR Loop A has just been placed into the Shutdown Cooling mode of operation using the normal return path. The cooldown rate is excessive. The Unit Supervisor directs you to reduce the cooldown rate.

Which one of the following is the correct action to reduce the cooldown rate?

A. Throttle close the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle open the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

B. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

C. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 2000-7100 gpm.

D. Throttle ESW flow through the RHR Heat Exchanger using RHR A HX'S ESW OUTLET VALVE, P45-F068A.

QUESTION 88

The plant is operating at 100% reactor power. RHR A HX'S BYPASS VALVE, E12-F048A, has failed in the fully open position.

Which mode(s) of RHR Loop A are/is OPERABLE?

A. Suppression Pool Cooling <u>only</u>

B. Low Pressure Coolant Injection <u>only</u>

C. Containment Spray <u>and</u> Suppression Pool Cooling

D. Containment Spray <u>and</u> Low Pressure Coolant Injection

QUESTION 89

Plant conditions are as follows:

- Core offload is in progress
- Refuel Bridge is stationed in the 'cattle chute' (Portable Refueling Shield) between the Reactor Pressure Vessel and the Dryer Storage Pool
- Refuel Bridge grapple is loaded with a new fuel bundle
- Reactor Mode Switch is in the SHUTDOWN position
- All control rods are fully inserted

Which one of the following describes the allowable direction(s) that the Refuel Bridge can travel in (i.e., travel direction will **not** be prevented by an interlock)?

A. The Refuel Bridge can move in either direction.

B. The Refuel Bridge can move towards the Dryer Storage Pool only.

C. The Refuel Bridge can move towards the Reactor Pressure Vessel only.

D. The Refuel Bridge cannot move in either direction

QUESTION 90

A plant transient resulted in a loss of extraction steam to FDW Heaters 5A and 5B. The Immediate Actions of ONI-N36, Loss of Feedwater Heating, have been completed. All remaining FDW Heaters are in operation. Current reactor power is 90%.

What is the Main Generator MWe limitation based on the current FDW Heater lineup?

The Main Generator Mwe Limitation Table from ONI-N36 is provided below for reference.

Heater	Number of Trains Lost	Side of Heater Lost	RFP Steam Supply Main	RFP Steam Supply Extraction	Basis
1 & 2	1	Condensate	1125 MWe	1188 MWe	1
5	2	Extraction	938 MWe	1000 Mwe	2
1 & 2, 3, 5, 6	2 Trains of the same heater	Condensate	563 MWe	625 MWe	3

A. 625 MWe

B. 938 MWe

C. 1000 MWe

D. 1188 MWe

QUESTION 91

The plant is operating at 90% reactor power with both Reactor Feed Pump Turbines (RFPTs) in operation. RFPT 'A' Flow Controller is in MANUAL and RFPT 'B' Flow Controller is in AUTOMATIC. The STARUP FDW PUMP SELECT SWITCH is in the MFP position.

Which one of the following describes the response of the Feedwater System if the speed of RFPT 'A' is slowly decreased?

A. RFPT 'B' flow rate will increase, RFPT 'A' flowrate will decrease, and total feedwater flow will remain the same.

B. RFPT 'B' discharge pressure will slightly decrease, RFPT 'A' discharge pressure will slightly increase, and total feedwater flow will remain the same.

C. RFPT 'B' flow rate will remain the same, RFPT 'A' flowrate will decrease, and total feedwater flow will decrease.

D. RFPT 'B' discharge pressure will slightly increase, RFPT 'A' discharge pressure will slightly decrease, and total feedwater flow will increase.

QUESTION 92

The plant is operating at 100% reactor power when Division 1 DC Bus ED-1-A is lost.

Which one of the following conditions will occur?

A. RCIC automatically initiates.

B. Recirculation Pumps 'A' and 'B' trip off.

C. Division 1 Diesel Generator automatically trips, if running.

D. Alarm window "ANN PWR SUPPLY FAIL" on H13-P680 energizes.

QUESTION 93

The plant is in a refueling outage and the M14 Containment Vessel and Drywell Purge System (CVDWP) is operating in the Refuel mode. Containment Ventilation Exhaust Radiation Monitor D17-K609C is in alarm due to a Downscale indication.

An I&C Technician is troubleshooting D17-K609C when the following alarms are received in the Control Room:

- CNTMT & DW PURGE EXHAUST FAN A FLOW LOW
- CNTMT & DW PURGE EXHAUST FAN B FLOW LOW
- CNTMT PURGE SUPPLY FAN A FLOW LOW
- CNTMT PURGE SUPPLY FAN B FLOW LOW
- DW PURGE SUPPLY FAN A FLOW LOW
- DW PURGE SUPPLY FAN B FLOW LOW

Which one of the following conditions is the probable cause for the current status of the CVDWP System?

A. Either Containment Ventilation Exhaust Radiation Monitor D17-K609A or D17-K609D is in an UPSCALE TRIP (HI-HI) condition due to a refueling accident in Containment.

B. Containment Ventilation Exhaust Radiation Monitor D17-K609B is in an UPSCALE TRIP (HI-HI) condition due to a refueling accident in Containment.

C. The I&C Technician inadvertently placed the MODE SWITCH for D17-K609D to the ZERO position.

D. The I&C Technician inadvertently placed the MODE SWITCH for D17-K609A to the TRIP TEST position.

QUESTION 94

Control Room HVAC and Emergency Recirculation (M25/26) Train 'A' has been manually shifted from the NORM mode to the EMERG RECIRC mode by placing CONT RM HVAC TRAIN A MODE SELECT, M25-S7, in the EMERG RECIRC position.

Assume no other operator actions were performed.

Which one of the following describes the current damper lineup for M25/26 Train 'A'?

	HVAC A OTBD SUPP DAMPER F010A	HVAC A INBD SUPP DAMPER F020B	EMG RCIRC DAMPER A F040A	HVAC A RETURN DAMPER F110A	HVAC A EXHAUST DAMPER F130A
A.	Open	Open	Closed	Open	Closed
B.	Open	Open	Closed	Closed	Open
C.	Closed	Closed	Open	Closed	Closed
D.	Closed	Open	Open	Closed	Closed

QUESTION 95

The plant is operating at 75% reactor power.

MSL B INBD MSIV B21-F022B control switch is in the TEST position. The Control Room Operator depresses the MSL B INBD MSIV TEST pushbutton 1B21H-S3B.

Which one of the following describes the response of MSL B INBD MSIV B21-F022B?

A. Instrument Air bleeds off the bottom portion of the MSIV air cylinder. and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

B. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

C. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

D. Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

QUESTION 96

Which one of the following would be the control rod movement sequence most likely to cause the phenomenon known as the 'reverse power effect'?

A. 1 or 2 notch withdrawal of a deep control rod.

B. 1 or 2 notch withdrawal of a shallow control rod.

C. 1 or 2 notch insertion of a shallow control rod.

D. 10 or 12 notch continuous withdrawal of a shallow control rod.

QUESTION 97

During refueling operations, a FPCC SURGE TANK A LEVEL HI/LO annunciator is received. The Control Room Operator reports that surge tank level is high. The FPCC SURGE TK FILL FROM CST VALVE, G41-F045, is verified closed.

Which one of the following could be the cause of the surge tank high level?

A. Steam Dryer was removed from the Dryer Storage Pool during RPV re-assembly.

B. Emergency makeup valve from the Service Water System (P41) is open or leaking by.

C. Fuel Transfer Tube Drain Tank is pumping down during fuel transfer operations.

D. FPCC flow to the lower fuel pools was increased.

QUESTION 98

Which one of the following load sets will be lost if Bus H11 becomes de-energized?

A. Hotwell Pump A, Hotwell Pump C, and Condensate Booster Pump B

B. Hotwell Pump B, Hotwell Pump C, and Condensate Booster Pump B

C. Hotwell Pump A, Condensate Booster Pump A, and Condensate Booster Pump C

D. Hotwell Pump C, Condensate Booster Pump A, and Condensate Booster Pump C

QUESTION 99

Which one of the following core components acts as a partition to force the majority of coolant and moderator flow into the control rod guide tubes, fuel support pieces, and to the fuel assemblies?

A. Baffle plate

B. Core shroud

C. Core plate

D. Control rod guide tube flow orifices

QUESTION 100

The reactor power 8-hour average limit is 3700 MWt.

What is the basis for this limitation?

A. To prevent exceeding the maximum steady state Main Generator real load.

B. To prevent exceeding the maximum steady state licensed reactor power level.

C. To minimize Recirculation Flow Control Valve (FCV) oscillations.

D. To minimize Main Turbine Control Valve oscillations.

Perry SRO Written Examination
Answer Key

Q#	Ans	K/A	Reference
1	B	295013AK2.01	TS 3.6.2.1, TS 3.6.2.3
2	A	295014AK2.03	GP Reactor Theory, USAR chapter 15.4
3	D	295030EK2.04	PIE bases document
4	B	295013AA2.01	LCO 3.6.2.1, PAP-0528
5	B	29516AA1.08	IOI-11, SDM-E12
6	C,D	295031EA2.04	PEI bases document, PEI-B13
7	B	295033EK3.02	PEI-N11 bases document
8	D	G2.1.12	LCO 3.1.7
9	A	217000K6.01	SDM-E51
10	B	G 2.4.25	PAP-1914, Attachment 4
11	B	G 2.1.22	TS 3.0.4, ORM section 2.0
12	A	G 2.1.2	PAP-0201, 10CFR 50.54 x and y, PAP-1604
13	C,D	G 2.1.7	PEI-B13
14	D	G 2.2.12	PAP-1105
15	C	G 2.2.26	IOI-9
16	D	G 2.2.22	LCO 3.6.10, LCO 3.7.8
17	B	G 2.2.20	PAP-0905
18	B	G 2.2.13	PAP-0205, PAP-1401
19	A	G 2.3.7	HPI-C0005
20	C	G 2.3.1	PAP-0123
21	D	G 2.4.38	EPI-B5
22	B	G 2.4.49	ONI-R42-4
23	DELETE		
24	D	G 2.4.6	PEI-B13, PEI bases document
25	D	500000EK1.01	PEI-M51/56
26	C	295004AA2.04	ON-R42-3, SDM-R10, ONI-E12-1, SDM-R42

Q#	Ans	K/A	Reference
27	B	295019AA2.02	ONI-P52
28	A	295003AK2.04	SDM-E22B
29	C	295006AA1.05	ONI-C71-1, GP Reactor Theory Text
30	B	295007AA2.03	AT&AA Text Chapter 2 (USAR 15B 5.2.2)
31	A	295009AA1.03	SDM-B33, SDM-B21
32	C	295010AA2.05	SDM-E31, Technical Specification 3.4.7
33	B	295015AK1.04	GP Reactor Theory Text, PEI Bases Document
34	D	295016AK2.02	SDM-C61
35	D	G 2.1.28	SDM-N27
36	D	295017AA1.06	SDM-N62, SDM-D17A
37	A	295023AK3.02	SDM-D17A, TS 3.3.6.1, USAR 15.7.6
38	B	295024EK3.06	SDM-C71, LCO 3.3.1.1
39	A	295025EK3.06	SDM-C22, LCO 3.3.1.1, LCO 3.3.4.2
40	C	295026EK1.01	GP Components Text, Suppression pool temperature PEI bases
41	C	295014AK3.01	IOI-4 Caution statement, PAP-0201 section 6.4.5
42	A	295027EK2.02	PEI Bases document
43	A	295031EK1.03	PEI bases document
44	C	295037EK2.02	SDM-C22
45	B	G 2.4.18	PEI-T23
46	B	295025EA1.03	SDM-B21/N11
47	C	295030EA2.02	PEI-T23, PEI-B13
48	A	295001AK3.01	AT&AA Text Chapter 5 (USAR 15.3.1)
49	B	295002AK2.11	ONI-N62
50	B	264000A3.01	SDM-R43
51	A	295005AA1.03	SDM-C11
52	D	295008AK1.01	SMD-C71, SDM-N32/C85, SDM-B21
53	C	295011AK2.01	PEI-T23, SDM-M11, SOI-M11
54	B	295012AK2.02	SDM-M13

Q#	Ans	K/A	Reference
55	D	295018AK2.01	SOI-B33, ONI-P43
56	A	G 2.1.32	SOI-C34, SOI-N27, ONI-N27
57	C	295020AK2.09	SMD-E12
58	A	295021AK1.04	IOI-12
59	B	295022AK1.02	ONI-C11-1, Technical Specification 3.1.5
60	A	295029EK3.01	PRI-T23, PEI bases document
61	D	295035EA1.02	SDM-M15
62	D	295036EK3.02	PEI-N11, PEI bases document
63	C	G 2.4.27	ONI-P54
64	C	201005A2.03	SDM-C11, SOI-C11
65	C,D	202002A3.02	SDM-B33, ARI-H13-P680-4
66	D	203000K4.14	IOI-11, Attachment 1, SDM-C61
67	D	209001K3.02	SDM-B21C
68	C	209002A4.15	SDM-E22A
69	B	212000K6.05	SDM-C71
70	D	215004A3.04	SDM-C51
71	DELETE		
72	C	216000K5.07	SDM-B21
73	C	400000K2.01	DCP 99-5019
74	D	218000A1.05	SDM-B21C
75	B	223001K5.01	SDM-M17
76	D	223002A3.02	SDM-B21(NS4)
77	B	226001K6.08	SDM-E12
78	B	239002A2.03	SDM-B21/N11, ONI-B21-1
79	D	259002A1.02	SDM-C34
80	B	261000K1.02	SDM-M15
81	B	262001A4.02	GP components text - chapter 5, IOI-3
82	B	264000K4.07	SDM-R43
83	B	212000A2.08	ONI-C71-1, PEI-B13, PEI bases document

Q#	Ans	K/A	Reference
84	C	201001K3.01	SDM-B33, SDM-C11
85	D	202001K4.15	SDM-B33
86	C	204000K1.15	SDM-E31, SDM-G33
87	B	205000A1.06	SOI-E12
88	B	G 2.1.33	SDM-E12, Technical Specification 3.5.1, 3.6.1.7, and 3.6.2.3
89	B	234000A3.02	SDM-F11/15
90	C	245000A2.06	ONI-N36
91	A	259001A4.02	SDM-C34, GP components text chapter 2
92	B	263000K3.03	SDM-R42, ONI-R42-1
93	B	272000K4.02	SDM-M14, SDM-D17A
94	D	29003A4.03	SDM-M25/26, SOI-M25/26
95	D	300000K1.05	SDM-B21/N11
96	B	201003K5.05	GP Reactor Theory Text, Chapter 5
97	C	233000A2.03	SDM-G41, ARI-H13-P970-1 (D3)
98	C	256001K2.01	SDM-N21/61
99	C	29002K4.02	SDM-B13
100	D	G 2.1.32	IOI-3

U.S. Nuclear Regulatory Commission
Written Examination January 2001
Senior Reactor Operator
Reference Materials

1. TS LCO 3.1.7
2. Simplified FDW System Diagram
3. PEI-SPI Supplement Figure 1a
4. PEI-M51/56 Flowchart
5. PAP-1914 Attachment 4

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

a. ≥ 0.38% $\Delta k/k$, with the highest worth control rod analytically determined; or

b. ≥ 0.28% $\Delta k/k$, with the highest worth control rod determined by test.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore primary containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore isolation capability in each required primary containment penetration flow path not isolated.	1 hour
	AND		
		-------NOTE-------- Entry and exit is permissible under administrative control. ------------------	
	D.4	Initiate action to close one door in each primary containment air lock.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		
	E.3	Initiate action to restore primary containment to OPERABLE status.	1 hour
	AND		
	E.4	Initiate action to restore isolation capability in each required primary containment penetration flow path not isolated.	1 hour
	AND		
		-------NOTE-------- Entry and exit is permissible under administrative control. ------------------	
	E.5	Initiate action to close one door in each primary containment air lock.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is: a. $\geq$ 0.38% $\Delta k/k$ with the highest worth control rod analytically determined; or b. $\geq$ 0.28% $\Delta k/k$ with the highest worth control rod determined by test.	Prior to each in vessel fuel movement during fuel loading sequence <u>AND</u> Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the monitored rod density and the predicted rod density shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored rod density and the predicted rod density is within ± 1% $\Delta k/k$.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement AND 1000 MWD/T thereafter during operation in MODE 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ A stuck rod may be bypassed in the Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9 if required to allow continued operation. -----------------------------	
	A.1 Disarm the associated control rod drive (CRD).	2 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than or equal to the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS).
	<u>AND</u>		
	A.3	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	---------NOTE--------- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ----------NOTE---------- Not applicable when THERMAL POWER > 19.0% RTP. ---------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. <u>OR</u> D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. -- Insert each fully withdrawn control rod at least one notch.	7 days
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. -- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 13 is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No OPERABLE control rod that is "slow" shall occupy a location adjacent to another OPERABLE control rod that is "slow" or a withdrawn control rod that is stuck.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------

During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 950 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel __AND__ (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	(continued)	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 950 psig.	120 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 950 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1
Control Rod Scram Times

-----------------------------------NOTES-----------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 13. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds)	
	REACTOR STEAM DOME PRESSURE(c) 950 psig	REACTOR STEAM DOME PRESSURE(c) 1050 psig
43	0.30	0.31
29	0.78	0.84
13	1.40	1.53

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < 950 psig are within established limits.

(c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 600 psig.	A.1	--------NOTE-------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------	
		Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 600 psig.	B.1 Restore charging water header pressure to ≥ 1520 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 1520 psig
	AND	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 600 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 1520 psig
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 --------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each control rod scram accumulator pressure is ≥ 1520 psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Rod Pattern

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 19.0% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1	--------NOTE--------- Affected control rods may be bypassed in Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9. ---------------------	
		Move associated control rod(s) to correct position.	8 hours
	<u>OR</u>		
	A.2	Declare associated control rod(s) inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Nine or more OPERABLE control rods not in compliance with BPWS.	B.1	--------NOTE-------- Affected control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9 for insertion only. -------------------- Suspend withdrawal of control rods.	Immediately
	AND		
	B.2	Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all OPERABLE control rods comply with BPWS.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.7.1	Verify available volume of borax-boric acid solution is within the limits of Figure 3.1.7-1.	24 hours
SR 3.1.7.2	Verify temperature of borax-boric acid solution is ≥ 70°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 70°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days
SR 3.1.7.5	Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or boron is added to solution AND Once within 24 hours after solution temperature is restored to ≥ 70°F

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 32.4 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after pump suction piping temperature is restored to ≥ 70°F

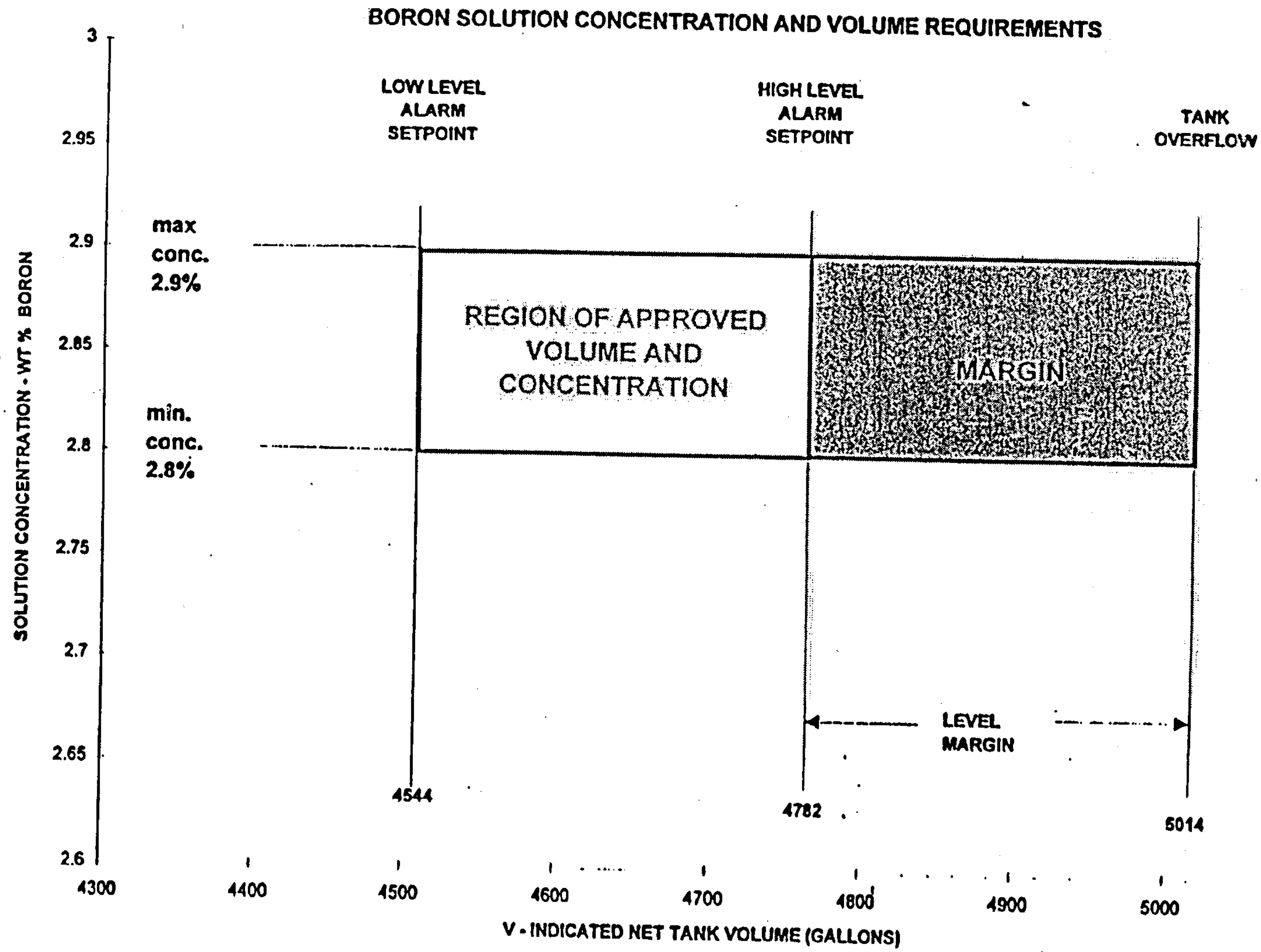


Figure 3.1.7-1
Boron Solution Concentration/Volume Requirements

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

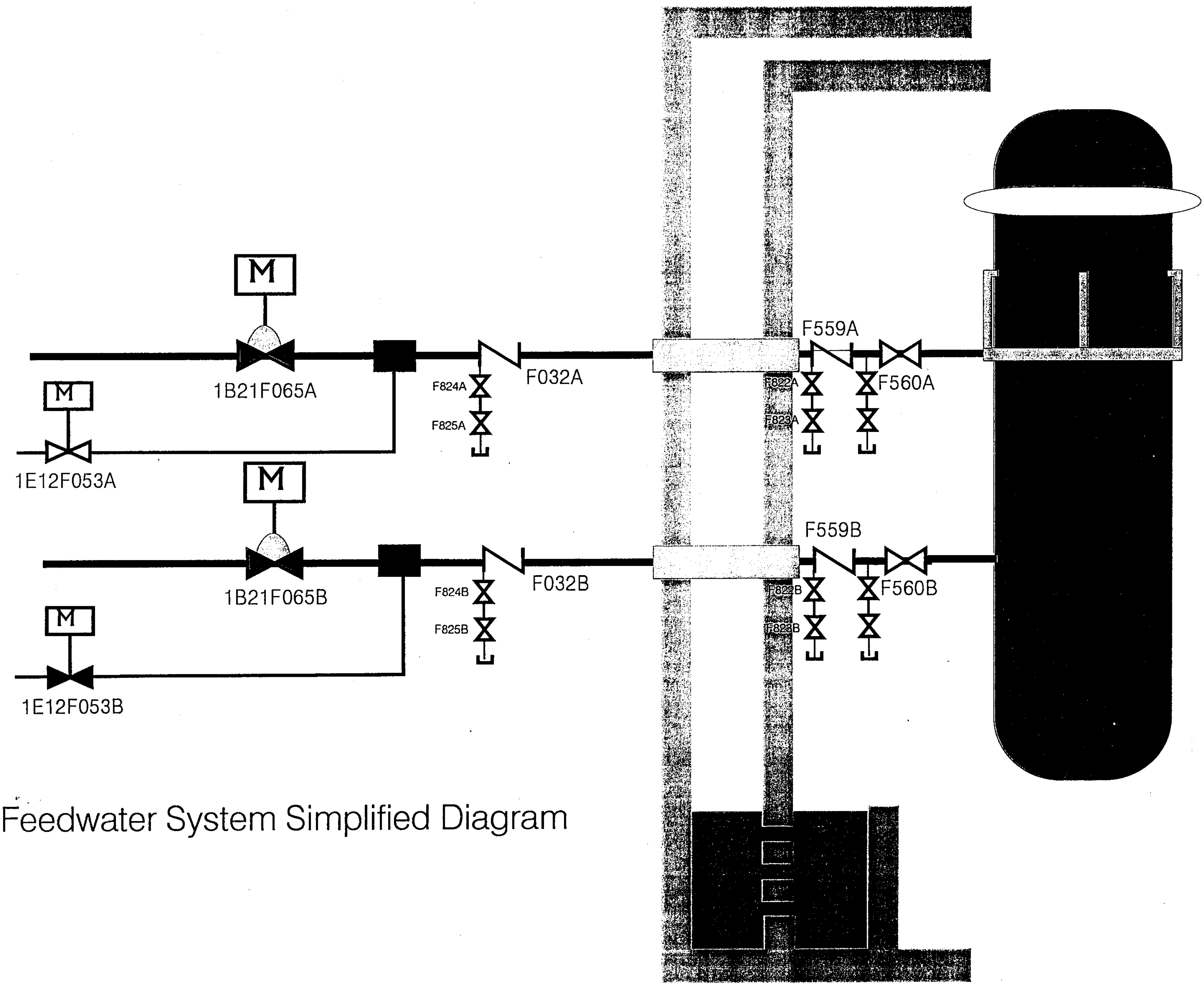
ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SDV vent and drain line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Restore valve to OPERABLE status.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 -----NOTE----- An isolated line may be unisolated under administrative control to allow draining and venting of the SDV. ----- Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

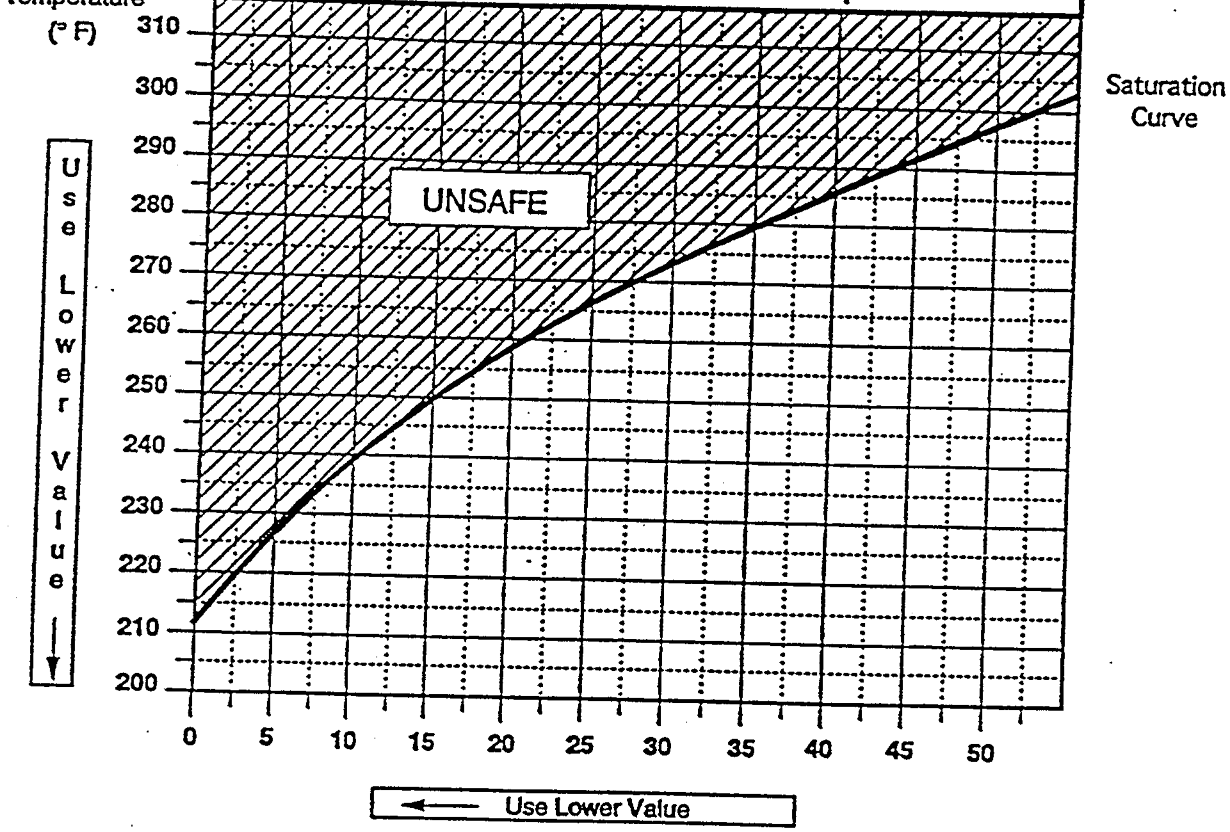
SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.8.1	-------------------NOTE------------------- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. --- Verify each SDV vent and drain valve is open.	31 days
SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days
SR 3.1.8.3	Verify each SDV vent and drain valve: a. Closes in ≤ 30 seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset.	24 months



Feedwater System Simplified Diagram

Figure 1a
RPV Low Pressure Saturation Temperature



Directions:

1.0 **IDENTIFY** RPV Pressure on the horizontal axis of the figure.

2.0 <u>IF</u> the value falls between marked lines on the figure,
<u>THEN</u> **USE** the <u>lower</u> value.

3.0 **IDENTIFY** the point formed by the intersection of the RPV Pressure and the Saturation Curve to determine RPV Saturation Temperature.

4.0 <u>IF</u> the value falls between marked lines on the figure,
<u>THEN</u> **USE** the <u>lower</u> value.

THEN proceed

Is Containment hydrogen below HDOL

NO

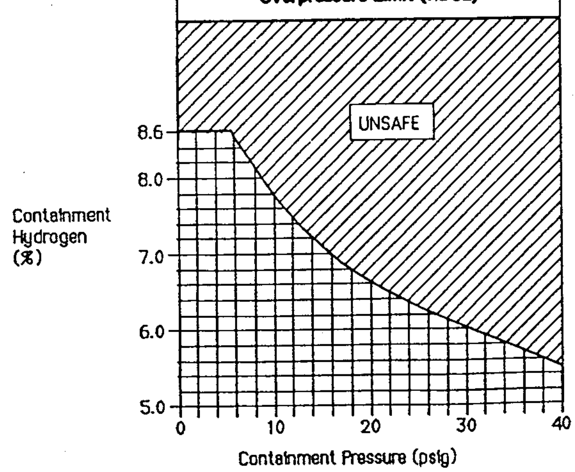
YES

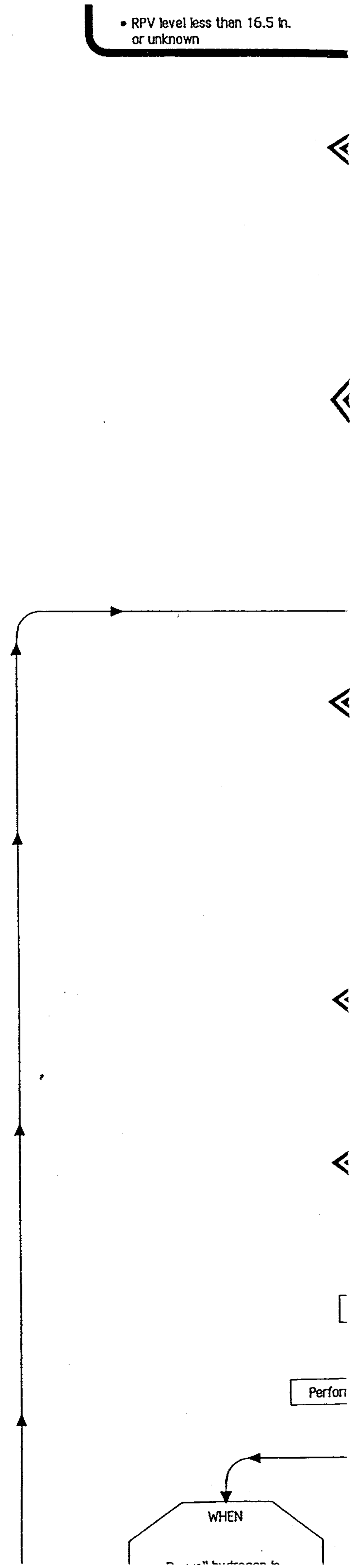
Start the Combustible Gas Mixing System

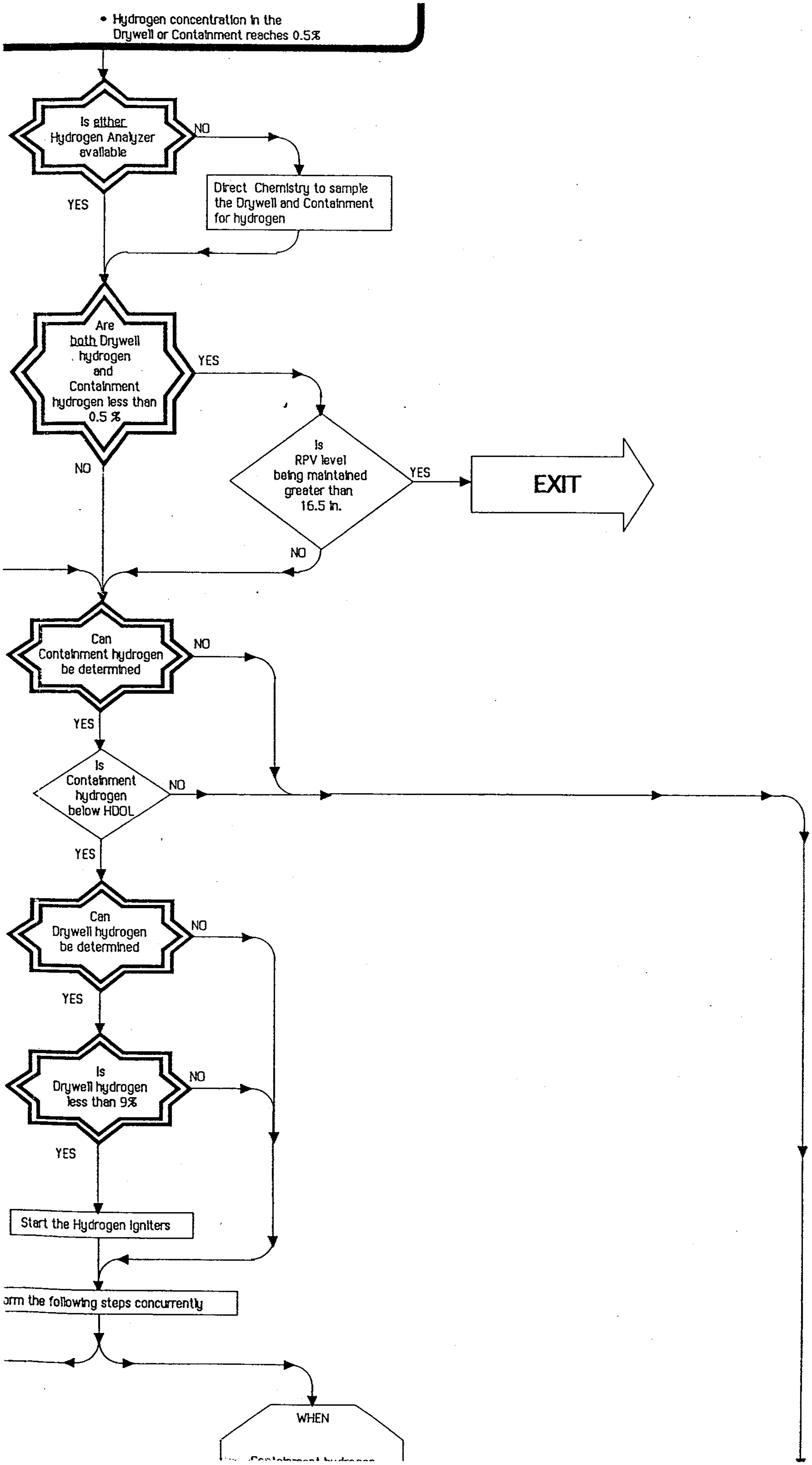
Prepar
Ri
Pr

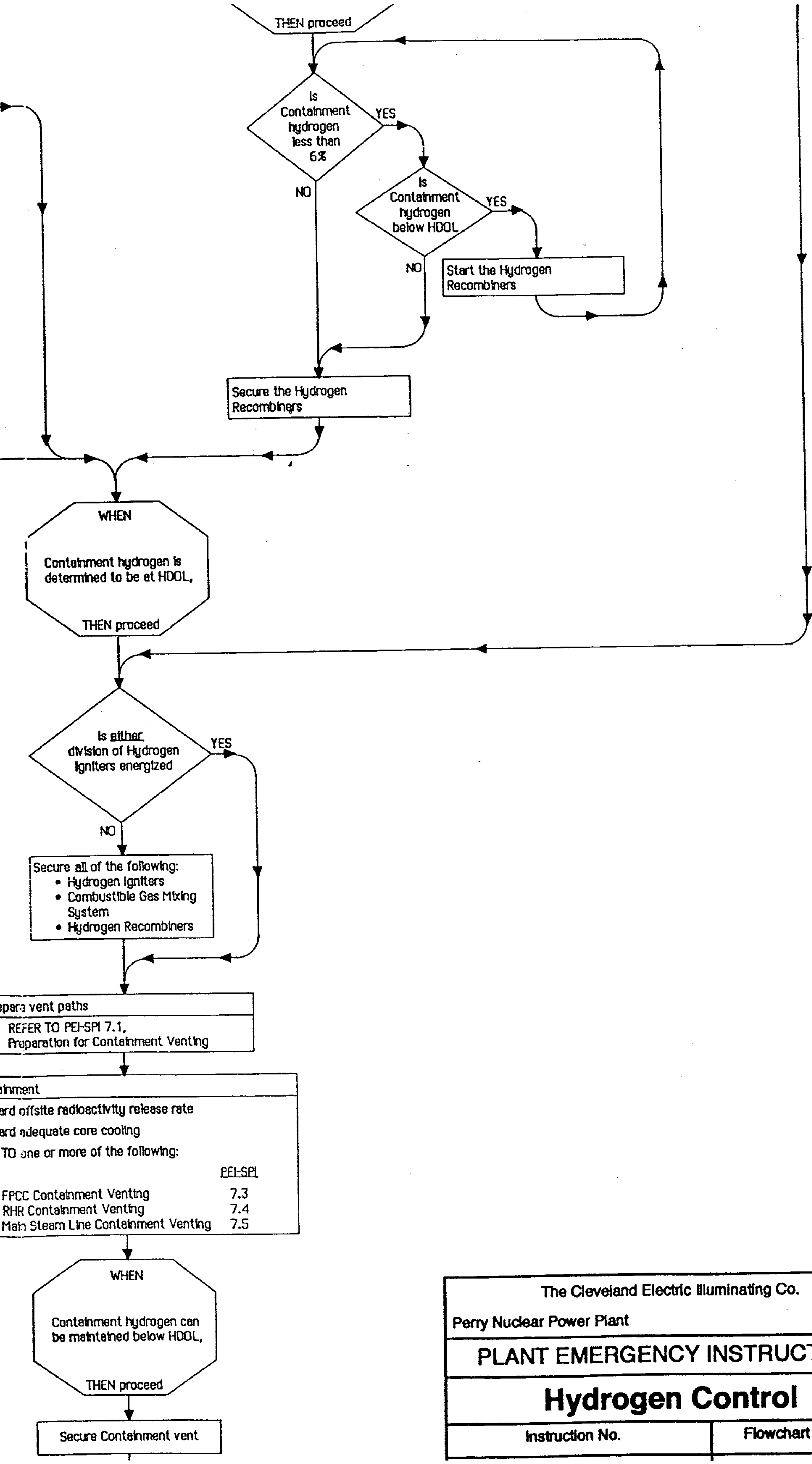
Vent Contain
Disregard
Disregard
REFER TO
a) FP
b) RH
c) Ma

Figure 7
Hydrogen Deflagration Overpressure Limit (HDOL)




• RPV level less than 16.5 in.
or unknown
Perforn
WHEN


• Hydrogen concentration in the Drywell or Containment reaches 0.5%
Is either Hydrogen Analyzer available
NO
YES
Direct Chemistry to sample the Drywell and Containment for hydrogen
Are both Drywell hydrogen and Containment hydrogen less than 0.5 %
YES
NO
Is RPV level being maintained greater than 16.5 in.
YES
NO
EXIT
Can Containment hydrogen be determined
NO
YES
Is Containment hydrogen below HDOL
NO
YES
Can Drywell hydrogen be determined
NO
YES
Is Drywell hydrogen less than 9%
NO
YES
Start the Hydrogen Igniters
orm the following steps concurrently
WHEN



The Cleveland Electric Illuminating Co.

Perry Nuclear Power Plant Unit 1

PLANT EMERGENCY INSTRUCTION

Hydrogen Control

Instruction No.	Flowchart Rev.

OPERABILITY REQUIREMENTS FOR FIRE PROTECTION SYSTEMS

General Notes

1. This Attachment provides a listing of the Operability Requirements, Applicability, Actions on Inoperable, and Surveillance Requirements of all Fire Protection Systems/Features/Components for the Perry Power Plant Unit 1 and Common Areas, Operable Unit 2 Systems, and Buildings/Areas within the Owner-Controlled Area.

2. The surveillance requirements delineated within this Attachment are minimum requirements. In some cases, Periodic Test Instructions are performed more frequently than required by this procedure for such purposes as scheduling, plant evolutions, for good practice, or the need to monitor equipment performance.

3. All surveillance frequencies are permitted a maximum allowable extension not to exceed 25% of the required interval. (Ref. T.S. 4.0.2 {SR 3.0.2}) |C-4

4. Contained within this Attachment and within the P54 Periodic Test Instructions is the pound (#) sign designation. A pound sign is utilized to designate those portions of the plant fire systems/features/components which satisfy requirements of 10CFR50 Appendix R pertaining to protection of or separation of safe shutdown circuits and components.

5. Also contained within this Attachment are dollar ($) sign, underlined dollar (<u>$</u>) signs, and cent (¢) sign designations. Dollar signs indicate items which satisfy previous NRC regulation and commitment. Underlined Dollar signs indicate items which were committed to during preparation of Revision 5 to this PAP and the associated Safety Evaluation or subsequent TCN's or Revisions. Cent signs indicate requirements which have been deemed appropriate by the Perry Fire Protection Staff in order to ensure operability of all fire systems. |TC-2

6. All hourly firewatches shall be on a frequency of 60 minutes with a margin of 15 minutes. The use of any part of the 15 minute margin does not alter the starting schedule for any subsequent patrol.

7. All continuous firewatches shall be continuous manned coverage. In the event that several large areas require continuous surveillance, or if the area requiring continuous surveillance is an ALARA concern, a patrol of each area once per 15 minutes with a margin of 5 minutes may be deemed acceptable. The use of any part of the 5 minute margin does not alter the schedule for any subsequent patrol.

8. For instances where the surveillance cannot be performed due to plant operability impact, personnel safety, or ALARA with the plant being in Operational Conditions {MODES} 1, 2, or 3, the surveillance shall be performed prior to restart following the next cold shutdown exceeding 72 hours. |C-4

9. Periodic Test Instructions not in compliance with this PAP upon PORC approval shall not be required to be revised until their next scheduled periodic review. The existing PTI shall take precedence should the PTI and this PAP conflict.

10. Compensatory actions identified within this Attachment do not require the application of <PAP-1402>.

(INTENTIONALLY BLANK)

ATTACHMENT 4 INDEX

System — Att. 4 Sheet

1 General Area Smoke / Heat / Flame Detection..............4

2 Smoke / Heat Detection for Fixed Suppression Systems.....8

3 Fire Suppression Water Supply (Motor/Diesel Pumps).......9

4 Fire Mains and Headers...................................12

5 Manual Fire Water Hose Stations..........................14

6 Yard Fire Hydrants and Hydrant Hose Houses...............16

7 Wet-Pipe Sprinkler Systems...............................18

8 Pre-Action Sprinkler, Water Spray, and Deluge Systems...20

9 Low Pressure CO2 Systems.................................25

10 CO2 Fire Hose Reels......................................27

11 CO2 Storage Tanks..28

12 Halon Suppression Systems (SB/TSC).......................29

13 Fuel Oil Pumphouse Foam System...........................31

14 Portable Fire Extinguishers..............................32

15 Fire Alarm Hand Stations.................................33

16 Fire Rated Assemblies (Doors, Barriers, Dampers)........34

17 Fire and Security System Computer and Peripherals.......37

18 Appendix R Emergency Lighting............................38

19 Owner-Controlled Area Fire Protection Systems...........39

1.A. SYSTEM: $ General Area Smoke / Heat / Flame Detection (Function A)

B. OPERABILITY REQUIREMENTS:

$ Fire Detection Instrumentation listed on Attachment 6 shall be deemed OPERABLE if they are capable of responding to an applied smoke, heat, or ultra-violet light sample and subsequently transmit an alarm to the SAS.

NOTE: Those detectors listed on Attachment 6 designated with a pound (#) sign protect equipment and areas which are safety related.

C. APPLICABILITY:

$ Fire Detection Instrumentation shall be OPERABLE whenever the equipment/fire zone they monitor is required to be OPERABLE.

EXCEPTION:

$ Fire Detection Instrumentation located within the Containment building ARE NOT required operable during the performance of Type A Containment Leakage Rate Tests

D. ACTIONS ON INOPERABLE:

TIME IF:

$ 1. One-Half or more of the total number of instruments on the detection zone or any two adjacent instruments are inoperable,

THEN:

1 HOUR $ a. If the instruments are located outside of the containment building, establish an HOURLY Firewatch Patrol of the affected area.

1 HOUR $ b.1 If the instruments are located in the containment building, remotely monitor the temperature of the affected area HOURLY for the following locations:

CONTAINMENT: (ERIS Screen 076)

Elev./Az: MPL #	720'6"/230° D23-N130A	720'6"/100° D23-N130B	689'4"/40° D23-N140A	689'4"/210° D23-N140B
Elev./Az: MPL #	647'0"/54° D23-N150A	645'6"/231° D23-N150B	613'0"/69° D23-N160A	613'0"/251° D23-N160B

DRYWELL: (ERIS SCREEN 076)

Elev./Az: MPL #	653'8"/315° D23-N100A	653'8"/135° D23-N100B	634'0"/308° D23-N110A
Elev./Az: MPL #	634'0"/145° D23-N110B	605'8"/308° D23-N120A	604'6"/143° D23-N120B

TC-1

Attachment 4
Sheet 5 of 39

OR:

1 HOUR $ b.2 If the Containment/Drywell areas are accessible, inspect the zone/area once every 8 HOURS by the performance of a physical walkdown.

IF:

$ 2. Less than one-half of the total number of instruments on the detection zone, but no two adjacent, are inoperable,

THEN:

14 DAYS $ a. Restore the inoperable instrument(s) to OPERABLE status

or within

1 HOUR $ b. Implement the requirements of 1.D.1 above.

IF:

¢ 3. The detection system is OPERABLE but will not transmit an alarm to the SAS,

THEN:

1 HOUR ¢ a. Establish an HOURLY Firewatch Patrol of the affected detection system control panel to verify no change in the status of the panel.

IF:

¢ 4. Duct smoke detector 1M38-N130 is inoperable,

THEN:

1 HOUR ¢ a. Establish an HOURLY Firewatch Patrol of the LPCS, HPCS, and RHR "C" pump rooms (AUX-574 - Rms. 2, 5, 7).

IF:

¢ 5. Duct smoke detector 1M15-N110A/B is inoperable,

THEN:

¢ a. Swap over to the opposite 1M15 exhaust train.

or within

1 HOUR $ b. Establish a Firewatch Patrol of the Reactor Building Annulus once every 8 hours.

TC-2

E. SURVEILLANCE REQUIREMENTS:

<u>FREQ</u>. 1. Each of the detection instruments located in the Control Room Panels, Cabinets, and Subfloor shall be demonstrated OPERABLE by:

6 MTHS <u>$</u> a.1 Performing a functional test of each smoke/heat detector located within all upright termination cabinets and control panels by applying a smoke or heat sample.

<u>$</u> a.2 Performing an alarm test of each subfloor panel/zone by installing a jumper within the detection control panels; and performing a supervised circuit test for each panel/zone by lifting a lead at the control panels.

1 R/O <u>$</u> b. For all detection instruments located within the Control Room Subfloor, performing a functional test by application of a smoke or heat sample as appropriate on 50% of the number of instruments in each detection zone.

2. For All Other Smoke Detectors listed on Attachment 6, perform subsection a or b as follows:

<u>$</u> a. Test per NFPA-72, 1993:

6 MTHS 1. Perform a visual inspection of each detector to ensure that it remains in good physical condition and that there are no changes such as building modifications, occupancy hazards, or environmental effects that would affect detector performance. TC-2

AND

12 MTHS 2. Functionally test each detector in place by the introduction of a smoke (approved test gas) sample and inspect the detector for dirt or blockages, cleaning as necessary. Perform a supervised circuit test for each detector panel/zone. TC-2

AND

2 YEARS 3. Perform a detector sensitivity check in accordance with the manufacturer's recommendations.

$ b. Test per PY-NRR/CEI-0272L:

6 MTHS 1. Perform a channel function test of each of the detectors by the introduction of a smoke (approved test gas) sample and perform a supervised circuit test for each panel/zone.

3. For Air Duct Mounted Detection Devices, in addition to tests described in 1.E.2.a or b above:

12 MTHS $ a. Test in accordance with the manufacturer's requirements to ensure that the device will properly sample the air stream.

4. For Each Restorable Heat Detector listed on Attachment 6 and accessible during Unit Operation, perform the following:

12 MTHS $ a. Perform a channel functional test of each of the detection loops (circuits) by applying a heat sample and perform a supervised circuit test for each panel/zone. TC-2

5. For Each Restorable Heat Detector listed on Attachment 6 and inaccessible during unit operation (Drywell Area Detection, Recirc Pump CO2 System, Turbine Bearings, Feed Pump Turbines), perform the following:

S/R $ a. Unless performed within the previous 12 months, perform a channel functional test of each of the detection loops (circuits) by applying a heat sample to one detector and perform a supervised circuit test for each zone every forced shutdown exceeding 72 hrs. TC-2

AND

R/O $ b. Perform a channel functional test of each of the detectors by applying a heat sample and perform a supervised circuit test for each panel/zone.

6. For each restorable heat detector located around an exterior oil-filled transformer:

18 MTHS OR 1 R/O ¢ a. Perform a channel functional test of each of the detectors by applying a heat sample and perform a supervised circuit test for each panel/zone during the performance of the deluge system functional test.

7. For each of the flame detectors listed on Attachment 6:

12 MTHS $ a. Perform a channel function test of each detector by applying a simulated ultra-violet light sample and perform a supervised circuit test for each panel/zone. TC-2

2.A. SYSTEM: $ Smoke / Heat Detection For Fixed Suppression Systems

B. OPERABILITY REQUIREMENTS:

$ Fire Detection Instrumentation listed on Attachment 6 which is used to automatically activate a fixed suppression system shall be deemed OPERABLE if they are capable of responding to an applied smoke or heat sample and subsequently initiate the suppression system as appropriate for the system design. Further required actions are defined in the Operability Requirements for each type of system.

NOTE: Those detectors listed in Attachment 6 designated with a pound (#) sign protect equipment and areas which are safety related.

C. APPLICABILITY:

$ Fire detection instrumentation associated with fixed suppression systems shall be OPERABLE whenever the system they activate is required OPERABLE.

D. ACTIONS ON INOPERABLE:

TIME IF:

$ 1. One or more of the instruments are inoperable,

THEN:

1 HOUR $ a. If the instruments are located outside of the containment building, establish an HOURLY Firewatch Patrol of the affected area.

1 HOUR $ b.1 If the instruments are located in the containment building (inside drywell for the Recirculation Pumps CO2 Systems) remotely monitor the temperature of the affected area HOURLY for the following locations:
DRYWELL: (ERIS Screen 076)

Elev./Az: MPL #	653'8"/315° D23-N100A	653'8"/135° D23-N100B	634'0"/308° D23-N110A
Elev./Az: MPL #	634'0"/145° D23-N110B	605'0"/308° D23-N120A	604'6"/143° D23-N120B

KC 1

OR:

1 HOUR $ b.2 If the Drywell area is accessible, inspect the zone/area once every 8 HOURS by the performance of a physical walkdown.

E. SURVEILLANCE REQUIREMENTS:

$ 1. All activating detectors shall be tested in accordance with Section 1.E of this attachment.

3.A. <u>SYSTEM: $ # Fire Suppression Water Supply (Motor/Diesel Pumps)</u>

B. **OPERABILITY REQUIREMENTS:**

$ The **Fire Suppression Water System** shall be deemed OPERABLE with:

$ a. Two fire suppression pumps, each rated at a minimum of 2500 gpm at 141 psi with the ability to take suction from the ESW forebay; and their discharges aligned to the fire suppression water header;

AND

$ b. The ability to automatically start upon a drop in system pressure to the individual pumps'.starting setpoint.

C. **APPLICABILITY:**

$ At All Times.

D. **ACTIONS ON INOPERABLE:**

<u>TIME</u> **IF:**

$ 1. **One Fire Pump** is inoperable,

THEN:

7 DAYS $ a. Restore the inoperable pump to OPERABLE status or provide an alternate backup pump.

IF:

$ 2. **Both Fire Pumps** are inoperable,

THEN:

24 HRS $ a.1 Establish a backup fire suppression water system.

OR

1 HOUR $ a.2 If a backup system cannot be established, implement the requirements of T.S. 3.0.3 and 3.0.4 {LCO 3.0.3 and LCO 3.0.4}.

E. **SURVEILLANCE REQUIREMENTS:**

FREQ. 1. The fire suppression water pumps shall be demonstrated OPERABLE:

7 DAYS $ a.1 By verifying the electrolyte level of the pilot cell of each diesel fire pump battery is above the cell plates.

$ a.2 By verifying each battery pilot cell specific gravity, corrected to 77°F is greater than or equal to 1.200.

$ a.3 The overall voltage for each battery bank is greater than or equal to 24 VDC.

31 DAYS $ b.1 By starting the electric motor fire pump by simulating a drop in system pressure and operating it for at least 15 minutes on recirculation flow.

$ b.2 By starting the diesel driven fire pump by simulating drop in system pressure from ambient conditions and operating it for at least 30 minutes on recirculation flow.
NOTE: Verification of starting setpoints in turn verifies that each pump would start sequentially to maintain the fire main pressure greater than 65 psig.

$ b.3 By verifying the diesel fuel storage tank contains a level of at least 150 gallons of fuel.

$ b.4 By verifying each manual valve in the pump discharge flow path and the diesel fuel supply is in its correct position.

92 DAYS $ c.1 By verifying that a sample of the diesel fuel from the diesel pump fuel oil storage tank, obtained in accordance with ASTM-D270-75, is within acceptable limits specified in Table 1 of ASTM-D975-77 when checked for viscosity, water, and sediment.

$ c.2 By verifying that each cell of each diesel pump battery specific gravity is appropriate for continued service of the battery.

18 MTHS $ d.1 By verifying that the diesel pump batteries and battery rack show no visual indication of physical damage or abnormal deterioration; and that the battery-to-battery and terminal connections are clean, tight, free or corrosion, and coated with an anti-corrosion material.

$ d.2 By performing a full functional test of each fire pump controller and a full flow test of each fire pump utilizing the guidance of NFPA 20-1990 and verifying:
   a. Deleted
   b. That each pump performs above its design demand performance curve as delineated within PTI-P54-P0036 "Diesel and Electric Fire Pumps Flow and Data Control Panel Functional Test". Note: If performance falls below the 10% degradation line, the Fire Protection System Engineer shall be contacted to determine increased frequency test requirements and the initiation of appropriate corrective measures.

c. Deleted
d. That NFPA-20 required alarms are received at the Secondary Alarm Station.

$ e. By subjecting the diesel fire pump to inspections and maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service:

3 MTHS By checking and performing a chemistry analysis of the engine motor oil; by checking the air filter oil bath, coolant levels, position of water filter shut-off valves, condition of all hoses and belts; inspecting the pump/engine assembly for evidence of external air, oil, or water leakage; and checking the throttle linkage.

1 YEAR Change the engine oil and filter; change the right angle drive cooling oil; change the fuel filter; drain and flush the coolant reservoir and check the heat exchanger anodes; check engine alternator and starting motor wiring and terminals; adjust fuel injectors, crossheads, and valves.

2 YEARS Perform turbocharger maintenance and check all exhaust and intake mounting hardware for tightness.

(INTENTIONALLY BLANK)

4.A. SYSTEM: $ Fire Mains and Headers

B. OPERABILITY REQUIREMENTS:

$ An OPERABLE **Flow Path** shall exist from the discharge header of the fire pumps through distribution piping and OPERABLE isolation, sectionalizing, or control valves to each yard hydrant, fixed suppression system, and hose standpipe which protects safety-related equipment/areas; and all other fire suppression and hose standpipe systems.

C. APPLICABILITY:

$ At All Times.

D. ACTIONS ON INOPERABLE:

TIME **IF:**

$ 1. Any portion of the fire main distribution piping is out of service,

THEN:

1 HOUR ¢ a. Verify that at least one OPERABLE flow path exists to each fire suppression system, yard hydrant, or hose standpipe. See Note Below.

OR:

24 HRS $ b. Establish back-up fire suppression water supply to the affected systems/areas through temporary jumpers.

NOTE: During the time period between the determination of loss of fire suppression water supply to a system/area and the establishment of a temporary supply, or if a temporary supply cannot be provided, all impaired systems shall be properly compensated for per the individual systems' requirements of this Attachment.

E. SURVEILLANCE REQUIREMENTS:

FREQ. 1. The Fire Suppression Water Distribution System shall be demonstrated OPERABLE:

31 DAYS $ a. By verifying that each manual valve in the flow paths are in their correct position. This includes flow path valves from the discharge of the fire service pumps through all yard and building distribution piping up to and including the isolation valve for each fixed water suppression system and each fire hose standpipe leg.

12 MTHS ¢ b. By performing a test of each valve supervisory position switch. TC-2

12 MTHS	$	c.	Cycle each testable valve, accessible during unit operation, in the flow paths through at least one complete cycle of full travel.
18 MTHS	$	d.1	Cycle each testable valve, inaccessible during unit operation, in the flow paths through at least one complete cycle of full travel.
	$	d.2	Performing a flush of the fire mains.
3 YRS	$	e.	Perform a full flow test of the system using the guidance of industry standards such as the National Fire Protection Association's Fire Protection Handbook. This test should divide the fire water distribution system into identifiable sections so that data can be trended to identify potential degradations of the fire mains due to partially shut valves, tuberculosis of the piping, etc.

(INTENTIONALLY BLANK)

5.A. **SYSTEM: $ Manual Fire Water Hose Stations**

B. **OPERABILITY REQUIREMENTS:**

$ **Manual Fire Water Hose Reels** shall be deemed OPERABLE if the hose reel/rack and isolation valve are accessible and the hose is capable of delivering water to the required point of application in safety-related areas; and all other plant areas.

C. **APPLICABILITY:**

$ Fire Water Hose Stations shall be OPERABLE whenever the equipment (systems) in the area is required to be OPERABLE.

D. **ACTIONS ON INOPERABLE:**

TIME **IF:**

$ 1. The fire water hose reel is the primary means of fire suppression in the ESWPH, CC, AX1, IB, FH, or RB1,

THEN:

1 HOUR $ a. Restore the hose stations to OPERABLE by performing the following:

1. Connect a gated wye on the nearest OPERABLE hose station(s).
2. Connect hose of OPERABLE station(s) to one outlet of the gated wye.
3. Connect hose of sufficient length to the second outlet of the gated wye and route the hose to the impaired area.
4. Where it can be demonstrated that the physical routing of the hose would result in a recognizable hazard to equipment, personnel, or to the hose itself, the hoses shall be stored at the OPERABLE hose station.

IF:

$ 2. The inoperable hose station is not the primary means of fire suppression in the ESWPH, CC, AX1, IB, FH, or RB1,

THEN:

24 HRS $ a. Complete the action described in 5.D.1.a above.

IF:

¢ 3. The inoperable hose station is not in the ESWPH, CC, AX1, IB, FH, or RB1,

THEN:

1 HOUR ¢ a. Ensure that the Fire Brigade is informed and advised to utilize an attack pack from the brigade equipment storage areas.

E. SURVEILLANCE REQUIREMENTS:

FREQ.	**1.**	All Fire Water Hose Stations shall be demonstrated OPERABLE by:
31 DAYS $	**a.**	Visually inspecting all fire hose stations accessible during plant operation to ensure all required equipment is at the station.
18 MTHS $	**b.1**	Visually inspecting all fire hose stations not accessible during plant operation to ensure all required equipment is at the station.
$	**b.2**	Removing the hose from its rack/reel and inspecting the hose and coupling gaskets; replacing as necessary.
3 YEARS $	**c.1**	Partially opening each hose station isolation valve to verify valve operability and no flow blockage.
$	**c.2**	Conducting a hydrostatic hose test at a test pressure of 250 psi.

(INTENTIONALLY BLANK)

6.A. SYSTEM: $ Yard Fire Hydrants and Hydrant Hose Houses

B. OPERABILITY REQUIREMENTS:

¢ Yard Fire Hydrants shall be deemed OPERABLE if they are accessible and capable of delivering adequate water flow.
Hose Houses shall be deemed OPERABLE if they contain the proper equipment as delineated in OM7C:PTI-P54-P0023, Hydrant Hose House Monthly Inspection.

$ The following hydrants and hose houses protect equipment and areas which are safety-related:

Hydrant 24 - Diesel Generator Building
Hydrant 23 - Diesel Generator Fuel Oil Storage Tanks
Hydrant 1 - Unit 2 Startup Transformer
Hydrant 3 - Unit 1 Startup Transformer

C. APPLICABILITY:

$ Fire hydrants and hose houses shall be OPERABLE whenever equipment in the areas protected is required to be OPERABLE.

D. ACTIONS ON INOPERABLE:

TIME IF:

$ 1. Any of the hydrants specifically identified in 6.B above is inoperable,

THEN:

24 HRS $ a. Route sufficient additional lengths of 2 1/2" or 3" fire hose from the nearest OPERABLE hydrant to provide protection to the unprotected area.

¢ Where it can be demonstrated that the physical routing of the hose presents a safety hazard to personnel or to the hose itself, store the hose in the nearest operable hose house and ensure that the fire brigade is notified.

IF:

¢ 2. Any other hydrant or hose house is inoperable,

THEN:

¢ a. Ensure that the fire brigade is aware of the status.

E. SURVEILLANCE REQUIREMENTS:

FREQ. 1. Fire hydrants and hose houses shall be demonstrated OPERABLE by:

31 DAYS $ a. Visually inspecting each hydrant hose house to verify that all required equipment is at the hose house.

6 MTHS **$** **b.** During the March-April-May and September-October-November time periods, visually inspect each fire hydrant to ensure that the hydrant barrels are dry and the hydrants are not damaged.

12 MTHS **$** **c.** Performing a flow check and flush of each hydrant.

1 YR **$** **d.1** Performing a hydrostatic hose test of all fire hose contained within each hose house at a test pressure of 250 psig. Kc-8

$ **d.2** Replacing all degraded gaskets within hose couplings.

(INTENTIONALLY BLANK)

7.A. SYSTEM: $ Wet-Pipe Sprinkler Systems

B. OPERABILITY REQUIREMENTS:

$ Wet-Pipe Sprinkler Systems shall be deemed OPERABLE if the system is lined up to automatically discharge water to the required point of application upon the fusing of one or more sprinkler heads and shall cause an alarm to be transmitted to the SAS upon the activation of at least one head.

C. APPLICABILITY:

$ Wet-pipe sprinkler systems shall be OPERABLE whenever the equipment they are designed to protect is required to be OPERABLE.

$ 1. The following sprinkler systems protect systems/areas which are safety-related, or contain redundant trains of safe shutdown equipment/circuits (#):
- Control Complex Elevations 599'0" and partial 574'10".
- Auxiliary Bldg 574'6" Reactor Core Isol Cooling Pump Room.
- Intermediate Bldg Elevation 599'0", Access Areas (see below for 574' areas)
- Intermediate Bldg Elevation 620'6", Access Areas.
- Auxiliary Bldg Elevation 620'6", West Access Area.

¢ 2. All other sprinkler systems:
- Intermediate Building 574'10' areas of IB 599'0" system
- Turbine Building East (Condenser Area)
- Turbine Building West
- Technical Support Center (SB 604')
- Service Building
- Primary Access Control Point
- Diesel Fire Pump Room (ESWPH 586')
- Fuel Oil Pumphouse
- Auxiliary Boiler Building
- Radwaste Building 620'/624'
- Service Building Annex
- Maintenance Building
- Maintenance Building Annex

D. ACTIONS ON INOPERABLE:

TIME	IF:		
$	1.	Any sprinkler system identified in 7.C.1 above will not deliver water to the hazard area,	
	THEN:		
1 HOUR	$	a.	Establish a continuous firewatch in the affected area with backup fire suppression equipment. OPERABLE hose stations in the area shall constitute backup fire suppression equipment.

IF:

¢ 2. Any **sprinkler system identified in 7.C.2** above will not deliver water to the hazard area,

THEN:

1 HOUR ¢ a. Establish an hourly firewatch of the affected area.

¢ **EXCEPTION:** Due to radiological concerns, the Turbine Building Condenser may not be accessible in conditions 1, 2 and 3. For this area, an hourly patrol to each condenser area access point may be performed to verify the absence of visual smoke within the condenser area.

E. SURVEILLANCE REQUIREMENTS:

FREQ. 1. Each wet-pipe sprinkler system shall be demonstrated OPERABLE by:

31 DAYS $ a. Verifying that each manual valve in the flow path is in its required position; including alarm line isolation valves.

3 MTHS $ b. Performing an Alarm Test of each sprinkler system Alarm Check Valve by use of the inspector's test valve, simulating flow from one sprinkler head and causing subsequent transmission of the fire alarm to the SAS per NFPA 72-1993. TC-2 TC-2

12 MTHS $ c. Cycling each valve in the flow path through at least one complete cycle.

18 MTHS $ d. For each system listed in 7.C.1 above, visually inspecting the sprinkler piping headers to verify integrity of the piping; and inspecting the sprinkler heads to verify no obstructions to spray patterns.

(INTENTIONALLY BLANK)

8.A. SYSTEM: $ Pre-Action Sprinkler, Water Spray, and Deluge Systems

B. **OPERABILITY REQUIREMENTS:**

$ **Pre-Action Sprinkler, Water Spray, and Deluge Systems** shall be deemed OPERABLE if the system is lined up to automatically open the deluge valve and initiate water flow upon detection signal and/or manual initiation as applicable and shall cause for an alarm to be transmitted to the SAS upon activation by any method. In addition, Charcoal Filter Plenum Water Spray Systems shall cause the associated plenum drain valve to open upon activation of the spray system; and the Iso Phase Buss Duct Deluge System shall automatically activate upon the initiation of any Main Transformer Deluge system.

C. **APPLICABILITY:**

$ Pre-action, water spray, and deluge systems shall be OPERABLE whenever the equipment/area protected is required to be OPERABLE. Note: {+ designates pre-action, = designates water spray, and ˆ designates deluge}

$ 1. The following systems protect equipment/areas which are safe shutdown (#):
+ Unit 1 Division 1 Cable Spreading/Penetration Rooms
+ Unit 1 Division 2 Cable Spreading/Penetration Rooms
= Unit 1 AEGTS Plenums 1M15-D001A and B
= Unit 1 CV&DWP Plenums 1M14-D001A and B
= CR Emergency Recirc Plenums M26-D001A and B
= Fuel Handling Exhaust Plenums M40-D001A, B, and C
= Unit 1 Aux Bldg Exhaust Plenum 1M38-D001

¢ 2. All other pre-action, water spray, and deluge systems:
= CA&MEA Ventilation Plenums M21-D001A and B
= Radwaste Building Exhaust Plenums M31-D001A and B
= Off Gas Building Exhaust Plenums 1M36-D001A and B
= IB Sub-Exhaust Plenum M33-D001
= TSC Emergency Recirc Plenum M52-D001
= Unit 2 Division 1 Cable Spreading/Penetration Rooms*
= Unit 2 Division 2 Cable Spreading/Penetration Rooms*
* Note: These two systems are manual initiation only.
+ Turbine Bearing and Turbine/Generator Skirt Area
+ Reactor Feed Pump Turbines
ˆ Hydrogen Seal Oil Unit
ˆ Unit 1 Main A/B/C Transformers
ˆ Unit 1 Start-up Transformer
ˆ Unit 1 Auxiliary Transformer
ˆ Unit 1 Iso Phase Buss Duct
ˆ Unit 1 Interbuss A/B/C Transformers
ˆ Unit 2 Start-up Transformer
ˆ Unit 2 Interbuss A/B/C Transformers

D. ACTIONS ON INOPERABLE

<u>TIME</u> IF:

$ 1. Either the **Unit 1 Div. 1 or Div. 2 Cable Spreading Pre-Action Spray System** will not deliver water to the hazard area, has an inoperable heat detection string, or will not automatically open the deluge valve upon any one of the activation methods,

THEN:

1 HOUR <u>$</u> a. Establish a continuous firewatch patrol throughout the affected area (IB Penetration Room, CC Vertical Chase, CC Cable Spreading Area) ensuring that each area is inspected at least once every 15 minutes; and provide backup fire suppression equipment such as verifying operability of hose reels in the area and providing additional portable extinguishers. Consult the P54 RSE for guidance.

IF:

¢ 2. Either the **Unit 2 Div. 1 or Div. 2 Cable Spreading Spray Systems** will not deliver water to the hazard area,

THEN:

1 HOUR ¢ a. Establish an Hourly firewatch patrol throughout the affected area (IB Penetration Room, CC Vertical Chase, CC Cable Spreading Area); and provide backup fire suppression equipment such as verifying operability of hose reels in the area and providing additional portable extinguishers. Consult the P54 RSE for guidance.

IF:

¢ 3. Any Automatic **Transformer Deluge System** will not deliver water to the hazard area, has an inoperable heat detection string, or will not automatically open the deluge valve upon any one of the activation methods,

<u>NOTE</u>:

- Unit 1 Main and Unit 1/Unit 2 Start-up transformers utilize a **manually** activated water spray deluge system with local heat detectors to provide control room annunciation for determining if manual initiation is necessary.
- The Auxiliary, Unit 1 Interbus, and Unit 2 Interbus transformers utilize a water spray deluge system that is activated **automatically** by a signal from the heat detectors located at each transformer.

(Reference - USAR Change 99-132 and Safety Evaluation 99-0105).

THEN:

1 HOUR ¢ a. Either transfer the transformer load to an alternate transformer;

OR

¢ b. Establish an hourly firewatch patrol of the affected transformer.

IF:

¢ 4. Either the **Turbine Bearing/Skirt Spray or Reactor Feed Pump Turbine Spray Systems** will not deliver water to the hazard has an inoperable heat detection string, or will not automatically open the deluge valve upon any one of the activation methods,

THEN:

1 HOUR ¢ a. Either establish an hourly firewatch patrol of the affected area if radiological conditions permit entry,

OR

¢ b. Notify the Reactor Operator to begin periodic monitoring of the Turbine/Feed Pump Turbine parameters as applicable.

IF:

$ 5. Any **Plenum Deluge System** can not deliver water to the hazard area or will not open the deluge valve upon any activation method,

THEN:

$ a. If the associated plenum temperature elements (thermister strips) are operable, no action is required.

OR

1 HOUR $ b. If the associated plenum temperature elements are not operable, establish an Hourly firewatch patrol of the affected plenum to check for smoke, excess heat, or plenum discoloration.

IF:

$ 6. Any system except plenum deluges will not transmit am alarm to the SAS,

THEN:

1 HOUR ¢ a. Establish an Hourly Check of the Riser for indications of system activation.

$ 7. Any **Plenum Deluge System** will not transmit am alarm to the SAS or cause the associated plenum drain valve to open upon water flow,

THEN:

1 HOUR ¢ a. Place the system in Secured Status per <SOI-P54(WTR)>. Apply the action statements of 8.D.5 above.

IF:

¢ 8. The supervisory air feature of any pre-action system is not operational, no action is necessary.

IF:

¢ 9. The **Hydrogen Seal Oil Unit Deluge System** will not deliver water to the hazard area, has an inoperable heat detection string, or will not automatically open the deluge valve upon any one of the activation methods,

THEN:

1 HOUR ¢ a. Establish an hourly firewatch of the area.

E. SURVEILLANCE REQUIREMENTS:

NOTE: Heat and Smoke Detectors which activate a fixed suppression system are tested per Section 1.B of this Attachment.

FREQ. 1. Each Pre-action, Spray and Deluge system shall be demonstrated OPERABLE by:

31 DAYS $ a. Verifying that each manual valve in the flow path is in its required position; including alarm line isolation valves.

3 MTHS $ b.1 Performing an Alarm Test of each system by use of the alarm test valve and causing subsequent transmission of the fire alarm to the SAS per NFPA 72-1993. TC-2

b.2 Deleted

¢ b.3 For each pre-action system equipped with supervisory air, performing a test of the supervisory low air alarm.

12 MTHS $ c. Cycling each valve in the flow path through at least on complete cycle.

18 MTHS $ d.1 For each system listed in 8.C.1 above, visually inspecting the sprinkler piping headers to verify integrity of the piping.

$ d.2 For each Unit 1 and 2 Cable Spreading system, inspecting the sprinkler heads to verify no obstructions to spray patterns.

$ d.3 For all systems, performing a full functional (trip) test, without actual water flow, to include verification that all automatic and manual trip activation methods cause the deluge valve to open and all automatic valves reposition as appropriate using the guidance of NFPA 15-1985.

$ d.4 Verifying that each plenum drain valve opens during the performance of d.3 above. TC-2

18 MTHS or 1 R/O	¢ e.	Performing a full flow test of each transformer deluge system using the guidance of NFPA 15-1985 and verifying spray nozzle orientation to prevent inadvertent tripping of an energized transformer by corrosion products causing arching at the transformer bushings.
5 YEARS ¢	f.	Performing a full flow test of the Hydrogen Seal Oil Unit Deluge System using the guidance of NFPA 15-1985.
AS REQ $	g.	Visually inspecting each charcoal filter plenum spray header/nozzles each time the charcoal is changed.

(INTENTIONALLY BLANK)

9A. SYSTEM: $ Low Pressure CO2 Systems

B. OPERABILITY REQUIREMENTS:

$ **Low Pressure CO2 Systems** shall be deemed OPERABLE if the system is lined up to deliver CO2 to the associated hazard area upon the design activation method(s) (automatic and/or manual); shall cause for associated fans to secure and dampers to close as applicable; shall cause for an alarm to be transmitted to the SAS upon activation; and shall have a minimum stored volume of CO2 in the associated storage tank as follows:

- Diesel Generator Systems: Tank P54-A008 / 2600 lbs.
- Control Room Subfloor: Tank P54-A009*/ 375 lbs.
- Process Computer Subfloor: Tank P54-A009 / 300 lbs.
- Turbine Lube Oil Rooms: Tank 1P54-A007 / 1500 lbs.
- Reactor Recirculation Pumps: Tank 1P54-A006 / 2460 lbs.

* The maximum amount of CO2 permitted in Tank P54-A009 is 1000 lbs.

C. APPLICABILITY:

$ Low Pressure CO2 Systems for the **Unit 1 Diesel Generator Rooms** (#) and the **Reactor Recirculation Pumps** shall be OPERABLE whenever the equipment they protect is required Operable.

$ Low Pressure CO2 Systems for the **Unit 1 Control Room Subfloor** (#), **Unit 1 Process Computer Room Subfloor**, and the **Turbine Lube Oil Storage and Purifier Rooms** shall be OPERABLE at all times.

D. ACTIONS ON INOPERABLE:

TIME IF:

$ 1. The **Control Room Subfloor CO2 Systems** will not deliver CO2 to the hazard area upon manual activation or has less than the required volume of CO2 in the storage tank as stated in 9.B above,

THEN:

1 HOUR $ a. Place 4 additional CO2 fire extinguishers within the control room, verify operablity of the CC 654' water hose stations, and if the subfloor area smoke/heat detection is inoperable, establish an additional (other than Control Room personnel) continuous firewatch patrol of the Control Room.

IF:

$ 2. Any of the **Diesel Generator or Turbine Lube Oil Rooms CO2** Systems are inoperable,

THEN:

1 HOUR $ a. Establish an hourly firewatch patrol of the affected area and provide backup fire suppression such as foam equipment or additional CO2 fire extinguishers.

IF:
¢ 3. The **Process Computer Room Subfloor CO2 System** is inoperable,

THEN:
1 HOUR ¢ a. Place two backup portable CO2 fire extinguishers in the area.

IF:
¢ 4. The **Reactor Recirculation Pumps CO2 System** is inoperable,

THEN:
1 HOUR ¢ a.1 Monitor Drywell Temperatures per 2.D.1.b.1 of this Attachment. | TC-1

OR
¢ a.2 With the Plant in OP CON {MODE} 4 or 5 and the Drywell accessible, establish a firewatch inspection of the Recirculation Pump(s) once every 8 hours. | TC-1 | C-4

IF:
¢ 5. **Any CO2 System** is in the Manual-Only Mode, such as the selector valve lock-out switches being activated for personnel safety,

THEN:
1 HOUR ¢ a. Implement the associated firewatch requirements of the system as described above, however no additional extinguishing equipment need be provided.

E. SURVEILLANCE REQUIREMENTS:

NOTE: Heat Detectors which activate CO2 Suppression Systems are tested per Section 1.B of this Attachment.

FREQ. 1. Each Low Pressure CO2 System shall be demonstrated OPERABLE by:

7 DAYS $ a. Verifying that the associated CO2 Storage Tank contains the minimum required volume of CO2 as delineated in Section 9.B of this Attachment.

31 DAYS $ b. Verifying that each valve in the flow path, which if it was not in its proper position would cause for system inoperability, is in its correct position.

18 MTHS $ c. Performing a full functional test of the system using the guidance of NFPA 12-1989 without actual CO2 discharge, to include verification that: all automatic and manual trip activation methods cause for the initiation of a simulated CO2 discharge; all discharge nozzles are clear of obstructions by discharging air through the nozzles; all automatic valves reposition as appropriate; all timer settings are correctly set; all associated ventilation secures as appropriate; verifying all remote/local alarms.

10.A. SYSTEM: ¢ C02 Fire Hose Reels

B. **OPERABILITY REQUIREMENTS:**

¢ C02 Fire Hose Reels shall be deemed OPERABLE when lined up to automatically initiate C02 flow from the storage tank to the hose nozzle upon automatic or manual activation of the system.

C. **APPLICABILITY:**

¢ C02 Fire Hose Reels shall be OPERABLE whenever the equipment in the area they protect is required to be OPERABLE.

D. **ACTIONS ON INOPERABLE:**

TIME		
		IF:
¢	1.	Any **Control Complex** C02 hose reel is inoperable,
		THEN:
1 HOUR ¢	a.1	If the Water hose reels in the area are operable, no action is required.
		OR
¢	a.2	If the Water hose reels in the area are not operable, then place a portable C02 fire extinguisher at each inoperable C02 hose reel.
		IF:
¢	2.	Any **Turbine Power Complex** C02 hose reel is inoperable,
		THEN:
1 HOUR ¢	a.	Verify all permanent plant fire extinguishers are in their designated location within the affected area.

E. **SURVEILLANCE REQUIREMENTS:**

FREQ.		
	1.	Each C02 Hose Reel shall be demonstrated OPERABLE by:
7 DAYS ¢	a.	Verifying that the associated C02 storage tank contains a volume of at least 500 lbs.
18 MTHS ¢	b.	Performing a "puff test" of each hose reel to verify all automatic functions and alarms activate and each reel discharges freely to verify no obstructions.
5 YRS ¢	c.	Performing a hydrostatic hose test per NFPA 12-1989.

11.A. **SYSTEM: $ CO2 Storage Tanks**

B. OPERABILITY REQUIREMENTS:

¢ CO2 Storage Tanks shall be deemed OPERABLE if the tank contains the minimum required capacity of the system(s) it supplies; and has a minimum pressure of 275 psig.

C. APPLICABILITY:

$ CO2 Storage Tanks shall be OPERABLE whenever the system(s) it supplies are required to be OPERABLE.

D. ACTIONS ON INOPERABLE:

TIME **IF:**

$ 1. Any CO2 storage tank is inoperable,

THEN:

1 HOUR $ a. Establish compensatory actions per Sections 9 and 10 of this attachment for the associated system(s):

TANK	SYSTEM(S)
1P54-A006	Reactor Recirculation Pumps
1P54-A007	Turbine Lube Oil Storage/Purifier Rooms Turbine Power Complex CO2 Hose Reels
P54-A008	Diesel Generator Rooms # Control Complex CO2 Hose Reels (Also inops main generator purge)
P54-A009	Control Room Subfloor Systems # Process Computer Room Subfloor

E. SURVEILLANCE REQUIREMENTS:

FREQ. 1. Each CO2 Storage Tank shall be demonstrated OPERABLE by:

7 DAYS $ a. Verifying that the tank storage pressure is greater than or equal to 275 psig and that the tank contains the minimum required volume for the system(s) it supplies.

(INTENTIONALLY BLANK)

12.A. <u>SYSTEM: ¢ Halon Suppression Systems (SB/TSC)</u>

B. OPERABILITY REQUIREMENTS:

¢ Halon Suppression Systems shall be deemed OPERABLE if the system is lined up to automatically initiate a discharge by all of the automatic and manual activation methods; shall cause for associated fans/dampers/louvers to secure as applicable; shall transmit the appropriate alarms to the SAS; and shall have at least one operable storage tank bank.

C. APPLICABILITY:

¢ Halon suppression systems shall be OPERABLE at all times.

D. ACTIONS ON INOPERABLE:

<u>TIME</u> **IF:**

¢ 1. The halon system will not automatically initiate a discharge upon detection or manual pull station activation but at least one string of detection is still operable,

THEN:

¢ a. Ensure the fire brigade is alerted to this fact so in the event of an alarm from the associated system control panel, they can perform manual activation by use of the automan strike button.

IF:

¢ 2. The halon system detection is inoperable,

THEN:

1 HOUR ¢ a. Establish an hourly firewatch patrol of the affected area.

IF:

¢ 3. Both halon storage banks are inoperable,

THEN:

1 HOUR ¢ a. Place two additional portable halon extinguishers in the affected area.

E. SURVEILLANCE REQUIREMENTS:

NOTE: Smoke Detectors which activate Halon Systems are tested per Section 1.B of this Attachment.

<u>**FREQ.**</u> 1. Each halon system shall be demonstrated OPERABLE by:

6 MTHS ¢ a. Verifying that the halon storage tanks are filled to within 5% of the design volume and that the storage tank pressure is within the operating band indicated on the pressure gauge.

12 MTHS ¢ b. Performing a full functional test of the system using the guidance of NFPA 12A-1987, without actual halon discharge, to include verification that all automatic and manual trip activation methods cause for the initiation of a simulated halon discharge; that all timer settings for predischarge times are correctly set; that all associated ventilation fans/dampers/louvers secure as appropriate; verifying the system integrity by visually inspecting piping, hangers, and control panels; and that all remote alarm indications function properly.

(INTENTIONALLY BLANK)

13.A. SYSTEM: ¢ Fuel Oil Pumphouse Foam System

B. **OPERABILITY REQUIREMENTS:**

¢ The Fuel Oil Pumphouse Foam Systems shall be deemed OPERABLE if the system is capable of delivering foam solution to either the fuel oil storage tank and dike area hose reel standpipes upon manual activation of the system; and shall transmit appropriate alarms to the SAS.

C. **APPLICABILITY:**

¢ The Foam System shall be OPERABLE whenever the fuel oil storage tank contains fuel oil and when fuel delivery operations is taking place at the FOPH.

D. **ACTIONS ON INOPERABLE:**

TIME			
	IF:		
¢	1.	The foam system is not capable of delivering foam solution to the tank and hose reels,	
	THEN:		
1 HOUR	¢	a.	Station portable foam equipment in either hydrant house 7 or 8, with a minimum foam concentrate supply of 15 gallons and a foam eductor or rover.
	IF:		
¢	2.	The fuel storage tank detection is inoperable,	
	THEN:		
1 HOUR	¢	a.	Establish an hourly firewatch patrol of the tank area.

E. **SURVEILLANCE REQUIREMENTS:**

FREQ.			
	1.	The Foam System shall be demonstrated OPERABLE by:	
12 MTHS ¢		a.1	Performing a functional test of the system per NFPA 11-1988 including discharge of foam solution through the hose connections.
¢		a.2	Sampling the foam concentrate to verify the specific gravity is acceptable for continued storage; and the foam solution to verify the concentration is appropriate.

14.A. SYSTEM: ¢ Portable Fire Extinguishers

B. OPERABILITY REQUIREMENTS:

¢ Portable Fire Extinguishers shall be deemed OPERABLE if they are at their designated location and accessible; its gauge indicates sufficient pressure; its agent weight or volume is within 5% of fill weight; and that it does not exhibit signs of physical damage or blocked nozzles. Portable extinguishers may be placed on temporary stands near their designated location if work activities would prohibit access to the designated location.

C. APPLICABILITY:

¢ Portable extinguishers shall be OPERABLE at all times.

D. ACTIONS ON INOPERABLE:

TIME		IF:
¢	1.	Any fire extinguisher is inoperable,
		THEN:
24 HRS ¢	a.	Replace the extinguisher with another of equivalent type and size.

E. SURVEILLANCE REQUIREMENTS:

FREQ.		
	1.	Each Portable Fire Extinguisher shall be demonstrated OPERABLE by:
31 DAYS ¢	a.	Performing a "Quick-Check" per NFPA 10-1988 and verifying that it is in its designated location.
12 MTHS ¢	b.	Performing an annual maintenance inspection per NFPA 10-1988.
5 YRS/ 6 YRS/ 12 YRS ¢	c.	Performing hydrostatic testing and internal inspections per NFPA 10-1988.

15.A. **SYSTEM: ¢ Fire Alarm Hand Stations**

B. **OPERABILITY REQUIREMENTS:**

¢ Fire Alarm Hand Stations shall be deemed OPERABLE if they are accessible and transmit the appropriate alarm to the SAS.

C. **APPLICABILITY:**

¢ Fire Alarm Hand Stations shall be OPERABLE at all times.

D. **ACTIONS ON INOPERABLE:**

TIME	**IF:**	
¢	1.	Any hand station is inoperable,
	THEN:	
1 HOUR	¢ a.	Verify portable extinguishers and communications equipment (Gai-Tronics, Telephones) in the area are operable.

E. **SURVEILLANCE REQUIREMENTS:**

FREQ.		
	1.	Each hand station shall be demonstrated OPERABLE by:
12 MTHS ¢	a.	Verifying the hand station will send an alarm to the SAS upon activation and is accessible per NFPA 72-1993.

(INTENTIONALLY BLANK)

16.A. **SYSTEM: $ Fire Rated Assemblies (Doors, Barriers, Dampers)**

B. OPERABILITY REQUIREMENTS:

$ Fire Rated Assemblies shall be deemed OPERABLE if they are capable of performing their applicable required design function(s):

Fire Doors: Unlocked doors shall be closed and latched; locked doors shall be locked closed and latched; all doors shall be free of obstructions and shall close freely and latch without assistance after opening.

Fire Barriers (Walls, Floors, Penetration Seals, Cable Tray and Conduit Wrap, Radiant Energy Shields, Structural Steel Pyrocrete): Shall not be degraded to a point where the design rating could not be maintained.

Fire Dampers: Shall be capable of freely closing upon electric or thermal activation of the fusible link as appropriate, including under air flow (dynamic).

C. APPLICABILITY:

$ All Fire Rated Assemblies shall be OPERABLE at all times.

D. ACTIONS ON INOPERABLE:

TIME IF:

$ 1. # Any Fire Rated Assembly which separates safety related fire areas or portions of redundant safe shutdown circuits/components is inoperable,

THEN:

1 HOUR $ a.1 **IF Operable Detection Exists** on at least one side of the barrier, establish an hourly firewatch patrol on one side of the affected barrier.

OR

1 HOUR $ a.2 **IF Operable Detection Does Not Exist** on at least one side of the barrier, establish a Continuous firewatch on one side of the barrier.

IF:

¢ 2. Any other Fire Rated Assembly is inoperable,

THEN:

1 HOUR ¢ a. Implement compensatory measures as directed by the P54 RSE, who shall use the guidance of various P54 PTI's, this PAP, and other regulatory or reference documents.

E. SURVEILLANCE REQUIREMENTS:

FREQ. 1. Each Fire Door shall be demonstrated OPERABLE by:

24 HRS $ a.1 Verifying that each Appendix R or BTP 9.5-1 unlocked/unsupervised fire door is closed and latched.

$ a.2 Verifying that each magnetically held open fire door is free of obstructions.

7 DAYS $ b.1 Verifying that each Appendix R or BTP 9.5-1 locked fire door is locked closed.

¢ b.2 Verifying that all other fire doors are closed and latched.

6 MTHS $ c.1 Performing a functional test and inspection of all fire doors; to include closures, latches, frame and hardware.

$ c.2 Performing a functional test of all magnetically held open fire doors and their releasing mechanisms.

NOTE: Each electrically supervised fire door is also a security door. These doors are not inspected weekly as locked doors or monthly for the alarm function via fire protection tests. These doors are verified locked closed at least daily by the Security Department; and are tested for the alarm function weekly via the Security Plan. Compensatory measures are required if inoperable.

$ 2. Each Fire Damper installed in an Appendix R or BTP 9.5-1 fire barrier shall be demonstrated OPERABLE by:

18 MTHS $ a. Performing a visual inspection of the fire damper and its associated hardware to include: proper orientation of the fusible link and its suspension; cleanliness of tracks; and proper stacking of blade package.

AS REQ $ b. Performing a static and dynamic functional test following any repair of the damper other than fusible link replacement.

$ 3. Each Appendix R or BTP 9.5-1 Fire Barrier Assembly shall be demonstrated OPERABLE by:

18 MTHS $ a. Performing a visual inspection of all conduit, cable, and cable tray fire wrap. Specific criteria for determining operability shall be as stated in the applicable Periodic Test Instruction.

10% per 18 MTHS $ b. Performing a Visual Inspection of 10 percent of each type of penetration seal. If apparent changes in appearance or abnormal degradation are found, an additional 10 percent of each type of penetration seal shall be inspected. This inspection shall continue until a 10 percent sample with no apparent deficiencies or degradations is found. Specific criteria for determining deficiencies or degradations shall be as stated in the applicable Periodic Test Instruction.

18 MTHS $ c. Performing a visual inspection of all Appendix R and BTP 9.5-1 fire barrier assemblies accessible during unit operation to ensure integrity; to include concrete and gypsum board walls and structural steel pyrocrete. Specific criteria for determining operability shall be as stated in the applicable Periodic Test Instructions.

1 R/O $ d. Performing a visual inspection of all radiant energy heat shields installed on Raceways within the Containment Building. Specific criteria for determining operability shall be as stated in the applicable Periodic Test Instruction.

1 R/O $ e. Performing a visual inspection of all Appendix R and BTP 9.5-1 fire barrier assemblies inaccessible during unit operation to ensure integrity; to include concrete and gypsum board walls and structural steel pyrocrete. Specific criteria for determining operability shall be as stated in the applicable Periodic Test Instructions.

(INTENTIONALLY BLANK)

17.A. <u>SYSTEM: ¢ Fire and Security System Computer and Peripherals</u>

B. **OPERABILITY REQUIREMENTS:**

¢ The Fire and Security System Computer and Peripherals shall be deemed OPERABLE if at least one monitoring loop per channel exists, and the system is capable of displaying an alarm for each device on the system.

C. **APPLICABILITY:**

¢ This system shall be OPERABLE at all times.

D. **ACTIONS ON INOPERABLE:**

<u>TIME</u> IF:

$ 1. The Computer system will not annunciate an alarm or supervisory trouble of a sprinkler or CO2 system pressure switch, fire pump alarm, or detection or suppression system control panel, and the device or panel itself is known to be operable,

THEN:

1 HOUR $ a. Establish an hourly firewatch patrol of the affected device or control panel to verify no alarm or abnormal conditions exist.

E. **SURVEILLANCE REQUIREMENTS:**

N/A. The Fire and Security Computer System in itself does not require any individual surveillances. The system is tested during field testing of its inputs devices. An "Off-Normal" report is generated at the beginning of each SAS shift to verify status of the system.

(INTENTIONALLY BLANK)

18.A. SYSTEM: $ Appendix R Emergency Lighting

B. **OPERABILITY REQUIREMENTS:**

$ Each Appendix R Emergency Lighting Unit identified in Table 9A.3-2 of the USAR shall be deemed OPERABLE if it automatically provides area lighting for a period of 8 hours upon loss of AC power and is properly orientated to provide illumination of/to the required area.

C. **APPLICABILITY:**

$ Each Appendix R Emergency Lighting Unit shall be OPERABLE At All Times.

D. **ACTIONS ON INOPERABLE:**

TIME **IF:**

1. Any Appendix R Emergency Lighthing Unit is inoperable,

THEN:

1 HOUR $ a. Generate a FI/BRP in order to provide documentation and Notify the Unit Supervisor to ensure that suitable sealed beam battery powered portable hand lights are available for response to the area.

NOTE: It is not necessary to generate a FI/BRP during the recharge period following performance of the battery discharge test.

E. **SURVEILLANCE REQUIREMENTS:**

12 MTHS $ 1. Perform the following:

a. Interrupt AC Power to each battery unit and verify that each lamp is ON after 8 hours of operation.

b. Verify that each lamp is ON after loss of AC power.

c. Verify that each lamp is properly orientated to provide lighting to/of the affected area.

19.A. **SYSTEM: ¢ Owner-Controlled Area Fire Protection Systems**

B. **OPERABILITY REQUIREMENTS:**
¢ OCA Fire Detection and Suppression systems shall be deemed OPERABLE if they are capable of performing their intended function.

The OCA systems shall include all systems outside of the Protected Area;, including the TEC/EOF.

C. **APPLICABILITY:**
All OCA fire systems should be OPERABLE at all times.

D. **ACTIONS ON INOPERABLE:**

NOTE: OCA Fire System Operability, is as determined by the P54 RSE. The actions required by this section are conservative and should be immediately implemented if the P54 RSE cannot be contacted. The P54 RSE may authorize compensatory measures which deviate from those suggested below.

TIME **IF:**
1. Any OCA sprinkler, halon, or CO2 system is inoperable,

THEN:
1 HOUR a. Establish an hourly firewatch of the affected area.

IF:
2. Any OCA building standpipes or hose reels are inoperable,

THEN:
a. Ensure the fire brigade is aware of the impairment.

IF:
3. Any OCA Fire System will not transmit an alarm via the OCA Fire Alarm System,

THEN:
a. Establish an hourly firewatch patrol of the affected component to ensure no abnormal conditions exist.

E. **SURVEILLANCE REQUIREMENTS:**
All OCA Fire Detection and Suppression Systems should be tested in accordance with the applicable NFPA Codes and Standards, or plant system equivalent requirements of this PAP. The systems are tested utilizing written instructions, guidelines, or listings prepared in accordance with accepted industrial practices; or by outside vendor service organizations, and are tracked through the Repetitive Task Program to ensure regularly scheduled performances. TC-2