Name: ________________________________

1. 007 EK2.02 001

Given the following plant conditions:

- The Plant is at 100% RTP.
- The reactor trips due to a RCS low flow condition.
- Generator lockout occurs immediately.
- Approximately 5 seconds after the generator lockout a fault on 4KV Bus 4 causes voltage on 480V Bus E-2 to drop to 290 VAC.
- Approximately 7 seconds after the generator lockout breaker 52/19, 4KV BUS 3-4 TIE BKR, trips open on overcurrent.

Which ONE (1) of the following identifies the current alignment of the Emergency buses?

A. 480V Bus E-1 is being powered via "A" EDG..
480V Bus E-2 is being powered via 4KV Bus 3

B. 480V Bus E-1 is being powered via 4KV Bus 2.
480V Bus E-2 is being powered via 4KV Bus 3

C. 480V Bus E-1 is being powered via "A" EDG.
480V Bus E-2 is being powered via "B" EDG.

D.✓ 480V Bus E-1 is being powered via 4KV Bus 2.
480V Bus E-2 is being powered via "B" EDG.

The correct answer is D.

A. Incorrect - 4KV Bus 3 will be energized by the SUT. Plausible if the students thinks that breaker 52/19 supplies power to the 4KV Bus 3. The Emergency bus lineup is correct.

B. Incorrect - The 4KV lineup is correct. 480V E-2 will be supplied power via "B" EDG. Breaker 52/28B will trip open on undervoltage at 328 VAC. In the stem of the question the voltage on E-2 drops to 290 VAC.

C. Incorrect - 4KV lineup is correct. The 480V E bus lineup is incorrect. 480V E-1 will be supplied power via the SUT. The student may think that since a SI was initiated that "A" EDG will energize E-1 since the "A" EDG is started on the SI sequencer.

D. Correct. 4KV lineup is correct. 4KV Buses 1, 2 and 3 will be powered from the SUT due to the fast, dead bus transfer. 4KV Buses 4 and 5 will be deenergized due to breaker 52/19 tripping open. 480V E-1 will remain energized from the SUT via 4KV Bus 2. 480V E-2 will be powered via "B" EDG. Breaker 52/28B will trip open on undervoltage at 290 VAC. In the stem of the question the voltage on E-2 drops to 320 Vac.

Question 1
Tier 1 / Group 1
K/A Importance Rating - RO 2.6 SRO 2.8

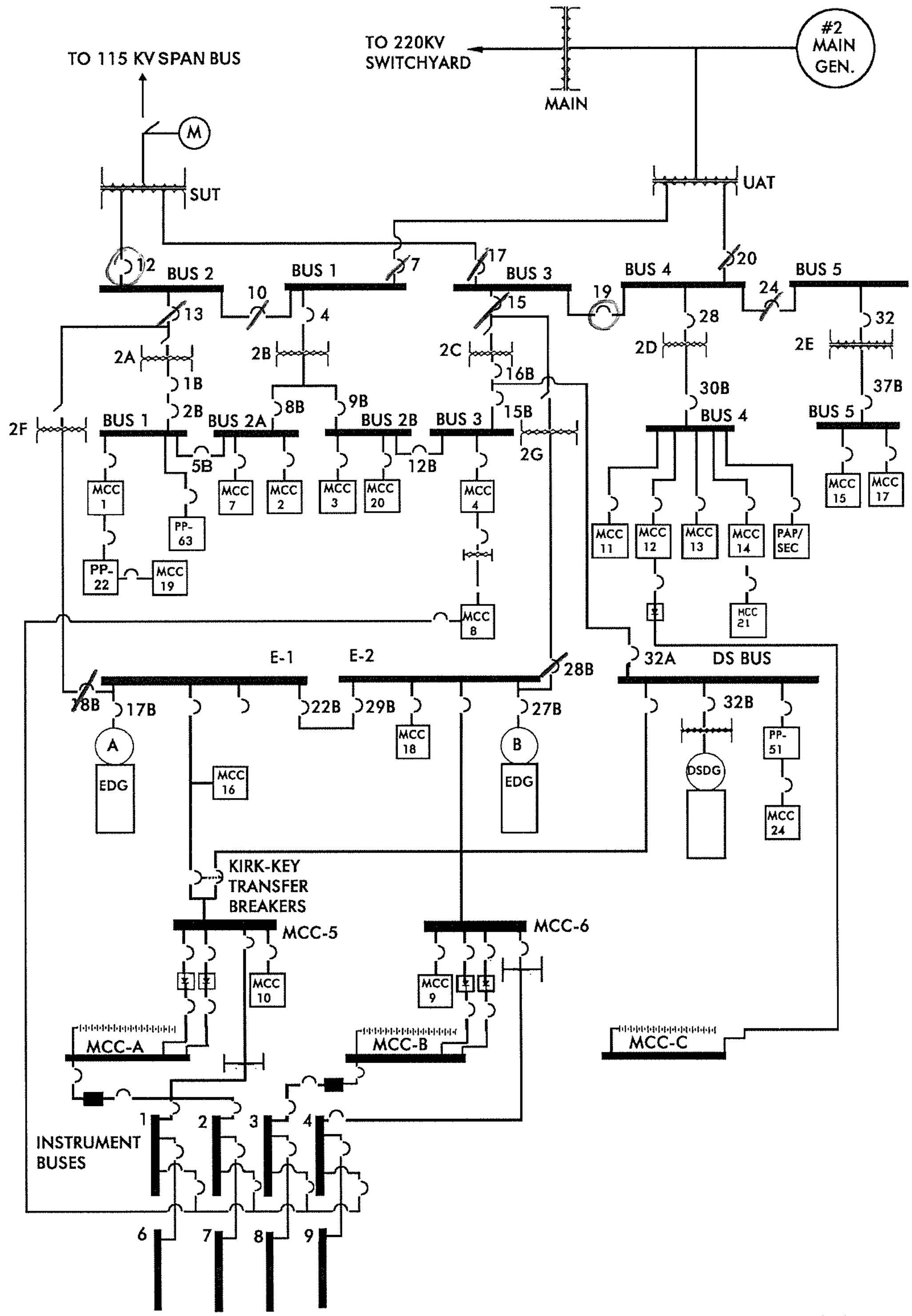
Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects

Reference(s) - Sim/Plant design, System Description, EPP-4
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-4-004
Question Source - NEW
Question History - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.7
Comments - Related to the March 28, 2010 event. K/A match because the candidate must know the effect of a generator lockout, emergency bus undervoltage condition and trip of normal supply breaker to a 4 KV bus. This will test the candidates knowledge of the electric plant arrangement and emergency power supplies.

① Normal AT-Power Lineup

ELECTRICAL DISTRIBUTION

KVAC-FIGURE-2

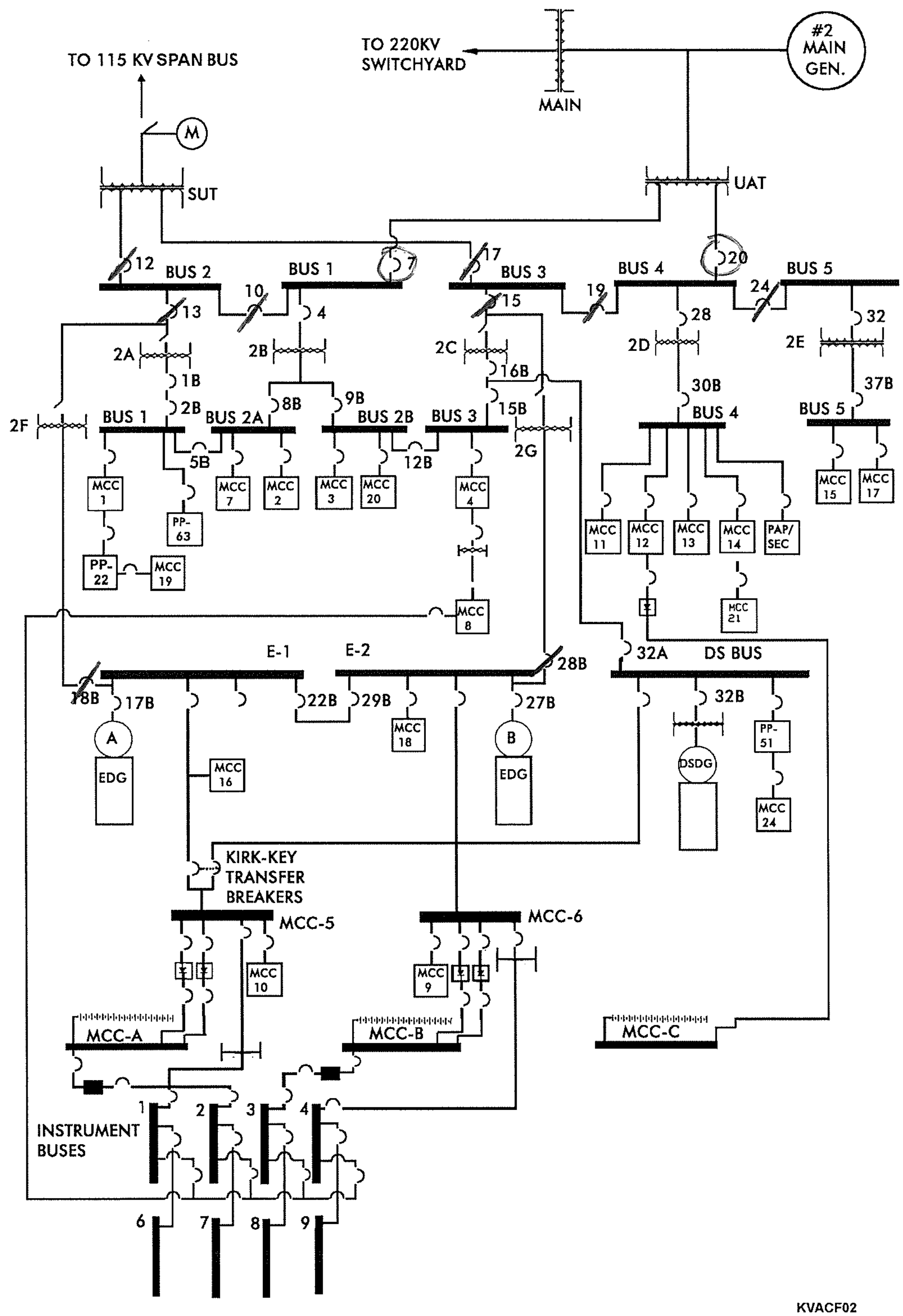


KVACF02

(2) POST FAST, DEAD Bus Transfer

ELECTRICAL DISTRIBUTION

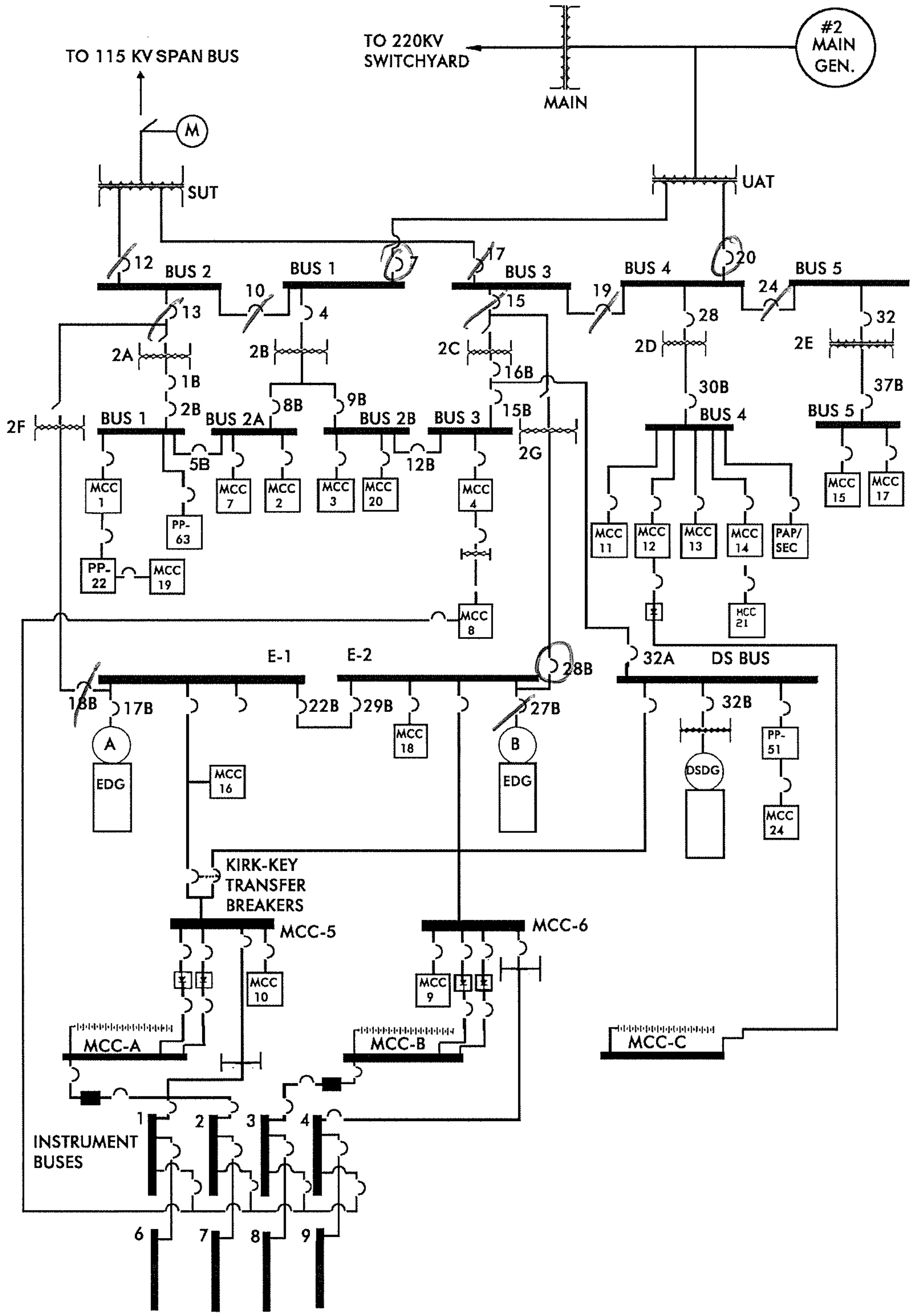
KVAC-FIGURE-2



③ POST UNDERVOLTAGE ON E-2

ELECTRICAL DISTRIBUTION

KVAC-FIGURE-2

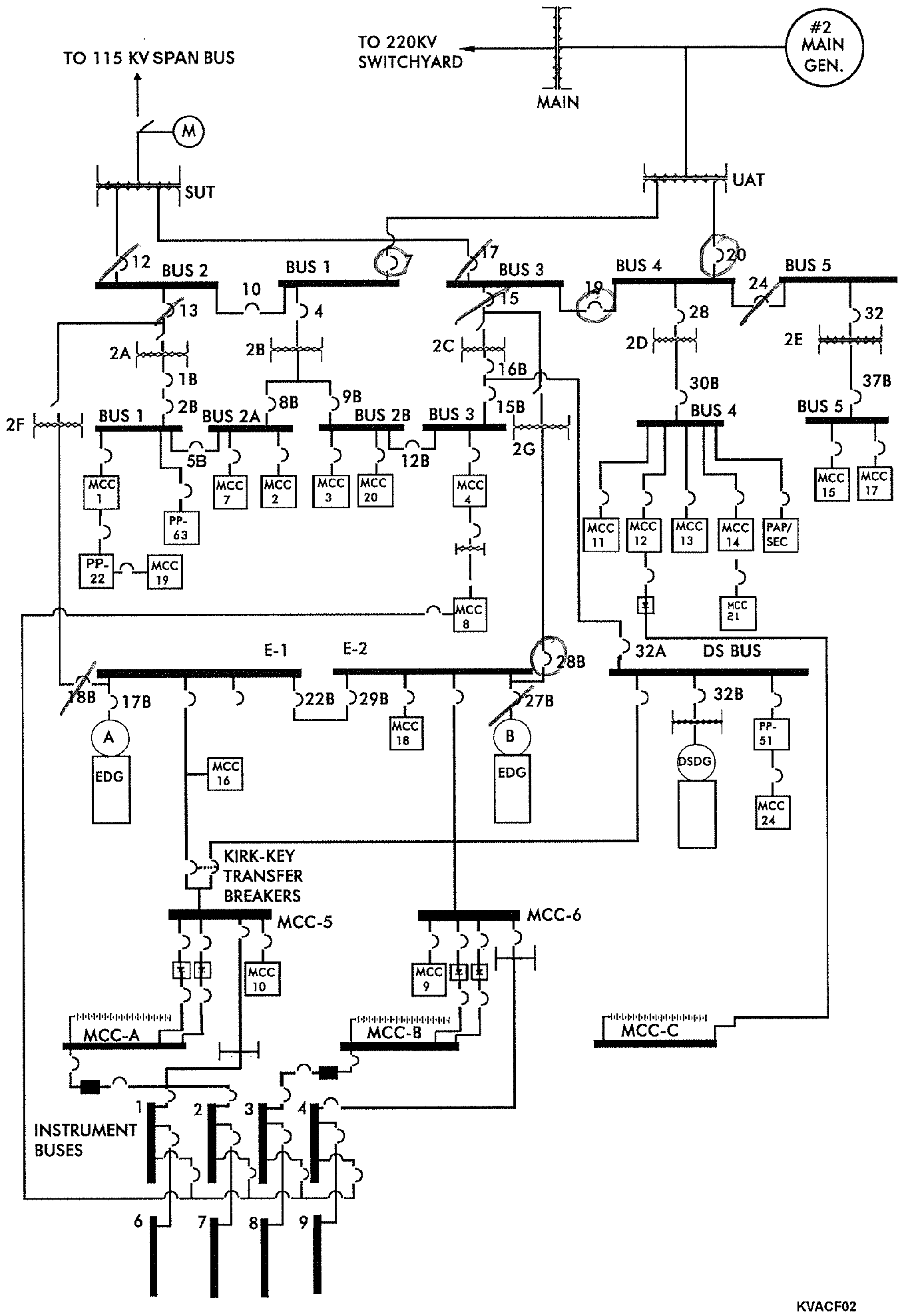


KVACF02

(4) POST 52/19 TRIP

ELECTRICAL DISTRIBUTION

KVAC-FIGURE-2



stored-energy type and uses a gear motor to charge a closing spring. Located on the ground floor southeast corner of the turbine building.

These Stored Energy Closing Mechanisms are charged automatically once the breakers are fully racked-in and DC control power is present. They are also recharged once the breakers are closed and DC control power is still present. If DC control power is lost, the Stored Energy Closing Mechanisms can be manually charged. The breakers can be manually closed and opened via the manual close and trip buttons on the breaker.

These breakers also have DC control circuits that provide anti-pumping of the breakers; i.e., if the breaker receives a close signal and then an open signal, the breaker will close and then open and will not re-close as long as the close signal remains. The breaker cannot be closed again until the control circuit resets. The control circuit will reset itself only after the close signal is removed.

These medium voltage circuit breakers have auxiliary contacts that change state with the open or closed position of the breaker main contacts. These auxiliary contacts are used in the breakers control circuit and to supply signals/ input to other control circuits.

4.0 INSTRUMENTATION

4.1 230KV Instrumentation and Control

Instrumentation and control of the 230KV transmission equipment is located on relay panels in the 230KV Switchyard Control Building. Controls for the unit North and South 230KV breakers (52/8 & 9) and indicators for the North and South 230KV bus voltages are located on the RTGB. To prevent inadvertent opening operation of 52/8 and 52/9 they are interlocked with the THINK button. Also interlocked with these two breakers is the exciter field breaker. The two OCBs must be open before the exciter field breaker can be opened.

The CP&L Skaale Energy Control Center (ECC) has parallel instrumentation and control of the 230KV switchyard via a supervisory control and data acquisition system. There is no remote control of the unit North and South 230KV breakers at Skaale ECC.

A synchroscope, synchroscope selector switch, and percent difference meter are provided on the RTGB (see Figure 7) to aid operators during operation of unit related equipment. Synchroscope incoming and running voltmeters and individual breaker synchroscope selector switches for the 230KV transmission equipment are located in the 230KV Switchyard Building to aid substation operators in OCB operation.

All 230KV breakers have counters associated with them to record operations. Each complete cycle of the OCB causes the counter to register a whole number in the counter

window. The main line breakers have automatic reclosers. The tiebreakers will remain open if they are tripped automatically.

Breaker status is provided on the ERFIS Computer.

A carrier relay system is used for transmission line protection. It utilizes a high frequency signal that is superimposed on the 60-Hertz transmission line. On a line fault the carrier relays, along with relays associated with CTs and PTs, provide line protection and breaker failure protection on the transmission lines.

The carrier system also acts to prevent overreach, in that only the necessary breakers open to isolate the fault and prevents OCBs further from the fault from opening.

The PTs and CTs are also utilized to obtain indications of load and voltage on each transmission line and to operate the synchroscope.
Breaker failure cut-off switches are provided on the respective relay panels in the 230KV Switchyard Building for each 230KV breaker. These switches are utilized to disable the breaker failure circuits for any breaker where accidental tripping could result during testing or modifications of switchyard relaying circuits.

Breaker failure lockout relays are provided on each 230KV breaker panel in the Switchyard Building to trip and block reclosing of the adjacent breakers of any failed breaker.

An oscillograph is located in the 230KV Switchyard Building to monitor the transmission system. If an alarm condition exists, the oscillograph will actuate causing an alarm on the 230KV annunciator panel located in the 230KV Switchyard Building. A digital fault recording system can be accessed remotely and analyzed via computer. A primary generator lockout relay (86P) and backup generator lockout relay (86BU) is provided on the Generator Protective Relay Panel located in the Unit 2 Control Room.

The 86P lockout is actuated by: (see Figure 8)

- Generator phase differential (87/G)
- Voltage controlled overcurrent relay (51-27)
- Generator ground (59N)
- Main transformer fault pressure (Any of the 3) (63/T1, 63/T2, 63/T3)
- Turbine exhaust hood hi-temp. (Above 225°F for 5 minutes)
- UAT phase differential
- UAT fault pressure (63/UT)
- Turbine trip (loss of auto stop oil pressure with 1 minute time delay)
- OCB 52/8 failure to open
- OCB 52/9 failure to open

The 86BU lockout is actuated by:

- OCB 52/8 failure to open
- OCB 52/9 failure to open
- Unit (main transformer) differential (87/GT)
- Negative sequence (46)
- Loss of (exciter) field (40)
- Turbine trip (governor/stop valves closed with 1 minute time delay)

The generator lockouts all cause a turbine trip to occur and shift the auxiliary load from the UAT to the SUT, open 52/8 & 52/9, and open the exciter field breaker. (See Figure 3)

There are two 230/115KV autotransformers used to connect the Unit 2 - 230KV switchyard to the Unit 1 - 115KV switchyard. #1 North autotransformer is connected between the North 230KV Bus and West 115KV Bus and #2 South autotransformer is connected between the South 230KV Bus and East 115KV Bus. The #1 Autotransformer is supplied from OCBs 52/5 and 52/6. The #2 Autotransformer is supplied from OCBs 52/1 and 52/2. Each transformer is equipped with a lockout relay. Operation of either relay will result in the loss of its respective transformer thereby severing that tie between switchyards.

There are two other lockouts. They are associated with the North and South 230KV buses. If the bus phase differential relay opens, it will open the OCBs adjacent to the respective bus. These lockouts are also activated by breaker-failure relays on each breaker connected to the bus.

4.2 4160V Instrumentation and Controls (See Figure 7)

The 4160V control instrumentation is located on the RTGB. The protective relays are located in their respective buses on the panel fronts. Protective relays for 4160V breakers that operate motors are located on the breaker front panels.

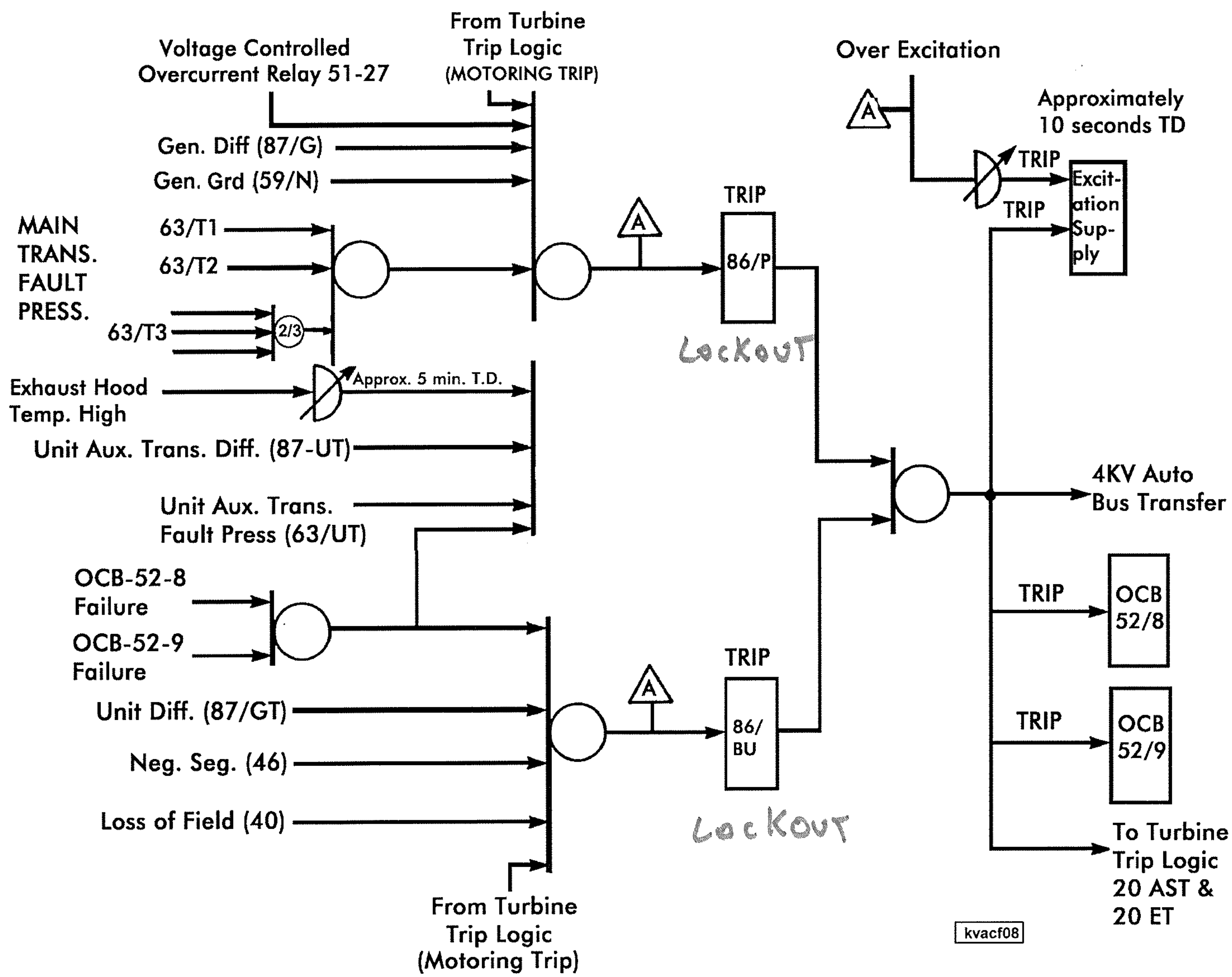
Instrumentation on the RTGB includes SUT amps (0-8000) and UAT amps (0-1200), 4160 bus voltage (0-5KVac) for buses 1, 2, 3, and 4 through selector switches.

The controls include indicating lights for breaker position, open/close control switches and synchroscope with key-operated switches, so that the operator can insure that the buses are in-phase before a breaker operation. For opening any line or tiebreaker except 52/24(bus 5 feeder) the THINK button must be depressed too.

The white "S/U transformer energized" light on RTGB indicates voltage on SUT. Power for the light is from breaker DC control power. Failure of this light could indicate failure of the relay necessary to initiate "Auto-Bus Transfer" and should be

GENERATOR TRIP LOGIC

KVAC-FIGURE-8



During a normal plant shutdown, the auxiliary load will be transferred to the SUT by using OP-603 (Sect. 7.1). Operators may manually transfer loads from the UAT to the SUT. This operation is a live bus, make-before-break transfer that automatically opens UAT supply breakers 52/7 and 52/20 after breakers 52/12 (SUT supply) and 52/19 (4KV bus 3-4 tie breaker) are manually closed.

6.2 Opening of Generator Output Breakers

If both generator output breakers (52/8 and 52/9) are opened manually, the UAT incoming line breakers to 4.16 KV Buses No. 1 and 4 (52/7 and 52/20) will open automatically, and then the SUT incoming line breaker (52/12) to 4.16 KV bus No. 2 and the 4.16 KV Bus No. 3 and 4 tie breaker (52/19) will automatically close. This operation is a dead bus transfer that first opens UAT supply breakers 52/7 & 52/20 and then closes breakers 52/12 & 52/19.

6.3 Reactor/Turbine Trip (See Figure 8)

A reactor and/or turbine trip actuation (with or without a concurrent SI signal) will initiate a fast, dead bus transfer of loads from the UAT to the SUT after a 60 second time delay (provided at least 1 generator breaker, 52/8 or 52/9, is closed during the 60 second period).
The transfer operation automatically trips breakers 52/7 and 52/20 while breakers 52/12 & 52/19 are simultaneously closed. The transfer is initiated by tripping either lockout relay 86P or 86BU (which will occur after the 60 second delay).

For a unit trip with a concurrent SI signal, the main generator is connected to the system (thereby acting as a synchronous motor) for 60 seconds until the fast dead bus transfer occurs. During the 60-second time delay, the steam generator feedwater pumps trip and the safeguards sequencing logic is initiated. The last safeguards load is sequenced on at approx. T = 39.5 seconds. At T = 60 seconds, the generator lockout relays (86P and/or 86BU) actuate with a resultant fast dead bus transfer to the SUT. Note that during the 60-second sequencing period, E1 is being powered from the UAT (with the main generator motoring) while E2 is already being powered from the SUT. For a unit trip without a concurrent SI signal, a similar scenario to that described above will take place except that the sequencing of safeguards loads will not occur.

6.4 Grid Instability Conditions (OMM-001-2)

Response to Grid System Alert Conditions

In the event that grid system conditions are severely challenged, the load dispatcher will declare either a System Contingency Alert or a System Reliability Alert. Based on the alert condition declared, plant work items will be allowed to continue, will be delayed, or will be rescheduled.

Control power is supplied to 480Vac buses by 125Vdc.

125Vdc MCC-A supplies Distribution Panel A for Buses E-1, 1, and 2A; 125Vdc MCC-B supplies Distribution Panel B for Buses E-2, 2B, and 3; 125Vdc MCC-(B-A) for Bus 4; Bus 5 uses AC control power from Bus 5.

Operation of the 480Vac DS breaker 52/32A for open and close is at the DS Bus and at the diesel panel. It will auto open for SI and UV on E-2 or UV on the DS Bus. The control power for its close and trip circuit is from DS Distribution Panel B. An UV trip attachment receives power from a potential transformer (PT) on the supply side of 52/32A

Operation of the generator breaker 52/32B open and close is at the DS Bus and at both diesel panels. Synchronization can only be done at the 4 KV Switchgear Room Panel. The control power for its trip circuit is from DS Distribution Panel A. The control power for its close circuit is from a PT from the DSDG. The charging springs for breakers 52/32A and 52/32B auto charge after they have discharged and control power is still available.

If breaker 52/32A trips, annunciation is provided in the control room via APP-036-H6. However, if breaker 52/32B trips, or if the control switch located in the diesel enclosure is placed in the "PULL-TO-LOCK" position, there is no direct indication in the control room.

The DS Bus breakers 33B, 33C, 33D & 34B get their AC control power from the bus through a CPT (Control Power Transformer). Breakers 34C & 34D are manual.

Control power is 120Vac for load breakers except MCC-5 and DS PP-51.

5.2 480Vac Bus Protection

The 480Vac bus main supply and tie breakers will trip on undervoltage at ≈70% of nominal voltage (328Vac) and overcurrent, long and short time delays.

Current limiters are provided on Buses E1 and E2 for the safety related loads. These devices are sized to open in the event that the fault current (short-circuit current) exceeds the interrupt capability of the DB-50/75 breakers. This ensures the DB-50/75 breakers do not rupture and damage other equipment on E1/E2. The current limiting devices are monitored by an indicating fuse/microswitch (KAZ Bussman) assembly; this assembly will trip the affected breaker and provide a visual indication of an open current limiter. The condition of the KAZ device can be observed through the Plexiglas window on the

rear of each breaker cubicle on E1/E2. If a current limiter fuse opens, the breaker would trip open and would not close until the current limiter fuse was replaced by Electrical Maintenance.

5.2.1 Bus Breaker Interlocks (See Figure 21)

The interlocks are to prevent a bus being powered from two separate sources. The one exception is when the EDG output breakers are closed.

The emergency bus interlocks for the incoming breaker 52/18B or 52/28B can't be closed unless the associated breaker 52/13 or 52/15 is open <u>or</u> associated EDG breaker is open. For breaker 52/13 or 52/15 they can't be closed unless the associated breaker 52/18B or 52/28B is open <u>or</u> associated EDG breaker is open. There is no interlock associated with the RTGB operation of the EDG output breakers (52/17B and 52/27B).

5.2.2 Undervoltage Protection

If undervoltage occurs on one or both emergency buses the bus will isolate by opening and locking out normal supply and tie breakers. Load shedding will occur with the exception of MCC-5, -10, and -16 for E1 and MCC-6, -9, and -18 for E2. The EDG will start and sequence loads back on to the emergency bus.

5.2.3 Degraded Grid Protection

This protection is needed because the setpoints of the bus undervoltage relays are different from the degraded grid undervoltage relays. Degraded grid undervoltage relays can be defeated by use of a key switch. In the NORMAL position protection is provided for a degraded system voltage with a 10-second time delay and a 430Vac setpoint. An orange light at the switchgear indicates a degraded grid condition exists. In the DEFEAT position, degraded grid undervoltage is bypassed. This position is used to prevent blackout protection actuation on start of an RCP. The switches are located at the emergency bus cubicles. Switch is key operated and the key cannot be removed in the DEFEAT position. An alarm DEGRADED GRID E-1 (E-2) PROT BYPD on APP-010-F5 (F6) notifies the CR when the key is in the DEFEAT position, bypassing the degraded grid protection. Improved Technical Specifications prohibits defeating degraded grid protection, for starting RCPs, when in Mode 1 (ITS 3.3.5).

6.0 SYSTEM OPERATION

6.1 Normal Operation

2. 008 AK2.03 001

Given the following plant conditions:

- The plant is in Mode 5.
- RCS Temperature: 170°F and stable.
- RCS Pressure: 350 psig and stable.
- PT-501, PZR Pressure, has failed low

Subsequently a loss of MCC-6 occurs resulting in the following:

- PZR PORV PCV-455C fails open.
- PT-501, PZR Pressure, fails low.

Which ONE (1) of the following actions are required IAW AOP-020, Loss of Residual Heat Removal (Shutdown Cooling) to address the Vapor Space Accident?

A. Close PCV-455C using its RTGB control switch.

B. Close RC-536, PORV Block Valve, using the RTGB control switch.

C✓ Place the associated LTOPP Arming Switch to the Normal position.

D. Take Manual Control of PC-444J and Reduce the Controller Output to Zero.

The correct answer is C

A. Incorrect. Plausible since closing PCV-455C would eliminate the release path. However, PCV-455C will have an open signal from the control circuit that will keep it in the open position as long as the LTOPP Arming Switch remains in the LOW PRESSURE position.

B. Incorrect. Plausible since closing RC-536 will isolate the PORV. However, RC-536 does not have power due to the loss of MCC-6.

C. Correct. Placing the LTOPP Arming Switch to the Normal position is directed in AOP-020 since it will remove the erroneous signal from the LTOPP controllers and cause PCV-455C to close.

D. Incorrect. Plausible since this action is taken in AOP-019, Malfunction of RCS Pressure Control, to manually control RCS pressure. In this case taking manually control of PC-444J will have no effect on PCV-455C.

Question 2
Tier 1 / Group 1
K/A Importance Rating - RO 2.5 SRO 2.4

Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Controllers and positioners

Reference(s) - Sim/Plant design, AOP-020, AOP-020 Basis Document, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-020-003
Question Source - NEW
Question History - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.7
Comments - K/A match since candidate must know that a loss of MCC-6 will cause an trip single to the PORV via the LTOPP controllers.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides the instructions necessary to mitigate the loss of RHR in all conditions for which RHR can be aligned to provide shutdown cooling. This includes loss of RHR cooling for reasons such as RCS leakage, loss of power, loss of Service Water or Component Cooling Water, RHR pump cavitation, and inadequate RHR flow or abnormal reductions in RHR cooling.

This procedure is applicable in Modes 4, 5, and 6 when fuel is in the vessel.

2. ENTRY CONDITIONS

Direct entry from any condition resulting in a loss of RHR pump(s), RHR pump cavitation, abnormal RHR flow or temperature control, or excessive loss of RCS inventory while RHR is aligned for shutdown cooling.

As directed by the following other procedures:

- AOP-005, Radiation Monitoring System, when a low level in the SFP exists due to an RCS leak with the SFP GATE VALVE open.
- AOP-014, Component Cooling Water System Malfunction, resulting in stopping of the RHR Pumps while in CSD.
- AOP-016, Excessive Primary Plant Leakage, if less than 200°F and leakage exceeds Charging Capacity.
- AOP-017, Loss Of Instrument Air, if the loss of Instrument Air has affected core cooling while on RHR.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Check RCS Level - LESS THAN -72 INCHES (69% FULL RANGE RVLIS)	IF RCS Level becomes less than -72 inches (69% FULL RANGE RVLIS), THEN verify BOTH RHR Pumps stopped. Go To Step 3.
2.	Verify BOTH RHR Pumps - STOPPED	
3.	Make PA Announcement For Procedure Entry	

NOTE

FRP-S.1 is NOT applicable for this event unless directed by the CSFSTs.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	From The RTGB, Verify Reactor Tripped As Follows: • REACTOR TRIP MAIN AND BYP - OPEN • Rod Position indication - ZERO • Rod Bottom lights - ILLUMINATED	IF the reactor does NOT trip, THEN dispatch an Operator to the Rod Drive MG Set Room to Open REACTOR TRIP BREAKERS A AND B.
5.	Check RCS Level - DECREASING: • Pressurizer level OR • RCS loop standpipe level OR • RVLIS OR • Refueling Cavity Watch report	IF either PZR PORV is failed open due to loss of input from PT-500 OR PT-501, THEN place the associated LTOPP Arming Switch to the NORMAL position. IF the event does NOT involve a loss of Inventory, THEN Go To Section E, Loss Of RHR Flow Or Temperature Control. IF RHR Pumps have been stopped due to loss of Inventory, THEN Go To Step 6.

5 This step checks the appropriate RCS level indication to determine if RCS inventory is decreasing. If RCS level is not decreasing, the RNO provides transition to address loss of RHR flow or cooling in Section E. The step give a variety of indications to allow for a variety of initial plant conditions. The RNO provides actions should a loss of power have occurred which causes a loss of the pressure transmitters that feed LTOPP. On a loss of power to these transmitters the associated PORV will fail open. If the loss of power also has deenergized MCC-6 the only easy to close the valves to place the LTOPP switches in the normal position.

In order to assure correct transition, if the pumps have already been tripped due to loss of inventory and this trip caused the inventory loss to stop, the RNO provides a transition to the subsequent step.

This step is not in the ARG and has been added here because the RNP procedure is designed for all losses of RHR rather than only those occurring while in Mid-Loop.

6 Letdown flow is verified isolated in this step to ensure inventory loss from the RCS or RHR System is only from the leak. This aids in conserving coolant volume in the system(s).

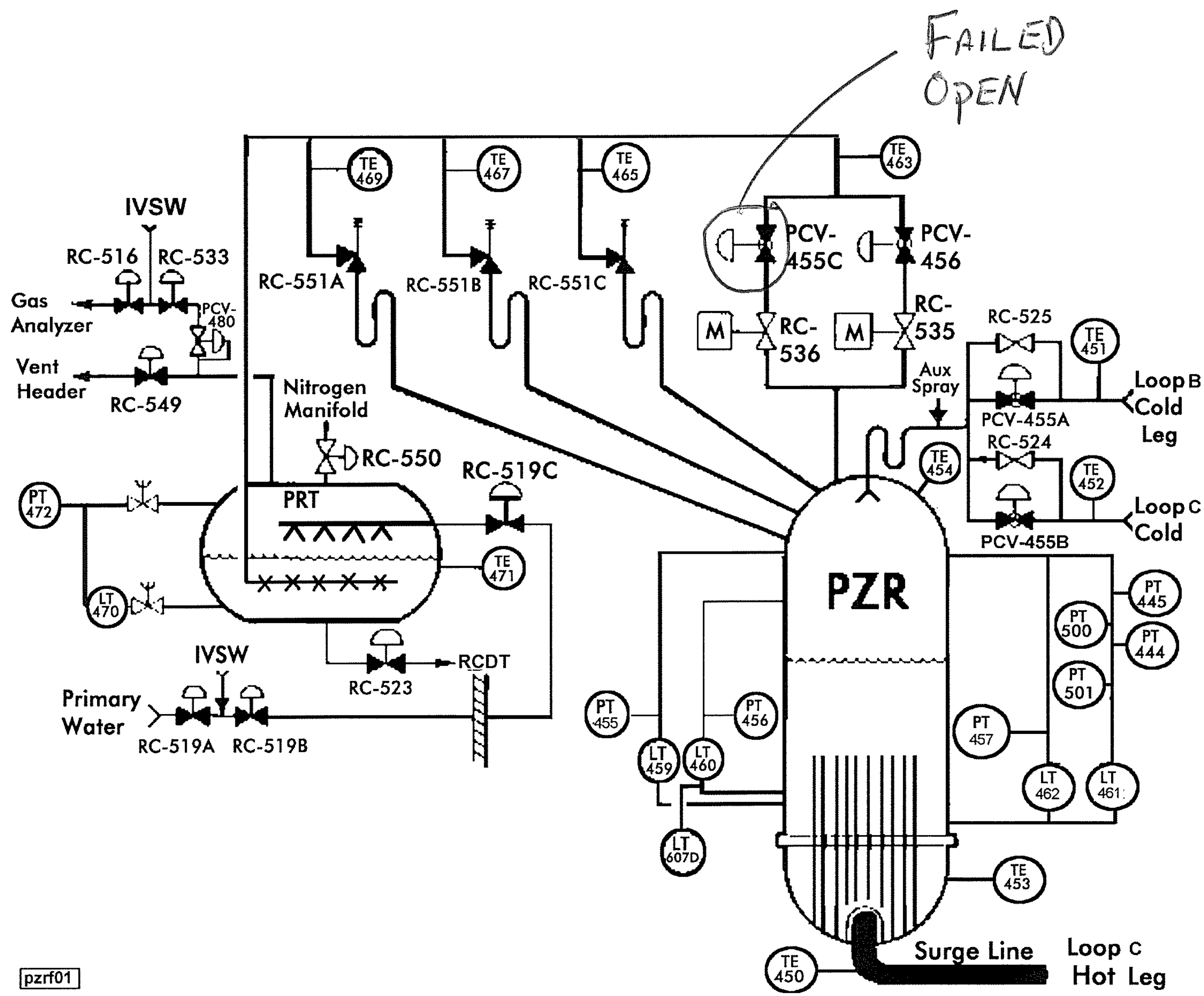
This step corresponds to ARG-2 step 2 and there are essentially no differences.

7 This step determines the current status of the Charging Pumps. If a pump is already running, the subsequent step to verify alignment of the suction supply is not required.

8 This step aligns charging flow to take a suction from the appropriate source and verifies a flow path of the discharge to the RCS. This may not always be the case during outage conditions. The RNO of this step incorporates the caution from ARG-1 (ARG-1, Step 7 CAUTION 2). This caution states that only borated water should be added to the RCS. The ARG caution could be interpreted to perform an action, therefore it is added as a step at this section of the procedure to assure that reactivity effects are considered when establishing makeup flow to the RCS to prevent the loss of core shutdown margin. This is of particular interest at reduced inventory conditions when additions of equal quantities of water would have a greater effect due to less inventory being diluted.

SYSTEM SIMPLIFIED DIAGRAM

PZR-FIGURE-1



INFORMATION USE ONLY

5.1.2 PZR Pressure Control Setpoint (PZR-Figure 8)

1.	PZR Pressure Controller (PC-444A)	
	Proportional gain	2
	Reset time constant	12 sec
	Rate time constant	off
	Pressure set point, Pref	2235 psig
2.	Spray Valve Controllers (PC-444C, PC-444D)	
	Proportional gain in % spray valve Lift per psi	2%/psi
	Set point where spray is initiated on compensated pressure signal from PC-444A	+25 psi (2260 psig)
	Setpoint where spray is full open	+75 psi (2310 psig)
3.	Variable Heater Controller	
	Proportional gain in % heating power	-3.33%/psi
	Set point where proportional heating is always full on, on signal from PC-444A	-15 psi (2220 psig)
	Setpoint where proportional heating is always full off	+15 psi (2250 psig)
4.	Power Relief Valve, PCV-455C operating on compensated pressure signal from PC-444A to PC-444B	+100 psi (2335 psig)
5.	Back-up heaters turned on, due to compensated pressure signal from PC-444A to PC-444F	-25 psi (2210 psig)
	Back-up heaters turned off	-15 psi (2220 psig)
6.	Power Relief Valve (PCV-456) actuated from actual pressure (PC-445A)	2335 psig

5.1.3 PZR PORV Control (PZR-Figure 7 & PZR-Figure 11)

The PZR PORVs have two modes of control, Normal and Low Temperature Overpressure Protection (LTOPP). In normal mode the PORVs have a permissive of 2000 psig to open

in Automatic. This permissive is supplied by the protection channels meeting a 2/3 logic. As stated before PCV-456 receives its signal from PT-445 set at 2335 psig and PCV-455C receives its signal from PC-444A at +100 psi which is nominally 2335 psig also. When the key switch for OVERPRESSURE PROTECTION on the RTGB is place in the LOW PRESSURE position (one switch for each PORV) the input to each PORV is shifted to the LTOPP controller.

5.1.4 Low Temperature Overpressure Protection Control (LTOPP) (PZR-Figure 11)

LTOPP control is required to be activated when the RCS is cooled down below 360 F to minimize Pressurized Thermal Shock (PTS) concerns. The LTOPP controller uses the lowest of TE-410, TE-420 and TE-430 to determine RCS temperature and pressure as sensed by PT-500 and PT-501. The lift setpoint is variable based upon auctioneered low RCS temperature. At an RCS temperature of 360°F, the pressure setpoint is 400 psig. The setpoint of the Comparators PC502 and PC503 are increased as RCS temperature is increased. The setpoint will not decrease below 400 psig.

There is one alarm associated with each channel of LTOPP. It actuates for 3 reasons: (1) RCS temperature is <360°F and LTOPP is not selected on the key switch for OVERPRESSURE PROTECTION, (2) The PORV has received an actuation signal based upon current pressure and temperature or (3) the associated Block valve is shut.

5.1.5 PZR Level Control (PZR-Figure 10)

PZR level is controlled by controlling charging pump speed. The level is programmed to ramp up as Tavg increases by LC-459G. This maintains approximately constant mass in the RCS as Tavg is increased and the coolant in the RCS expands. Level program is 22.2% at Tavg of 547°F and 53.3% at Tavg of 575.9°F.

There are 3 PZR level channels LT-459, LT-460 and LT-461. LC-459G the PZR level controller is normally fed by level channel LT-459 but can be replaced by LT-461 with a selector switch on the RTGB. The output of LC-459G is then fed to the charging pump speed controllers to control speed of the charging pump if their controllers are selected to Auto.

If PZR level increases 5% above program LC-459D will turn on the backup heaters and sound an annunciator for High Level Heaters on.

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

3. 009 EK3.23 001

Given the following plant conditions:

- Small Break LOCA occurred inside containment.
- The crew is currently implementing PATH-1.
- Containment pressure peaked at 4.2 psig and lowering.
- Core Exit T/Cs indicate 510°F and rising.
- RCS pressure is 1100 psig and lowering.
- Safety injection flow to the core is 500 gpm and steady.

Which ONE (1) of the following completes the statement below?

Based on these plant conditions, the Reactor Coolant Pumps will ____________.

A.✓ be secured due to inadequate subcooling.

B. NOT be secured due to adequate subcooling.

C. be secured due to loss of component cooling flow.

D. NOT be secured due to lower than expected SI flow.

The correct answer is A.

A. Correct. RCS subcooling is approximately 47 - 48°F, which is above the normal setpoint (35°F) but less than the adverse setpoint (55°F). Action is required based on plant conditions.

B. Incorrect. Plausible since the RCS subcooling is approximately 47 - 48°F, which is above the normal setpoint (35°F) but less than the adverse setpoint (55°F). Would be chosen if student fails to realize adverse numbers in affect or forget the adverse number setpoint.

C. Incorrect. Plausible if the student thinks that a Phase B signal has been received. Pressure peaked at 4.2 psig. The Phase B setpoint is 10 psig. A safety injection signal is initiated when Containment pressure exceeds 4 psig.

D. Incorrect. SI flow is adequate. Foldout "A" RCP trip criteria states that all RCPs are to be stopped if RCS subcooling less than 35°F [55°F] AND SI Pumps - At least one running AND capable of delivering flow. However, based on given conditions the student may realize that subcooling is low but think that SI flow is inadequate based on their understanding of the Foldout criteria. Therefore, it is plausible for the student to decide not to secure the RCPs due to lower than expected SI flow.

Question 3
Tier 1 / Group 1
K/A Importance Rating - RO 4.2 SRO 4.3

Knowledge of the reasons for the following responses as the apply to the small break LOCA: RCP tripping requirements

Reference(s) - Sim/Plant design, EPP Foldouts, Steam Tables
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-FOLDOUTS-005
Question Source - NEW
Question History - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 41.10 / 45.6 / 45.13
Comments - K/A match since candidate must analyze the conditions and know the RCP trip criteria and its basis.

CONTINUOUS USE

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PART 4

END PATH PROCEDURE

EPP-Foldouts

FOLDOUTS

REVISION 29

CONTINUOUS USE

FOLDOUT A

(Page 1 of 7)

1. RCP TRIP CRITERIA

   a. IF BOTH conditions below are met, THEN stop all RCPs:

   - SI Pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW TO THE CORE
   - RCS Subcooling - LESS THAN 35°F [55°F]

   b. IF the PHASE B Isolation Valves are Closed, THEN stop all RCPs.

2. SI ACTUATION CRITERIA

   IF EITHER condition below occurs, THEN Actuate SI and Go To PATH-1, Entry Point A:

   - RCS Subcooling - LESS THAN 35°F [55°F]
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [32%]

3. MSR ISOLATION CRITERIA

   Perform the following to isolate the MSRs:

   a. IF ANY Purge OR Shutoff Valve does not indicate fully closed, THEN place the associated RTGB Switch to CLOSE.

   b. IF ANY Purge OR Shutoff Valve can NOT be closed from the RTGB AND RCS temperature is less than 540°F and lowering, THEN close the MSIVs AND MSIV BYPs.

   c. IF a loss of power prevents isolation of the MSRs, THEN close the MSIVs AND MSIV BYPs.

4. 011 EK1.01 001

Which ONE (1) of the following describes reflux cooling?

Steam produced in the core rises in the hot legs to the S/Gs where it is condensed and returned to the vessel via the __(1)__ legs and to enhance this method of cooling, the __(2)__.

A. (1) cold

(2) S/G levels must be maintained.

B. (1) cold

(2) steaming from ALL S/Gs must be secured.

C.✓ (1) hot

(2) S/G levels must be maintained.

D. (1) hot

(2) steaming from ALL S/Gs must be secured.

The correct answer is C.

A. Incorrect - Flow returns via the hot leg.

B. Incorrect - Flow returns via hot legs and steaming from the steam generators is preferred to enhance the cooling effect.

C. Correct - Steam flows along the top of the hot legs to the steam generator where it condenses and returns to the core via the bottom of the hot leg piping. The S/G level must be maintained to cause condensation.

D. Incorrect - Steaming from the steam generators is preferred to enhance the cooling effect.

Question 4
Tier 1 / Group 1
K/A Importance Rating - RO 4.1 SRO 4.4

Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA : Natural circulation and cooling, including reflux boiling.

Reference(s) - PATH-1 Basis Document
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-C.1-003
Question Source - NEW
Question History - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.8 / 41.10 / 45.3
Comments - K/A match since the candidate must understand the concept of reflux boiling and the plant conditions necessary to enhance its effectiveness.

Case A has been analyzed to a long term stable condition. For breaks in this category, the establishment of an equilibrium pressure where pumped SI equals break flow constitutes a safe and stable condition for the long term, provided that the steam generator heat sink is maintained until such time that the break flow and SI sensible heat can remove all the decay heat. Once equilibrium pressure was established, the core was covered and adequate flow existed to remove decay heat through the steam generator with a small amount of voiding. This stable and safe condition could go on without interruption for a long period of time. The only change in the primary system conditions through the transient for this case is a gradual decrease in fluid temperatures which is beneficial, since it indicates that adequate core cooling is being maintained.

The equilibrium pressure condition is stable for the long term provided that SI and auxiliary feedwater are available. Since the RCS pressure at the equilibrium condition is determined by a balance between break and SI flowrate, in order to depressurize to a cold shutdown condition it is necessary to cool the primary fluid further while stepping down the SI flowrate. Long-term cooldown/depressurization of the plant is performed using guideline ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

Breaks 3/8" < diameter <~ 1", maximum safety injection

Maximum safety injection (all high-head SI pumps operating) for a break in this range of size would have little impact on the results given for the previous case (Category 2) with minimum safety injection. The main effect would be a slightly higher equilibrium pressure where safety injection flow matches break flow. Therefore, this case was not included in the analysis.

Breaks ~ 1" < diameter <~ 13-1/2" (1FT^2)

For break sizes of one to two-inch in equivalent diameter, the RCS will rapidly depressurize early in the transient, and an automatic reactor trip and safety injection signal will be generated based on low pressurizer pressure. During the early stages of the depressurization, when the system is still full of two-phase liquid, the break flow, which also will be mostly liquid, is not capable of removing all the decay heat. Therefore, the early depressurization is limited by energy removal considerations, and the RCS pressure will temporarily hang up above the steam generator safety valve set pressure, assuming no steam dump is available. The RCS pressure stays at this level in order to provide a temperature difference from primary to secondary so that core heat may be removed by the steam generator. At this energy-balance controlled pressure, however, pumped safety injection flow is less than the break flow, and there is a net loss of mass in the RCS. Voiding throughout the primary side occurs and eventually the RCS begins to drain, starting from the top of the steam generator tubes. The rate of RCS drain is determined by the net loss of liquid inventory, a function of both SI flow and break size.

Prior to the occurrence of draining, heat is removed from the steam generator through continuous two-phase natural circulation, with two-phase mixture flowing over the top of the steam generator tubes. As the draining continues, the natural circulation mode of heat removal as just defined ceases, and core heat is removed through condensation of steam in the steam generator. This method of heat removal is called reflux and is discussed in Reference 2.

The condensation mode of heat removal is almost as efficient as continuous two-phase natural circulation in removing heat. However, condensation heat transfer coefficients may be lower than continuous two-phase natural circulation heat transfer coefficients. Thus, as the steam generator tubes drain, a slight increase in primary system pressure occurs to give a greater delta T from primary to secondary in order to remove all the decay heat. The steam generator secondary side pressurizes to the safety valve set pressure early in the transient, and remains there throughout the natural circulation and steam condensation heat removal modes. Eventually the mixture level on the primary side may drop completely below the steam generator tubes and begin to drain other regions in the RCS. Depending on the location of the break, the draining may partially uncover the core.

For example, for a cold leg break liquid in the crossover leg region (loop seal) will block steam from the break, and the core must partially uncover in order to create a vent path for steam to exit from the core, upper plenum, hot legs and steam generators through to the break. The RCS draining occurs until such time that the break location uncovers, and break flow switches from two-phase to all steam. For hot leg or pressurizer vapor space breaks, however, the steam vent path exists without the need for the crossover leg region (loop seal) to clear of all liquid. Thus, no core uncovery is predicted.

5. 015 AK1.04 001

The plant is at 25% RTP with GP-005, Power Operation, power ascension in progress when the "A" RCP trips on overcurrent.

Which ONE (1) of the following identifies the effect on Main Feedwater to the S/Gs from steady-state to steady-state conditions?

Feedwater flow...

A. lowers to all S/Gs.

B. remains constant.

C. to S/G "A" rises. Feedwater flow to S/Gs "B" and "C" lowers.

D✓ to S/G "A" lowers. Feedwater flow to S/Gs "B" and "C" rises.

The correct answer is D.

A. Incorrect. S/Gs with increased heat removal requirements will have increased feed requirements.

B. Incorrect. Plausible if the student assumes feed control is in manual at low power.

C. Incorrect. Opposite effect.

D. Correct. Less heat input, less feed flow.

Question 5
Tier 1 / Group 1
K/A Importance Rating - RO 2.9 SRO 3.1

Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow.

Reference(s) - System Design / Simulator
Proposed References to be provided to applicants during examination - None
Learning Objective - FW009
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.8 / 41.10 / 45.3
Comments - K/A match because the candidate must understand the effect of a loss of one RCP on affected loop S/G level and the feedwater flow to the non-affected loops from steady-state to steady-state.

1.0 INTRODUCTION

The Reactor Coolant System (RCS) consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator (S/G), a pump, loop piping and instrumentation. The pressurizer (PZR) surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary.

The principal heat removal systems which are interconnected with the RCS are the Steam and Power Conversion, Safety Injection (SI), and Residual Heat Removal (RHR) System. The RCS is dependent upon the S/Gs, and the steam, feedwater, and condensate systems for normal and residual heat removal from the normal operating conditions to a reactor coolant temperature of approximately 350°F.

The RCS transfers the heat generated in the core to the S/Gs where steam is generated to drive the turbine generator. Borated demineralized water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance required for safe operation. Water also acts as a neutron moderator and reflector.

The RCS provides a boundary which contains the coolant under normal operating temperature and pressure conditions. It serves to confine radioactive material and limits contamination to acceptable values in the event of an uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the systems heat capacity attenuates thermal transients generated by the core or extracted by the S/Gs. The RCS accommodates coolant volume change within the protection system criteria.

2.0 GENERAL DESCRIPTION

2.1 System Purpose

The purpose of the RCS is to transfer heat from the reactor core to the S/Gs to produce steam for the turbine.

The RCS transfers heat from the core to the S/Gs utilizing the three heat transfer loops connected in parallel to the reactor vessel, each containing a Reactor Coolant Pump (RCP), a S/G, and associated instrumentation.

2.2 Design Basis

The RCS design & operating parameters are listed below with the safety, power operated relief and PZR spray valves setpoints, and the protection system setpoint pressures.

remainder of the flow includes the flow through the rod cluster control guide thimbles, the leakage across the reactor pressure vessel nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant then flows out of the vessel through exit nozzles.

The coolant then passes into the hot leg and flows to the S/G. The hot leg contains three active and three spare dual element temperature detectors mounted 120 degrees apart around the hot leg piping. Additionally there is a wide range temperature detector located in the hot leg down stream of the three dual element detectors.

The coolant passes into the inlet side of the S/G lower head and flows up through the tube sheet into the S/G U-tubes. The coolant passes out of the U-tubes to the outlet side of the lower head. It then passes out of the S/G to the intermediate leg.

The coolant leaves the S/G and flows to the RCP suction through the intermediate leg. Located at the elbow in the intermediate leg are three differential pressure detector flow element sensing points. These differential pressure detectors sense the low pressure at the inside of the elbow, with a separate penetration for each detector and a common penetration for the high side of the differential pressure detector.

The coolant leaves the RCP and flows to the reactor vessel inlet via the cold leg. Both wide range and narrow range temperature detectors are located in the cold leg.

2.4 System Description (Figures 1-4, 19)

The RCS consists of three heat transfer loops connected in parallel to the Reactor Vessel, each containing a RCP, the tube side of a S/G, and the associated connecting piping. The system also includes the PZR, PZR Relief Tank (PRT), a Reactor Vessel flange leak detection line, valves and piping, and instrumentation and controls independent of the Reactor Protection System, Process Instrumentation and Control System, or other control systems outside the RCS boundary. All RCS equipment, except for the nitrogen pressure regulator and a few isolation valves located in the auxiliary lines serving the PRT, is located inside the Reactor Containment Building (CV).

During normal operation the RCPs circulate the pressurized water through the reactor vessel and the heat transfer loops. The water, serving as both coolant and moderator/reflector, flows to the S/Gs, where the heat is transferred to the secondary steam system, and back to the pumps to repeat the cycle.

RCS pressure control is accomplished by means of a PZR that contains adequate water/steam volume to accommodate volume changes in the RCS due to temperature changes; PZR spray and heaters to maintain saturation conditions in the PZR at the required operating pressure; and a relief system to accommodate design load transients. RCS components utilized for RCS pressure control are automatically controlled by the

6. 015 AK2.10 001

The plant is at 25% RTP with GP-005, Power Operations, power ascension in progress when the following occurs:

- Annunciator APP-001-D2, RCP #1 SEAL LEAKOFF HI FLOW, alarms.
- The RO reports #1 seal leakoff is 5.8 gpm for RCP "A", and #1 seal leakoff for RCP "B" and RCP "C" has lowered from 3.0 to 2.0 gpm.

Which ONE (1) of the following identifies the required operator actions and proper sequence IAW AOP-018, Reactor Coolant Pump Abnormal Conditions?

A.✓ Trip the reactor and then stop RCP "A",

Go to PATH-1 while continuing with AOP-018 and then shut the #1 seal leakoff isolation valve (within 3 to 5 minutes).

B. Trip the reactor and then stop RCP "A".

Immediately shut the #1 seal leakoff isolation valve and then Go to PATH-1 while continuing with AOP-018.

C. Stop RCP "A" and then trip the reactor,

Go to PATH-1 while continuing with AOP-018 and then shut the #1 seal leakoff isolation valve (within 3 to 5 minutes).

D. Stop RCP "A" and then trip the reactor.

Immediately shut the #1 seal leakoff isolation valve and then Go to PATH-1 while continuing with AOP-018.

The correct answer is A.

A. Correct. AOP-018, Section A, steps 1 - 6.

B. Incorrect. Since reactor power is below P-8, stopping the RCP prior to manually tripping the reactor is possible without causing a direct reactor trip. However, experience on the simulator has shown that stopping the RCP prior to tripping the reactor is a challenge to safety features of the plant. (See AOP-018 Basis Document)

C. Incorrect. AOP-018 clearly states that PATH-1 must be continued with the performance of AOP-018. Per OMM-022, Emergency Operating Procedures User's Guide, certain procedures that direct tripping the Reactor, tripping an RCP and then going to Path-1 (such as AOP-018) states that the RCP shall be tripped after the reactor has been verified tripped and prior to entering Path-1. All remaining steps of AOP-018 are to be performed after the immediate actions of Path-1 are performed and verified.

D. Incorrect. See description of "C" above.

Question 6
Tier 1 / Group 1
K/A Importance Rating - RO 2.8 SRO 2.8

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls

Reference(s) - System Design / Simulator, AOP-018
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-018-004
Question Source - RNP Bank - Procedure
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.7
Comments - K/A match because candidate must analyze information given from RCP related indicators (seal leakoff flow, annunciator alarms) and determine the appropriate actions to mitigate. Candidate must also remember the sequence for closing the seal leakoff isolation valve which is a use of RCP controls.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-018

REACTOR COOLANT PUMP ABNORMAL CONDITIONS

REVISION 20

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides instructions to protect the Reactor Coolant Pumps after a component failure or support system malfunction.

2. ENTRY CONDITIONS

This procedure is entered for any of the following events:

- Loss of Seal Injection flow to any RCP
- High Vibration on an RCP during steady state conditions (Confirmed valid IAW APP-001-B5)
- Indication of any RCP Seal malfunction
  - RCP #1 Seal leakoff flow greater than 5 gpm
  - RCP #1 Seal leakoff flow less than 1 gpm
  - RCP #2 Seal leakoff flow greater than 0.5 gpm
  - Indication of RCP #3 Seal malfunction (repeat low standpipe level at normal pressure)
- As directed by AOP-016, Excessive Primary Plant Leakage

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Make PA Announcement For Procedure Entry.	

NOTE

- The RCP malfunctions in the Table below are listed in order of priority.
- This procedure is NOT applicable during implementation of EPP-1, Loss Of All AC OR any of its recovery procedures.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Evaluate Plant Conditions AND Go To The Appropriate Section For RCP Malfunction Not Yet Addressed:	Return to procedure and step in affect.

MALFUNCTION	SECTION
Reactor Coolant Pump Seal Failure	Section A
High Reactor Coolant Pump Vibration	Section B
Loss of Seal Injection	Section C

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 1 of 12)

* 1.	Check Any RCP #1 Seal Leakoff Flow - GREATER THAN 5.7 GPM	IF seal leakoff exceeds 5.7 gpm, THEN Go To Step 2. Go To Step 10.
2.	Check Either Of The Following Conditions Exist: • RCP #1 Seal Leakoff Flow On Unaffected RCP(s) - REDUCED YES OR • RCP Thermal Barrier ΔP On Affected RCP(s) - REDUCED	Perform the following: a. Perform cross-check of all RCP parameters to determine cause of indicated high leakoff flow. b. Observe The NOTE Prior To Step 2 and Go To The Main Body, Step 2 Of This Procedure

****

CAUTION

To prevent damage to the RCP Seal Stack, the affected RCP Seal Leakoff Isolation valve must be closed between 3 minutes and 5 minutes of stopping the RCP.

****

3.	Check Plant Status - MODE 1 OR MODE 2 yes	Stop the affected RCP(s) Observe the CAUTION prior to Step 5 and Go To Step 5.
4.	Perform The Following: a. Trip the reactor b. Trip the affected RCP(s) c. Go To Path-1 while continuing with this procedure.	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 2 of 12)

****

CAUTION

Restart of a RCP that has been stopped due to a seal malfunction, prior to the cause of the seal malfunction being determined and corrected, could cause catastrophic failure of all 3 RCP Seals.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Check Time Elapsed Since Stopping The Affected RCP(s) - GREATER THAN 3 MINUTES	WHEN at least 3 minutes have elapsed since tripping the affected RCP(s), THEN Go To Step 6.
6.	Close Seal Leakoff Valve(s) For Affected RCP(s):	

RCP	VALVE
A	CVC-303A
B	CVC-303B
C	CVC-303C

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Check SI - ACTUATED	Go To Step 27

7. 022 AA1.09 001

Given the following plant conditions:

- Plant is in Mode 3 at 547°F.
- The Charging Line in Pipe Alley has ruptured and sprayed water on FCV-626,THERM BARRIER OUTLET.
- An electrical short causes FCV-626 to travel to the closed position.
- The Crew has secured ALL Charging Pumps IAW AOP-018, Reactor Coolant Pump Abnormal Conditions.

How much time is allowed to restore RCP seal cooling AND what constitutes seal cooling restoration IAW AOP-018?

A.✓ 15 minutes to restore EITHER thermal barrier cooling OR seal injection.

B. 15 minutes to restore BOTH thermal barrier cooling AND seal injection.

C. 30 minutes to restore EITHER thermal barrier cooling OR seal injection.

D. 30 minutes to restore BOTH thermal barrier cooling AND seal injection.

The correct answer is A.

A. Correct.

B. Incorrect. 15 minutes is the correct time but both are not required to be restored for seal cooling.

C. Incorrect. 30 minutes is an incorrect time. Restoring either seal injection or thermal barrier cooling is correct.

D. Incorrect. 30 minutes and having to restore both seal injection and thermal barrier cooing is incorrect.

Question 7
Tier 1 / Group 1
K/A Importance Rating - RO 3.2 SRO 3.3

Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Pump Makeup: RCP seal flows, temperatures, pressures, and vibrations.

Reference(s) - System Design / Simulator, AOP-018, OMM-022.
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-018-004
Question Source - RNP Bank - Procedure
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5 / 45.6
Comments - K/A match because the candidate must analyze the information provided and determine that a complete loss of RCP seal cooling has resulted. Candidate must remember the time limit for seal cooling restoration and what constitutes seal cooling restoration.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 1 of 12)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Check APP-001-D1, RCP THERM BAR COOL WTR LO FLOW alarm - ILLUMINATED	IF APP-001-D1 ILLUMINATES, THEN observe the CAUTION prior to Step 2 and Go To Step 2. Observe the NOTE prior to Step 11 and Go To Step 11.

****

CAUTION

IF more than 15 minutes elapses without RCP Seal Cooling, THEN Seal Cooling (CCW AND Seal Injection) MUST be isolated before starting CCW OR Charging or Seal Damage could occur.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 2.	Check Elapsed Time Since All RCP Seal Cooling Was Lost - GREATER THAN 15 MINUTES	IF RCP Seal Cooling is NOT OR can NOT be restored in less than 15 minutes, THEN Go To Step 3. Go To Step 10.
3.	Check Plant Status - MODE 1 OR MODE 2	Stop the affected RCP(s) Go To Step 5.
4.	Perform The Following: a. Trip the reactor b. Trip the affected RCP(s) c. Go To Path-1 while continuing with this procedure	

8.3.22 RCP Seal Cooling Restoration

1. In the event that RCP Seal Cooling is completely lost (both CCW and Seal Injection) cooling from at least one source must be restored in less than 15 minutes.

2. **IF** at least one form of seal cooling can **NOT** restored within 15 minutes, **THEN** the RCPs should be tripped **AND** the seals should be isolated **PRIOR TO** restoring flow from Charging **OR** CCW by closing the following valves:

   - FCV-626, THERM BARRIER OUTLET **OR** CC-736, CC FROM RCP "A", "B", "C" THERMAL BARRIER.
   - RCP SEAL WATER FLOW CONTROL VALVEs
     1) CVC-297A
     2) CVC-297B
     3) CVC-297C

3. These actions should only be required in EPP-1 and DSP-002 for extraordinary circumstances, however, since there are an infinite number of events that can occur, this blanket requirement is included in this procedure.

8.3.23 Failure Of A Pump To Stop On Demand (CR 154571)

1. Throughout the EOPs and AOPs directions are provided for stopping pumps.

2. **IF** at any time a pump fails to stop on demand from the RTGB, **THEN** an operator should be dispatched to locally trip the pump.

3. The operator should use judgment for the urgency in performing the local action based on the event in progress at the time.

8. 026 AA1.06 001

Given the following plant conditions:

- The plant is at 100% RTP.
- CV Spray Pump "A" operating during the performance of OST-352-1, Containment Spray Component Test - Train A.
- The following alarm is received:
  - APP-002-E1, CV SRY PMP COOL WTR LO FLOW

Which ONE (1) of the following identifies the alarm setpoint and the action required IAW APP-002-E1?

The alarm indicates that CCW flow to CV Spray Pumps is less than......

A.✓ 30 GPM. Stop CV Spray Pump "A".

B. 30 GPM. Continued operation of CV Spray Pump "A" is allowed.

C. 7 GPM. Stop CV Spray Pump "A".

D. 7 GPM. Continued operation of CV Spray Pump "A" is allowed.

The correct answer is A.

A. Correct. APP-002-E1 correct setpoint and action is to stop the operating CV Spray pump if it is being operated for testing purposes.

B. Incorrect. Proper setpoint but wrong action to be taken. APP-002-E1 will direct the pump to be stopped if operating under non-emergency conditions OR long term recirculation.

C. Incorrect. Setpoint for low flow alarm is wrong and wrong action is specified. 7 GPM is the low flow alarm setpoint for the RHR pump cooler. The action specified in this distractor is correct.

D. Incorrect. Setpoint for low flow alarm is wrong and wrong action is specified. 10 GPM is the low flow alarm setpoint for the RHR pump cooler. APP-002-E1 will direct the pump to be stopped if operating under non-emergency conditions OR long term recirculation.

Question 8
Tier 1 / Group 1
K/A Importance Rating - RO 2.9 SRO 2.9

Ability to operate and/or monitor the following as they apply to the Loss of CCW:
Control of flow rates to components cooled by the CCW System

Reference(s) - System Design / Simulator, APP-002-E1, APP-001-B7, 5379-00376 Sheet 4
Proposed References to be provided to applicants during examination - None
Learning Objective - CS 008
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5 / 45.6
Comments: K/A match because the candidate must know the flow rates to the CV Spray pumps and the necessary operator actions for a loss of CCW cooling.

ALARM

CV SRY PMP COOL WTR LO FLOW

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Misaligned valve or leak in lines supplying CCW to or from the CV Spray Pump(s).
2. Loss of CCW.
3. Removal of CV Spray Pump from service.

OBSERVATIONS

1. None applicable (see actions)

ACTIONS CK (✓)

1. **IF** long term post accident recirculation is **NOT** in progress, **THEN DISPATCH** personnel to check CCW flow to the CV Spray Pump seal coolers (FIC-657). ______
2. **IF** Containment Spray Pumps are operating under non-emergency conditions **OR** long term recirculation, **THEN STOP** the pumps. ______
3. **IF** a loss of CCW has occurred, **THEN REFER TO** AOP-014. ______
4. **IF** valve alignment is **NOT** correct, **THEN VERIFY** correct valve alignment using OP-306. ______
5. **IF** a CCW leak is present, **THEN ISOLATE** the leak. ______
6. **IF** CCW flow has been isolated to a CV Spray Pump removed from service, **THEN PERFORM** the following:
   a. **VERIFY** CCW flow to the operable CV Spray Pump is ≥20 gpm **AND LOG** CCW flow shiftly. ______
   b. **IF** CCW flow to the operable CV Spray Pump can **NOT** be maintained ≥20 gpm, **THEN DECLARE** both CV Spray trains inoperable **AND ENTER** ITS LCO 3.0.3 in MODES 1, 2, 3, or 4. ______

DEVICE/SETPOINTS

1. FIC-657 / 30 gpm

POSSIBLE PLANT EFFECTS

1. Loss of Containment Spray Pump Seals. (If in recirculation mode)
2. Possible entry into TECH SPECS LCO

ALARM

RHR PMP A COOL WTR LO FLOW

NOTE:	If CCW is **NOT** available to the RHR pump seal coolers, the RHR pumps shall **NOT** be operated with pump discharge temperature greater than 135°F. With CCW available to the RHR pump seal coolers there is no time limit for running a single pump with flow only through the heatup recirculation line. It will be necessary to rotate the RHR pumps to avoid exceeding the 50°F ΔT limit between RHR loops as stated in GP-007. Based on the investigation performed for NCR 00222886, the normal operating value for CCW cooling flow to the RHR pump seal should be controlled to 10 gpm minimum. Based on the above, the minimum seal cooling flow for operability of the RHR pump is 5 gpm. Note that this is based on the WCAP guidance as well as some margin (approximately 2 gpm) to the vendor stated requirements.

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Loss of Component Cooling Water
2. Break/leak in line to pump

OBSERVATIONS

1. Annunciator APP-001-F5, CCW PMP LO PRESS
2. CCW Surge Tank Level (LI-614B)

ACTIONS

		CK (✓)
1.	**IF** a loss of CCW has occurred, **THEN REFER TO** AOP-014.	_____
2.	**IF** required, **THEN DISPATCH** an operator to check Component Cooling Flow to RHR Pumps (FI-638) **AND** supply/return lines.	_____
3.	**IF** a leak has occurred in the CCW line to/from RHR Pump A **AND** RHR Pump A is in service, **THEN PLACE** RHR Pump B in service, **STOP** RHR Pump A, **AND ISOLATE** the leak.	_____
4.	**IF** RHR Pump A is in service **AND** a leak has occurred in the CCW line from RHR Pump A **AND** no other RHR Pump is available, **THEN INITIATE** efforts to patch the leak **AND** restore RHR Pump B.	_____

DEVICE/SETPOINTS

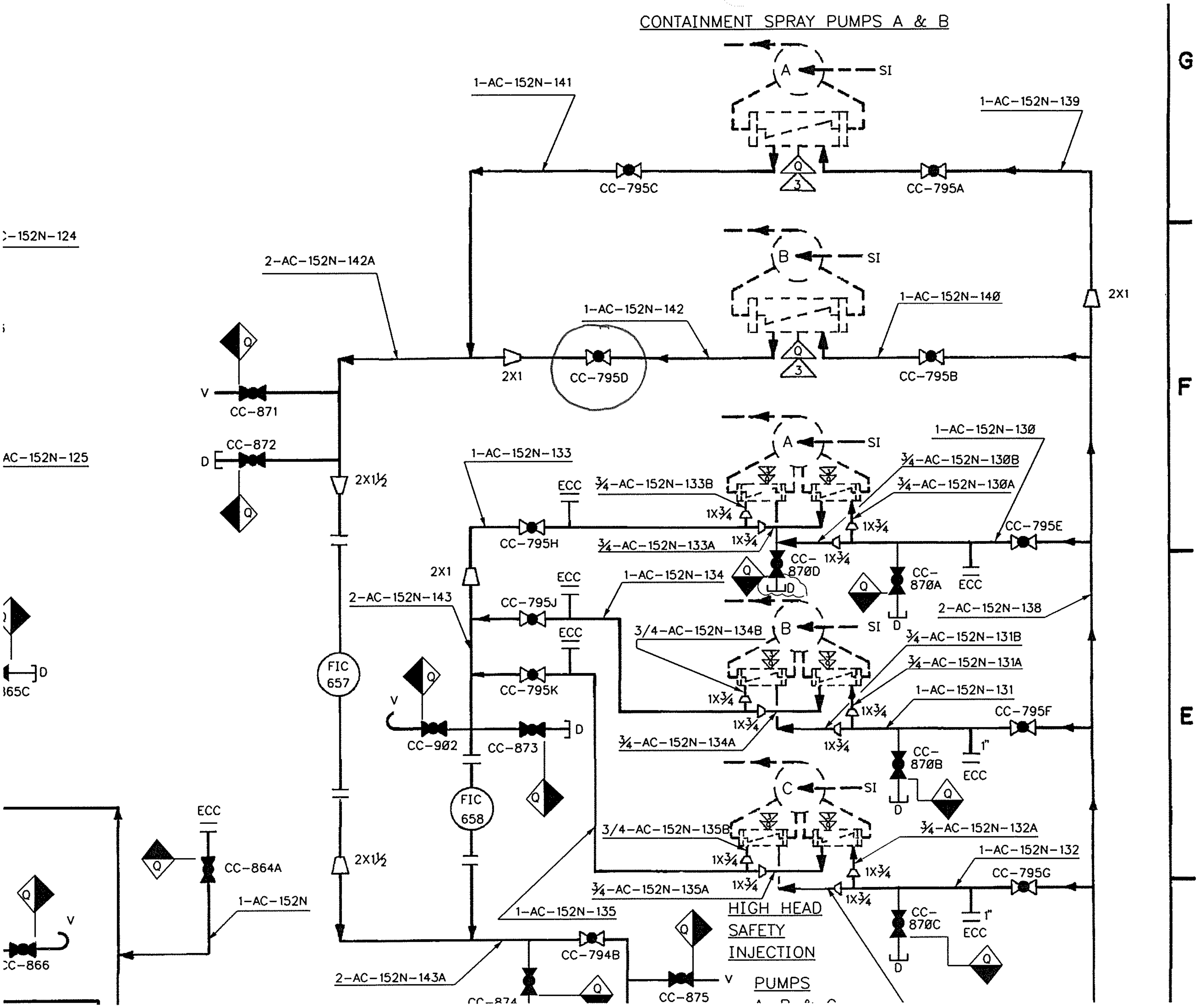
1. FC-638 / 7 gpm

POSSIBLE PLANT EFFECTS

1. Failure of RHR Pump Seal

REFERENCES

1. ITS LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.5.2, LCO 3.5.3, LCO 3.9.4 and LCO 3.9.5
2. AOP-014, Component Cooling Water System Malfunction
3. CWD B-190628, Sheet 489, Cable M



CONTAINMENT SPRAY PUMPS A & B
SI
1-AC-152N-141
1-AC-152N-139
CC-795C
CC-795A
2-AC-152N-142A
1-AC-152N-142
1-AC-152N-14Ø
2X1
CC-795D
CC-795B
CC-871
CC-872
1-AC-152N-133
1-AC-152N-13Ø
¾-AC-152N-13ØB
¾-AC-152N-13ØA
¾-AC-152N-133B
¾-AC-152N-133A
CC-795H
CC-795E
1X¾
CC-870D
CC-870A
ECC
2-AC-152N-143
1-AC-152N-134
2-AC-152N-138
CC-795J
CC-795K
3/4-AC-152N-134B
¾-AC-152N-131B
¾-AC-152N-131A
1-AC-152N-131
¾-AC-152N-134A
CC-795F
CC-870B
FIC 657
FIC 658
CC-9Ø2
CC-873
2X1½
3/4-AC-152N-135B
¾-AC-152N-132A
1-AC-152N-132
¾-AC-152N-135A
CC-795G
1-AC-152N-135
CC-870C
HIGH HEAD SAFETY INJECTION PUMPS
CC-864A
1-AC-152N
CC-866
2-AC-152N-143A
CC-794B
CC-874
CC-875
G
F
E

9. 029 EG2.4.21 001

Which ONE (1) of the following plant conditions requires entry into FRP-S.1, "Response To Nuclear Power Generation/ATWS" ?

A. Source Range startup rate of +0.1 dpm.

B. Intermediate Range startup rate of -0.1 dpm.

C. Power Range indicates 3% with an Intermediate Range startup rate of -0.2 dpm.

D✓ Power Range indicates 3% with an Intermediate Range startup rate of +0.2 dpm.

The correct answer is D.

A. Incorrect. A positive SR SUR would meet the entry conditions for FRP-S.2 (YELLOW).

B. Incorrect. A -0.1 dpm IR SUR would be an entry condition for FRP-S.2 (YELLOW).

C. Incorrect. Power must be greater than 5% to enter FRP-S.1 without having positive IR SUR indication.

D. Correct. Even though Power Range NIs indicate less than 5%, the fact that IR SUR is positive will lead you to transition to FRP-S.1 (ORANGE).

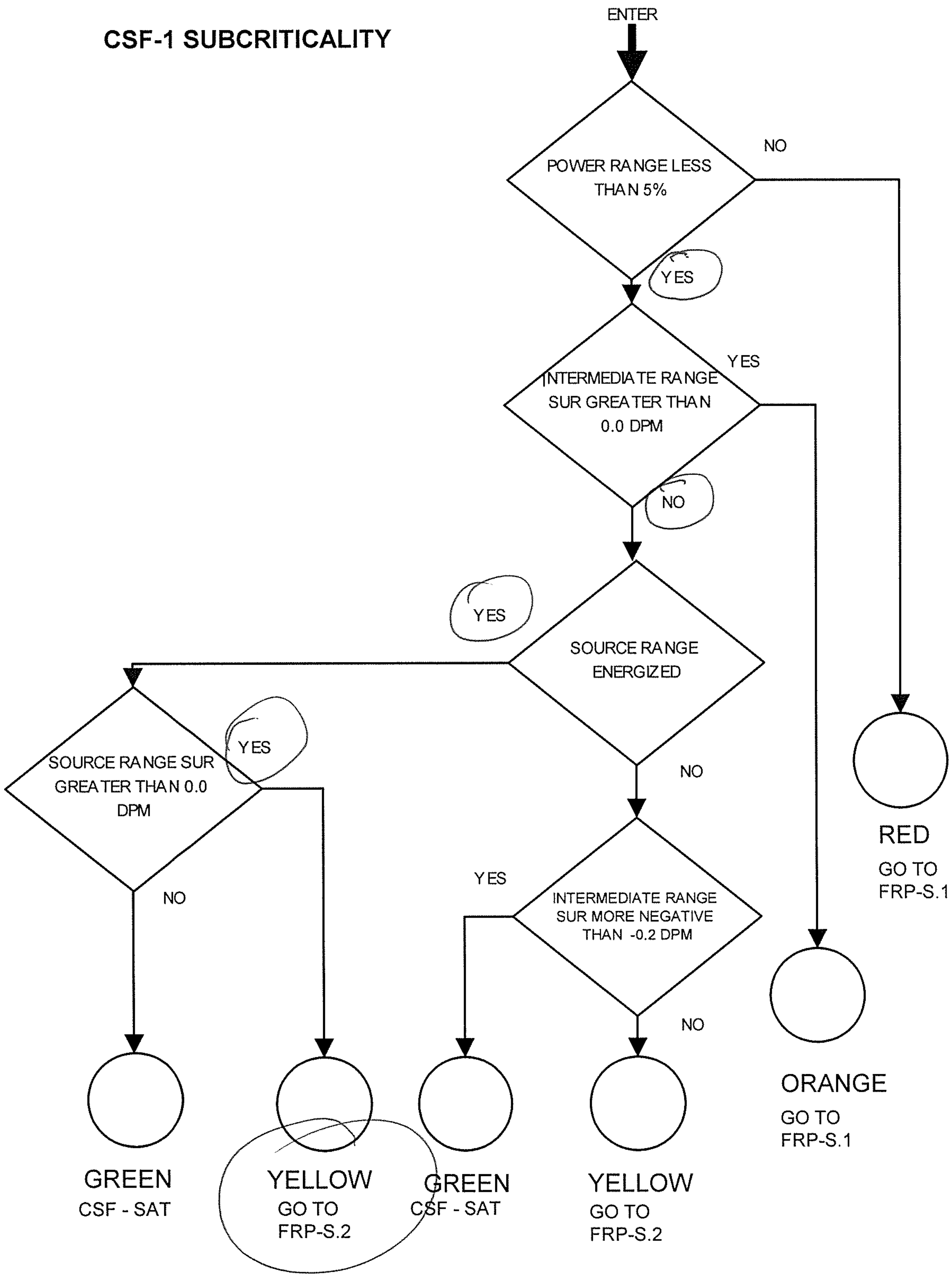
Question 9
Tier 1 / Group 1
K/A Importance Rating - RO 4.0 SRO 4.6

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
Anticipated Transient Without Scram (ATWS)

Reference(s) - System Design / Simulator, CSFST, FRP-S.1 Basis Document.
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-S.1-002
Question Source - RNP Bank - Procedure
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 43.5 / 45.12
Comments: K/A match because candidate must know the parameter setpoints and proper logic to meet the entry requirements for FRP-S.1, Response to Nuclear Power Generation / ATWS.

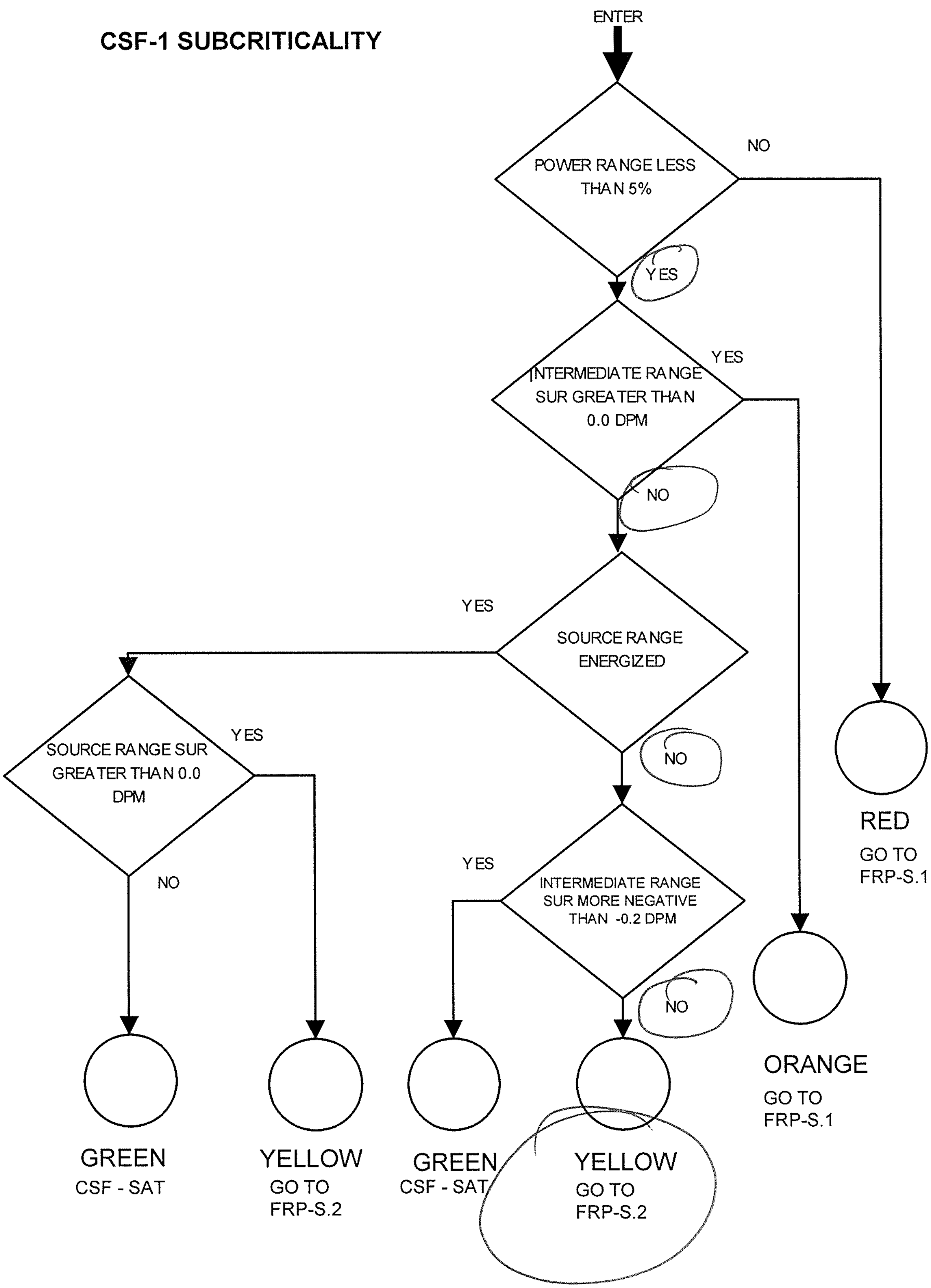
A.

CSF-1 SUBCRITICALITY



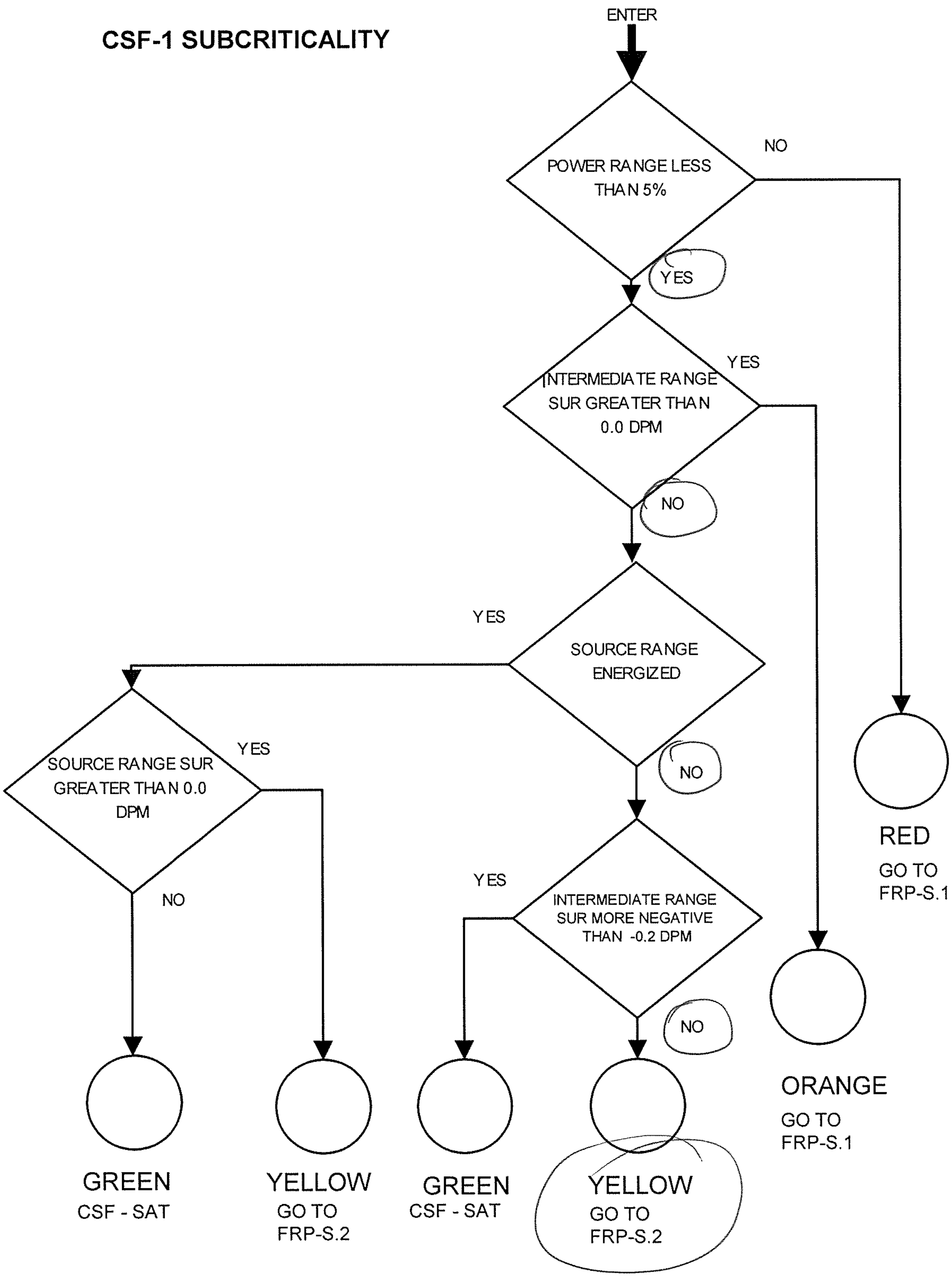
B.

CSF-1 SUBCRITICALITY



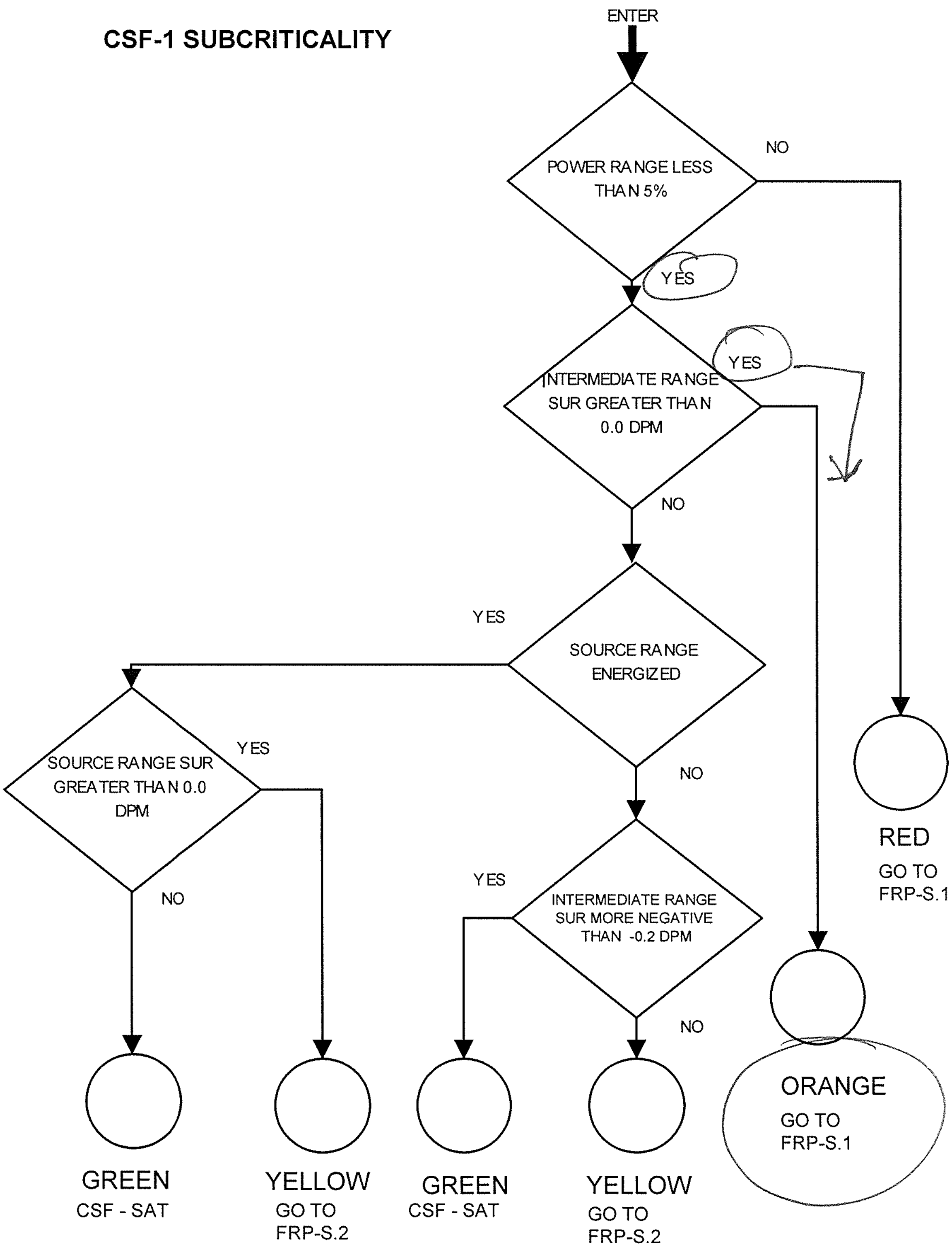
C.

CSF-1 SUBCRITICALITY



D.

CSF-1 SUBCRITICALITY



10. 040 AK3.02 001

Which ONE (1) of the following states the ESFAS signal that will provide protection for a main steam line break adjacent to a Main Steam Safety Valve and the basis?

High Steam Line..........

A.✓ Delta P; Provides protection for a steam line break upstream of the MSIVs.

B. Delta P; Provides protection for a steam line break downstream of the MSIVs.

C. Flow with Low Tave; Provides protection for a steam line break downstream of the MSIVs.

D. Flow with Low Tave; Provides protection for a steam line break upstream of the MSIVs.

The correct answer is A.

A: Correct.

B: Incorrect. High Steam Line Delta P is correct. Providing protection downstream of the MSIVs is incorrect. High Steam Line Delta P is protection for steam line break upstream of the MSIVs and is sensed from the steam header to the individual line pressures. In the present configuration, "A" steam line pressure would have to lower 100 psig below the steam header pressure for the SI to actuate.

C: Incorrect. High Steam Line Flow with Low Tave SI is for steam break protection downstream of the MSIVs and has to sense 2 steam lines with high flow to satisfy the logic for SI to actuate. The failure is limited to 1 steam line in this situation and that failure is upstream of the MSIVs.

D: Incorrect. High Steam Line Flow with Low Tave SI is for steam break protection down stream of the MSIVs and has to sense 2 steam lines with high flow to satisfy the logic for SI to actuate. The failure is limited to 1 steam line in this situation.

Question 10
Tier 1 / Group 1
K/A Importance Rating - RO 4.4 SRO 4.4

Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: ESFAS initiation

Reference(s) - System Design / Simulator, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - ESF-005
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 41.10 / 45.13
Comments: K/A match because candidate must know the ESFAS signal that provides protection for a steam line break and the basis for this response.

MSIVs.

4.1.4 Steam Line Pressure (ESF-Figure-1 & 3)

High Steam Line Flow w/ Low TAve

Steam Line Pressure measurement is utilized for steam line break protection. Low steam line pressure (614 psig) in two of three main steam lines or Low Tavg (543°F) in two of three loops, coincident with high steam line flow in two-of-three main steam lines, will initiate the Steam Line Isolation and Safety Injection signals. This is to protect against: a steam line break downstream of the main steam check valves, a feed line break, and/or an inadvertent opening of a SG safety.

In addition, each steam line pressure measurement is compared with a main steam header pressure measurement to determine if a high steam line differential pressure exists. A coincidence of two-of-three steam line differential pressures (100 psid) in any one steam line, that is, steam line pressure lower than main steam header pressure, will initiate a Safety Injection signal.

D/P

The steam header pressure is electronically limited to a minimum value of 585 psig. Therefore, this SI signal must be blocked before a plant cooldown is started to prevent SI actuation when S/G pressures drop below 485 psig(approximately 467°F). The steam line differential pressure circuit detects faults upstream of the MSIVs. Since the steam line check valves prevent reverse flow to the faulted S/G, excessive steam line differential pressure does not close the MSIVs.

4.1.5 Containment Pressure (ESF-Figure-4 & 5)

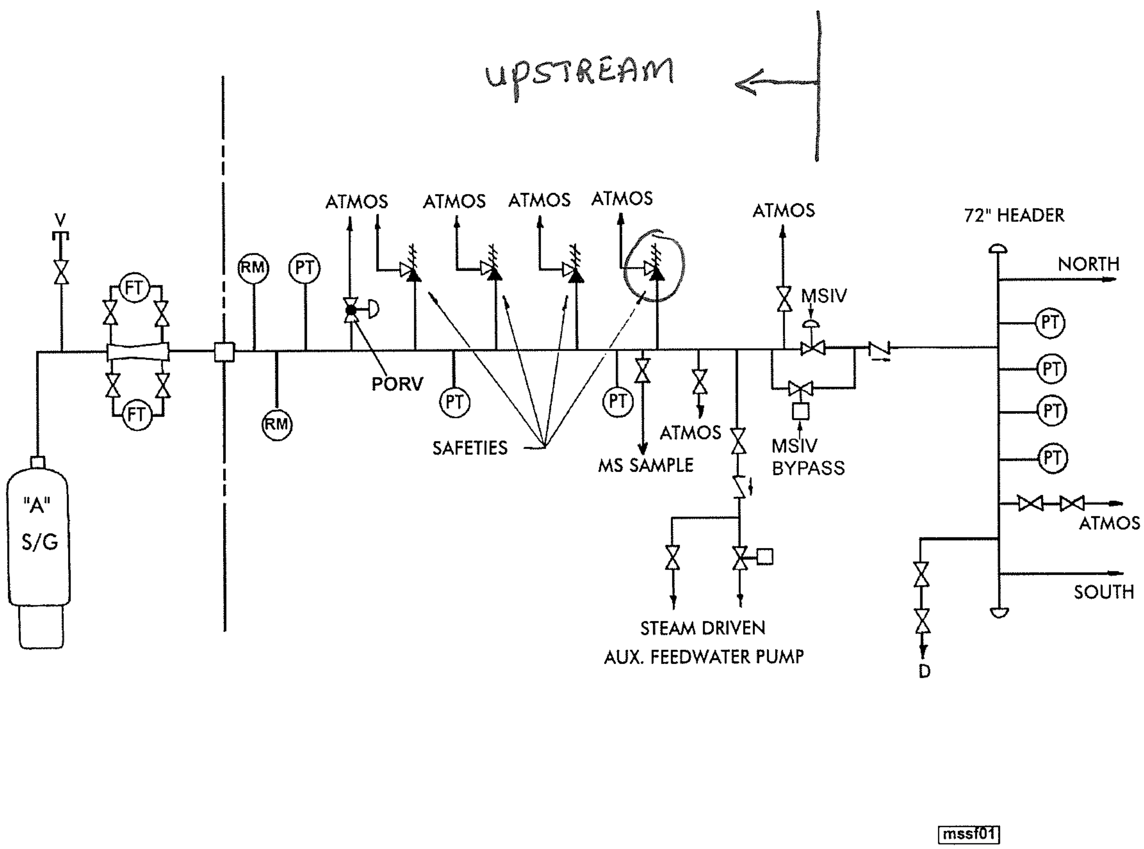
Containment Pressure measurement is utilized to initiate Emergency Core Cooling in response to a Loss of Coolant Accident (LOCA), and to provide containment pressure protection for either a LOCA, a feed line break inside containment, or a Main Steam Line Break inside containment. Nine pressure comparators, with inputs from six pressure transmitters, are used as inputs to ESFAS. Three pressure comparators provide an input for Hi Containment Pressure at 4 psig increasing pressure. Six pressure comparators provide an input for Hi-Hi Containment Pressure at 10 psig.

A coincidence of two of three Hi Containment Pressure (4 psig) will initiate a Safety Injection signal.

A coincidence of two separate two-of-three Hi-Hi Containment Pressure signals (10 psig), i.e., two-of-three twice, will initiate the following signals:

SYSTEM DIAGRAM (S/G TO 72" HEADER)

MSS-FIGURE-1 (Rev 0)



INFORMATION USE ONLY

11. 054 AA2.08 001

The plant is at 70% RTP when the BOP observes the following indications for all 3 S/Gs from the FF/SF Trend Recorders and Feed Reg. Valve Controllers:

- S/G level is LOWERING.
- Steam Flow is STABLE.
- Feed Flow is LOWERING.
- Feed Reg Valve positions are all OPENING.

Which ONE (1) of the following identifies the cause of the event in progress?

A. Heater Drain Pump Trip.

B.✓ Main Feedwater Pump Trip.

C. LCV-1530A, HDT Level Control Valve, fails OPEN.

D. LCV-1530B, HDT Level Control Valve, fails OPEN.

The correct answer is B.

A-Incorrect. The trip of a HDP will cause the suction pressure of the Main Feed Pump to lower. HCV-1459 should open to compensate for the loss of the HDP. There would be no long term impact on S/G Level and Feed Flow.

B-Correct. At 70% power a loss of one main feed pump will leave inadequate feed flow to maintain steam generator levels within the program level.

C-Incorrect. LCV-1530A failing OPEN would result in low Heater Drain Tank level that would trip the operating Heater Drain Pump. The trip of a HDP will cause the suction pressure of the Main Feed Pump to lower. HCV-1459 should open to compensate for the loss of the HDP. There would be no long term impact on S/G Level and Feed Flow.

D-Incorrect. LCV-1530B failing OPEN would result in low Heater Drain Tank level that would trip the operating Heater Drain Pump. The trip of a HDP will cause the suction pressure of the Main Feed Pump to lower. HCV-1459 should open to compensate for the loss of the HDP. There would be no long term impact on S/G Level and Feed Flow.

Question 11
Tier 1 / Group 1
K/A Importance Rating - RO 2.9 SRO 3.3

Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Steam flow-feed trend recorder.

Reference(s) - System Design / Simulator, AOP-010, AOP-010 Basis Document, APP-006-A3, - D1
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-010-002
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 43.5 / 45.13
Comments: K/A match because the candidate must interpret Steam flow - feed trend recorder data and determine what caused the given given trends.

ALARM

S/G A NAR RANGE LO/LO-LO LEVEL *** WILL REFLASH ***

AUTOMATIC ACTIONS

1. Reactor trip **AND** AFW startup on 2/3 Lo-Lo Level

CAUSE

1. Instrument Channel Failure:
    1) Steam Flow
    2) Feedwater Flow
    3) Steam Generator Level
2. Level Control System Failure
3. Loss of Feedwater Pump
4. Failure of Feedwater Control Valve
5. Load Rejection

OBSERVATIONS

1. Steam Generator Level
2. Steam Flow
3. Feedwater Flow
4. Feed Reg. Valve position
5. FW Pump Breaker indication
6. Auxiliary Feedwater Pumps Running

ACTIONS	CK (✓)
1. **IF** an Instrument Channel supplying SGLC has failed, **THEN** refer to AOP-025.	______
2. **IF** Feedwater Control failure, **THEN** refer to AOP-010.	______
3. **IF** an Instrument Channel **NOT** supplying SGLC has failed, **THEN** removed from service using OWP-027.	______

DEVICE/SETPOINTS

1. LC-474A, LC-474B, LC-475A, LC-475B, LC-476A / 35% of span
2. LC-474A, LC-475A, LC-476A / 16% of span

POSSIBLE PLANT EFFECTS

1. Reactor trip.

REFERENCES

1. ITS LCO 3.3.3 Table 3.3.3-1 Item 13, LCO 3.3.8 Table 3.3.8-1 Item 1, LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, Table 3.3.1-1 Item 13
2. AOP-010, Main Feedwater/Condensate Malfunction
3. AOP-025, RTGB Instrument Failure
4. OWP-027, Steam Generator Level
5. CWD B-190628: Sheet 415, Cables P & N; Sheet 418, Cables L, M, & R; Sheet 440, Cables BM & BQ

ALARM

S/G A LVL DEV

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Instrument Channel Failure:
   1) Steam Flow
   2) Feedwater Flow
   3) Steam Generator Level
   4) Turbine First Stage Pressure
2. Level Control System Failure
3. Secondary System Transient

OBSERVATIONS

1. Steam Generator Level
2. Programmed Steam Generator Level Setpoint
3. Steam Flow
4. Feedwater Flow
5. Turbine First Stage Pressure

ACTIONS	CK (✓)
1. **IF** Instrument Channel failure, **THEN** refer to AOP-025.	_______
2. **IF** Feedwater Control failure, **THEN** refer to AOP-010.	_______

DEVICE/SETPOINTS

1. LC-478D / 5% Deviation from Programmed Level (39 to 52% Level from 0 to 20% power; 52% Level at greater than 20% power)

POSSIBLE PLANT EFFECTS

1. S/G level increase to high level annunciator **OR** decrease to low level annunciator.

REFERENCES

1. AOP-010, Main Feedwater/Condensate Malfunction
2. AOP-025, RTGB Instrument Failure
3. CWD B-190628, Sheet 418, Cables N, P

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-010

MAIN FEEDWATER/CONDENSATE MALFUNCTION

REVISION 26

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides the instructions to mitigate abnormal conditions in the Main Feedwater and Condensate System, including Main FWP trips, Condensate Pump trips, Heater Drain Pump trips, control system malfunctions and valve failures.

2. ENTRY CONDITIONS

Any abnormal condition in Main Feedwater/Condensate system resulting in a system flow transient, with the exception of Instrument Failures.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Step 1 is an immediate action step.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Feedwater Regulating Valves - OPERATING PROPERLY (MANUAL OR AUTO): • FCV-478 • FCV-488 • FCV-498	Perform the following: a. Verify FRV for affected S/G(s) in manual control. b. Attempt to stabilize S/G level using FRV and/or FRV Bypass Valves by matching steam flow with feed flow. c. Stop any load change in progress. d. IF unable to control S/G level, THEN trip the Reactor AND Go To PATH-1. e. Go To Step 37.
* 2.	Check Reactor Trip Setpoint - BEING APPROACHED	IF a Reactor Trip Setpoint is approached, THEN trip the Reactor and Go To Path-1. Go To Step 4.
3.	Trip The Reactor And Go To Path-1.	
4.	Make PA Announcement For Procedure Entry	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Go To The Appropriate Step from The Table Below:	

EVENT	STEP
Main Feed Pump Trip	Step 6
Condensate AND Feed Pump Trip	Step 10
Condensate Pump Trip Without MFP Trip	Step 48
Heater Drain Pump Trip	Step 15
Pipe Break / Leak	Step 21
HCV-1459 Failed Open	Step 35
Other	Step 24

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Check Reactor Power - LESS THAN 70%	Trip the Reactor and Go To Path-1.
7.	Check Reactor Power - GREATER THAN 60%	Go To Step 13.

NOTE

Rapid power reductions may result in the axial flux difference exceeding the operating band values and require a power reduction to less than 50% to comply with ITS 3.2.3 Condition C.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Reduce Turbine Load At 1%/MIN To 5%/MIN To Achieve Less Than 60% Reactor Power Using Attachment 1	
9.	Go To Step 13	
10.	Check Reactor Power - LESS THAN 70%	Trip the Reactor and Go To Path-1.
11.	Check Reactor Power - GREATER THAN 50%	Go To Step 13.

Progress Energy

NON
POM

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AOP-010-BD

BASIS DOCUMENT, MAIN FEEDWATER/CONDENSATE MALFUNCTION

REVISION 26

DISCUSSION:

This procedure provides the direction necessary for abnormal conditions that arise in the Feed and Condensate systems that could affect feed flow to the Steam Generators. Entry conditions are based on the operator recognizing that an abnormal condition has occurred in the Feedwater/Condensate System that has resulted in a flow transient or the inability of the system to maintain S/G level at program.
Specifically, the following problems are addressed:

- Feed Pump trip
- Condensate Pump trip
- Heater Drain Pump trip
- HCV-1459, LP heaters Bypass, fail open
- Feed Reg Valve failure
- Feed Reg Bypass Valve failure
- LCV-1530A, HDT LEVEL CONTROL VALVE, fail open or closed
- LCV-1530B, HEATER DRAIN PUMPS SUCTION DUMP TO CONDENSER, fail open or closed
- Heater Drain Pump shaft shear
- Condensate Pump shaft shear
- S/G level control circuit failure
- Condensate or Feedwater System leak

Note that S/G instrumentation is not listed as a possible entry condition. These are covered under AOP-025, RTGB Instrument Failure.

Since the severity of a reduction in feedwater flow is directly proportional to the chances of surviving such a transient without a reactor trip, and since this transient could occur in a variety of plant conditions that would also factor into the survivability of such a transient, the approach taken to this procedure was to write actions that would result in a stable plant if the severity were to allow it.

The actions performed for these transients do not involve maneuvering the plant at rates in excess of the normal maximum of 1% to 5%. Should events occur that require a power reduction at rates in excess of these, the plant should be tripped. The actions in this procedure for large transients are based on recommendation 2 of INPO SOER 94-01 which calls for reactor scram and turbine trip criteria.

DISCUSSION (Continued)

At no time in the procedure should the operator fail to trip the plant should conditions arise that warrant a reactor trip. If the operator should realize that due to power level and conditions causing the transient a trip is unavoidable, then the plant should be tripped.

The paragraphs below will summarize the actions and strategy for the various entry conditions to this procedure.

Feed Pump Trip

The most severe of the transients is a Main Feed Pump Trip coincident with a Condensate Pump trip from full power. The mitigation strategy for this event is to trip the unit. The power level reduction required is very large and the time available very short. In order to reduce power below the point needed for matching steam flow and feed flow, power must be reduced at a rate large enough to actuate steam dump. Steam dump actuation compounds the problem; therefore, in order to prevent the trip steam dump would be required to be placed in a condition to prevent operation. This in turn will increase primary plant pressure from excessive Tavg swings. Rather than potentially challenge the PZR PORVs and to provide conservative guidance, the reactor will be tripped for any MFP trip greater than 70%, or Condensate Pump and Feed Pump trip greater than 70%. (CR 98-0204)

Feed Pump trips at a power level less than these values will be mitigated by a power reduction within the normal maneuvering rates of 1% to 5%.

Condensate Pump Trip

If only one Condensate Pump is running and it trips, the standby pump should start. If it does not, then the Feed Pump will trip, resulting in a total loss of feed water. If two Condensate Pumps are running and only one Feed Pump, then a Condensate Pump trip will not result in a significant feed transient unless reactor power is above 50%. If two Feed Pumps are running, one will trip when the Condensate Pump trips, which is covered by the Feed Pump trip scenario (see above).

DISCUSSION (Continued)

Heater Drain Pump Trip

The effects of a Heater Drain Pump trip will be seen as an effect on Feed Pump suction pressure. If suction pressure decreases to too low a value, Feed Pump trips will occur. If one Feed Pump trips, the Heater Drain Pump trip is moot and actions for the transient are covered by the direction for a Feed Pump trip discussed above. If a Feed Pump trip does not occur, then a power reduction may or may not be required dependent on the initial power level and equipment running, in order to maintain feedwater requirements within system capabilities. If only one Feed Pump is running, then power could not be high enough to require any Heater Drain Pumps to be running so suction pressure would not be a concern. If two Feed Pumps are running, then HCV-1459 would be depended on to ensure adequate margin of suction pressure to the trip setpoint.

Just as a Feed pump trip could be caused by a Condensate Pump trip, a trip of a Heater Drain Pump (on low tank level) could be caused by a failed open discharge or high level dump valve (LCV-1530A is designed to fail "as is", however a malfunction of the controller could cause it to go full open.). In any case, actions to mitigate a trip of a Heater Drain Pump must include checks for other failures that could have resulted in the pump trip. For a HDP trip the actions are to reduce power to a level dependent on the number of running pumps and dispatch personnel to investigate the potential causes of the trip.

HCV-1459, LP heaters Bypass, Fails Open

HCV-1459 auto opens if two Feed Pumps are running, any Heater Drain Pump is running, and suction pressure decreases to 300 psi, or if no Heater Drain Pumps are running, 350 psi. If HCV-1459 is open, then it must be determined if it is due to failure of HCV-1459 or for other reasons.

If LCV-1530A fails closed, the effect on Feed Pump suction pressure would be the same as a trip of Heater Drain Pumps. If LCV-1530B fails open, it would most likely result in a trip of the Heater Drain Pump on low tank level.

However, it is possible that LCV-1530A could close to compensate for the failure of LCV-1530B such that low tank level does not occur. In this case, the affects would be seen on Feed Pump suction pressure.

DISCUSSION (Continued)

Heater Drain Pump or Condensate Pump Shaft Shear

If a Heater Drain or Condensate Pump shaft shears, Feed Pump suction pressure will decrease but pump trip will not be indicated. If shaft shear is indicated, then direction to trip the pump will not have any effect on suction pressure. A Condensate Pump shaft shear is not likely survivable from full power. Since these events can not be diagnosed from the Control Room, the intent of the procedure is to commence power reduction at the normal maneuvering rate while investigation commences. If the event occurs at high power it will become evident to the operator that the plant will not recover and a manual reactor trip will be in order. A HDP shaft shear is a survivable event. A plant power reduction within the normal maneuvering rates should recover S/G level.

LCV-1530A or LCV-1530B fail

If LCV-1530A or LCV-1530B fail open, there should be no effect unless Heater Drain Pumps are running. In this case, HDT low level trip of the Heater Drain Pumps may occur, which would be covered in the Heater Drain Pump trip scenario.
If LCV-1530B fails open when Heater Drain Pumps are not running, this would be indicated by a low level alarm and handled by the APP.

If 1530A fails closed, then 1530B should open to compensate. If neither valve opens the Heater Vent and Drain System will rapidly fill to a water solid condition. This will be characterized by severe water hammer on the secondary and water relief through the heater safety valves. In the event that this occurs the strategy is to trip the unit. For this reason, if maintenance is to be performed on the controller for LCV-1530A and B, contingency plans must be put in place with personnel pre-staged to perform actions such as opening the bypass around LCV-1530A should it go closed. If the actions can be completed in a rapid manner prior to the system filling water solid the safety of personnel and equipment will not be jeopardized.

The basis design of LCV-1530B calls for air to both open and close the valve, therefore it is possible for the valve to fail “as is”. “As is” is the CLOSED position, however, in order to assure that the valve will fail to the OPEN position, as was originally designed, a pilot trip valve arrangement has been added to the operator such that on decreasing air pressure, the valve will fail open. This does not prevent the valve from failing closed due to an electronic failure in the controller.

INDIVIDUAL STEP DESCRIPTION:

Step	Description
1	This step checks if the FRV is controlling properly in manual or automatic control and is placed early in the procedure to address a malfunction of the automatic control circuitry. Early action to match steam and feed flows will stabilize S/G levels to allow diagnosis of the failure. Load changes are stopped to aid the operator in maintaining level control. If FRV or FRV Bypass valve failures will not allow the operator to obtain control, then the plant is tripped. This step is an immediate action step.
2-3	These steps trip the reactor should a reactor trip setpoint be approached. This is a continuous action step so that should it become apparent to the operator that the event is not survivable or that a trip is imminent, a manual trip should be performed.
4	This step directs making a PA announcement for procedure entry. The PA announcement should include the procedure being entered and the reason for entry. This will help to alert personnel that may be called upon for assistance for the event. Since this is an informational announcement that does not direct any local or plant evacuations or warn of any hazardous personnel situations, it is not necessary to use the VLC switch when making this announcement.
5	This step provides transitional guidance to the operator based on the event causing entry to the procedure. This procedure is designed to mitigate Feedwater Transients resulting from numerous possible causes. Transitioning via a table has been shown to be more efficient and simpler to use rather than numerous transitions within the procedure. This will improve the human factors of the procedure by reducing the total transitions required.
N8	A new note is being added which states: “Rapid power reductions at the beginning of core life could result in the axial flux difference exceeding the operating band values and require a power reduction to < 50% to comply with ITS 3.2.3 Condition C.” This is being added in response to the Reactivity Management Peer Panel comments on the investigation of the AFD response to a loss of "A" Heater Drain Pump event.

INDIVIDUAL STEP DESCRIPTION:

Step	Description
6-9	These steps are reached if a Main Feedwater Pump trip has occurred. First power is checked between (and including) 60% to 80%. If power is greater than 70% the reactor will be tripped. 70% is chosen based on Management discretion for conservative decision making. Larger power reductions require more rapid maneuvering and may challenge the primary side PORVs. If power is below 60% no power reduction will be required. For power levels greater than 60% power will be reduced at rate chosen by the CRS, between 1% and 5% per minute. The closer the plant is to 60%, the lower the rate necessary to match steam and feed flow. 5% per minute is the maximum normal maneuvering rate of the plant. If a power reduction is required at rates above this value, the reactor should be tripped.
N12	A new note is being added which states: "Rapid power reductions at the beginning of core life could result in the axial flux difference exceeding the operating band values and require a power reduction to < 50% to comply with ITS 3.2.3 Condition C." This is being added in response to the Reactivity Management Peer Panel comments on the investigation of the AFD response to a loss of "A" Heater Drain Pump event.
10-12	These steps are reached if both a Condensate Pump and a Main Feed Pump have tripped. This is the most severe of the transients. These steps are the same as for a Main Feed Pump trip, with the exception that the target power levels are 70% and 50% due to the more severe transient. BASIS DOCUMENT, MAIN FEEDWATER/CONDENSATE MALFUNCTION
13	This step is reached from the section for Main Feed Pump trips. The intent of the step is to provide direction for MFP trips at low power levels from which an automatic trip does not occur. Normally if only one Condensate Pump and Feed Pump are running and one was to trip, the standby pump would automatically start and the operator would only be required to stabilize the plant. Should be standby pump not start, or was unavailable, action will be dependent on power level. If greater than 10% the reactor will be tripped. If less than 10% the Turbine will be tripped and AOP-007 entered. No attempt will be made to manually start a Main Feed Pump or Condensate Pump in the middle of a transient.
14	This step completes the series of steps associated with Main Feed Pump and Condensate Pump trips. It provides transitional guidance to the end of the procedure for plant stabilization and AOP exit.

INDIVIDUAL STEP DESCRIPTION:

Step	Description
N15	A new note is being added which states: "Rapid power reductions at the beginning of core life could result in the axial flux difference exceeding the operating band values and require a power reduction to < 50% to comply with ITS 3.2.3 Condition C." This is being added in response to the Reactivity Management Peer Panel comments on the investigation of the AFD response to a loss of "A" Heater Drain Pump event.
15	This step is reached a HDP has tripped. It establishes the appropriate unit power level for the current Feedwater configuration following a loss of a Heater Drain Pump. Since the Heater Drain Pumps supply a portion of the available feedwater flow, power must be reduced in the event that a pump, or pumps, is lost. The table provides a listing of allowable power levels for combinations of Feedwater, Condensate, and Heater Drain Pumps. These power levels are based on past plant operating experience. As with the trip of the MFPs, power is reduced within the normal maximum maneuvering rates.
16	This step checks the number of Feedwater Pumps in service. If two pumps are running, the Heater Drain Pump trip could cause a substantial decrease in Feedwater Pump net positive suction head (NPSH). If only one Feedwater Pump is running, loss of one Heater Drain Pump will probably not challenge Feedwater Pump NPSH requirements. If NPSH requirements are challenged subsequent steps will check HCV-1459 operation.
N17	Informational note identifies the interlock associated with HCV-1459 and QCV-10624. When HCV-1459 is open, the secondary bypass for the Condensate Polishers will also open to ensure NPSH for the Feedwater Pumps.
17	HCV-1459, LP HEATERS BYPASS, will automatically open to ensure Feedwater Pump NPSH if: – Feedwater Pump suction pressure decreases to less than 300 psig with a Heater Drain Pump running. OR – Feedwater Pump suction pressure decreases to less than 350 psig with no Heater drain pumps running. This continuous action step checks the position of HCV-1459 and monitors Feedwater Pump suction pressure if the bypass valve has not opened.

INDIVIDUAL STEP DESCRIPTION:

Step	Description
1N18	The note is used to alert the Operator as to a possible cause of the Heater Drain Pump trip on low Heater Drain Tank level.
2N18	This note reminds the operator that LCV-1530B fails open on a loss of air.
3N18	This note explains the expected transient relative to LCV-1530B should LCV-1530A fail open (the intent of the step). This will assist the operator in understanding actions required in the subsequent step.
4N18	This note provides diagnostic information. This note and the series of notes above it are designed to assist the operator with the diagnosis of LCV-1530 operation. These valve could be in a variety of positions from the HDP trip or because their failure caused the HDP trip.
18	A possible cause for the Heater Drain Pump trip is due to low Heater Drain Tank level. This step is used to assist in diagnosing the reason for the Heater Drain Pump trip by checking if Heater Drain Tank HI/LO annunciator was received. If an abnormal condition is experienced in the Heater Drain Tank, an Operator is dispatched to check operation of level control valves. LCV-1530A could have failed open due to a malfunction of the controller, LC-1530. In any event, I&C assistance will be required for this malfunction because LCV-1530A can not be operated in local manual. If an extended period of malfunction is expected, the SSO may elect to effect local manual control of the bypass around LCV-1530A, however, this is not included as part of this procedure. If that is the desired option, power should be stabilized and a plan formulated with appropriate pre-job briefings held outside of the AOP framework.
19	HCV-1459, LP Heaters Bypass, may have received a signal to automatically open when the Heater Drain Pump tripped in order to ensure adequate NPSH to the Feedwater Pumps. This step checks the position of HCV-1459. If HCV-1459 is open, the RNO directs the Operator to continue reducing power until Feedwater Pump suction pressure increases to 400 psig, at which time HCV-1459 will close.
20	At this point all applicable actions have been completed for a HDP trip and the Operator is directed to steps which will prepare to exit the procedure.

12. 056 AG2.4.47 001

Given the following plant conditions:

- The plant is at 100% RTP when offsite power is lost.
- The crew has implemented EPP-5, Natural Circulation Cooldown.
- RCS Temperature is 288°F and lowering.
- RCS Pressure is 345 psig and stable.

Which ONE (1) of the following describes the indications that are directed to be used IAW EPP-5 to determine if a Steam Void is present in the Reactor Vessel?

Using ICCM A and B and LR-459, PZR Level, voids are determined to be present in the Reactor Vessel by observing ....

A.✓ RVLIS upper range, PZR level changes

B. RVLIS upper range, Core Exit Thermocouple Temperatures

C. subcooling, PZR level changes

D. subcooling, Core Exit Thermocouple Temperatures

The correct answer is A.

A. Correct. Indications are listed in step 28 of EPP-5. PZR Level is checked for large unexpected variations OR RVLIS upper range indication is checked to see if it is less than 100%. Either indication would be indication of a steam void in the reactor vessel. If a void is present then steps would be taken to collapse the void IAW EPP-6.

B. Incorrect. RVLIS upper range is correct. CETC temperatures is an indication of heat remaining in the reactor vessel but is not an indication of steam void formation. Subcooling and CETC temperatures are monitored during natural circulation cooldown but are not indications that can definitively show steam void formation in the vessel.

C. Incorrect. Subcooling is used during RCS cooldown and depressurization as limit for stopping the depressurization. When 25°F/hr rate is used the subcooling limit is 50°F. When 10°F/hr rate is used the subcooling limit is 100°F. PZR level changes is used to determine steam voids.

D. Incorrect. Subcooling is used during RCS cooldown and depressurization as limit for stopping the depressurization. When 25°F/hr rate is used the subcooling limit is 50°F. When 10°F/hr rate is used the subcooling limit is 100°F. CETC temperatures is an indication of heat remaining in the reactor vessel but is not an indication of steam void formation. Subooling and CETC temperatures are monitored during natural circulation cooldown but are not indications that can definitively show steam void formation in the vessel.

Question 12
Tier 1 / Group 1
K/A Importance Rating - RO 4.2 SRO 4.2

Loss of Offsite Power: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Reference(s) - System Design / Simulator, EPP-5
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-5-004
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.12
Comments: K/A match because candidate is given a loss of offsite power situation and must know which instruments are utilized to determine is a void is forming in the RV Head while on natural circulation.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

END PATH PROCEDURE

EPP-5

NATURAL CIRCULATION COOLDOWN

REVISION 14

ANSWER ON PAGE 15

LIST OF EFFECTIVE PAGES

EFFECTIVE PAGES	REVISION
Cover Sheet	14
LEP	14
3 through 24	14

PRR 165804

Summary of Changes

Step 13: Reformatted to be similar to step 24.

Note 18: Added new note for Supplement K.

Step 33 RNO; Added sub-step reminding Operator that SI-855 should be closed.

Step 36: Reformatted step to provide by-pass around step for Seal Injection if seals have been isolated.

Step 37: Deleted sub-step for PORV keys.

Step 42: Changed wording slightly.

Attachment 1: New attachment for C/A steps.

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides actions to perform a natural circulation cooldown and depressurization to cold shutdown, with no accident in progress, under requirements that will preclude any upper head void formation.

2. ENTRY CONDITIONS

a. EPP-2, Loss Of All AC Power Recovery Without SI Required, after the plant conditions have been stabilized following the restoration of AC emergency power.

b. Outside the EOP Network when a natural circulation cooldown is required.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*1.	Check SI Signal - INITIATED	IF SI initiation occurs during this procedure, THEN Go To Path-1, Entry Point A. Go To Step 3.
2.	Go To Path-1, Entry Point A	
3.	Open Foldout A	
4.	Perform The Following:	
	a. Reset SPDS	
	b. Initiate monitoring of Critical Safety Function Status Trees	
*5.	Determine RCP Seal Status:	
	a. Check RCP Seal Cooling - PREVIOUSLY LOST	a. Go To Step 6.
	b. Contact Plant Operations Staff for seal status evaluation while continuing with this procedure	
	c. Check seal status evaluation - COMPLETED	c. WHEN seal status evaluation is completed, THEN Go To Step 5.d. Observe CAUTION prior to Step 9 and Go To Step 9.
	d. Check RCP status - SEALS INTACT AND RCPS MAY BE RESTARTED	d. Observe CAUTION prior to Step 9 and Go To Step 9.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Try To Restart An RCP As Follows:	
	a. Establish conditions for starting an RCP using OP-101, Reactor Coolant System and Reactor Coolant Pump Startup and Operation, while continuing with this procedure	
	b. Check conditions for starting an RCP - ESTABLISHED	b. IF conditions for starting an RCP can be established during this procedure, THEN Go To Step 6. Verify natural circulation using Supplement E. Observe CAUTION prior to Step 9 and Go To Step 9.
	c. Run RCPs in the priority order of C, B, A to provide PZR spray	
	d. Start one RCP using OP-101, Reactor Coolant System and Reactor Coolant Pump Startup and Operation	d. Verify natural circulation using Supplement E. Observe CAUTION prior to Step 9 and Go To Step 9.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Establish Normal PZR Spray As Follows:	
	a. Check RCP status - ONLY ONE RUNNING	a. Stop all but one RCP.
	b. Check RCP C - RUNNING	b. Place PCV-455B, PZR Spray Valve Controller, in MAN <u>AND</u> adjust controller output to ZERO.
	c. Check RCP B - RUNNING	c. Place PCV-455A, PZR Spray Valve Controller, in MAN <u>AND</u> adjust controller output to ZERO. Go To Step 8.
	d. Maintain PZR level between 30% and 40%	
8.	Go To Appropriate Plant Procedure As Directed By The SSO <u>OR</u> CRSS	

****

<u>CAUTION</u>

PC-444J output signal will increase over time when system pressure is greater than the controller setpoint and may cause PCV-455C to open below 2335 psig.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Check PZR Heaters - POWER AVAILABLE	Energize 150 KW Of PZR Heaters Using EPP-21, Energizing Pressurizer Heaters From Emergency Busses, While Continuing With This Procedure
10.	Contact Chemistry To Obtain Boron Samples Of The RCS <u>AND</u> PZR	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

In the step below, the RCS must be over-borated to compensate for the effects of a PZR outsurge.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
11.	Ensure Adequate Shutdown Margin Exists As Follows:	
	a. Contact Plant Operations Staff to calculate the RCS Boron concentration required for Cold Shutdown allowing for a PZR outsurge	
	b. Check calculations - COMPLETED	b. WHEN the calculations are completed, THEN Go To Step 11.c.
	c. Check RCS Boron concentration - GREATER THAN REQUIRED	c. Perform the following: 1) Borate RCS to cold shutdown boron concentration using OP-301, Chemical And Volume Control System (CVCS). 2) Steam all intact S/Gs for at least 30 minutes prior to sampling to ensure adequate chemical mixing. 3) Contact Chemistry to obtain periodic boron samples of the following: • Reactor Coolant System • Pressurizer 4) WHEN the RCS is at cold shutdown boron concentration, THEN Go To Step 12.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
12.	Align RCS Makeup System As Follows: a. Verify FCV-113A, BORIC ACID FLOW Controller: 1) POT set per curves 4.1 <u>AND</u> 5.10 of the Plant Curve Book 2) In AUTO b. Verify FCV-113A, BA TO BLENDER Switch - IN AUTO c. Verify RCS MAKEUP MODE Switch - IN AUTO d. Momentarily place RCS MAKEUP SYSTEM Switch to START	
*13.	Verify Both CRDM Cooling Fans - RUNNING • HVH-5A • HVH-5B	<u>IF</u> any non-running CRDM Cooling Fan becomes available, <u>THEN</u> start the fan.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Excessive steam dump using the steam line PORVs may initiate a high steam line ΔP SI.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
14	Initiate RCS Cooldown To Cold Shutdown As Follows:	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 25°F IN THE LAST 60 MINUTE	
	b. Maintain RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
	c. Check steam dump to Condenser - AVAILABLE	c. Dump steam using STEAM LINE PORVs. Go To Step 14.e.
	d. Dump steam to Condenser	
	e. Control feed flow to maintain S/G levels - BETWEEN 39% <u>AND</u> 50%	
15	Check RCS Hot Leg Temperatures - LESS THAN 543°F	<u>WHEN</u> RCS hot leg temperatures less than 543°F, <u>THEN</u> Go To Step 16.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16.	Restore Steam Dumps As Follows:	
	a. Check steam dump to Condenser - AVAILABLE	Continue RCS cooldown using STEAM LINE PORVs. Observe NOTE prior to Step 17 and Go To Step 17.
	b. Momentarily place STEAM DUMP CONTROL Switch to BYPASS T-AVG INTLK position	
	c. Check APP-006-F5, STEAM DUMP ARMED - ILLUMINATED	
	d. Continue RCS cooldown using Steam Dump to Condenser	

NOTE

Low Tavg SI initiation circuits will automatically unblock if Tavg increases to greater than 543°F.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
17.	Defeat Low Tavg Safety Injection Signal As Follows:	
	a. Momentarily place SAFETY INJECTION T-AVG Selector Switch to BLOCK position	
	b. Verify LO TEMP SAFETY INJECTION BLOCKED status light - ILLUMINATED	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Supplement K is available for optimizing Auxiliary Spray below.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
18.	Depressurize RCS To 1950 PSIG As Follows:	
	a. Establish Auxiliary Spray as follows:	a. Use one PZR PORV.
	1) Establish Letdown using OP-301, Chemical And Volume Control System (CVCS)	
	2) Use CVC-311, AUX PZR SPRAY	
	b. Check RCS pressure - LESS THAN 1950 PSIG	b. <u>WHEN</u> RCS pressure less than 1950 psig, <u>THEN</u> Go To Step 18.c.
	c. Stop RCS depressurization	

NOTE

Low Pressure SI initiation circuits will automatically unblock if PZR pressure increases to greater than 2000 psig.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
19.	Defeat Low Pressure Safety Injection Signal As Follows:	
	a. Momentarily place PZR PRESS/HI STM LINE DP Switch to BLOCK position	
	b. Verify LO PRESS SAFETY INJECTION BLOCKED status light - ILLUMINATED	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
20.	Maintain The Following RCS Conditions: • RCS pressure - APPROXIMATELY 1950 PSIG • PZR level - BETWEEN 30% <u>AND</u> 40% • Cooldown rate in RCS cold legs - LESS THAN 25°F IN THE LAST 60 MINUTE • RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
21.	Monitor RCS Cooldown: • Check Core exit T/Cs - DECREASING • Check RCS hot leg temperatures - DECREASING • Check RCS subcooling - INCREASING	Increase steam dump from intact S/Gs.
*22.	Check Cooldown Rate Required - GREATER THAN LIMITS	<u>IF</u> the cooldown rates must be increased to greater than those allowed in this procedure, <u>THEN</u> Go To EPP-6, Natural Circulation Cooldown With Steam Void In Vessel. Observe ~~NOTE prior~~ to Step 24 and Go To Step 24.
23.	Go To EPP-6, Natural Circulation Cooldown With Steam Void In Vessel	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

If less than two CRDM Cooling Fans are running, maintaining RCS subcooling at 100°F and cooldown less than 10°F/hr precludes the formation of a steam void in the vessel upper head.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*24.	Verify CRDM Cooling Fans - BOTH RUNNING • HVH-5A • HVH-5B	Maintain RCS pressure approximately 1950 psig until RCS subcooling greater than 100°F. <u>IF</u> any non-running CRDM Cooling Fan becomes available, <u>THEN</u> start the fan. Go To Step 27.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
25.	Continue RCS Cooldown And Depressurization As Follows:	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 25°F IN THE LAST 60 MINUTE	
	b. Maintain RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
	c. Maintain RCS subcooling - GREATER THAN 50°F	c. Stop depressurization <u>AND</u> establish subcooling.
	d. Check steam dump to Condenser - AVAILABLE	d. Dump steam using STEAM LINE PORVs. Go To Step 25.f.
	e. Dump steam to Condenser	
	f. Control feed flow to maintain S/G levels - BETWEEN 39% AND 50%	
	g. Establish auxiliary spray as follows: 1) Establish Letdown using OP-301, Chemical And Volume Control System (CVCS) 2) Use CVC-311, AUX PZR SPRAY	g. Use one PZR PORV.
26.	Go To Step 28	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
27.	Continue RCS Cooldown And Depressurization As Follows:	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 10°F IN THE LAST 60 MINUTE	
	b. Maintain RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
	c. Maintain RCS subcooling - GREATER THAN 100°F	c. Stop depressurization <u>AND</u> establish subcooling.
	d. Check steam dump to Condenser - AVAILABLE	d. Dump steam using STEAM LINE PORVs. Go To Step 27.f.
	e. Dump steam to Condenser	
	f. Control feed flow to maintain S/G levels - BETWEEN 39% AND 50%	
	g. Establish auxiliary spray as follows: 1) Establish Letdown using OP-301, Chemical And Volume Control System (CVCS) 2) Use CVC-311, AUX PZR SPRAY	g. Use one PZR PORV.
28	Determine If Steam Void Present In Reactor Vessel: • Check PZR level - LARGE UNEXPECTED VARIATIONS <u>OR</u> • RVLIS upper range indication - LESS THAN 100%	Go To Step 30.

13. 058 AA2.01 001

The plant is at 100% RTP.

Battery Charger "B" is the In-Service Battery Charger and Battery Charger "B-1" is in Standby.

The following annunciators are received:

- APP-036-D3, BATT A/B LOW VOLT
- APP-036-D2, BATT CHARGER B/B-1 TROUBLE
- DC Bus B voltage indicates 123 volts and lowering.

Which ONE (1) of the following describes the status of the DC Bus and the Operator action that will be required IAW OP-601, DC Power?

Battery Charger B-1 ___(1)___. The "B" Batteries will be placed on ___(2)___ on the B-1 Battery Charger.

A.✓ (1) must be manually placed in service.

(2) float setting

B. (1) must be manually placed in service.

(2) equalize setting

C. (1) will Auto-Start to become the In-Service Battery Charger.

(2) float setting

D. (1) will Auto-Start to become the In-Service Battery Charger.

(2) equalize setting

The correct answer is A.

A. Correct - DC Bus voltage will lower slowly if the charger fails or trips. DC Bus operability is restored when voltage is greater than the limits and is connected to a full capacity charger and battery. Battery Charger B-1 must be manually placed in service IAW OP-601, DC Supply System, and will be placed on "float."

B. Incorrect - Distractor is correct with the exception of the equalize setting. The equalize setting is only used by Maintenance personnel to periodically apply an equalizing charge at an elevated voltage to restore the battery to full capacity.

C. Incorrect - EC 69420 has been implemented to ensure that In-Service battery chargers will return to service automatically when AC power is restored following a loss of AC. In this scenario AC power was not lost. Based on the indication Battery Charger B tripped. Therefore, Battery Charger B-1 would need to be manually placed in service IAW OP-601, DC Supply System. The candidate may be confused with the purpose and function of the newly installed EC and think that it will automatically place the standby charger in service trips.

D. Incorrect - See discussion above. Equalize setting is used to raise the charger output to charge the battery after a test or maintenance. The Equalize position is not utilized by Operations personnel in OP-601, DC Supply System.

Question 13
Tier 1 / Group 1
K/A Importance Rating - RO 3.7 SRO 4.1

Ability to determine and interpret the following as they apply to the Loss of DC Power: That a loss of DC power has occurred; verification that substitute power sources have come on line.

Reference(s) - System Design / Simulator, TS 3.8.4, APP-036, OP-601
Proposed References to be provided to applicants during examination - None
Learning Objective - DC 004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments: K/A match because the candidate must analyze conditions given from the DC bus and determine what action must be taken. In this case the standby battery charger must be placed in service to provide a new power source to the batteries.

ALARM

BATT CHARGER B/B-1 TROUBLE

AUTOMATIC ACTIONS/CAUSE

NOTE: Following a Loss of Off-Site Power (LOOP), the In Service Battery Charger will automatically restart. The annunciator MAY alert to momentary DC overvoltage condition during the battery charger start.

CAUSES	CHARGER	AUTO ACTION
DC Overvoltage	B and B-1	Charger trips
AC Input Failure	B and B-1	**IF** the battery charger is "In Standby", **THEN** it will trip. **IF** the battery charger is "In Service", **THEN** it should restart.
Fuse blown	B	Charger trips
DC Output Breaker open	B-1	Charger trips
Ground Test	B	None
DC Ground	B and B-1	None
DC Undervoltage	B-1	None
Fan Failure	B	None (See NOTE and ACTION 8)
High temperature	B	None

OBSERVATIONS

1. ERFIS Points APV3023A, DC MCC-B VOLT and APV3025A, DC MCC-B CURRENT

ACTIONS CK (✓)

1. **IF** a loss of MCC-6 has occurred, **THEN REFER** to AOP-024. _____
2. **IF** required, **THEN DISPATCH** an operator to check the Battery Charger. _____
3. **IF** the in service charger is determined to be inoperable **AND** the standby charger is available, **THEN PLACE** the standby charger in service IAW OP-601. _____
4. **IF** cause is **NOT** due to loss of power supply **OR** Charger failure, **THEN CHECK** for DC ground using OMM-035. _____
5. **IF** a ground is indicated, **THEN IDENTIFY AND ISOLATE** the grounded equipment. _____
6. **IF** a ground is suspected to exist on the battery, **THEN CONTACT** the Engineering to evaluate the possibility of isolating the battery with Battery Charger B **OR** B-1 carrying the bus load (either battery charger can be used to support this step). _____
7. **IF** Battery "B" discharged to less than 110 Volts **OR** overcharged to greater than 150 Volts, **THEN CONTACT** I&C to perform the applicable sections of MST-903 within 24 hours as required by ITS SR 3.8.6.2. _____

ACTIONS (Continued) CK (✓)

NOTE:	This Note and Step 8 applies to Battery Charger B only. Battery Charger B is **NOT** inoperable with one failed fan. The receipt of a high temperature alarm **OR** a failure of the remaining fan would render the battery charger inoperable.

8. **IF** a fan failure occurs, **THEN PERFORM** the following as applicable:

   a. **IF** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601 (this will reduce the frequency of the required compensatory action to check the local temperature alarm and remove the load from the charger with only one fan.) _____

   b. **IF** a standby battery charger is **NOT** available, **THEN CHECK** the local alarm display **AND** operating cooling fan on the In Service battery charger at least every two hours until the condition is resolved **OR** the other battery charger becomes available and is placed in service. _____

   c. **IF** the In Service battery chargers were transferred in Step 8.a above, **THEN CHECK** the local alarm display **AND** operating cooling fan on the standby battery charger for proper operation at least every two hours until the condition is resolved **OR** the battery charger is removed from service for Maintenance. _____

   d. **IF** a local high temperature alarm is observed **OR** the remaining operating fan fails on the In Service battery charger without a standby battery charger available, **THEN REFER TO** ITS 3.8.4 or 3.8.5. _____

   e. **IF** a local high temperature alarm is observed **OR** the remaining operating fan fails on the standby battery charger, **THEN SHUTDOWN** the standby battery charger per OP-601. _____

   f. **INITIATE** actions to have the failed fan repaired. _____

9. **IF** high temperature indication is present **AND** both fans are operating, **THEN PERFORM** the following as applicable:

   a. **IF** air flow under and over the In Service charger is blocked **THEN RESTORE** air flow under and over the In Service charger. _____

   b. **IF** there is no air obstruction, **THEN PLACE** the standby battery charger in service per OP-601 **AND SHUTDOWN** the charger with high temperature per OP-601. _____

10. **IF** blown fuse indication is present, **THEN PLACE** the standby battery charger in service per OP-601 **AND SHUTDOWN** the charger with blown fuse indication per OP-601. (The unit with the blown fuse indication is to be considered inoperable.) _____

ACTIONS (Continued) CK (✓)

11. **IF** low DC output voltage indication is present, **THEN PERFORM** the following as applicable:

    a. **IF** charger in float, **THEN ADJUST** float adjustment until output voltage is 133-135 V. _____

    b. **IF** charger in equalize, **THEN ADJUST** equlize adjustment until output voltage is 138-139 V. _____

    c. **IF** voltage can **NOT** be satisfactorily adjusted **AND** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601. _____

12. **IF** high or low voltage shutdown indication **AND** high DC output voltage is illuminated, **THEN PLACE** the standby battery charger in service per OP-601 if available. _____

13. **IF** low AC input voltage is illuminated **AND** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601. (When unit with low ac input voltage is placed in standby, it WILL trip.) _____

DEVICE/SETPOINTS

NOTE:	This note is applicable to Battery Charger B-1 and **NOT** for Battery Charger B. Ground alarm circuit cards are not calibrated to a certain pickup voltage. No set point or tolerances are specified during testing, only that the ground relay energizes and de-energizes by adjusting a 10 k ohm resistor on the test board. Per Power Conversion Products, the ground will actuate when a system ground of 13 k ohms is sensed.

1. Battery Charger B- X314/X311 (X305, X306, X307, X308, X309, X310, X312, X313) / Variable
2. Battery Charger B-1 - K1, K2, K4, K5, K7/ Variable

POSSIBLE PLANT EFFECTS

1. Low Battery voltage

REFERENCES

1. ITS LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6
2. OMM-035, Ground Isolation
3. OP-601, DC Supply System
4. MST-903, Station Battery Charge
5. AOP-024, Loss of Instrument Bus
6. CWD, B-190628, Sheet 956, Cable C
7. VTM 762-209-181

ALARM

BATT A/B LO VOLT

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Sustained loss of Battery Charger
2. Failure of a Battery or cell

OBSERVATIONS

1. ERFIS Points APV3022A, DC MCC-A VOLT and APV3024A, DC MCC-A CURRENT ERFIS Points APV3023A, DC MCC-B VOLT and APV3025A, DC MCC-B CURRENT

> **NOTE:** ITS SR 3.8.6.2 requires verification of battery cell parameters within 24 hours after a battery discharge to less than 110 Volts. This is accomplished by MST-903.
>
> **NOTE:** Minimum battery voltage for proper inverter operation is 109VDC. The inverter will trip when an input voltage of 100 VDC is sensed at the inverter.

ACTIONS		CK (✓)
1.	**CHECK** Battery voltage immediately.	_____
2.	**IF** required, **THEN REENERGIZE** in service battery charger **OR PLACE** standby battery charger in service IAW OP-601, DC Supply System.	_____
3.	**IF** a battery voltage has dropped below 110VDC, **THEN CONTACT** I&C to perform the applicable sections of MST-903 within 24 hours as required by ITS SR 3.8.6.2.	_____
4.	**IF** battery voltage drops below 109VDC, **THEN** consideration should be given to supplying the affected instrument bus from MCC-8.	_____
6.	**IF** Battery "A" voltage drops below 109VDC, **THEN** consideration should be given to switching APP-036 power supply to Distribution Panel C in accordance with OP-601.	_____
7.	**IF** the battery is declared inoperable, **THEN REFER** to ITS 3.8.6.	_____
8.	**IF** plant trip occurs, **THEN REFER** to EOP Network.	_____

DEVICE/SETPOINTS

1. 27/MCC-A / 123 VDC (Reset at 126.5 VDC)
2. 27/MCC-B / 123 VDC (Reset at 126.5 VDC)

POSSIBLE PLANT EFFECTS

1. Plant trip
2. Loss of DC

REFERENCES

1. ITS LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6
2. OP-601, DC Supply System
3. MST-903, Station Battery Charge
4. CWD, B-190628, Sheet 955, Cable M
5. EOP Network
6. Memo - APP-009-40, "A-B Battery Low Volts" Revision, Robinson File No. 5235, Serial: RNPD/91-2874
7. ESR 97-00243, Battery Terminal Voltages
8. PM-182, A" and "B" Station Batteries Undervoltage Alarm Check and Alignment
9. Tech. Manual 728-132-23

INIT VERI

8.2.3 Shifting IN-SERVICE Battery Charger on MCC-B From "B" to "B-1"

1. Initial Conditions

a. This revision has been verified to be the latest revision available. _______ ____
Date

b. The "B-1" Battery Charger is in Standby IAW Section 8.1.4. ____

c. A Safety Function Determination has been completed IAW OMM-007 **AND** no loss of Safety Function exists **OR** the loss of Safety Function is acceptable for the time the battery chargers will be disconnected from the battery bus. ____

2. Instructions

a. **VERIFY** "B-1" Battery Charger Float/Equalize switch on FLOAT. ____ ____

b. **VERIFY** equalize timer on Battery Charger "B-1" on zero. ____ ____

NOTE:	The following step will place the plant in Required Actions of ITS LCO 3.8.4 **AND** 3.8.9 (if in Modes 1, 2, 3, **OR** 4) **OR** ITS LCO 3.8.5 (if in Modes 5 **OR** 6) since neither battery charger will be aligned to the battery. (CR 97-00299) APP-036-D2, BATT CHARGER B/B-1 TROUBLE, and/or APP-036-D3, BATT A/B LO VOLT, may illuminate during the following step but should clear when the other battery charger breaker is closed.

c. On MCC-B, **OPEN** the Battery Charger breaker for "B" Battery Charger. ____ ____

d. **RECORD** time action statement entered. Time ________ ____

8.2.3.2 (Continued) INIT VERI

e. On MCC-B, **CLOSE** the Battery Charger breaker for “B-1” Battery Charger. ____ ____

NOTE:	EC 69420 is changing the label of the SW3 switch on Battery Charger “B-1”. If the label has **NOT** been changed it will be labeled “CHARGER B-1 IN SERVICE”. If the label has been changed it will be labeled “CHARGER IN SERVICE”.

f. **POSITION** SW3 switch on Battery Charger “B-1” to CHARGER “B-1” IN SERVICE (CHARGER IN SERVICE). ____ ____

g. **POSITION** S304 switch on Battery Charger “B” to CHARGER IN STANDBY. ____ ____

h. **RECORD** time action statement exited. Time _______ ____

i. **CHECK** Battery Charger “B-1” for DC Ground Indication. ____

j. **WHEN** battery charger has been in-service for 5 minutes, **THEN VERIFY** voltage between 133 and 135 Volts DC as indicated on the battery charger VOLTMETER. ____

	Initials	Name (Print)	Date
Performed by:	________	____________________	________
	________	____________________	________
	________	____________________	________

Approved by: ______________________________ __________
Shift Manager Date

14. 062 AA2.01 001

Given the following plant conditions.

- The plant is at 100% RTP.
- APP-008-F7, SOUTH SW HDR LO PRESS, and APP-008-F8, NORTH SW HDR LOW PRESS, alarms are received.
- A Security Guard reports a large amount of water flowing to the storm drains from the area between the Aux. Building and Radwaste Building.
- IAO identifies rupture at SW piping elbow on east side of Auxiliary Building.

Which ONE (1) of the following identifies the Service Water Header that is ruptured and the components than must be "VERIFIED STOPPED" or taken out of service IAW AOP-022, Loss of Service Water?

The __(1)__ Header has ruptured and the __(2)__ must be removed from service.

A. (1) South

(2) "B" EDG

B.✓ (1) North

(2) "B" EDG

C. (1) South

(2) "A" EDG

D. (1) North

(2) "A" EDG

The correct answer is B.

A. Incorrect - "A" EDG is cooled from the South Header. The North Header runs between the Aux. Building and the Radwaste Building above ground. The South Header is underground outside of the Auxiliary Building. The South Header comes out of the ground in the CCW Pump Room.

B. Correct

C. Incorrect - "A" EDG is cooled from the South Header. The North Header runs between the Aux. Building and the Radwaste Building above ground. The South Header is underground outside of the Auxiliary Building. The South Header comes out of the ground in the CCW Pump Room.

D. Incorrect - "A" EDG is cooled from the South Header. The North Header runs between the Aux. Building and the Radwaste Building above ground. The South Header is underground outside of the Auxiliary Building. The South Header comes out of the ground in the CCW Pump Room.

Question 14
Tier 1 / Group 1
K/A Importance Rating - RO 2.9 SRO 3.5

Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: Location of a leak in the Service Water System.

Reference(s) - System Design / Simulator, AOP-022, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-022-005
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 43.5 / 45.13
Comments: K/A match because candidate must analyze given conditions to determine the location of a Service Water leak and then determine the leak locations impact on plant components.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-022

LOSS OF SERVICE WATER

REVISION 34

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides instructions in the event of a break of either the North or South Service Water Headers upstream OR downstream of check valves SW-541 or SW-545 or flooding in the Intake Area Service Water Pits.

2. ENTRY CONDITIONS

   This procedure is entered whenever there is an indication that a break of a Service Water Header has occurred.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Step 1 is an immediate action step.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check The Following Alarms - EXTINGUISHED: • APP-008-E7, S SW HDR STRAINER PIT HI LEVEL • APP-008-E8, N SW HDR STRAINER PIT HI LEVEL	Perform the following: a. Close the following SW X-CONN Valves: • V6-12B • V6-12C b. Go To Section F.
* 2.	Check SW - ANY AVAILABLE	IF a total loss of Service Water occurs due to hostile action, THEN Go To EPP-28, Loss of Ultimate Heat Sink.
3.	Make PA Announcement For Procedure Entry	

NOTE

A SW Header leak may be identified by observing the sequence in which SW Header low pressure alarms are received, and evaluating SW Header pressure indications.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 4.	Check Leak Location - IDENTIFIED	Perform local inspections as necessary to determine leak location. WHEN the leak location is identified, THEN observe the NOTE prior to Step 5 and Go To Step 5. Observe the CAUTION prior to Step 6 and Go To Step 6

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

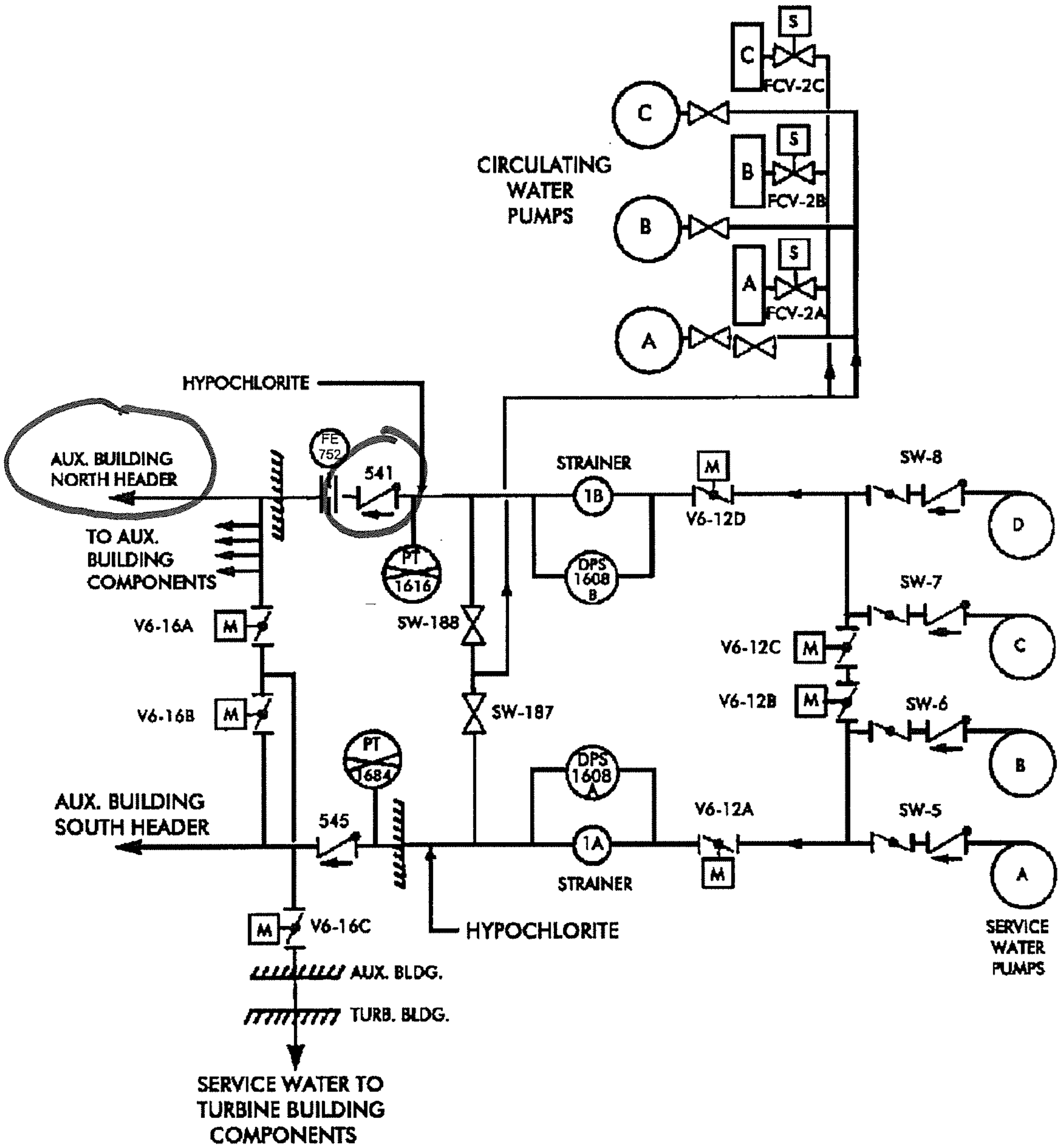
NOTE

- Turbine Building Service Water will automatically isolate following a Turbine Trip coincident with low Service Water pressure after a 60 seconds time delay.
- Supplement M, Component Alignment For Loss Of SW To Turbine Building is available in the EOP Network for securing components.
- AOP-022 does not provide guidance for leaks in the common return line in the Auxiliary building with the exception of controlling the flooding with Attachment 4.

5. Perform Appropriate Section For Leak Location:

LEAK LOCATION	**SECTION**
North Service Water Header Upstream of Check Valve SW-541	Section A
South Service Water Header Upstream of Check Valve SW-545	Section B
North Service Water Header Downstream of Check Valve SW-541	Section C
South Service Water Header Downstream of Check Valve SW-545	Section D
Turbine Building Service Water Header	Section E
Service Water Pits Flooding in Intake Area	Section F

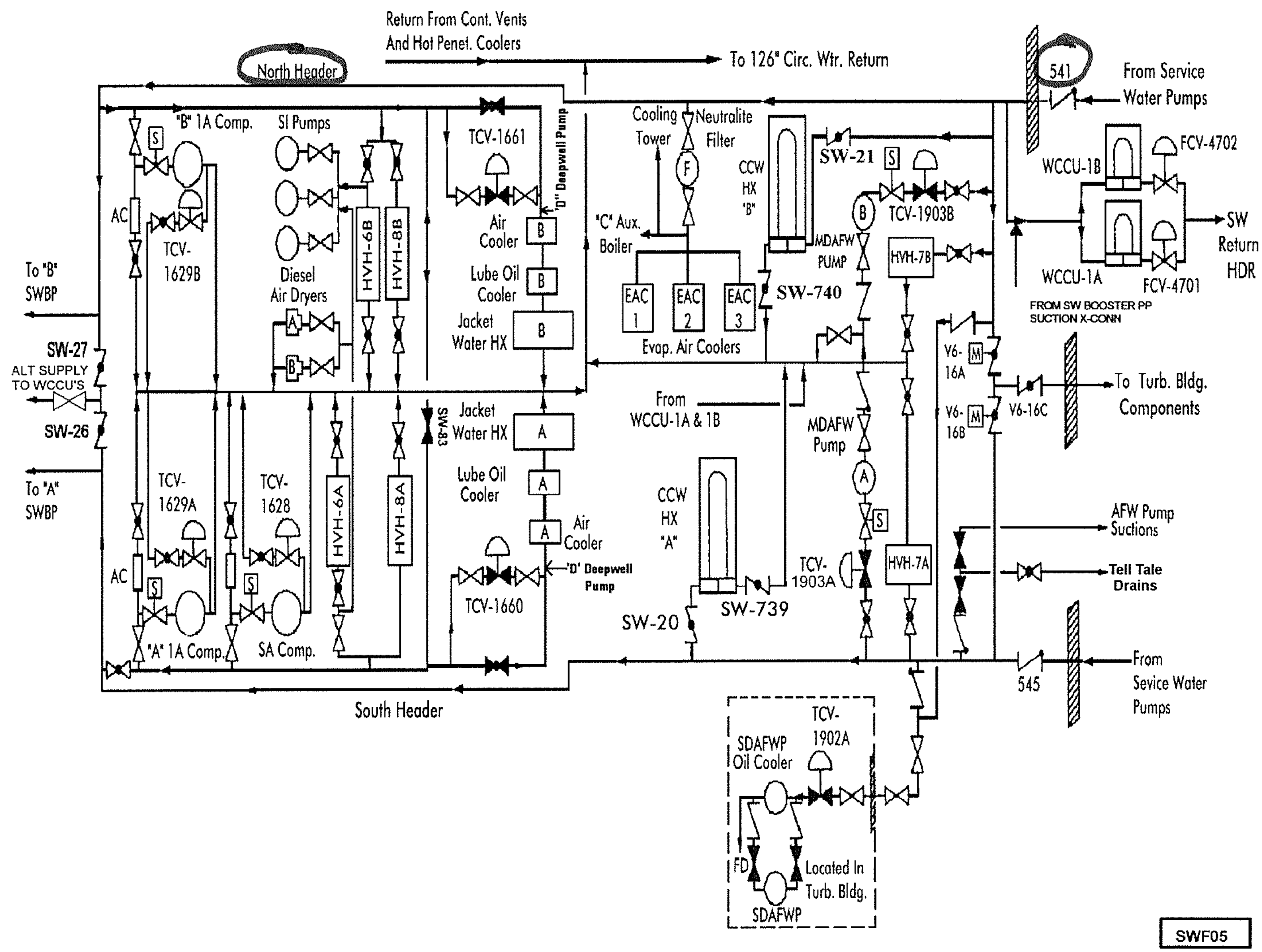
SERVICE WATER PUMPS
SW-FIGURE-3



INFORMATION USE ONLY

AUXILIARY BUILDING SERVICE WATER

SW-FIGURE-5



INFORMATION USE ONLY

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 1 of 10)

1. Close V6-12D, SW NORTH HDR ISO, to isolate the Service Water Header

2. Close V6-16A, SW TURB BLDG SUPPLY, to isolate the Turbine Building from the affected Service Water Header

3. Verify The Following Valves - OPEN:

   - V6-12A, SW SOUTH HDR ISO
   - V6-12B, SW X-CONN
   - V6-12C, SW X-CONN
   - V6-16B, SW TURB BLDG SUPPLY
   - V6-16C, SW TURB BLDG ISO

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 2 of 10)

****

CAUTION

Containment Fan Cooler Motors may operate for up to fifteen (15) minutes with a COMPLETE LOSS of cooling water flow during non-accident conditions. (Ref. ESR 95-00700)

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 4.	Verify The Following SW BOOSTER PUMP Alignment:	
	a. SW BOOSTER PUMP A - RUNNING	a. WHEN plant conditions support starting SW BOOSTER PUMP A, THEN Verify SW BOOSTER PUMP A - RUNNING. IF a SW BOOSTER PUMP is not started within 15 minutes, THEN Stop the running HVH Units.
	b. SW BOOSTER PUMP B - STOPPED	

NOTE

- Isolation of the North Service Water Header places the unit in Technical Specification Action Statement 3.7.7.A.
- IF a SW Header is out of service for maintenance, THEN SW-83, DIESEL CROSS-CONNECT, is likely to be open and may need to be closed to isolate a leak in the North Header.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Perform Attachment 6 While Continuing With This Procedure	Removes 'B' CCW HX From Service

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 3 of 10)

****

CAUTION

Electrically rated boots and gloves must be worn in the Auxiliary Building where flooding is in progress.

****

NOTE

- A ruler can be obtained from the AUX BUILDING FLOODING TOOL BAG.
- RHR Pit flooding may occur when Auxiliary Building first floor water level exceeds 6 inches.
- Dielectric rubber boots and low voltage electrical gloves are available in the AOP/DSP/EOP Equipment Storage Locker inside the Auxiliary Building.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 6.	Determine If Actions For Auxiliary Building Flooding Are Necessary As Follows:	
	a. Check for any of the following indications of flooding: • Water level on Auxiliary Building first floor - GREATER THAN 6 INCHES BY LOCAL INDICATION OR • APP-001-E4, RHR PIT A HI LEVEL - ILLUMINATED OR • APP-001-E5, RHR PIT B HI LEVEL - ILLUMINATED	a. IF at any time flooding is indicated, THEN perform Attachment 4 while continuing with this procedure. Go To Step 7.

(CONTINUED NEXT PAGE)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 4 of 10)

6. (CONTINUED)

   b. Perform Attachment 4 while continuing with this procedure

7. Verify The Following Equipment - STOPPED:

   - EMERG DIESEL GENERATOR B
   - AFW PUMP B

****

CAUTION

Confined Space entry requirements must be observed to access the North SW Strainer Pit.

****

NOTE

- SW-188, NORTH HDR SUPPLY TO SCRN WASH & CW PMP GLAND SEAL, is located in the North SW Strainer Pit.
- SW-839 and SW-845, NORTH SW HEADER CHEMICAL INJECTION, are located above the North SW Strainer Pit on the North side.
- Key #91 OR the Security Key is required to access the SW Strainer Pits.

8. Verify The Following Valves At The Intake Structure - CLOSED:

   - SW-188
   - SW-839
   - SW-845

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 5 of 10)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Check Circulating Water Pump Status - ANY RUNNING	Go To Step 12.
*10.	Check SW-188 - CLOSED	WHEN SW-188 is closed, THEN perform Step 11. Go To Step 12.
11.	Determine If Adequate Seal Water Is Available To Circulating Water Pumps As Follows: • APP-008-E4, CW PMP A SEAL WTR LOST - EXTINGUISHED AND • APP-008-E5, CW PMP B SEAL WTR LOST - EXTINGUISHED AND • APP-008-E6, CW PMP C SEAL WTR LOST - EXTINGUISHED	Perform Attachment 5 while continuing with this procedure.
12.	Check Steps 1 And 2 Of Attachment 6 - COMPLETE	WHEN Steps 1 and 2 of Attachment 6 are complete, THEN Go To Step 13.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 6 of 10)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
13.	Check South SW Header Pressure On PI-1684 - GREATER THAN 40 PSIG	Perform one or both of the following to restore South SW Header pressure to between 40 psig and 50 psig: • Start additional SW Pump(s). OR • Throttle SW flow from CCW Heat Exchanger A as follows: a. Throttle SW-739, CCW HEAT EXCHANGER "A" RETURN, to establish SW pressure between 40 psig and 50 psig as indicated by PI-1619A.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

<u>SECTION C</u>

<u>LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING</u>

(Page 7 of 10)

****

<u>CAUTION</u>

CCW temperature is limited to 105°F when any of the following equipment is in service: Post Accident Sampling Heat Exchanger, and Excess Letdown Heat Exchanger.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
14.	Determine Maximum Allowable CCW Temperature As Follows:	
	a. Check RCS temperature - LESS THAN <u>OR</u> EQUAL TO 350°F	a. Maintain CCW Heat Exchanger outlet temperature indicated on TI-607 less than or equal to 105°F. Go To Step 15.
	b. Check all the following equipment - SECURED: • Post Accident Sampling Heat Exchanger • Excess Letdown Heat Exchanger	b. Maintain CCW Heat Exchanger outlet temperature indicated on TI-607 less than or equal to 105°F. Go To Step 15.
	c. Maintain CCW Heat Exchanger outlet temperature indicated on TI-607 less than or equal to 125°F	
*15.	Check Step 11 Of Attachment 6 - COMPLETE	<u>WHEN</u> Steps 11 of Attachment 6 is complete, <u>THEN</u> observe the <u>CAUTION</u> prior to Step 16 and perform Step 16. Observe the <u>CAUTION</u> prior to Step 17 and Go To Step 17

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 8 of 10)

****

CAUTION

Containment Fan Cooler Motors may operate for up to fifteen (15) minutes with a COMPLETE LOSS of cooling water flow during non-accident conditions. (Ref. ESR 95-00700)

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16.	Determine If A SW Booster Pump Should Be Started:	
	a. Check SW Booster Pumps - ALL STOPPED	a. Verify ONE SW Booster Pump is running. Observe the CAUTION prior to Step 17 and Go To Step 17
	b. Start one SW Booster Pump	

****

CAUTION

Operation of the Safety Injection Pumps without adequate cooling water to the thrust bearing could result in pump damage.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
17.	Check Service Water Cooling From SI Pumps To Drain Header - FLOW OBSERVED	Perform Attachment 1 while continuing with this procedure.
18.	Align Fire Water To The Control Room HVAC Using Attachment 3 While Continuing With This Procedure	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 9 of 10)

****

CAUTION

Normal cooling water supply to MOTOR DRIVEN AFW PUMP B is unavailable. Pump operation prior to establishing emergency cooling could result in pump damage.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
19.	Align Emergency Cooling Water To MOTOR DRIVEN AFW PUMP B Using Attachment 2 While Continuing With This Procedure	
20.	Check AUXILIARY BOILER C - SHUTDOWN	Notify Chemistry personnel that cooling water has been lost to Auxiliary Boiler C Sample Cooler.
21.	Perform The Following:	
	a. Inspect the area of the leak	
	b. Report findings to the Control Room	
	c. Identify and isolate the source of the SW leak	
*22.	Check Attachment 6 - COMPLETED	WHEN Attachment 6 has been completed, THEN perform Step 23 Go To Step 24
23.	Check SW to Instrument Air Compressor B - ISOLATED	IF needed to supply Instrument Air, THEN Start Instrument Air Compressor B using OP-905 while continuing with this procedure. Go To Step 24
24.	Refer To Technical Specifications For Any Applicable LCOs	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF NORTH SERVICE WATER HEADER IN AUXILIARY BUILDING

(Page 10 of 10)

25. Implement The EALs

26. Return To Procedure And Step In Effect

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 6

ISOLATION OF NORTH SW HEADER IN THE AUXILIARY BUILDING

(Page 1 of 4)

****

CAUTION

Electrically rated boots and gloves must be worn in the Auxiliary Building where flooding is in progress.

****

NOTE

Dielectric rubber boots and low voltage electrical gloves are available in the AOP/DSP/EOP Equipment Storage Locker inside the Auxiliary Building.

1. Close One Of The Following Valves In The Auxiliary Building Hallway:

- SW-18, NORTH & SOUTH SUPPLY HDR X-CONN (CHAIN OPERATED)

  OR

- SW-19, NORTH & SOUTH SUPPLY HDR X-CONN (CHAIN OPERATED)

NOTE

SW-102 is located overhead East of CCW PUMP B on top of horizontal header supported by adjustable pipe support.

2. Close The Following Valves In The CCW Pump Room:

- SW-740, CCW HEAT EXCHANGER "B" RETURN

- SW-102, HVH-7B SUPPLY

REMOVES 'B' CCW HX FROM SERVICE

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 6

ISOLATION OF NORTH SW HEADER IN THE AUXILIARY BUILDING

(Page 2 of 4)

3. Notify Control Room Personnel That Steps 1 And 2 Of Attachment 6 Are Complete

4. Place INSTRUMENT AIR COMPRESSOR B Control Switch In The OFF Position

****

CAUTION

Subsequent actions to cross-connect cooling water to plant components should not be performed if the leak will be reinitiated.

****

NOTE

SW-53, NORTH SUPPLY HDR TO "B" TRN COMPONENTS IN AUX BLDG, is located in Auxiliary Building hallway, midway between INSTRUMENT AIR DRYERS A and B, above third cable tray.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Check Leak Location - DOWNSTREAM OF SW-53	Go To Step 9
6.	Close SW-53	
7.	Notify Control Room personnel that SW can NOT be cross-connected to supply the following equipment: • EDG B • INSTRUMENT AIR COMPRESSOR B • HVH-6B • HVH-8B	
8.	Go To Step 14.	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 6

ISOLATION OF NORTH SW HEADER IN THE AUXILIARY BUILDING

(Page 3 of 4)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Check Leak Location - UPSTREAM OF SW-25, NORTH HDR SUPPLY TO SW BOOSTER PUMPS	Close SW-25 Notify Control Room personnel that SW can NOT be cross-connected to supply SW BOOSTER PUMP B. Observe the NOTE prior to Step 12 and Go To Step 12

NOTE

- SW BOOSTER PUMP B may be operated as required after the suction path is established from the South Service Water Header.
- Closing SW-503, SW PUMP SUPPLY TO IVSW TANK, also isolates SW to the Penetration Coolers.

10. Cross-Connect SW Booster Pump Suction Supply As Follows:

    a. Close SW-25, NORTH HDR SUPPLY TO SW BOOSTER PUMPS

    b. Open SW-26, SW BOOSTER PUMP SUCTION CROSS-CONNECT

    c. Open SW-27, SW BOOSTER PUMP SUCTION CROSS-CONNECT

    d. Close SW-503, SW PUMP SUPPLY TO IVSW TANK

    e. Open SW-200A, SWBP "A" TO IVSW TANK SUPPLY

11. Notify Control Room Personnel That Step 11 Is Complete And SW Has Been Cross-Connected To Supply SW BOOSTER PUMP B

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 6

ISOLATION OF NORTH SW HEADER IN THE AUXILIARY BUILDING

(Page 4 of 4)

NOTE

- SW-83, DIESEL SUPPLY CROSS-CONNECT, is located at the North end of EMERGENCY DIESEL GENERATOR A.
- Cross-connecting Service Water to the Emergency Diesel Generators will also cross-connect Service Water to HVH-6B, HVH-8B, and INSTRUMENT AIR COMPRESSOR B.
- A ladder is required for access to SW-53.

12. Cross-Connect Service Water Supply To EDGs As Follows:

    a. Close SW-53

    b. Open SW-83

13. Notify Control Room Personnel That SW Has Been Cross-Connected To Supply The Following Equipment:

    - EDG B
    - INSTRUMENT AIR COMPRESSOR B
    - HVH-6B
    - HVH-8B

14. Notify Control Room Personnel That Attachment 6 Is Complete

- END -

15. 077 AG2.4.45 001

Given the following plant conditions:

- The plant is operating at 85% RTP.
- APP-009-D7, GEN LO FREQ, was received 30 seconds ago.
- AOP-026, GRID INSTABILITY, has been entered.
- Grid frequency is at 58.3 Hz and slowly lowering following a load rejection.

.

The crew has determined that the load rejection was 110 MWe.

Which ONE (1) of the following actions will be taken?

A. Trip the Reactor and Go to PATH-1 ONLY.

B.✓ Trip the Reactor, Stop the RCPs and Go to PATH-1.

C. Transition to AOP-015, Secondary Load Rejection ONLY.

D. Transition to AOP-015, Secondary Load Rejection, while continuing with AOP-026.

The correct answer is B.

A. Incorrect. This is plausible if the student does not remember that the RCPs are tripped due to the low system frequency to prevent an automatic trip of RCPs at 58.2 Hz.

B. Correct.

C. Incorrect. Plausible if the student fails to recognize that the trip criteria of 58.4 Hz has been exceeded. Also, student fails to recall that AOP-015 is performed while continuing in AOP-026 if a load rejection of greater than 100 MWe occurs.

D. Incorrect. Plausible if the student fails to recognize that the trip criteria of 58.4 Hz has been exceeded.

Question 15
Tier 1 / Group 1
K/A Importance Rating - RO 4.1 SRO 4.3

Grid Disturbances: Ability to prioritize and interpret the significance of each annunciator or alarm.

Reference(s) - System Design / Simulator, AOP-026
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-026-003
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.10 / 43.5 / 45.3 / 45.12
Comments: K/A match because candidate is given information relative to a grid disturbance and must analyze the indications. Based on the analysis the candidate must determine the appropriate actions to take.

ALARM

GEN LO FREQ

AUTOMATIC ACTIONS

1. Turbine Governor will attempt to pickup load (auto action of governor)

CAUSE

1. Grid frequency drop due to inadequate generation to supply system demand.

OBSERVATIONS

1. Frequency Meter (RTGB)
2. Generator Load
3. Reactor power

ACTIONS — CK (✓)

1. **IF** Generator frequency is less than 59.8 Hz, **THEN REFER TO** AOP-026. (SOER 99-01, R2a) ______

DEVICE/SETPOINTS

1. Under Frequency Relay on Generator Output / 59.8 Hz

POSSIBLE PLANT EFFECTS

1. Reactor trip
2. Loss of off-site power

REFERENCES

1. AOP-026, Grid Instability
2. CWD, B-190628, Sheet 908, Cable R
3. SOER 99-01, Loss Of Grid, recommendation 2a

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-026

GRID INSTABILITY

REVISION 9

CORRECT TRANSITIONS

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides instructions in the event a grid instability condition occurs.

NOTE

Notification from the Load Dispatcher that adequate voltage support is NOT available in the event of a LOCA requires entry into ITS 3.8.1.

2. ENTRY CONDITIONS

- Observed frequency less than 59.8 Hz
- Load Dispatcher informs the Control Room a low frequency conditions exists.
- Load Dispatcher informs the Control Room that a condition exists in which adequate voltage support can NOT be provided by the system grid in the event of a LOCA.

\- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Determine If Reactor Trip Is Required As Follows:	
	a. Check Plant Status - MODE 1 OR 2	a. Go To Step 2.
	b. Check system frequency - LESS THAN 58.4 HZ YES	b. IF either of the following conditions occur, THEN perform steps 1.c through 1.e. • Frequency decreases to less than 58.4 Hz OR • Frequency remains between 58.4 Hz and 59.0 Hz for greater than 5 minutes Go To Step 3.
	c. Trip the Reactor	
	d. Stop the RCPs	
	e. Go To Path-1	

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-026

GRID INSTABILITY

REVISION 9

INCORRECT TRASITION

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides instructions in the event a grid instability condition occurs.

NOTE

Notification from the Load Dispatcher that adequate voltage support is NOT available in the event of a LOCA requires entry into ITS 3.8.1.

2. ENTRY CONDITIONS

- Observed frequency less than 59.8 Hz
- Load Dispatcher informs the Control Room a low frequency conditions exists.
- Load Dispatcher informs the Control Room that a condition exists in which adequate voltage support can NOT be provided by the system grid in the event of a LOCA.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Determine If Reactor Trip Is Required As Follows:	
	a. Check Plant Status - MODE 1 OR 2	a. Go To Step 2.
	b. Check system frequency - LESS THAN 58.4 HZ	b. IF either of the following conditions occur, THEN perform steps 1.c through 1.e. • Frequency decreases to less than 58.4 Hz OR • Frequency remains between 58.4 Hz and 59.0 Hz for greater than 5 minutes Go To Step 3.
	Wrong Transition	
	c. Trip the Reactor	
	d. Stop the RCPs	
	e. Go To Path-1	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

With Unit #2 separated from the grid, system frequency can be obtained from the Load Dispatcher OR Unit #1 when it is connected to the grid.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 2.	Determine If Reactor Coolant Pump Trip In Modes 3, 4, OR 5 Is Required As Follows:	
	a. Check system frequency - LESS THAN 58.4 HZ	a. IF frequency decreases to less than 58.4 Hz, THEN perform Steps 2.b and 2.c . IF all running RCPs trip on low frequency, THEN perform Step 2.c. Go To Step 3.
	b. Stop Running RCPs.	
	c. Verify one of the following methods of core cooling available: • RHR aligned for core cooling OR • Natural circulation using Supplement E, Natural Circulation Verification.	
3.	Make PA Announcement For Procedure Entry	
4.	Check ANY of the following Diesel Generators - PARALLELED TO THE GRID. • EDG A • EDG B • DS Diesel	Go To Step 6.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Shutdown paralleled Diesel Generator using applicable operating procedure in effect.	
6.	Check Turbine Status - ON LINE	Go To Step 28.
7.	Check Turbine Load - INCREASED DUE TO DROP IN SYSTEM FREQUENCY	<u>IF</u> Turbine load increases due to decrease in frequency, <u>THEN</u> perform Steps 8 through 13. Go To Step 14.
8.	Check Reactor Power - LESS THAN <u>OR</u> EQUAL TO 100%	Reduce Turbine load using the Valve Position Limiter to maintain Reactor power less than 100%.
9.	Limit Turbine Load Increase, Based On Current Operating Equipment, Using The Valve Position Limiter	
10.	Check S/G Levels - TRENDING TO PROGRAM LEVEL	Control feed flow to restore S/G levels to programmed level.
11.	Check T_{avg} - TRENDING TO T_{ref}	Restore T_{avg} within -1.5 to +1.5°F of T_{ref} as follows: • Place ROD BANK SELECTOR to M (Manual) <u>AND</u> manually position rods. <u>OR</u> • Dilute the RCS using OP-301, Chemical And Volume Control System (CVCS), while continuing with this procedure.
12.	Check PZR Pressure - TRENDING TO 2235 PSIG	Control PZR Heaters and Spray Valves to restore PZR pressure between 2220 psig and 2250 psig.
13.	Check PZR Level - TRENDING TO REFERENCE LEVEL	Control charging and letdown to restore programmed PZR level.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*14.	Check Turbine Load Reduction - • IN PROGRESS OR • HAS OCCURRED	IF a Turbine load reduction occurs, THEN perform Steps 15 through 19. Go To Step 20.
15.	Check Turbine Load Reduction - LESS THAN 100 MWE	Go To AOP-015, Secondary Load Rejection Or Turbine Runback, while continuing with this procedure. Go To Step 20.
16.	Check S/G Levels - TRENDING TO PROGRAM LEVEL	Control feed flow to restore S/G levels to programmed level.
17.	Check T_{avg} - TRENDING TO T_{ref}	Restore T_{avg} within -1.5 to +1.5°F of T_{ref} as follows: • Place ROD BANK SELECTOR to A (Auto) AND check auto rod insertion. OR • Place ROD BANK SELECTOR to M (Manual) AND manually insert rods. OR • Borate the RCS using OP-301, Chemical And Volume Control System (CVCS), while continuing with this procedure.
18.	Check PZR Pressure - TRENDING TO 2235 PSIG	Control PZR Heaters and Spray Valves to restore PZR pressure between 2220 psig and 2250 psig.
19.	Check PZR Level - TRENDING TO REFERENCE LEVEL	Control charging and letdown to restore programmed PZR level.
20.	Check Additional Main Generator Capacity - AVAILABLE	Go To Step 24.

BASIS FOR DISTRACTOR C ; D

16. W/E 04 EK3.2 001

Given the following plant conditions:

- The crew is performing the actions in EPP-20, LOCA Outside Containment.
- Auxiliary Building radiation levels are lowering.
- Safety Injection flow is 80 GPM and lowering.
- PZR level is off scale LOW.
- RCS pressure is 1450 psig and rising.

Which ONE (1) of the following identifies the leak status IAW EPP-20?

The leak is.....

A.✓ isolated because RCS pressure is rising.

B. NOT isolated because SI pump flow still exists.

C. NOT isolated because PZR level is not on scale.

D. isolated because Auxiliary Building radiation levels are lowering.

The correct answer is A.

A: Correct. EPP-20 basis supports that the most likely break location is in the low pressure piping of the RHR system. RCS pressure increase is the indicator used to ensure that the leak has been isolated.

B: Incorrect. At 1450 psig, SI pump will be injecting into the RCS, but very close to shutoff head of the pump.

C: Incorrect. RCS pressure is used to ensure that isolation has occurred. RCS inventory in the PZR may be delayed, depending on how much inventory was lost during the event.

D: Incorrect. RCS pressure increase is the indicator used for verifying isolation, not Auxiliary Building radiation levels dropping.

Question 16
Tier 1 / Group 1
K/A Importance Rating - RO 3.4 SRO 4.0

Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment): Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment).

Reference(s) - System Design / Simulator, EPP-20, EPP-20 Basis Document
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-20-004
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 41.10 / 45.6 / 45.13
Comments: K/A match because the candidate must understand how to determine if the leak is isolated during the performance of EPP-020, LOCA Outside Containment.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

END PATH PROCEDURE

EPP-20

LOCA OUTSIDE CONTAINMENT

REVISION 8

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to identify and isolate a LOCA outside Containment.

2. ENTRY CONDITIONS

   Path-1, when there is abnormal radiation in the auxiliary building due to a loss of RCS inventory outside Containment.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Verify RHR Loop Supply - ISOLATED	
	a. At MCC-5, locally close the breaker for RHR-750, RHR PUMP SUCTION FROM RCS (CMPT-12C)	
	b. Verify CLOSED RHR-750, RHR LOOP SUPPLY.	b. Perform the following: 1) At MCC-6, locally close the breaker for RHR-751, RHR PUMP SUCTION FROM RCS (CMPT-8M) 2) Verify CLOSED RHR-751, RHR LOOP SUPPLY.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Verify The Following Valves - CLOSED:	
	a. SI-869, SI HOT LEG HDR	a. Verify CLOSED Loop 2 and Loop 3 Hot Leg Inj Valves: • SI-866A • SI-866B
	b. LTDN LINE STOP CVC-460A & B	b. Verify CLOSED the following valves: 1) LTDN ORIFICE Valves: • CVC-200A • CVC-200B • CVC-200C 2) LTDN LINE ISO Valves: • CVC-204A • CVC-204B

3.	Verify The Following Sample Valves At The SAMPLING CONTROL PANEL - CLOSED:
	• PZR STEAM SPACE
	  • PS-956A
	  • PS-956B
	• PZR LIQUID SPACE
	  • PS-956C
	  • PS-956D
	• RCS HOT LEGS
	  • PS-956E
	  • PS-956F	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4	Determine If Break Is In Cold Leg Injection Piping As Follows:	
	a. Reset SAFETY INJECTION	
	b. Close RHR COLD LEG INJ Valves while monitoring for an RCS pressure increase: • RHR-744A • RHR-744B	
	c. Check RCS pressure - INCREASING	c. Open RHR COLD LEG INJ Valves: • RHR-744A • RHR-744B Go To Step 5.
	d. Maintain RHR COLD LEG INJ Valves - CLOSED • RHR-744A • RHR-744B	
	e. Notify Plant Operations Staff that the break has been isolated and to determine further recovery actions	
	f. Go To Step 8	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Determine If Break Is In Hot Leg Injection Piping As Follows:	
	a. Check either LOOP 2 OR LOOP 3 HOT LEG INJ Valves - OPEN • SI-866A OR • SI-866B	a. Go To Step 6.
	b. Place the Control Power Key Switches for the HOT LEG INJ valves in the NORMAL position • SI-866A, LOOP 3 HOT LEG INJ • SI-866B, LOOP 2 HOT LEG INJ	
	c. Close LOOP 2 and LOOP 3 HOT LEG INJ Valves while monitoring for an RCS pressure increase: • SI-866A • SI-866B	
	d. Check RCS pressure - INCREASING	d. Place the Control Power Key Switches for the HOT LEG INJ valves in the DEFEAT position • SI-866A • SI-866B Go To Step 6.
	e. Maintain LOOP 2 and LOOP 3 HOT LEG INJ Valves - CLOSED • SI-866A • SI-866B	

(CONTINUED NEXT PAGE)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	(CONTINUED)	
	f. Place the Control Power Key Switches for the HOT LEG INJ valves in the DEFEAT position	
	g. Notify Plant Operations Staff that the break has been isolated and to determine further recovery actions	
	h. Go To Step 8	
6.	Determine If Break Is In Seal Return Piping As Follows:	
	a. Close CVC-381, SEAL WTR RTRN ISO while monitoring for an RCS pressure increase	
	b. Check RCS pressure - INCREASING	b. Open CVC-381, SEAL WTR RTRN ISO. Go To Step 7.
	c. Maintain CVC-381, SEAL WTR RTRN ISO - CLOSED	
	d. Notify Plant Operations Staff that the break has been isolated and to determine further recovery actions	
	e. Go To Step 8	
7.	Check The Following Using Available Indications And Local Inspections - INTACT: • Charging line from Charging Pumps to CV Penetration • Seal Injection lines from Charging Pumps to CV Penetration • SI Cold Leg Injection lines from SI Pumps to CV Penetration	Perform the following: a. Isolate the leak. b. Notify Plant Operations Staff that the break has been isolated and to determine further recovery actions.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Determine If Break Is Isolated:	
	a. Check RCS pressure - INCREASING	a. Go To EPP-15, Loss Of Emergency Coolant Recirculation.
	b. Go To Path-1, Entry Point C	

- END -

Progress Energy

NON POM

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

EPP-20-BD

EPP-20 BASIS DOCUMENT

REVISION 8

RNP STEP	WOG STEP	BASIS/DIFFERENCES

RNP DIFFERENCES/REASONS

The EPP provides a plant specific list of valves as directed by the ERG. The EPP provides instructions to keep the valves closed once the break is identified to prevent reinitiating the LOCA. The steps also notify the Plant Operations Staff, since safety related flowpaths may be rendered inoperable and alternate options will have to be determined.

Step 4.c restores the valves to their previous position if the leak is not isolated. This is done due to the fact that these valves could be open OR closed, depending on plant conditions.

SSD DETERMINATION

This is an SSD per criterion 4 and 10.

8 3 WOG BASIS

PURPOSE: To determine if the LOCA outside containment has been isolated from previous actions

BASIS:

This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

KNOWLEDGE:

It should be noted that for some breaks SI flow may cause an RCS pressure increase independent of break isolation. It should also be noted that for larger breaks, RCS repressurization may be delayed following break isolation. Additionally, if the RCS is saturated or a cooldown is in progress, RCS repressurization will proceed more slowly. Other means of verifying break isolation should be checked. For example, increasing RVLIS trend due to injection flow, decreasing trends in local abnormal conditions and local observation (if practical) may be useful.

Comment [DGS2]: DW-03-016

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

17. W/E 05EK1.2 001

Given the following plant conditions:

- A Reactor Trip and Safety Injection have occurred.
- The crew has performed the actions of PATH-1.
- AFW flow cannot be established.
- All S/G NR levels are off-scale low.
- The crew has entered FRP-H.1, Response to Loss of Secondary Heat Sink.
- RCS Pressure is 175 psig and stable.
- Intact S/G pressures are 300 psig and trending down.

Which ONE (1) of the following completes the statement below?

A ___(1)___ Break LOCA is in progress AND a Secondary Heat Sink ___(2)___.

A. (1) Large

(2) is required.

B. (1) Small

(2) is required.

C.✓ (1) Large

(2) is NOT required.

D. (1) Small

(2) is NOT required.

The correct answer is C.

A-Incorrect. Large break LOCA is correct. Secondary heat sink is not required if S/Gs are at a higher pressure than the RCS. They act as a heat source.

B-Incorrect. Small break LOCA is not correct. Secondary heat sink is not required if S/Gs are at a higher pressure than the RCS. They act as a heat source.

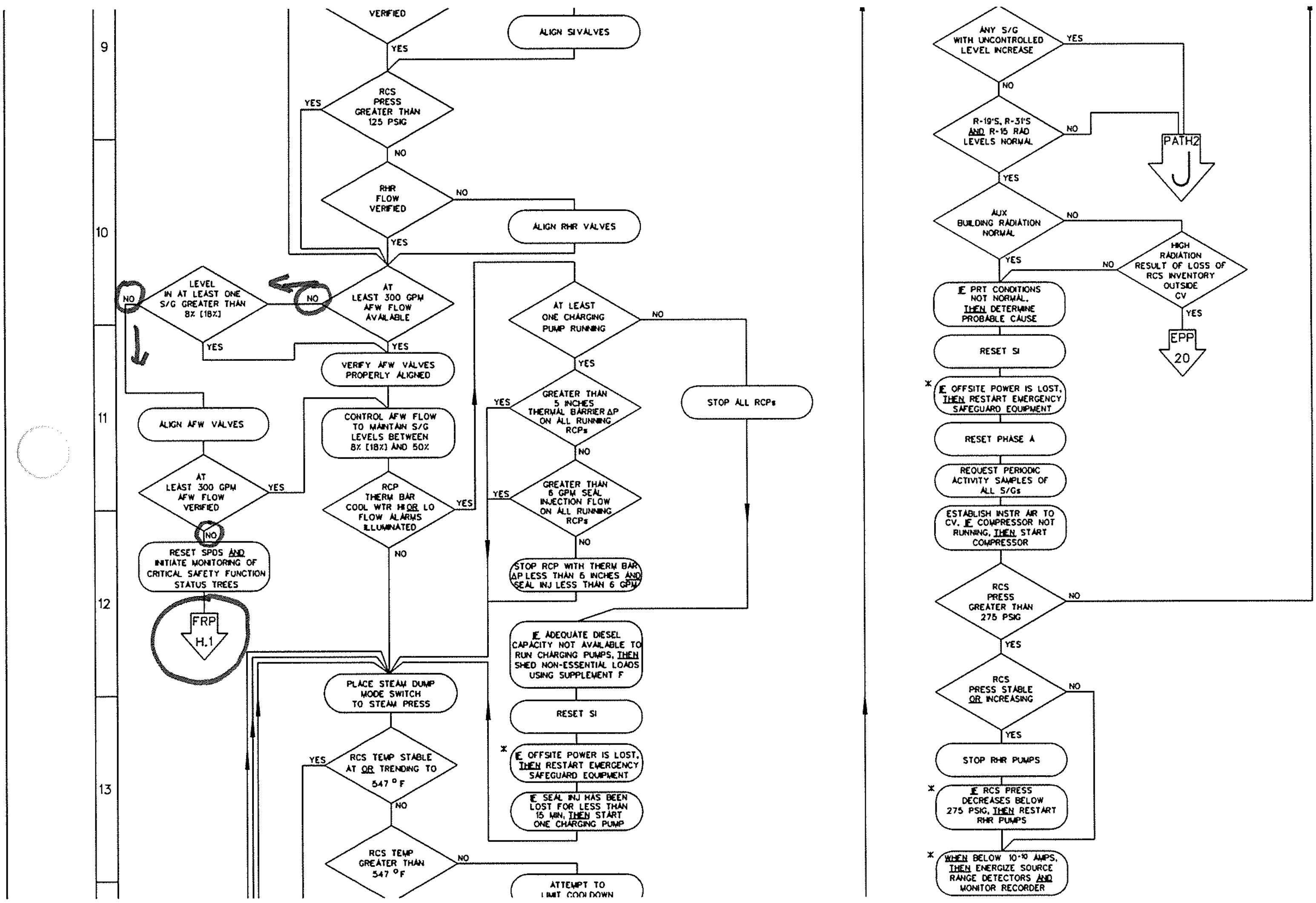
C-Correct. LBLOCA, RCS less than S/G pressure. Heat sink is not required.

D-Incorrect. LBLOCA. Heat sink is not required.

Question 17
Tier 1 / Group 1
K/A Importance Rating - RO 3.9 SRO 4.5

Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink): Normal, abnormal and emergency operating procedures associated with (Loss of Secondary Heat Sink).

Reference(s) - System Design / Simulator, FRP-H.1 and basis document.
Proposed References to be provided to applicants during examination - None
Learning Objective - PATH-1-007, FRP-H.1-003
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.8 / 41.10 / 45.3
Comments: K/A match because candidate must analyze the given indications and determined at a Large Break LOCA has occurred. Candidate must the operational implications associated with a Large Break LOCA in progress and that a Secondary Heat Sink is not required.


9
10
11
12
13
VERIFIED
ALIGN SI VALVES
YES
RCS PRESS GREATER THAN 125 PSIG
YES
NO
RHR FLOW VERIFIED
NO
ALIGN RHR VALVES
YES
LEVEL IN AT LEAST ONE S/G GREATER THAN 8% [18%]
NO
NO
AT LEAST 300 GPM AFW FLOW AVAILABLE
YES
YES
VERIFY AFW VALVES PROPERLY ALIGNED
ALIGN AFW VALVES
CONTROL AFW FLOW TO MAINTAIN S/G LEVELS BETWEEN 8% [18%] AND 50%
AT LEAST 300 GPM AFW FLOW VERIFIED
YES
NO
RESET SPDS AND INITIATE MONITORING OF CRITICAL SAFETY FUNCTION STATUS TREES
FRP H.1
RCP THERM BAR COOL WTR HI OR LO FLOW ALARMS ILLUMINATED
YES
NO
AT LEAST ONE CHARGING PUMP RUNNING
NO
YES
STOP ALL RCPs
GREATER THAN 5 INCHES THERMAL BARRIER ΔP ON ALL RUNNING RCPs
YES
NO
GREATER THAN 6 GPM SEAL INJECTION FLOW ON ALL RUNNING RCPs
YES
NO
STOP RCP WITH THERM BAR ΔP LESS THAN 6 INCHES AND SEAL INJ LESS THAN 6 GPM
IF ADEQUATE DIESEL CAPACITY NOT AVAILABLE TO RUN CHARGING PUMPS, THEN SHED NON-ESSENTIAL LOADS USING SUPPLEMENT F
RESET SI
* IF OFFSITE POWER IS LOST, THEN RESTART EMERGENCY SAFEGUARD EQUIPMENT
IF SEAL INJ HAS BEEN LOST FOR LESS THAN 15 MIN, THEN START ONE CHARGING PUMP
PLACE STEAM DUMP MODE SWITCH TO STEAM PRESS
RCS TEMP STABLE AT OR TRENDING TO 547 °F
YES
NO
RCS TEMP GREATER THAN 547 °F
NO
ATTEMPT TO LIMIT COOLDOWN
ANY S/G WITH UNCONTROLLED LEVEL INCREASE
YES
NO
R-19'S, R-31'S AND R-15 RAD LEVELS NORMAL
NO
PATH2
YES
AUX BUILDING RADIATION NORMAL
NO
YES
HIGH RADIATION RESULT OF LOSS OF RCS INVENTORY OUTSIDE CV
NO
YES
EPP 20
IF PRT CONDITIONS NOT NORMAL, THEN DETERMINE PROBABLE CAUSE
RESET SI
* IF OFFSITE POWER IS LOST, THEN RESTART EMERGENCY SAFEGUARD EQUIPMENT
RESET PHASE A
REQUEST PERIODIC ACTIVITY SAMPLES OF ALL S/Gs
ESTABLISH INSTR AIR TO CV. IF COMPRESSOR NOT RUNNING, THEN START COMPRESSOR
RCS PRESS GREATER THAN 275 PSIG
NO
YES
RCS PRESS STABLE OR INCREASING
NO
YES
STOP RHR PUMPS
* IF RCS PRESS DECREASES BELOW 275 PSIG, THEN RESTART RHR PUMPS
* WHEN BELOW 10⁻¹⁰ AMPS, THEN ENERGIZE SOURCE RANGE DETECTORS AND MONITOR RECORDER

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

FUNCTION RESTORATION PROCEDURE

FRP-H.1

RESPONSE TO LOSS OF SECONDARY HEAT SINK

REVISION 23

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides actions to respond to a loss of secondary heat sink in all Steam Generators.

2. ENTRY CONDITIONS

a. PATH-1, when minimum AFW flow is not verified AND narrow range level in all S/Gs is less than 8% [18%].

b. CSF-3, Heat Sink Critical Safety Function Status Tree on a RED condition.

\- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Feed flow is not re-established to any faulted S/G if an intact S/G is available.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Total Feed Flow - LESS THAN 300 GPM DUE TO OPERATOR ACTION	Go To Step 3.
2.	Reset SPDS And Return To Procedure And Step In Effect	
3.	Determine If Secondary Heat Sink Is Required As Follows:	
	a. Check RCS pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE	a. Reset SPDS and Go To PATH-1, Entry Point C.
	b. Check RCS temperature - GREATER THAN 350°F [310°F]	b. Perform the following: 1) Place RHR System in service using Supplement I. 2) WHEN adequate cooling with RHR is established, THEN reset SPDS and return to procedure and step in effect.
* 4.	Check Any Two S/G Wide Range Levels - LESS THAN 10% [19%]	IF any two S/G Wide Range Levels decrease to less than 10% [19%], THEN Go To Step 5. Go To Step 6.
5.	Perform The Following: a. Stop all RCPs b. Observe CAUTION prior to Step 31 and Go To Step 31	

RNP STEP	WOG STEP	BASIS/DIFFERENCES

RNP DIFFERENCES/REASONS

The RNP procedure places the caution or note in an action step to prevent actions within cautions and noted as required by the writer's guide.

SSD DETERMINATION

This is an SSD per criterion 11.

3 1 WOG BASIS

BASIS:

Before implementing actions to restore flow to the steam generators, the operator should check if secondary heat sink is required. For larger LOCA break sizes, the RCS will depressurize below the intact steam generator pressures. The steam generators no longer function as a heat sink and the core decay heat is removed by the RCS break flow. For this range of LOCA break sizes, the secondary heat sink is not required and actions to restore secondary heat sink are not necessary. For these cases, the operator returns to the guideline and step in effect.

Comment [COMMENT5]: DW-95-050

Since Step 8 directs the operator to return to Step 1 if the loss of secondary heat sink parameters are not exceeded, break sizes that take longer to depressurize the RCS will be detected on subsequent passes through Step 1.

If RCS temperature is low enough to place the RHR System in service, then the RHR System is an alternate heat sink to the secondary system. Therefore, an attempt is made to place the RHR System in service in parallel to the attempts to reestablish feedwater flow. RCS pressure must be below normal RHR System pressure limits.

KNOWLEDGE:

The operator must be able to place the RHR System in service before the pressure and temperature limits are exceeded to make this alternate heat sink a valid option. Efforts to restore feedwater flow to the SGs should not be delayed if the RHR System is not a valid option.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

18. W/E 11EA1.3 001

Given the following plant conditions:

- A SBLOCA has occurred. Containment Pressure is 2.3 psig.
- CCW Pumps are not available.
- The operating crew has transitioned to EPP-15, Loss of Emergency Coolant Recirculation.

Which ONE (1) of the following correctly describes the reason for depressurizing the RCS and the terminating setpoint?

To __(1)__, depressurize the RCS until PZR level is > 71% [60%] __(2)__ RCS Subcooling is between 35°F [55°F] and 45°F [65°F].

A.✓ (1) minimize RCS leakage

(2) OR

B. (1) minimize RCS leakage

(2) AND

C. (1) inject the SI accumulators

(2) OR

D. (1) inject the SI accumulators

(2) AND

The correct answer is A.

A. Correct.

B. Incorrect. The first half of answer is correct. The RCS is depressurized until pressurizer level is >71% OR RCS Subcooling is between 35 and 45°F.

C. Incorrect. The steam generators are depressurized to inject the SI accumulators. The question is asking for the reason for depressurizing the RCS. The second half of the answer is correct.

D. Incorrect. See previous discussions. Both sections of the answer are incorrect.

Question 18
Tier 1 / Group 1
K/A Importance Rating - RO 3.7 SRO 4.2

Ability to operate and / or monitor the following as they apply to the Loss of Emergency Coolant Recirculation: Desired operating results during abnormal and emergency situations.

Reference(s) - System Design / Simulator, EPP-15
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-15-003
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5 / 45.6
Comments: K/A match because candidate will be given a Loss of Emergency Coolant Recirculation scenarios and must know the reason for taking actions to depressurize the RCS and what indications are monitored for depressurization termination.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- The upper head region may void during RCS depressurization if RCPs are NOT running. This will result in a rapidly increasing PZR level.
- Letdown is NOT required for use of auxiliary spray.
- Supplement K is available for optimizing Auxiliary Spray flow below.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*40.	Depressurize RCS To Minimize RCS Leakage As Follows:	
	a. Check RCS Subcooling - GREATER THAN 45°F [65°F]	a. Go To Step 41.
	b. Use normal PZR Spray to depressurize the RCS	b. Use one PZR PORV. IF no PZR PORV is available, THEN use Auxiliary Spray.
	c. Check EITHER of the following conditions satisfied: • PZR LEVEL - GREATER THAN 71% [60%] OR • RCS SUBCOOLING - BETWEEN 35°F [55°F] AND 45°F [65°F]	c. WHEN the either of the listed conditions are satisfied, THEN perform Step 40.d. Go To Step 41.
	d. Stop RCS depressurization	

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

EPP-15-BD

EPP-15 BASIS DOCUMENT

REVISION 17

For Guideline ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, the following specific design features of the reference plant design should be noted since certain steps in this guideline have been written based upon inclusion of these design features:

- The reference design uses both containment spray pumps and containment fan coolers for containment heat removal subsequent to an accident. The design basis containment heat removal capability is accomplished by certain combinations of spray pumps and fan coolers, which is described in the affected steps.
- In the SI System of the reference plant design, the sump recirculation isolation valves do not open automatically upon a coincident low level (switchover setpoint) in the RWST and an "S" signal. Manual switchover is required on low RWST level.
- In the SI System design of the reference plant, separate and distinct piping and valving exist between the containment sump and the suction of the containment spray pumps. This piping and valving is completely separate from that which connects the containment sump to the suction of the low-head heat removal pumps.

The impact of these design features on the loss of ECR guideline, ECA-1.1, are discussed in detail under the specific steps in Section 4.1, "Detailed Description of Steps, Notes, and Cautions".

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline ECA-1.1 is to restore ECR capability, delay depletion of the RWST, and depressurize the RCS.

The following subsections provide a summary of the major categories of operator actions and the key utility decision points for guideline ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

3.1 High Level Action Summary

A high level summary of the actions performed in ECA-1.1 is given in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN ECA-1.1

- Continue Attempts to Restore ECR
- Increase/Conserve RWST Level
- Initiate Cooldown to Cold Shutdown
- Depressurize RCS to Minimize RCS Subcooling
- Try to Add Makeup to RCS from Alternate Source
- Depressurize SGs to Cool Down and Depressurize RCS
- Maintain RCS Heat Removal

o Continue Attempts to Restore ECR

The first instruction for the operator is to try to restore the equipment needed for ECR in order to avoid performing any extreme recovery actions. Attempts to restore ECR should, therefore, be continued throughout the guideline. IF ECR capability is restored, the operator is instructed to return to the guideline in effect for further recovery actions.

o Increase/Conserve RWST Level

Makeup is added to the RWST to extend the time the SI pumps and containment spray pumps can take suction from the RWST. In addition, RWST outflow is minimized by stopping any unnecessary containment spray pumps and decreasing the SI pump flowrate. If, however, the RWST is empty, the operator stops all pumps taking suction from it.

o Initiate Cooldown to Cold Shutdown

A 100°F/hr cooldown rate is established to cool the RCS to cold shutdown conditions. This permits a controlled depressurization of the RCS to limit coolant leakage. The cooldown rate is limited to 100°F/hr to minimize thermal stresses in the reactor vessel and remain within the ORANGE and RED priority limits of the Integrity Status Tree.

o Depressurize RCS to Minimize RCS Subcooling

The RCS is depressurized to minimize RCS subcooling. Reducing RCS pressure also results in reducing break flow from the LOCA.

o Try to Add Makeup to RCS from Alternate Source

At this point in ECA-1.1, there is no SI flow injecting into the RCS since the RWST is empty and ECR capability is lost. Therefore, the operator should provide makeup from any alternate source to cool the core.

o Depressurize SGs to Cool Down and Depressurize RCS

A controlled RCS cooldown to cold shutdown is initiated early to decrease the overall temperature of the RCS coolant and metal in order to reduce the need for heat removal from supporting plant systems and equipment. The SGs are further depressurized to decrease the RCS pressure and temperature for the following reasons: 1) to inject the SI accumulators; 2) to minimize break flow from a LOCA; and 3) to reach RHR System conditions.

o Maintain RCS Heat Removal

The RHR System (if it was placed in service) or the dumping of steam is used to maintain RCS heat removal, since there is no SI flow to the RCS at this time. The plant engineering staff should be consulted for further instructions.

3.2 Key Utility Decision Points

There are two key utility decision points in guideline ECA-1.1 when the utility must determine an appropriate course of action. First, in Step 22 the plant engineering staff is consulted to determine if the RHR System should be placed in service. At this time the plant engineering staff should determine the availability of the RHR System. Second, when the guideline is completed, the last step (Step 37) directs the operator to consult the plant engineering staff who will determine any further course of action.

STEP SPECIFIC DESCRIPTION AND RNP DIFFERENCES

The following pages will provide the RNP step number, the ERG step number, the WOG basis for each step where applicable, the differences between the ERG and RNP step, and the Category of deviation (SSD).

RNP STEP	WOG STEP	BASIS/DIFFERENCES
PEC	PEC	WOG BASIS N/A, there is no WOG basis description for the PEC, other than the general description. RNP DIFFERENCES/REASONS No significant differences. SSD DETERMINATION This is not an SSD.
1-2	1C1	WOG BASIS PURPOSE: To alert the operator to continue with guideline in effect if ECR capability is restored BASIS: This caution instructs the operator to continue with guideline and step in effect for further recovery action if ECR capability is restored during the loss of ECR guideline. RNP DIFFERENCES/REASONS The RNP procedure places the caution or note in an action step to prevent actions within cautions and noted as required by the writer's guide. SSD DETERMINATION This is an SSD per criterion 11.
3	N/A	WOG BASIS N/A, this step is not in the WOG ERG.

19. 001 AA2.04 001

Given the following plant conditions:

- The plant is operating at 50% RTP while supporting MFP "A" maintenance.
- Rod Control is in MANUAL.
- The RO withdraws Bank 'D' two steps for Tavg control
- When the Rod Control switch is released, Bank 'D' rods continue to move until stopped by performing the immediate actions of AOP-001, Malfunction of Reactor Control System.

Which ONE (1) of the following describes the effect on Reactor Power?

(Assume no additional operator actions have occurred.)

A.✓ Rise initially, then trend back to 50% RTP.

B. Remain stable due to no change in steam demand.

C. Rise and stabilize at a value greater than 50% RTP.

D. Rise initially, then trend back to a power level less than 50% RTP.

The correct answer is A.

A. Correct. Reactor Power will increase and then trend back to 50% RTP since there was no change in steam demand.

B. Incorrect. Plausible, since power will return to the original value.

C. Incorrect. Plausible. The initial increase in reactivity due to rod withdrawal will cause NI power to initially increase. However, reactor power will decrease back to original value due to MTC.

D. Incorrect. MTC will add negative reactivity, but only enough to offset the temperature increase. Power should return to the original value. Conceptual error.

Question 19
Tier 1 / Group 2
K/A Importance Rating - RO 4.2 SRO 4.3

Continuous Rod Withdrawal - Ability to operate and interpret the following as they apply to Continuous Rod Withdrawal: Reactor power and its trend.

Reference(s) - Sim/Plant design, AOP-001. GFES - Reactor Theory
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-001-004
Question Source - RNP Bank - Modified
Question Cognitive Level - F
10 CFR Part 55 Content - 43.5 / 45.13
Comments - K/A match because candidate is given a rod withdrawal event and must determine the expected response of Reactor Power from event and back to stead-state.

20. 036 AK2.01 001

Given the following plant conditions:

- The plant is in Mode 6 with a core offload in progress.
- The CV Manipulator has just removed a fuel assembly from core location H-8 and is in transit to the CV upender.
- The CV manipulator operator accidently places the gripper switch to the "DISENGAGE" position before the fuel assembly is lowered into the CV upender.
- The fuel assembly releases from the gripper while in the full up position and lands on the CV upender and damages several fuel rods.

Which ONE (1) of the following describes the interlock(s) that failed to prevent this action?

A. ONLY the "Gripper Lock" Interlock.

B. ONLY the "Gripper-Weight Indicator" Interlock.

C. The "Gripper-Weight Indicator" AND the "Hoist/Gripper" Interlocks.

D.✓ The "Gripper-Weight Indicator" AND the "Gripper Lock" Interlocks.

The correct answer D.

A. Incorrect. Plausible since the "Gripper Lock" Interlock is a mechanical interlock that cannot be bypassed but can fail. However, the "Gripper-Weight Indicator" Interlock will maintain pneumatic pressure to the gripper to ensure that the assembly will not drop.

B. Incorrect. Plausible since the "Gripper-Weight Indicator" Interlock can be bypassed or can fail. However, the "Gripper Lock" Interlock will ensure that the assembly does not drop.

C. Incorrect. The "Gripper Lock" Interlock will ensure that the fuel assembly does not fail. A failure of the "Hoist/Gripper" Interlock could allow the Hoist to move in either the up or down direction without being in the proper location but would not cause the fuel assembly to fall.

D. Correct. A failure of both these interlocks could result in the fuel assembly being released from the gripper.

Question 20
Tier 1 / Group 2
K/A Importance Rating - RO 2.9 SRO 3.5

Knowledge of the interrelations between the Fuel Handling Incidents and the following: Fuel handling equipment

Reference(s) - Sim/Plant design, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - FHS 006
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.7
Comments - K/A match because can know what interlocks associated with fuel handling equipment had to fail to result in given fuel handling incident.

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OPERATIONS TRAINING

H.B. Robinson Plant
System Description

FUEL HANDLING SYSTEM

SD-008

Revision 9

Recommended By:	**D. Knight**	**Date:**	**12/3/08**
	Instructor/Developer		
Concurrence By:	**Marty Arnold**	**Date:**	**12/9/08**
	Operations Management		
Approved By:	**J. F. Jones**	**Date:**	**1/12/09**
	Superintendent/Supervisor Training		

INFORMATION USE ONLY

(This adjustable setpoint is determined in EST-030 and is set prior to each refueling.)

When the inner mast assembly is being raised or lowered, the operator should continuously watch the load cell digital readout. In the event a large or unexplained change in load appears, the operator should immediately STOP the equipment and evaluate the situation.

e. Bridge Position Indication - the bridge position is provided by notches cut into the bridge tracks. When the pointer on the southeast leg of the bridge is aligned with a particular notch, the bridge is in position.

4.1.3 Interlocks (Figure 35)

a. Bridge - Trolley - Hoist - The bridge, trolley, and hoist switch controls are mutually interlocked through the use of the same control panel such that only one drive is operable at a time. When one of the control handles is moved from neutral, the other drive circuits are opened. There is no way of bypassing this interlock.

b. Gripper Tube Up - The gripper tube must be at its top stop position to energize the bridge or trolley drive circuits.

c. Bridge - Trolley Position - bridge and trolley positions are mutually interlocked to limit the mast to a path of travel that clears the guide studs in the vessel and the Upper Internals in their storage position. With the mast over the core area, trolley travel toward the transfer system centerline is stopped at the last row of fuel assemblies by a limit switch to protect the guide stud on this side of the vessel. Bridge travel toward the Transfer System is stopped when the mast passes over the end core coordinate. At this point the trolley drive is able to drive in one direction to permit the mast to be positioned over the transfer system centerline. When the trolley is on the transfer system centerline a limit switch is actuated enabling the bridge drive, it then can be driven toward the Transfer System loading point. During travel down the Transfer Canal, the trolley drive is locked out to keep the mast on the Transfer Canal centerline. On the return trip from the Transfer System to the core, the interlocks keep the mast following the same travel path in reverse.

d. Hoist - Mast Position - The position of the mast is interlocked with the Hoist Control Circuit such that the mast must be in the proper position relative to the Bridge position -- over the core area or the Fuel Transfer System -- to enable operation of the hoist.

WRONG

e. Hoist - Gripper - The gripper must be in the either the engaged or disengaged position as indicated by ENGAGED or DISENGAGED indicating lamps to enable operation of the Hoist in either the up or down direction.

f. Gripper - Weight Indicator - The air line to the Gripper Control Knob is closed by a solenoid valve until the load sensed by the Weight Indicator drops below 1200 pounds indicating that there is no fuel assembly suspended from the gripper.

g. Hoist - Weight Indicator - The Weight Indicator is connected into the Hoist Drive Circuit to stop the hoist in the up direction and light the Overload Lamp when the sensed weight is greater than ~2250 pounds. In the down direction, the weight indicator stops the Hoist Drive and illuminates the Slack Cable Lamp when the suspended weight drops to less than 300 pounds. The Overload Interlock, in the upward direction can be bypassed through the Overload Interlock Bypass Switch. The Slack Cable Switch, in the downward direction cannot be bypassed.

BOTH HAD TO FAIL

h. Hoist Slow Zones - The Hoist Drive Circuit is interlocked through geared limit switches in the Hoist to cut out the Main Hoist Control and transfer control to the jog controls at critical elevations of gripper travel. There is no bypass provided to cut out the Slow Zones. The Upper Slow Zone is approximately 20" and is set to operate as the bottom of a fuel assembly, with the gripper engaged, is entering or leaving the top of the core or upender. This slow zone is operable when the gripper is engaged indicating that there is a fuel assembly in the gripper. The Lower Slow Zone is effective, both raising and lowering, for approximately 10" above the full down position of the fuel in the core or upender. This slow zone operates with the Gripper either engaged or disengaged.

i. Gripper Lock - The Gripper Mechanism contains a mechanical spring locking device that locks the fingers in either the engaged or disengaged position unless the gripper is supported on top of a fuel assembly and the Gripper Tube weight is exerting a downward force of about 450 pounds. This lock will prevent operation of the gripper even if the interlock, in the air supply line, is bypassed and the operating cylinder is pressurized.

4.1.4 Interlock Bypass Switches

The Interlock Bypass Switches are located in the bottom of the Manipulator Crane Control Console.
The 6 Interlock Bypass Switches are as follows:

a. Bridge Right Bypass (TS-4) permits bridge motion toward the Transfer Canal:
   1. When the trolley is off the transfer system centerline.
   2. When Gripper Tube is not at top limit.

b. Bridge Left Bypass (TS-3) permits bridge motion in opposite direction with Gripper Tube not at top limit and also bypasses the Bridge Travel Limit Switch.

c. Trolley Bypass (TS-5) permits trolley motion in either direction with Gripper Tube not at top limit. Also permits trolley motion when the bridge is beyond the core area and in the Transfer Canal Area.

21. 037 AG2.4.9 001

Given the following plant conditions:

- The plant is at 3% RTP with T_{ave} at 549°F.
- A 50 gpm tube leak has been identified in "A" S/G.

Which ONE (1) of the following describes the appropriate actions that should be taken IAW AOP-035?

AOP-035, S/G Tube Leak
GP-006, Normal Plant Shutdown From Power Operation to Hot Shutdown
Supplement G, Steam Generator Isolation

A.✓ Shutdown the plant IAW GP-006 and then Isolate "A" S/G IAW AOP-035.

B. Isolate "A" S/G IAW AOP-035 and then commence a plant shutdown IAW GP-006.

C. Trip the reactor and go to PATH-2. Isolate "A" S/G IAW Supplement G concurrent with PATH-2.

D. Trip the reactor and go to PATH-1. Transition to PATH-2 and isolate "A" S/G IAW Supplement G concurrent with PATH-2.

The correct answer is A.

A. Correct. AOP-035 will direct the operator to conduct a plant shutdown. The plant is to be less than 50% in 1 hour and in Mode 3 in 3 hours. AOP-035 will direct isolation of the affected S/G.

B. Incorrect. Isolation of the S/G is not a prerequisite to commencing the plant shutdown.

C. Incorrect - The threshold for performing a reactor trip has not been reached. The charging pumps are more than capable of maintaining pressurizer level with a 50 gpm tube leak. If the tube leak developed into a tube rupture that was beyond the capability of the charging pumps then PATH-2 would ultimately be entered.

D. Incorrect - The threshold for performing a reactor trip has not been reached. The charging pumps are more than capable of maintaining pressurizer level with a 50 gpm tube leak. PATH-1 would be entered if a reactor trip is initiated.

Question 21
Tier 1 / Group 2
K/A Importance Rating - RO 3.8 SRO 4.2

Steam Generator (S/G) Tube Leak: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Reference(s) - Sim/Plant design, AOP-035
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-035-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13
Comments - K/A match because candidate must know the S/G tube leak rate limits that would either result in a Rx Trip or a plant shutdown.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-035

S/G TUBE LEAK

REVISION 22

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides the direction necessary to respond to a Steam Generator tube leak when the EOP network has not been entered.

> **NOTE**
>
> - A warning alarm on RR-1 is confirmed if any diverse OR redundant indication shows primary to secondary leakage.
> - Alarms on R-15, R-19, R-24, and R-31 are confirmed in accordance with APP-036, Auxiliary Annunciator.

2. ENTRY CONDITIONS

Any of the below conditions:

- Confirmed R-24 leakage greater than or equal to 30 gpd.

OR

- Confirmed alarm on radiation monitors R-15, R-19, OR R-31.

OR

- Confirmed warning alarm on RR-1, RMS RECORDER, for radiation monitors R-15, R-19, OR R-31.

OR

- Notification by Chemistry Personnel that a PSAL-1, PSAL-2, OR PSAL-3 condition exists for primary to secondary leakage.

OR

- Indications of Primary to Secondary leakage as determined by AOP-016, EXCESSIVE PRIMARY PLANT LEAKAGE.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Determine If Reactor Trip Needed As Follows:	
	a. Check the following: • PZR Level - LESS THAN 10% OR • RCS Subcooling - LESS THAN 35°F	a. IF PZR Level can NOT be maintained greater than 10% OR Subcooling can NOT be maintained greater than 35°F, THEN trip the Reactor and Go To Path-1. Go To Step 2.
	b. Trip the Reactor and Go To Path-1	
2.	Make PA Announcement For Procedure Entry.	

NOTE

Use of the RWST for RCS Makeup will add negative reactivity.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 3.	Check VCT Level - LESS THAN 12.5 INCHES	IF VCT level lowers to less than 12.5 inches, THEN perform Step 4. Go To Step 5.
4.	Align Charging Pump Suction From The RWST As Follows:	
	a. At The RTGB, Verify LCV-115B, EMERG MU TO CHG SUCT - OPEN	a. Open CVC-358, RWST TO CHARGING PUMP SUCTION, prior to continuing.
	b. Verify LCV-115C, VCT OUTLET - CLOSED	
	c. Check Plant Mode - MODE 1	c. Go To Step 5
	d. Trip the Reactor and Go To PATH - 1	
5.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Adjust Charging Flow As Follows:	
	a. Check Charging Pump Status - AT LEAST TWO RUNNING	a. Start one additional Charging Pump.
	b. Place running Charging Pumps Speed Controllers in MAN AND adjust output to maximum	
7.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
8.	Check Letdown - IN SERVICE	Go To Step 11.
9.	Verify All Letdown Flowpaths Isolated As Follows: • LCV-460A & B, LTDN LINE STOP Valves - CLOSED • HIC-137, EXCESS LTDN FLOW Controller - ADJUSTED TO 0% • CVC-387, EXCESS LTDN STOP - CLOSED	
10.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
11.	Trip The Reactor AND Go To PATH-1	
12.	Control Charging Flow To Maintain Desired RCS Level	
*13.	Check RCS Leakage - GREATER THAN RUNNING CHARGING FLOW	IF leakage exceeds Charging flow, THEN Go To Step 6. Go To Step 15.
14.	Go To Step 6	
15.	Notify Chemistry Personnel To Periodically Sample All S/Gs For Activity And Boron Concentration	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16	Check Assistance To Open S/G Sample Valves- REQUESTED	WHEN assistance to open S/G Sample valves is requested, THEN observe the NOTE prior to Step 17 and perform Step 17 Observe the NOTE prior to Step 18 and Go To Step 18

NOTE

Operation of the S/G Blowdown OVERRIDE OPEN key switches will result in an ITS 3.6.3 entry.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
17.	WHEN Requested By Chemistry Personnel To Support S/G Samples, THEN Perform The Following:	
	a. Check R-19 Monitor - IN ALARM OR EXPECTED TO ALARM	a. Observe the NOTE prior to Step 18 and Go To Step 18
	1) Place the S/G Blowdown OVERRIDE OPEN key switch for the affected S/G(s) in the OVERRIDE OPEN position • S/G A BLOWDOWN SGB-1933A & SGB-1933B • S/G B BLOWDOWN SGB-1934A & SGB-1934B • S/G C BLOWDOWN SGB-1935A & SGB-1935B	
	2) Within ONE Hour restore the S/G Blowdown OVERRIDE OPEN key switch for the affected S/G(s) to the NORMAL position (ITS 3.6.3. Condition B)	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Radiation Monitor R-24 does not provide an accurate determination of leakage until S/G samples have been obtained and the monitor has been calibrated for the optimal node for leakage location.

18. Determine Leak Rate Using At Least One Of The Following Methods:

- R-24 Recorder
- Perform OST-051, Reactor Coolant System Leakage Evaluation
- Perform a Charging versus Letdown balance
- Notify Chemistry personnel to perform isotopic analysis of S/G samples for leak rate determination
- Use R-15 to monitor for low level Primary-to-Secondary leakage using the OP-504, Condenser Air Removal section "Using R-15 to Monitor for Low Level Primary to Secondary Leakage"
- Use CP-014 Conversion Factors to correlate R-15 to leakage

19. Check Leak Rate Determination - COMPLETE

20gpm

RESPONSE NOT OBTAINED: WHEN leak rate determination is complete, THEN observe the NOTE prior to Step 20 and Go To Step 20.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- ITS LCO 3.4.13 provides a primary to secondary leakage limit of 75 gpd through any one S/G.
- Total leakage is assumed to be coming from a single S/G when unable to determine leakage from the individual S/Gs.
- Normally performed steps in GP-006, such as placing S/G Blowdown to the Flash Tank may require Release Permits.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*20.	Check Leak Rate - GREATER THAN <u>OR</u> EQUAL TO 100 GPD FOR A SINGLE S/G YES 20gpm	<u>IF</u> the leak rate exceeds the limit, <u>THEN</u> Go To Step 21. Go To Step 22.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

It is important to perform GP-006 and AOP-035 concurrently to the extent possible in order to minimize secondary contamination.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
21.	Perform The Following Power Reduction:	
	a. Notify Chemistry that a PSAL-3 event has occurred	
	b. Check Reactor Status - MODE 1 OR MODE 2	b. Observe the NOTE prior to Step 24 and Go To Step 24
	c. Initiate Plant Shutdown To Mode 3 Using GP-006, Normal Plant Shutdown From Power Operation To Hot Shutdown, While Continuing With This Procedure	
	d. Adhere to the following time limits: • Be less than 50% power within 1 hour of declaring PSAL-3 • Be in Mode 3 within 3 hours of declaring PSAL-3	
	e. Observe the NOTE prior to Step 24 and Go To Step 24	
*22.	Check Leak Rate - GREATER THAN OR EQUAL TO 75 GPD FOR A SINGLE S/G	IF the leak rate exceeds the limit, THEN Go To Step 23. Go To Step 105.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

It is important to perform GP-006 and AOP-035 concurrently to the extent possible in order to minimize secondary contamination.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
23.	Perform The Following Power Reduction:	
	a. Notify Chemistry that a PSAL-2 event has occurred	
	b. Check Reactor Status - MODE 1 OR MODE 2	b. Observe the NOTE prior to Step 24 and Go To Step 24
	c. Initiate Plant Shutdown To Mode 3 Using GP-006, Normal Plant Shutdown From Power Operation To Hot Shutdown, While Continuing With This Procedure	
	d. Be in Mode 3 within 6 hours of declaring PSAL-2	
	e. Observe the NOTE prior to Step 24 and Go To Step 24	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Radiation Monitor R-24 does not provide an accurate determination of leakage until S/G samples have been obtained and the monitor has been calibrated for the optimal node for leakage location.

24. Identify Leaking S/G Using At Least One Of The Following Methods:

'A' S/G

- Evaluate indications on R-24 Recorder

  OR

- Evaluate indications on RI-19A, RI-19B, and RI-19C, STM GEN BLOW DN Radiation Monitors

  OR

- Evaluate indications on R-31A, R-31B, and R-31C, STEAMLINE RADIATION MONITORs

  OR

- Chemistry analysis of S/G samples for boron and activity

25. Implement The EALs

26. Review Technical Specification LCOs

- ITS LCO 3.4.13
- ITS LCO 3.7.15
- ITS LCO 3.6.3

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
27.	Initiate Monitoring RCS Leak Rate As Follows:	
	a. Check Radiation Monitor R-24 - IN SERVICE	a. Go To Step 27.d.
	b. Log R-24 readings at 15 minute intervals	
	c. Go To Step 28	
	d. Check RCS Leak Rate - LESS THAN 10 GPM	d. Log RCS leakage estimates at 15 minute intervals. Go To Step 28.
	e. Monitor R-15 Trends Using Attachment 5, R-15 Monitoring	
28.	Contact An Operator To Bypass The Condensate Polishers As Follows:	
	a. Place the SECONDARY BYPASS Switch to the OPEN position	
	b. Depress the OFF pushbutton for each in service demineralizer	
29.	Perform Attachment 4, Controlling Secondary Contamination, While Continuing With This Procedure	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
30	Isolate Non-essential Flowpaths From The Affected S/G As Follows:	
	a. Verify SDAFW Pump STEAM SHUTOFF from affected S/G - CLOSED: • V1-8A OR • V1-8B OR • V1-8C	
	b. Perform Attachment 1, Local S/G Isolation, while continuing with this procedure	

NOTE

At high primary to secondary leak rates the ability to cooldown will be limited by RCS makeup capability. IF a rapid RCS cooldown is required AND difficulty is experienced maintaining PZR level, THEN consideration should be given to initiation of Safety Injection.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
31.	Check Turbine - TRIPPED	WHEN the Turbine is tripped as directed by GP-006, THEN Go To Step 32.
*32.	Check The Following Rod Banks - FULLY INSERTED:	WHEN the rod banks are fully inserted, THEN perform Step 33.
	• All Control Banks • Shutdown Bank B • Shutdown Bank A	Go To Step 34.

22. 051 AA1.04 001

Given the following plant conditions:

- The plant is at 8% RTP with "B" MFP under clearance for motor bearing repairs.
- The Operating crew has entered AOP-012, Partial Loss of Condenser Vacuum or Circulating Water Pump Trip due to rising Condenser Back Pressure.
- The Main Turbine has been manually tripped IAW AOP-012.
- "A" MFP is subsequently secured due to an identified ground.

Which ONE (1) of the following actions is required by AOP-007, Turbine Trip Below P-8?

A.✓ Position the control rods to reduce Reactor power to less than or equal to 3%.

B. Position the control rods to maintain Reactor power between 7% and 10%.

C. Trip the Reactor and go to PATH-1 while continuing with AOP-012.

D. Trip the Reactor and go to PATH-1.

The correct answer is A.

A. Correct - Since power is less than 10% with no Main Feed Pumps available direction is given in AOP-007 to control S/G levels using AFW pumps and reduce power to less than 3%.

B. Incorrect - The actions listed would be correct if a Main Feed Pump is available. The stem of the question states that neither are available. Therefore, reactor operation up to 10% power is not to be supported by AFW pumps.

C. Incorrect - The reactor would be tripped if the Reactor Trip Block P-7 status light was extinguished. The initial conditions of 8% RTP would mean that the light would be illuminated. Additionally, AOP-012 is not a continuous use procedure and would not be continued if the reactor was tripped.

D. Incorrect - The reactor would be tripped if the Reactor Trip Block P-7 status light was extinguished. The initial conditions of 8% RTP would mean that the light would be illuminated.

Question 22
Tier 1 / Group 2
K/A Importance Rating - RO 2.5 SRO 2.5

Ability to operate and / or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position

Reference(s) - Sim/Plant design, AOP-007, AOP-012
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-012-005
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5 / 45.6
Comments - K/A match because the candidate is given a Loss of Condenser Vacuum condition at low power with a loss of main feed water. The candidate must determine the necessary actions relative to control of the reactor with control rods based on given conditions.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
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PART 5
ABNORMAL OPERATING PROCEDURE

AOP-012

PARTIAL LOSS OF CONDENSER VACUUM OR CIRCULATING WATER PUMP TRIP

REVISION 22

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides instructions for a partial loss of Condenser vacuum or Circulating Water Pump trip.

2. ENTRY CONDITIONS

   This procedure is entered whenever a partial loss of Condenser vacuum or Circulating Water Pump trip occurs.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
	NOTE Steps 1 and 2 are Immediate Action steps.	
1.	Check Circulating Water Pump - ANY TRIPPED	Go To Section A, Partial Loss of Condenser vacuum.
2.	Verify The Tripped Circulating Water Pump Discharge Valve - CLOSED OR CLOSING • V6-50A, CIRC WATER PMP "A" DISCH OR • V6-50B, CIRC WATER PMP "B" DISCH OR • V6-50C, CIRC WATER PMP "C" DISCH	
3.	Start Any Available Circulating Water Pump	
4.	Make PA Announcement For Procedure Entry	
5.	Check Liquid Waste Batch Release - IN PROGRESS	Go To Step 7.
6.	Stop Any Liquid Waste Batch Release In Progress As Follows: a. Stop the applicable running waste release pump. b. Isolate the waste release flowpath.	
7.	Check Condenser Status - VACUUM PREVIOUSLY ESTABLISHED	Implement the EALs. Return To Procedure and Step in effect.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section A

Partial Loss of Condenser Vacuum

(Page 1 of 5)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Condenser Status - VACUUM PREVIOUSLY ESTABLISHED	Implement the EALs. Return To Procedure and Step in effect.
2.	Verify All Available Vacuum Pumps - RUNNING	
3.	Verify The Following VACUUM BREAKER Valves - CLOSED • MS-70A • MS-70B	
4.	Make PA Announcement For Procedure Entry	
5.	Check Plant Conditions - IN MODES 1 OR 2	Go To Step 20.
6.	Check Condenser Back Pressure On PI-1312 AND PI-1313 - APPROACHES RESTRICTED REGION OF ATTACHMENT 3, CONDENSER BACKPRESSURE LIMIT CURVE	IF Condenser Backpressure approaches the restricted region of Attachment 3, Condenser Backpressure Limit Curve, THEN Go To Step 7. Go To Step 9.
7.	Check REACTOR TRIP BLOCK P-7 Status Light - ILLUMINATED	Perform the following: a. Trip the Reactor. b. Go To Path-1.
8.	Perform Turbine Trip Actions As Follows: a. Manually trip the Turbine b. Go To AOP-007, Turbine Trip Below P-8	

CONTINUOUS USE

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ABNORMAL OPERATING PROCEDURE

AOP-007

TURBINE TRIP BELOW P-8

REVISION 10

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides the instructions for response to a Turbine trip below the P-8 interlock (40% NI Power).

2. ENTRY CONDITIONS

   Upon receiving indication that a Turbine trip is required or has occurred below P-8 (40% NI Power).

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Turbine Trip As Follows: • BOTH Turbine Stop Valves - CLOSED OR • All Governor Valves - CLOSED	Perform the following: a. Manually trip the Turbine by simultaneously depressing the THINK and TURBINE TRIP Pushbuttons. b. IF the Turbine will NOT trip, THEN run back Turbine at maximum rate until the Governor Valves are closed. c. IF the Turbine can NOT be run back, THEN perform one of the the following: • IF the REACTOR TRIP BLOCK P-7 status light is EXTINGUISHED, THEN Trip the Reactor and Go To Path-1 OR • IF the REACTOR TRIP BLOCK P-7 status light is ILLUMINATED, THEN perform the following: • Verify closed ALL MSIVs • Verify closed ALL MSIV BYPs • Go To Step 9

NOTE

In the Steam Pressure Mode, the Steamline PORVs may actuate until Reactor Power is reduced.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Check Steam Dump To Condenser - ACTUATED	IF RCS Tavg exceeds 566°F and Steam Dump to the condenser is not actuated, THEN Trip the Reactor and Go To Path-1

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
3.	Check Main FW Pump Status - ANY RUNNING	~~<u>IF</u> the REACTOR TRIP BLOCK P-7 status light is EXTINGUISHED, <u>THEN</u> Trip the Reactor and Go To Path-1~~ 8% <u>IF</u> the REACTOR TRIP BLOCK P-7 status light is ILLUMINATED, <u>THEN</u> perform the following: a. Verify MDAFW Pumps - RUNNING b. Start SDAFW Pump as necessary c. Control AFW to maintain S/G levels 39% to 52% d. Perform one or both of the following to reduce Reactor power to less than or equal to 3% while continuing. • Manually insert control rods. • Borate using OP-301, Chemical and Volume Control System (CVCS) Section " RCS Boration Quick Checklist" e. Go To Step 9
4.	Check Reactor Power When Turbine Tripped - LESS THAN 20%	Go To Step 7

23. 059 AK3.01 001

Given the following plant conditions:

- "A" Monitor Tank is currently at 40% and being released IAW OP-705, Waste Liquid Release and Recirculation.
- The Inside AO reports a nonisolable leak on "A" Monitor Tank.

Which ONE (1) of the following identifies the correct actions IAW AOP-008, Accidental Release of Liquid Waste?

Stop the release and transfer contents of the "A" Monitor Tank to __(1)__ to minimize leakage to __(2)__.

A. (1) Monitor Tank "B"

(2) storm drains

B.✓ (1) CVCS Holdup Tank

(2) storm drains

C. (1) Monitor Tank "B"

(2) Aux Building Sump

D. (1) CVCS Holdup Tank

(2) Aux Building Sump

The correct answer is B.

A. Incorrect. Procedure steps do not exist to transfer contents from one monitor tank to another. AOP-008 directs the transfer of contents to the CVCS HUT. Storm drains is correct since the Monitor Tanks are outside the Aux. Building.

B. Correct.

C. Incorrect. AOP-008 directs the transfer of contents to the CVCS HUT. The Monitor Tanks are located outside the Aux. Building and leakage would not be directed to the Aux. Building Sump.

D. Incorrect. First part of answer is correct. The Monitor Tanks are located outside the Aux. Building and leakage would not be directed to the Aux. Building Sump.

Question 23
Tier 1 / Group 2
K/A Importance Rating - RO 3.5 SRO 3.9

Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Termination of a release of radioactive liquid

Reference(s) - Sim/Plant design, AOP-008
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-008-004
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 41.10 / 45.6 / 45.13
Comments - K/A match because candidate must know the appropriate actions for a leak in a Monitor Tank and potential consequences should these actions not be taken.

CONTINUOUS USE

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
PLANT OPERATING MANUAL
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ABNORMAL OPERATING PROCEDURE

AOP-008

ACCIDENTAL RELEASE OF LIQUID WASTE

REVISION 9

Effective Date: 8/30/01

RECOMMENDED BY:	Procedure Sponsor	8/9/01 Date
APPROVED BY:	Ken Jung for Manager - Operations	8/24/01 Date

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

   To provide the instruction necessary to respond to a leak from the RWST, Monitor Tank A or B, or Waste Condensate Tank C, D, or E.

2. ENTRY CONDITIONS

   Any unexplained indication of a decrease in any of the following tanks or upon receiving a report that leakage has developed from any of the following tanks:

   - RWST
   - Monitor Tank A or B
   - Waste Condensate Tank C, D, or E

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Leak Status - CONFIRMED BY LOCAL VISUAL INSPECTION	Perform the following: a. Perform a local visual inspection for leakage prior to continuing. b. <u>IF</u> external leakage is found, <u>THEN</u> Go To Step 2. c. <u>IF</u> leakage is <u>NOT</u> found, <u>THEN</u> perform the following: 1) Contact I&C to determine the problem with level indication. 2) Return to procedure and step in effect.
2.	Locally Identify The Source Of Leakage To Determine If It Is Isolable	
3.	Evacuate Unnecessary Personnel From The Affected Area As Follows: a. Place the VLC switch in EMERG b. Place and hold the EVACUATION ALARM switch in the LOCAL position for 15 seconds c. Make a PA System announcement for all unnecessary personnel to stand clear of the affected area due to a leak in progress d. Place and hold the EVACUATION ALARM switch in the LOCAL position for 15 seconds e. Repeat the announcement f. Place the VLC switch in NORM	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	Notify RC Personnel To Survey The Affected Area And Establish Access Control	
5.	Notify Chemistry To Take Samples Of Leakage For Activity	
6.	Implement The EALs	
7.	Transition To Steps For The Affected Tank Below:	

TANK	STEP
RWST	Step 8
Monitor Tank	Step 17
Waste Condensate Tank	Step 21

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Determine Leak Location As Follows:	
	a. Check leak location - IDENTIFIED	a. <u>WHEN</u> the leak location is identified, <u>THEN</u> Go To Step 8.b.
	b. Check leak location - INSIDE AUXILIARY BUILDING	b. Go To Step 10.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
17.	Stop Additions To The Affected Tank As Follows:	
	a. Check status of waste processing to affected tank - IN PROGRESS	a. Go To Step 18.
	b. At the WDBRP, stop the Gas Stripper Feed Pumps	
	c. Locally close the affected Monitor Tank inlet valve located on the south side of the tank: • CVC-1280, MONITOR TANK "A" INLET OR • CVC-1278, MONITOR TANK "B" INLET	
	d. Notify Chemistry that processing has been stopped	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
18.	Determine Leak Status As Follows:	
	a. Check leak location - IDENTIFIED	a. WHEN the leak location is identified, THEN Go To Step 18.b.
	b. Check leak - DETERMINED ISOLABLE	b. Perform the following: 1) Transfer contents from the leaking Monitor Tank to the CVCS Holdup Tank(s) using OP-305-1, Boron Recycle Process (Infrequent Operation). 2) Notify Maintenance personnel to initiate repair activities. 3) Go To Step 32.
	c. Check leak location - LOCATED IN MONITOR TANK PUMP TRANSFER LINE	c. Isolate the leak as close to the source as possible. Go To Step 31.
19.	Isolate The Leak As Follows:	
	a. At the WDBRP, verify Monitor Tank Pumps - STOPPED	
	b. Locally close the discharge valve on the leaking Monitor Tank located on the south side of the tank: • CVC-1281, MONITOR TANK "A" DISCHARGE OR • CVC-1282, MONITOR TANK "B" DISCHARGE	
20.	Go To Step 31	

24. 074 EA2.07 001

Given the following plant conditions:

- PATH-1 has been entered.
- SI or RHR flow cannot be established.

Time	CETC Temp	RVLIS Full Range
1000	675°F	51%
1005	695°F	39%
1010	705°F	37%
1015	750°F	30%
1030	1000°F	29%
1045	1150°F	28%
1100	1205°F	25%

Based on the above indications, what is the earliest time at which entry conditions for FRP-C.1, Response to Inadequate Core Cooling, would be present?

A. 1005

B.✓ 1010

C. 1015

D. 1100

The correct answer is B.

A. Incorrect - The criteria for RVLIS has been met for FRP-C.2, Response to Degraded Core Cooling, but not for FRP-C.1 since CETC temperature is below 700°F.

B. Correct.

C. Incorrect. The criteria was met at 1010. This is plausible because RVLIS value is equivalent to the lowest RVLIS Dynamic Range that would meet the entry requirements for FRP-C.2.

D. Incorrect. The criteria was met at 1010. This is plausible because exceeding 1200°F CETC will by itself meet the entry conditions for FRP-C.1.

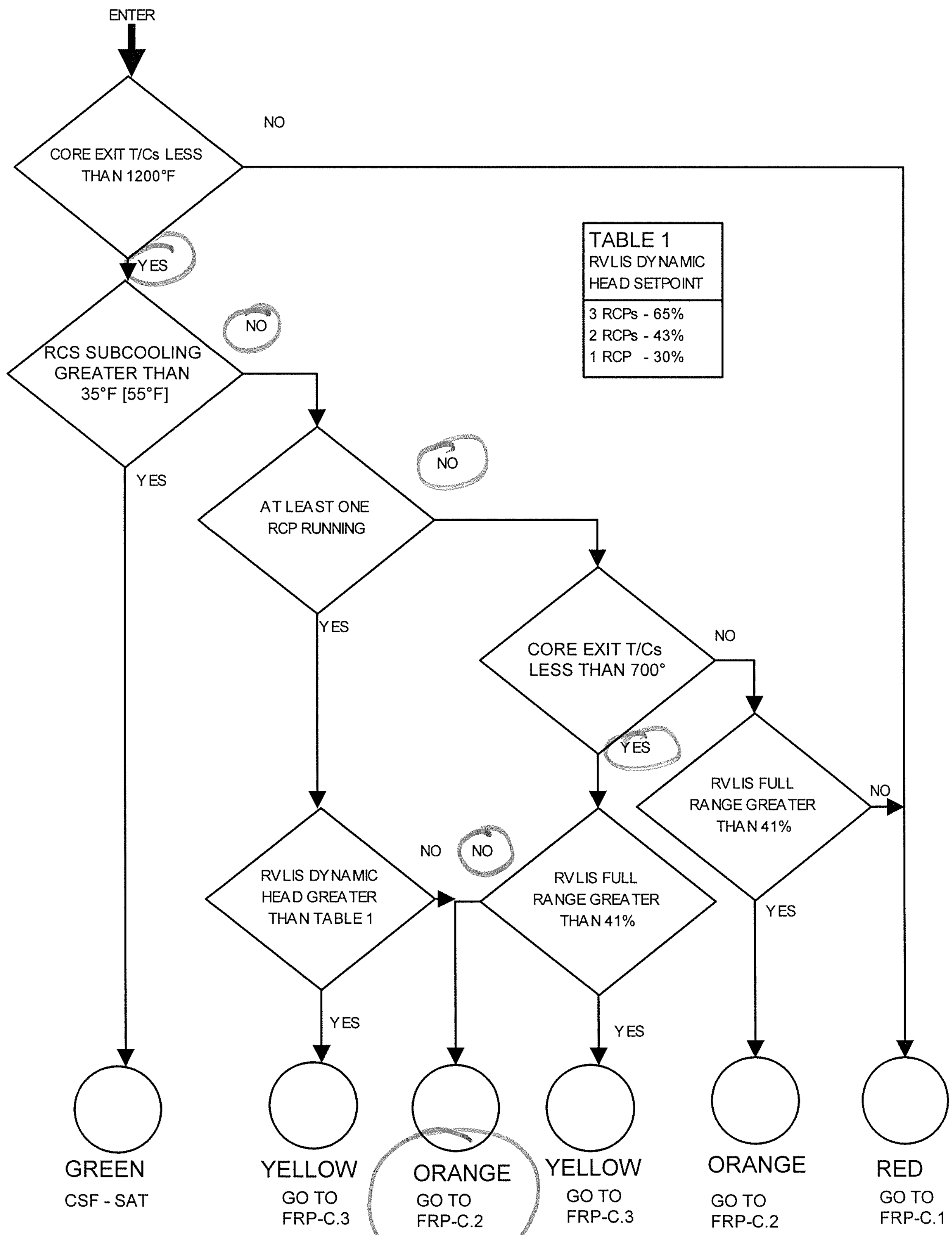
Question 24
Tier 1 / Group 2
K/A Importance Rating - RO 4.1 SRO 4.7

Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: The difference between a LOCA and inadequate core cooling, from trends and indicators

Reference(s) - Sim/Plant design, FRP-C.1, CSFST
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-C.1-003
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments - K/A match because the candidate must interpret given data and determine when an Inadequate Core Cooling condition exists.

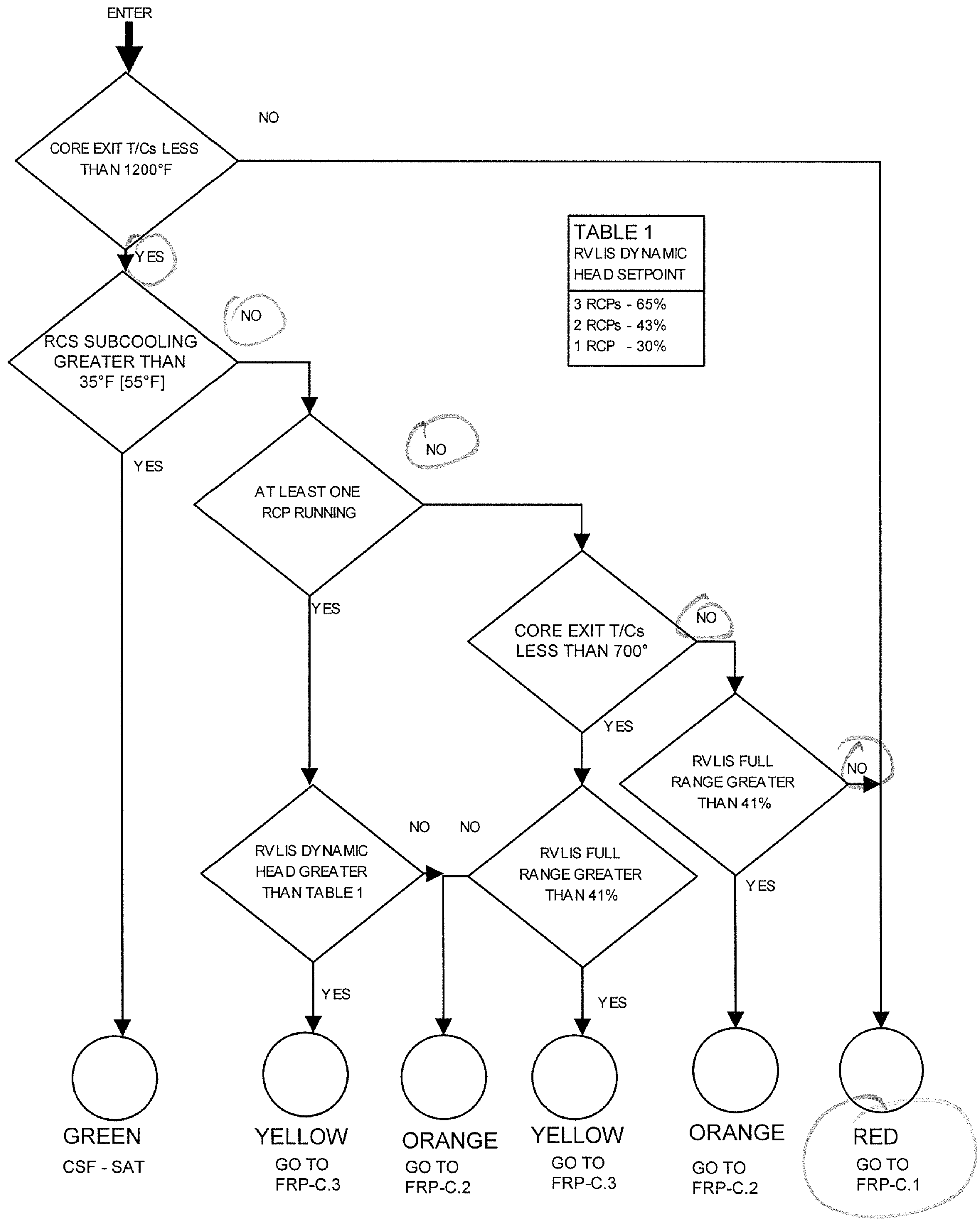
1005

CSF-2, CORE COOLING



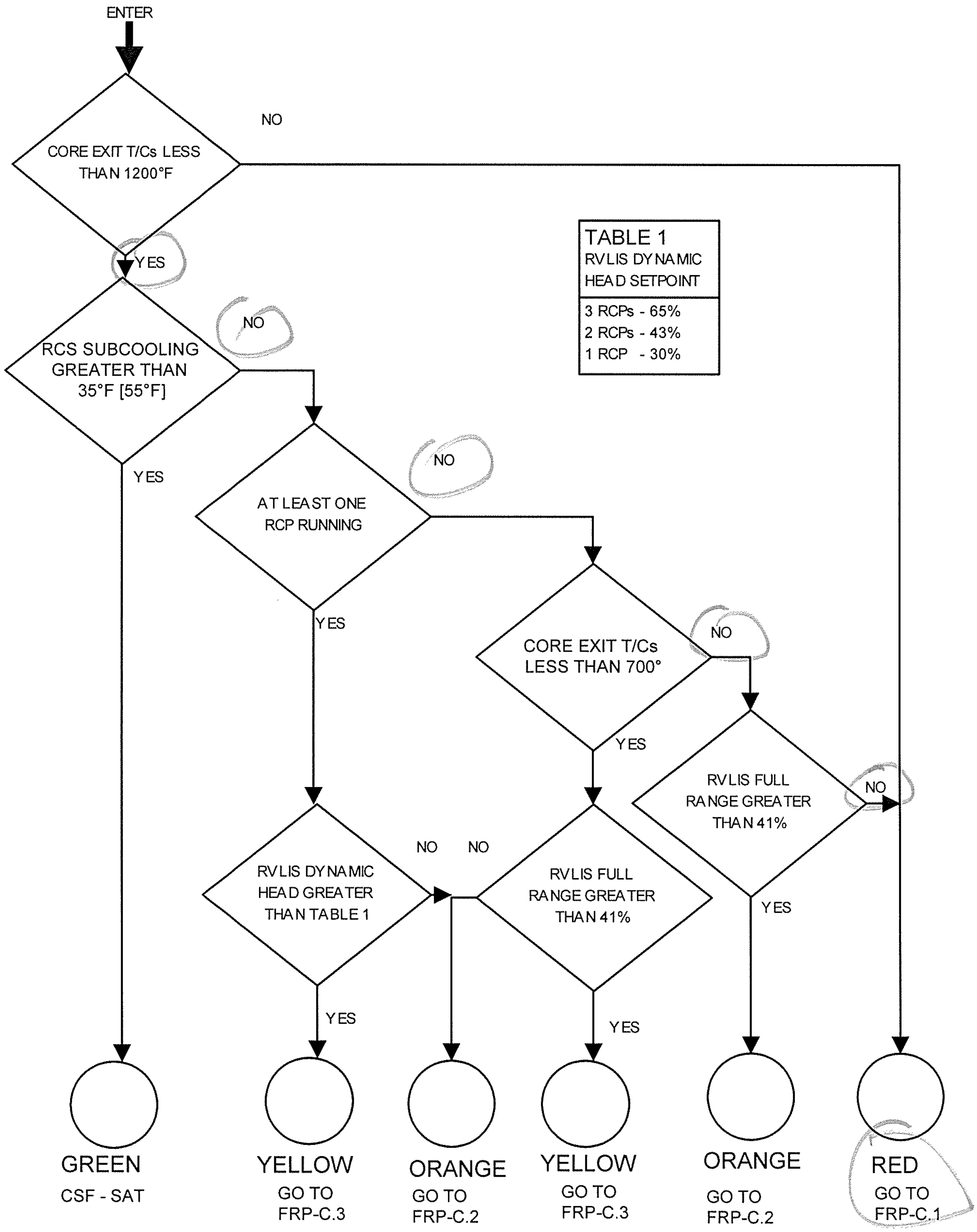
1D10

CSF-2, CORE COOLING



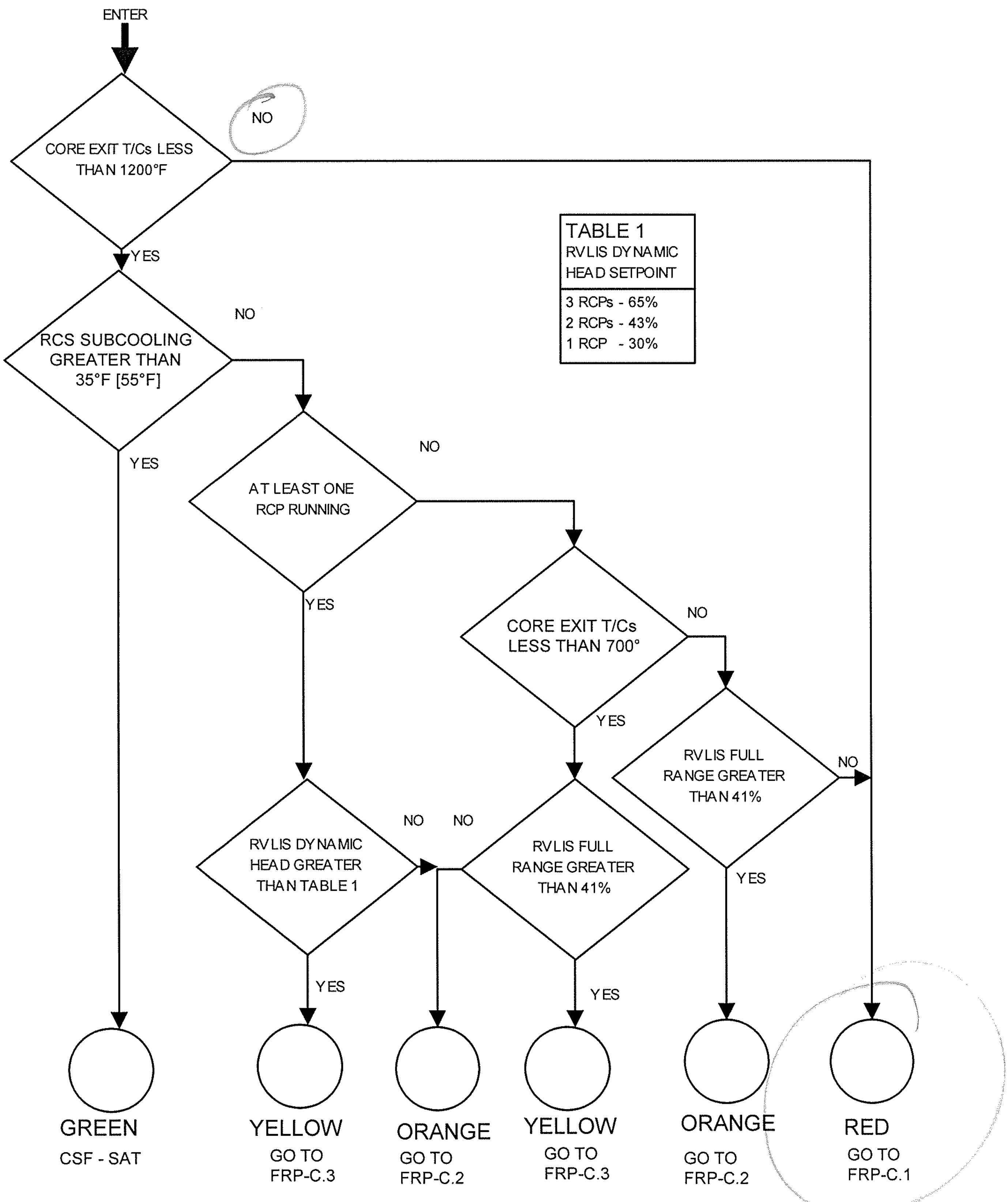
1015

CSF-2, CORE COOLING



1100

CSF-2, CORE COOLING



CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

FUNCTION RESTORATION PROCEDURE

FRP-C.1

RESPONSE TO INADEQUATE CORE COOLING

REVISION 17

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to restore core cooling.

2. ENTRY CONDITIONS

   CSF-2, Core Cooling Critical Safety Function Status Tree on a RED condition.

- END -

25. W/E 03 EK1.1 001

The crew is implementing EPP-8, Post-LOCA Cooldown and Depressurization, following a Small Break LOCA

- Containment pressure peaked at 3.8 psig
- Steps have been taken to reduce SI flow by stopping "C" SI Pump.
- "A" SI Pump remains running.
- All charging pumps are running with maximum charging flow.

After "C" SI pump is stopped the following plant conditions are observed:

- RCS subcooling is 32°F and lowering.
- PZR level is 22% and lowering.

Which ONE of the following;
(1) identifies the actions that will be taken
AND
(2) why will this action be taken?

A.✓ (1) Start both SI pumps.

(2) Due to low subcooling.

B. (1) Start both SI pumps.

(2) Due to low pressurizer level.

C. (1) Energize PZR heaters.

(2) Due to low subcooling.

D. (1) Energize PZR heaters.

(2) Due to adequate pressurizer level.

The correct answer is A.

A. Correct. Foldout criteria for SI reinitiation criteria is if any of the following occurs - RCS subcooling < 35°F (currently 32°F) or PZR level can not be maintained> 10% (currently 22%). With low subcooling then start both SI Pumps.

B. Incorrect. Plausible because the action is correct for PZR level not being maintained greater than 10%. However PZR level is 22%. If CV Pressure had exceeded 4 psig, the PZR Level requirement would have been 32%.

C. Incorrect. EPP-8 has several steps that energize PZR Heaters to help maintain a steam bubble in the pressurizer or heat pressurizer insurge to saturation. However, in this case foldout criteria requires that both SI pumps be started.

D. Incorrect. EPP-8 has several steps that energize PZR Heaters to help maintain a steam bubble in the pressurizer or heat pressurizer insurge to saturation. PZR level is adequate to energize heaters per the control logic but not in accordance with EPP-8. EPP-8 requires level to be greater than 71% to energize heaters.

Question 25
Tier 1 / Group 2
K/A Importance Rating - RO 3.4 SRO 4.0

Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown and Depressurization): Components, capacity, and function of emergency systems.

Reference(s) - Sim/Plant design, EPP- Foldout "B" and EPP-8,
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-8-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.8 / 41.10 / 45.3
Comments - K/A match because candidate is given the conditions following a SBLOCA after a SI flow reduction has been attempted. Candidate must know the operational requirements for SI reinitiation.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

END PATH PROCEDURE

EPP-8

POST LOCA COOLDOWN AND DEPRESSURIZATION

REVISION 17

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to cooldown and depressurize the RCS to Cold Shutdown conditions following a loss of reactor coolant inventory.

2. ENTRY CONDITIONS

   Path-1, when RCS pressure is greater than the shutoff head of the RHR Pumps.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Open Foldout B	
* 2.	Determine If RHR Pumps Should Be Stopped:	
	a. Check RCS pressure: • GREATER THAN 275 PSIG [400 PSIG] AND • STABLE OR INCREASING	a. Go To Step 3.
	b. Verify RHR PUMPs - STOPPED	
	c. Check RCS pressure - LESS THAN 275 PSIG [400 PSIG]	c. IF RCS pressure decreases below 275 psig [400 psig] during this procedure, THEN restart RHR Pumps. Go To Step 3.
	d. Restart the RHR Pumps	
* 3.	Check Emergency Busses - ANY ENERGIZED BY EMERGENCY DIESEL	IF offsite power is lost, THEN perform Steps 4 and 5. Go To Step 6.

CONTINUOUS USE

FOLDOUT B

(Page 1 of 2)

1. SI TERMINATION CRITERIA

   IF ALL conditions below occur, THEN Go To EPP-7, SI Termination:

   a. RCS subcooling - GREATER THAN 35°F [55°F]

   b. Heat Sink established as follows:

      - Total feed flow to intact S/Gs - GREATER THAN 300 GPM OR $0.2x10^6$ PPH

      OR

      - Level in at least one intact S/G - GREATER THAN 8% [18%]

   c. RCS pressure -

      - GREATER THAN 1650 PSIG [1750 PSIG]

      AND

      - STABLE OR INCREASING

   d. PZR level - GREATER THAN 10% [32%]

2. SI REINITIATION CRITERIA

   IF EITHER condition below occurs, THEN start both SI Pumps:

   - RCS Subcooling - LESS THAN 35°F [55°F] >@ 32°F
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [32%]

3. SECONDARY INTEGRITY CRITERIA

   IF EITHER condition below occurs, THEN Go To EPP-11, Faulted Steam Generator Isolation, unless the faulted S/G is already isolated:

   - Any S/G pressure is decreasing in an uncontrolled manner.
   - Any S/G has completely depressurized.

26. W/E 14EA1.1 001

Given the following plant conditions:

- A LOCA with a loss of off-site power has occurred.
- The supply breaker to MCC-6 tripped open prior to CV Spray Actuation.
- CV Pressure is 15 psig and both Containment Spray Pumps are operating.

Which ONE (1) of the following completes the statements below?

__(1)__ is the status of the Containment Spray System AND __(2)__ is the current status of the Component Cooling Water Pumps.

(Assume no local operator actions have been taken.)

A. (1) There is no discharge path aligned for CV Spray Pump "B". The discharge path is aligned from CV Spray Pump "A".

(2) "A" CCW Pump is running.

B. (1) CV Spray Pumps "A" and "B" are providing spray flow.

(2) "A" CCW Pump is running.

C. (1) There is no discharge path aligned from CV Spray Pump "B". The discharge path is aligned from CV Spray Pump "A".

(2) "B" and "C" CCW Pumps are secured.

D✓ (1) CV Spray Pumps "A" and "B" are providing spray flow.

(2) "B" and "C" CCW Pumps are secured.

The correct answer is D.

A. Incorrect. Both CV Spray pumps have one train of discharge valves that has power to reposition to provide a flowpath. "A" CCW Pump would be running if the DS Bus was energized. However, the stem of the question does not address the DS Bus and no local operator action has been taken to start the DSDG.

B. Incorrect. First half of answer is correct. "A" CCW Pump would be running if the DS Bus was energized. However, the stem of the question does not address the DS Bus and no local operator action has been taken to start the DSDG.

C. Incorrect. Both CV Spray pumps have one train of discharge valves that has power to reposition to provide a flowpath. The second half of the answer is correct.

D. Correct. CV Spry Pump "B" discharge valves are powered from MCC-5 and MCC-6. The valve powered from MCC-5 has power available to travel open and provide a flow path.

Question 26
Tier 1 / Group 2
K/A Importance Rating - RO 3.7 SRO 3.7

Ability to operate and / or monitor the following as they apply to the (High Containment Pressure): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference(s) - Sim/Plant design, System Description, EDP-003
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-J.1-004
Question Source - NEW
Question History - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5 / 45.6
Comments - This question tests the candidates knowledge of the power supplies to the containment spray discharge valves and the system flow paths, and how they are impacted upon a loss of power. This question also tests the candidates knowledge of interlocks that prevent operation of the CCW pumps during a Large Break LOCA (CV Spray signal) and a loss of off-site power.

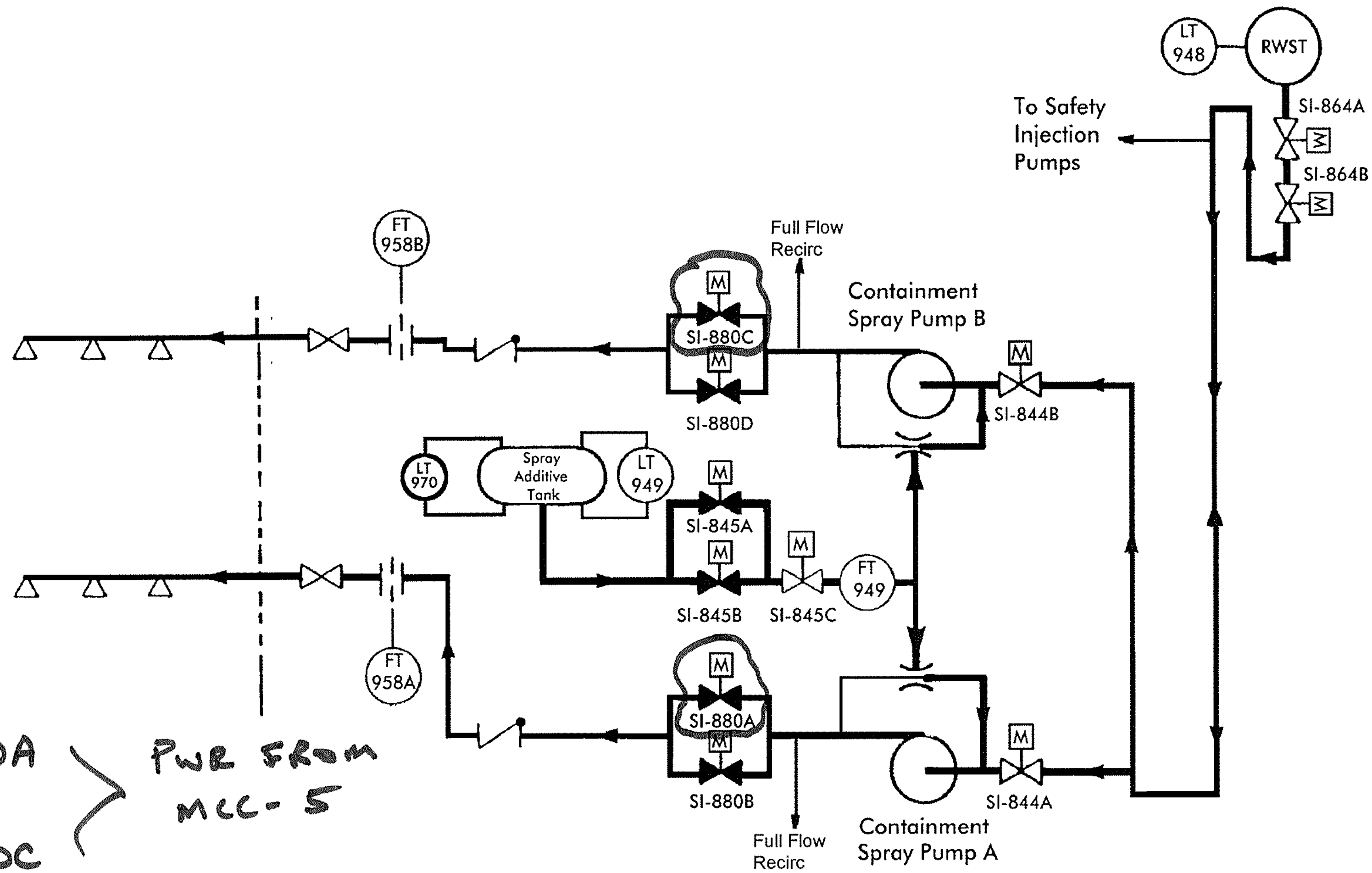
MCC-5 POWER SUPPLY: 480V BUS E-1 (52/21A) LOCATION: AUX BLDG HALLWAY			
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
8B	CV IODINE REMOVAL UNIT, HVE-3 HVE-3	521	52/MCC-5(8B)
8D	BLANK N/A	N/A	N/A
8F	SI PUMP ROOM RECIRC FAN, HVH-6A HVH-6A	551	52/MCC-5(8F)
8J	MOV-350, BORIC ACID TO CHARGING PUMP SUCTION HEADER MOV-350	195	52/MCC-5(8J)
8M	CVC-381, RCP SEAL WATER RETURN ISOLATION CVC-381	165	52/MCC-5(8M)
9C	CC-735, RCP A,B,C THERMAL BARRIER COOLING WATER ISOLATION CC-735	230	52/MCC-5(9C)
9F	SI-865C, ACCUMULATOR C DISCHARGE SI-865C	284	52/MCC-5(9F)
9J	SI-880C, CV SPRAY PUMP B DISCHARGE SI-880C	291	52/MCC-5(9J)
9M	SI-867A, BIT INLET SI-867A	243	52/MCC-5(9M)
10A	SPARE N/A	N/A	N/A
10CL	BORIC ACID HEAT TRACE SECONDARY PANEL BA-HT-SEC-PNL	N/A	52/MCC-5(10CL)
10CR	FEED TO LP-29 (ALT POWER) LP-29	N/A	52/MCC-5(10CR)

MCC-5 POWER SUPPLY: 480V BUS E-1 (52/21A) LOCATION: AUX BLDG HALLWAY			
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
12M	SI-845A, SPRAY ADDITIVE TANK DISCHARGE SI-845A	295	52/MCC-5(12M)
13C	SI-844A, CV SPRAY PUMP A SUCTION SI-844A	293	52/MCC-5(13C)
13F	SI-863A, RHR PUMP A DISCHARGE TO SI PUMP SUCTION SI-863A	280	52/MCC-5(13F)
13J	SI-862A, RHR LOOP RWST ISOLATION SI-862A	248	52/MCC-5(13J)
13M	SI-845C, SPRAY ADDITIVE TANK OUTLET THROTTLE SI-845C	297	52/MCC-5(13M)
14C	RHR-759A, RHR HEAT EXCHANGER A OUTLET RHR-759A	215	52/MCC-5(14C)
14F	SI-865A, ACCUMULATOR A DISCHARGE SI-865A	282	52/MCC-5(14F)
14J	SI-880A, CV SPRAY PUMP A DISCHARGE SI-880A	288	52/MCC-5(14J)
14M	SI-861A, CV SUMP SUCTION SI-861A	268	52/MCC-5(14M)
15M	CURRENT LIMITING REACTOR CLRX/MCC-5	1187	N/A
16A	BLANK N/A	N/A	N/A
16C	EDG FUEL OIL TRANSFER PUMP A DG-FO-XFER-PMP-A	947	52/MCC-5(16C)

MCC-6			
POWER SUPPLY: 480V BUS E-2 (52/23C)		LOCATION: E-1/E-2 ROOM	
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
CAUTION MCC 6, Breaker 8M for RHR-751 is normally OPEN iaw Appendix R requirements – Reference (92/18 Criteria) - Label C-423.			
8M	RHR-751, RHR PUMP SUCTION FROM RCS RHR-751	213	52/MCC-6(8M)
9C	BORIC ACID TRANSFER PUMP B BA-XFER-PMP-B	192	52/MCC-6(9C)
9F	CC-716B, RCP COOLING WATER INLET ISOLATION CC-716B	232	52/MCC-6(9F)
9J	CC-730, RCP BEARING COOLING WATER OUTLET ISOLATION CC-730	233	52/MCC-6(9J)
9M	SI-866A, SI PUMP DISCHARGE HOT LEG INJECTION SI-866A	241	52/MCC-6(9M)
10A	BLANK N/A	N/A	N/A
10C	SDAFW PUMP AUX OIL PUMP SDAFWP-OIL-PMP	634	52/MCC-6(10C)
10F	CC-749B, RHR HEAT EXCHANGER B COOLING WATER OUTLET CC-749B	219	52/MCC-6(10F)
10J	SI-865B, ACCUMULATOR B DISCHARGE SI-865B	283	52/MCC-6(10J)
10M	SI-867B, BIT INLET SI-867B	244	52/MCC-6(10M)
11C	V2-6C FW HEADER DISCHARGE FW-V2-6C	640	52/MCC-6(11C)
11F	SI-880B, CV SPRAY PUMP A DISCHARGE SI-880B	289	52/MCC-6(11F)

MCC-6 POWER SUPPLY: 480V BUS E-2 (52/23C) LOCATION: E-1/E-2 ROOM			
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
11J	SI-864B, RWST DISCHARGE SI-864B	236	52/MCC-6(11J)
11M	SI-862B, RHR LOOP RWST ISOLATION SI-862B	249	52/MCC-6(11M)
12C	SI-844B, CV SPRAY PUMP B SUCTION SI-844B	294	52/MCC-6(12C)
12F	SI-880D, CV SPRAY PUMP B DISCHARGE SI-880D	292	52/MCC-6(12F)
12J	RHR-744B, RHR LOOP TO RCS COLD LEG RHR-744B	221	52/MCC-6(12J)
12M	SI-863B, RHR PUMP B DISCHARGE TO SI PUMPS SUCTION SI-863B	281	52/MCC-6(12M)
13C	RHR-759B, RHR HEAT EXCHANGER B OUTLET RHR-759B	217	52/MCC-6(13C)
13F	SI-845B, SPRAY ADDITIVE TANK DISCHARGE SI-845B	296	52/MCC-6(13F)
13J	SI-860B, CV SUMP RECIRC SUCTION SI-860B	267	52/MCC-6(13J)
13M	SI-870B, BIT OUTLET COLD LEG INJECTION SI-870B	245	52/MCC-6(13M)
14C	CC-832, CCW MAKE-UP FROM PRIMARY WATER CC-832	203	52/MCC-6(14C)
14F	LCV-115C, VOLUME CONTROL TANK DISCHARGE LCV-115C	160	52/MCC-6(14F)
14J	SI-861B, CV SUMP SUCTION SI-861B	269	52/MCC-6(14J)

CONTAINMENT SPRAY FLOWPATH
CSS-FIGURE-1



SI-880A
&
SI-880C

PWR FROM
MCC-5

cssf01

INFORMATION USE ONLY

generator is started and its output breaker closes when output voltage builds up (within 10 seconds). There is a two-second time delay that ensures that a voltage dip does not result in closure unless E2 is de-energized. The closing of the output breaker renews the block to ensure that "B" auxiliary feed pump starts at the proper time. When the loss of voltage on the E2 Bus is sensed, the timing for the blackout sequence starts. After twenty (20) seconds "C" Service Water Pump starts, a start signal is sent to both Service Water Booster Pumps, and both Service Water Booster Pumps start when the Service Water Header pressure builds up to 30 PSIG. After twenty (25) seconds "D" Service Water Pump starts and a start signal is sent to both Service Water Booster Pumps. Both Service Water pumps are started to ensure sufficient flow exist to the EDG if one of the pumps is inoperable. After thirty (30) seconds "C" Component Cooling Pump will start. After thirty-nine and one-half (39.5) seconds "B" Auxiliary Feed Pump will start and reset the blackout sequence after a 2 second time delay.

If sufficient voltage does not exist on E2 at the end of one cycle, the control circuit will lock-out the loss of voltage signal and the blackout sequence will stop. In this case, the E2 motor loads must be manually started upon voltage restoration. This lock-out feature is installed to prevent excessive cycling of equipment and to remove additional battery loading during a sustained loss of power to E2.

6.3.3 Loss of Power to the Emergency Buses With An SI Signal present

The same action that occurred in Item 6.3.1 and 6.3.2 will take place except the blackout sequence loads will be blocked by the S-Signal. During the bus clearing 52/1B and 52/2B will also be opened on the E1 side and 52/15B and 52/32A and 52/16B will be open on the E2 side. This is a redundant measure to ensure the Emergency Busses are isolated from the rest of the plant electrical system. After the diesel generator output breakers are closed the safeguard sequence will take place. If a Containment Spray signal (P-signal) is present, the Containment Spray pumps will start anytime power is available to its respective emergency bus.

NOTE: "B" and "C" Charging Pumps will be tripped if a SI signal is present and power is being supplied by diesel generators.

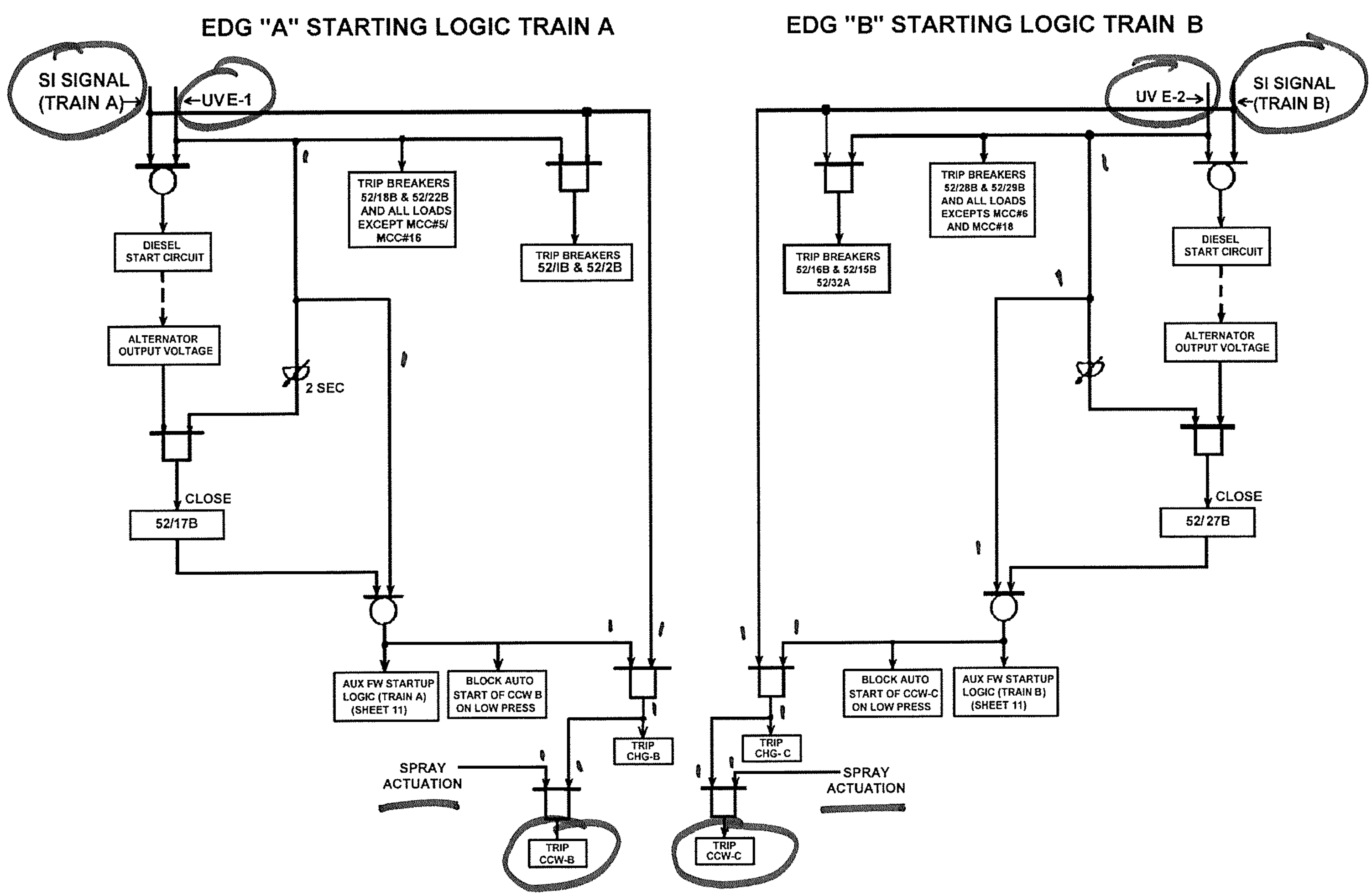
NOTE: "B" and "C" Component Cooling Pumps will be tripped if a SI and a spray signal are present and power is being supplied by diesel generators.

NOTE: "B" and "C" Component Cooling Pumps will not start automatically on low pressure if the diesel generator is supplying its respective bus.

NOTE: "A" Component Cooling Pump will start anytime on low pressure if power is available. "A" pump is on the DS Bus.

6.4 Actions That Can Be Initiated By Other Signals

EDG STARTUP LOGIC
ESF-FIGURE-11



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27. W/E 15EK3.1 001

Which ONE (1) of the following identifies the operational concern associated with Containment Flooding?

Containment Flooding could lead to .....

A.✓ Critical system transmitter failures needed to ensure an orderly safe plant shutdown.

B. Submersion of the Reactor Coolant Drain Tank resulting in damage to the Gas Analyzer.

C. Water potentially leaking through Electrical Penetrations resulting in electrical shorts and/or ground.

D. Submersion of CVC-200A, B, C, Letdown Orifice Isolations resulting in a loss of normal letdown capability.

The correct answer is A.

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, provides actions to respond when the containment level is greater than design flood level. This level is significant since the critical systems and components, which are necessary to ensure an orderly safe plant shutdown and provide feedback to the operator regarding plant conditions, are normally located above the design flood level.

A. Correct. The concern is that flooding would affect critical systems which are necessary to ensure an orderly safe shutdown and provide feedback to the operator.

B. Incorrect. Submersion of the RCDT will occur during a design basis LOCA. Also, the RCDT is not needed to safely place the plant in a shutdown condition. The gas analyzer does sample the RCDT. The gas analyzer can be damaged if excessive moisture is sampled.

C. Incorrect - The electrical penetrations are air tight and thus water tight. By design no water will leak from the containment to the outside. Electrical short and/or grounds can occur if electrical connection are exposed to water. Should not occur in CV since cables are designed for harsh environment operation.

D. Incorrect - CVC-200A,B,C will be flooded during normal design basis LOCA. During a LOCA letdown will be isolated so the position of CVC-200A,B,C is not critical to maintain the plant in a shutdown condition.

Question 27
Tier 1 / Group 2
K/A Importance Rating - RO 2.7 SRO 2.9

Knowledge of the reasons for the following responses as they apply to the (Containment Flooding): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Reference(s) - Sim/Plant design, FRP-J.2 Basis Document.
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-J.2-002
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 41.10 / 45.6 / 45.13
Comments - K/A match because the candidate must know the consequences associated with Containment Flooding.

DISCUSSION (From the WOG FR-Z.2 Basis Document)

1. INTRODUCTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, is a Function Restoration Guideline (FRG) that provides procedural guidance when the containment level is greater than flood level.

There is only one explicit transition to guideline FR-Z.2. It is from the Critical Safety Function Status Tree F-0.5, CONTAINMENT, on an ORANGE priority when containment sump level is greater than flood level.

After all the actions in guideline FR-Z.2 are completed, the operator is instructed to return to the guideline and step in effect.

2. DESCRIPTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, provides actions to respond when the containment level is greater than design flood level. This level is significant since the critical systems and components, which are necessary to ensure an orderly safe plant shutdown and provide feedback to the operator regarding plant conditions, are normally located above the design flood level. Therefore, the guideline FR-Z.2 is entered from the Containment Status Tree on an ORANGE priority when this design flood level is exceeded.

The primary purpose of the containment sump area is to collect the water injected into the containment or spilled from the reactor coolant system following an accident. The water collected in the containment sump is then available for long term core and/or containment cooling via the emergency core cooling or containment spray recirculation systems. In addition, the containment sump collects the injected or spilled water into areas such that vital systems or components will not be flooded and thus rendered inoperable.

The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators. This water volume approximates the maximum water volume introduced into the containment following a LOCA plus a steamline or feedline break inside containment.

An indicated water level in the containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes have been introduced into the containment. Also, the high water level provides an indication that potential flooding of critical systems and components needed for plant recovery may occur.

The actions in this guideline attempt to identify any unexpected source of water and isolate it if possible. Beyond that the plant engineering staff is consulted to determine if transfer of containment sump water to other tanks is appropriate.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-Z.2 is to provide actions to respond to containment flooding.

The following subsections provide a summary of the major action categories of operator actions and the key utility decision points for guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-Z.2 is given below in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN FR-Z.2

- o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible
- o Notify Plant Engineering Staff of Sump Level and Activity Level

o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible

The first action in this guideline is to try to identify the source of water which is causing containment flooding and isolate it. The concern regarding flooding is that critical plant components needed for plant recovery could be damaged and rendered inoperable.

o Notify Plant Engineering Staff of Sump Level and Activity Level

By knowing the sump level and activity level, the plant engineering staff can determine if the excess water can be transferred to storage tanks located outside containment.

28. 003 A3.04 001

Given the following plant conditions:

- The plant is at 50% RTP.
- FT-434 RCS LOOP "C" FLOW TRANSMITTER low pressure sensing line develops a leak to the CV atmosphere.

Which ONE (1) of the following describes the expected plant response to this event?

A. All Loop C flow channels read low.

B. All Loop C flow channels read high.

C.✓ Flow indication will rise on ONLY the affected Loop C flow channel.

D. Flow indication will lower on ONLY the affected Loop C flow channel.

The correct answer is C

A. Incorrect - Misconception on loop flow detector arrangement and operation. This fault will cause flow indication to rise and will only affect one flow channel since there are three low pressure taps and one high pressure tap.

B. Incorrect - Misconception on loop flow detector arrangement and operation. This fault will cause flow indication to rise but will only affect one flow channel since there are three low pressure taps and one high pressure tap.

C. Correct

D. Incorrect - Misconception on loop flow detector arrangement and operation. This fault will cause flow indication to rise.

Question 28
Tier 2 / Group 1
K/A Importance Rating - RO 3.6 SRO 3.6

Ability to monitor automatic operation of the RCPS, including: RCS flow

Reference(s) - Sim/Plant design, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - RCS 012
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5
Comments - K/A match because candidate must know the impact on RCS flow indication with a leak on the low pressure sensing line.

(OTΔT) and overpower ΔT (OPΔT) setpoints. Through a two out of three (2/3) matrix it supplies a signal to steam break protection and the steam dump interlock.

Protection ΔT is used for high ΔT alarm. The loop ΔT is compared to its OPΔT and OTΔT setpoints for a reactor trip function. A switch on the RTGB allows the operator to select one of the ΔTs to be recorded on TR-412.

4.1.3 Flow

4.1.3.1 Reactor Coolant Loops

The flow rate in each reactor coolant loop is obtained from three differential pressure measurements at a piping elbow in the reactor coolant pump suction of each loop:

Loop 1 - FT-414, 415 & 416
Loop 2 - FT-424, 425 & 426
Loop 3 - FT-434, 435 & 436

There are 3 Low Pressure taps and 1 High Pressure tap. This ensures that a failure of 1 Low Pressure tap would not cause a Reactor trip nor prevent a trip from occurring.

These measurements provide a low flow signal to the Reactor Protection System to actuate a reactor trip. Each flow measurement has a flow indicator on the RTGB.

4.1.4 Pressure

RCS loop pressure is sensed by a narrow and a wide range pressure transmitter on Loop 3. The wide range pressure signal (PT-402) provides indication at PI-402 and is recorded at PR-444 on the RTGB.

The narrow range pressure signal (PT-403) provides signals to the RHR system for valve interlocks. (Figure 18)

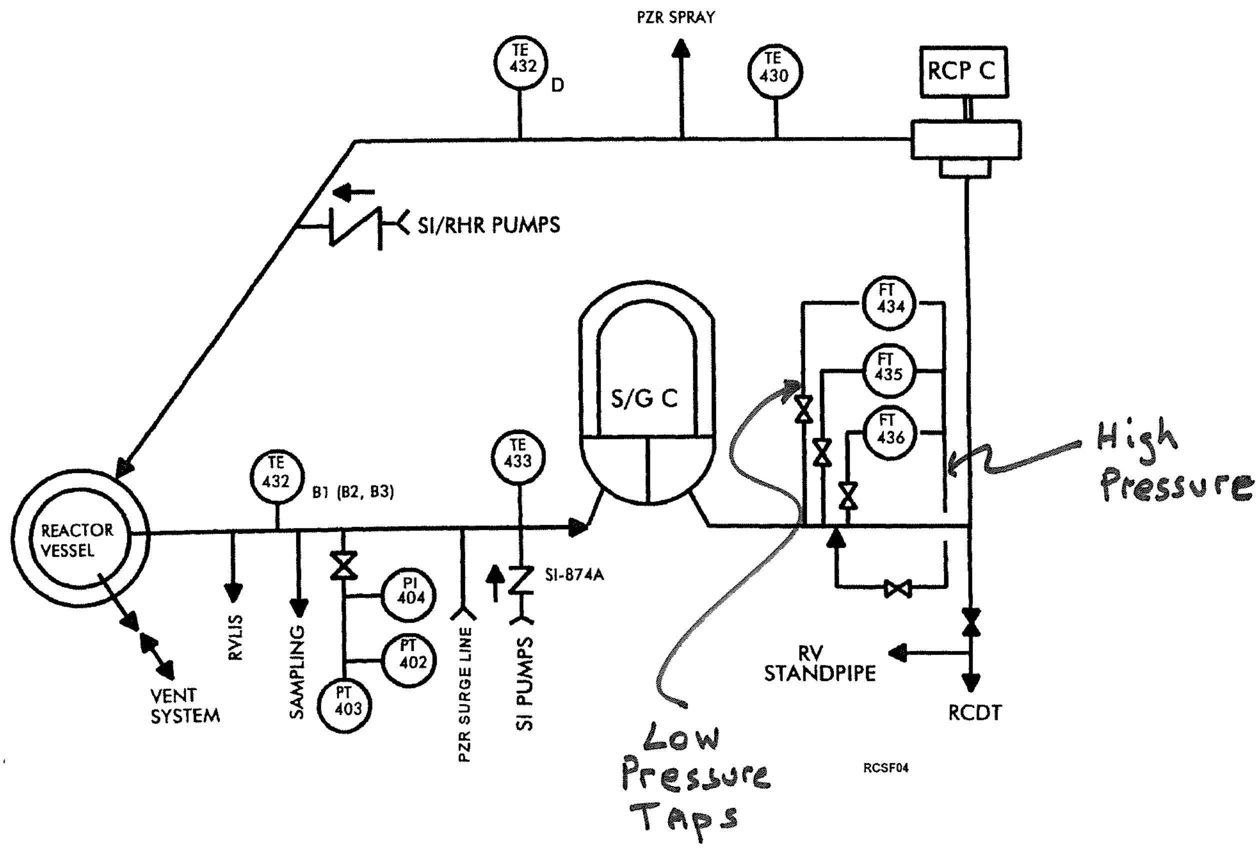
Maximum pressure gauges are connected to Loop 2 (PI-405), and Loop 3 (PI-404). Each has two needles. One indicates actual pressure and the other will stay at the maximum pressure that is obtained. These gauges are located inside the CV outside of the missile barrier (OMB).

4.1.5 Level

Level in the RCS during refueling and other cold shutdown activities is sensed at the Loop 2 and Loop 3 crossover leg drain lines by LT-403 (Loop 2) and LT-404 (Loop 3). These measurements provide indication on the RTGB, and a low level alarm for use during RCS drain down and midloop operations. There are also local level indicators

LOOP 3 SIMPLIFIED DRAWING

RCS-FIGURE-4



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29. 004 A3.15 001

Given the following plant conditions:

- The plant is in Mode 5.
- The RCS is solid with pressure being automatically maintained.
- A clearance error results in PZR Backup Group A Heaters becoming energized.

Which ONE (1) of the following describes the FIRST (INITIAL) response to the rising Pressurizer temperature and pressure?
(Assume NO operator actions.)

A. Annunciator APP-001-D6, LP LTDN LN HI PRESS, actuates.

B. Annunciator APP-001-E6, LP LTDN RELIEF HI TEMP, actuates.

C. Letdown flow rises as HCV-142, RHR TO LETDOWN & PURIFICATION FLOW CONTROL , opens.

D.✓ Letdown flow rises as PCV-145, LOW PRESSURE LETDOWN PRESSURE CONTROL VALVE, opens.

The correct answer is D.

A. Incorrect. Plausible because this would eventually occur at 400 psig if PCV-145 did not initially control the rising pressure. Incorrect because, in this case, no adjustments to charging and letdown have been made and PCV-145 would act first and automatically maintain pressure until it is 100% open.

B. Incorrect. Plausible if normal letdown was in service. If normal letdown was in service CVC-203A would lift at 500 psig if PCV-145 did not initially control a rising pressure condition. When aligned for solid plant operation the normal letdown path, and thus CVC-203A are removed from service.

C. Incorrect. Plausible because flow will increase due to PCV-145 opening.

D. Correct. PCV-145 controls RCS Pressure with the system solid. With the term "automatically" specified in Bullet 2, PCV-145 is set at 350 psig. Until PCV-145 is 100% open, system pressure will be maintained.

Question 29
Tier 2 / Group 1
K/A Importance Rating - RO 3.5 SRO 3.6

Ability to monitor automatic operation of the CVCS, including: PZR pressure and temperature

Reference(s) - Sim/Plant design, System Description, GP-007, Plant Cooldown From Hot Shutdown to Cold Shutdown.
Proposed References to be provided to applicants during examination - None
Learning Objective - CVCS 009
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5
Comments - Meets K/A by considering PCV-145 alignment and response to a change in PZR temperature and therefore pressure since the plant is solid.

ATTACHMENT 10.3
Page 3 of 4

RHR AND CVCS OPERATION WITH SOLID RCS CONDITIONS

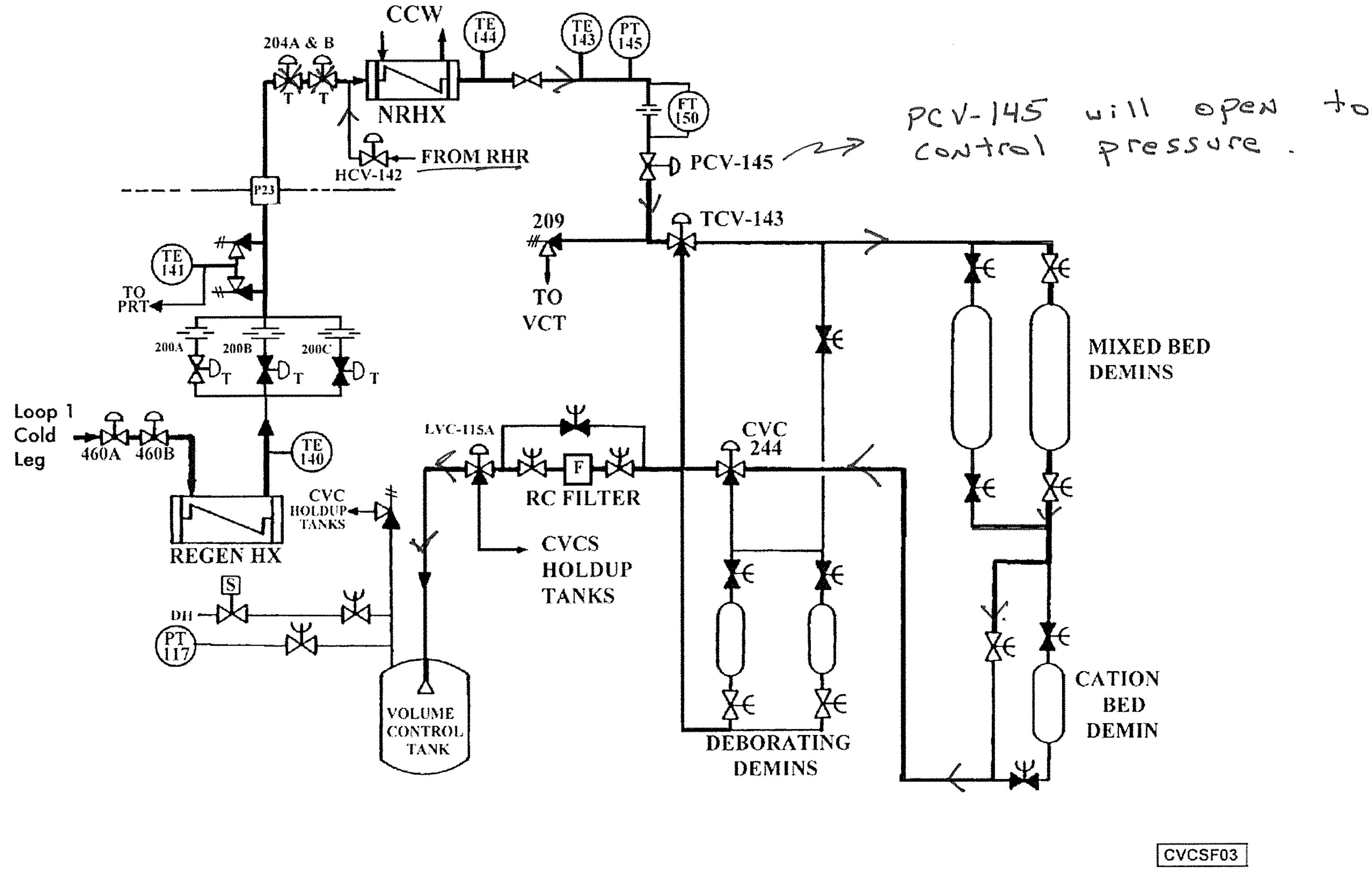
6. Close HCV-758
The effect of closing HCV-758 is equal and opposite as that of opening HCV-758. Initially RCS pressure will increase, then steady state to steady state, an increase in letdown flow will stabilize pressure to offset the increased water volume due to the heat up. → SAME AS ENERGIZING HEATERS IN PZR.

7. Stop the running RHR pump
When the running RHR pump is stopped, letdown flow essentially drops to zero. At the same time that letdown flow goes to zero, RCS temperatures began to rise due to no flow through the coolers. Both of these factors will rapidly increase RCS pressure. If left with no Operator action, RCS pressure would eventually rise to a point to again establish letdown flow equal to charging flow. If charging flow is terminated, RCS pressure will still rise to a point sufficient to give letdown flow through HCV-142 and PCV-145 to offset increased RCS water volume. The problem is that this pressure is greatly above the setpoint for LTOPP operation and the RCS pressure/temperature limitation curve.

8. Letdown demineralizer or RCS filter become clogged (Loss of L/D flow)
For all of the above examples we have considered that demineralizer and filter Delta - P are constant. Over long periods of time the head loss across these devices increases. As the Delta - P across the demineralizers and RC Filter increase, letdown flow decreases. As letdown flow decreases, RCS pressure and therefore letdown pressure increase. As letdown pressure increases PCV-145 opens to restore pressure. This action restores letdown flow to its original value. Since this generally occurs over a long period of time, the only perceptual change that the Operator would notice is a gradual opening of PCV-145 over a period of time. If the demineralizers or RC Filter clog rapidly, as may occur during a CRUD burst or inadvertent valve closure, letdown flow would initially decrease. As flow decreases RCS pressure and letdown pressure will increase. PCV-145 will open. However, in this case since there is no flow path for letdown downstream of PCV-145, PCV-145 will have little or no effect on pressure. If left with no action, RCS pressure will continue to rise until LTOPP operation occurs. Relief CVC-209, located downstream of PCV-145 will operate to relieve pressure at 200 psig. However, it is doubtful if this pressure when combined with the head loss up stream is sufficiently low to prevent LTOPP operation. This same type of effect is seen if RHR-759 A & B and RHR-758 are closed simultaneously except that CVC-209 will not lift. LTOPP operation should handle this event.

NORMAL LETDOWN

CVCS-FIGURE-3



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INIT

8.2.56 **IF** the RCS will be cooled to less than 150°F, **THEN** prior to reaching 150°F, collapse the PZR Bubble as follows:

> **NOTE:** The PZR will fill via spray flow as long as the PZR Surge Line temperature is approximately equal to the PZR Water temperature. This will require operation of the PZR Spray valves as PZR Spray bypass flow will be insufficient to maintain an effective fill rate.
>
> Decreasing PZR Surge Line temperature is an indication of an insurge when the temperature decrease is greater than the normal temperature decrease associated with depressurization.
>
> LI-459, LI-460 and LI-461 shall be used to monitor PZR level in the following steps.
>
> TRM 3.5 limits concerning RCS oxygen concentration are not applicable once the RCS **AND** the Pressurizer are both <250°F. Hydrogen Peroxide additions to the RCS may be performed while filling the Pressurizer when Pressurizer Temperature is <250°F.

1. Verify VCT hydrogen concentration less than 4%. ____% ____
2. Increase charging flow to fill the PZR to 100% level. ____
3. Slowly adjust PC-444J, PZR PRESS 444J, in Manual to throttle open the PZR Spray Valves **OR** throttle open Pressurizer Spray Valves in Manual ____
4. Monitor PZR Surge Line temperature on TI-450. ____
5. **IF** PZR Surge Line temperature decreases **AND** is not due to the normal temperature decrease associated with depressurization, **THEN** decrease charging **OR** spray flow slightly. ____

> **NOTE:** Starting a second Charging Pump before the PZR is solid will minimize the perturbation on the RCS and reduce the potential for lifting LTOP.

6. **IF** it is desired to maximize purification while the PZR is water solid, **THEN** verify two Charging Pumps RUNNING. ____

8.2.56 (Continued) INIT

7. Control Charging flow **AND** PZR Heaters, as necessary, to maintain RCS pressure below 375 psig but above the pressure necessary to maintain at least 210 psid across the RCP No. 1 seal. ____

NOTE:	When the water level rises near the PZR spray nozzle, the spray will lose its effectiveness and pressure will begin to increase as the rising water level compresses the steam bubble. Continued operation of the RCPs during filling of the PZR is required to maintain an effective spray flow. PZR Heaters should be operated as necessary to maintain pressure control while going solid. This includes reducing heaters if the outsurge of the PZR is adversely affecting RHR such that the cooldown cannot be maintained. A PZR outsurge is still required. PZR Steam Space temperature will decrease when PZR water level rises above the PZR Steam Space RTD.

8. Prior to filling the PZR water solid with RCS average temperature less than 260°F, verify at least one RCP is STOPPED. (ACR 93-083) ____

9. **WHEN** the PZR is completely filled, as indicated by an increase in letdown flow to exceed Charging flow, **THEN** perform the following:

   a. Control Charging Pump speed to maintain the desired letdown flow. ____

   b. Verify CLOSED PCV-455A, LOOP 2 PRESSURIZER SPRAY VALVE, by placing PCV-455A, PZR SPRAY 444G, controller in Manual and adjusting the demand signal to 0%. ____

   c. Verify CLOSED PCV-455B, LOOP 3 PRESSURIZER SPRAY VALVE, by placing PCV-455B, PZR SPRAY 444H, controller in Manual and adjusting the demand signal to 0%. ____

8.2.56.9 (Continued) INIT

NOTE:	The pressure reading on PI-145, LOW PRESS LTDN PRESS, will be approximately the additive of the RCS Pressure and the running RHR Pump Discharge Pressure when the RCS is in solid plant pressure control. A PC-145 potentiometer setting of approximately 7.5 will cause PC-145 to control at approximately 450 psig on PI-145 which will control RCS pressure at approximately 350 psig.

d. Control PCV-145, PRESSURE, controller as necessary for pressure control. ____

e. **WHEN** RCS pressure is under control using PCV-145 **AND** Charging Pump speed, **THEN** perform the following:

NOTE:	PZR heaters should be deenergized one set at a time, pausing between each set.

1) Deenergize **ALL** PZR heaters. ____

2) Place a clearance on all PZR heaters.

– PZR Control Group Heaters ____

– PZR Backup Group "A" Heaters ____

– PZR Backup Group "B" Heaters ____

Clearance # __________

10. Adjust HIC-121, CHARGING FLOW, to maintain the Labyrinth Seal dp at greater than 5 inches. (RSPO-93-052) ____

INIT

8.2.57 **WHEN** the PZR is solid, **THEN** perform the following to cooldown the PZR:

NOTE:	The maximum allowable PZR cooldown rate is 200°F/hr. (TRMS 3.4) The PZR shall be cooled by opening the operating loop Spray Valve in 5% increments until the desired cooldown rate is established.

1. Open the operating RCP Loop Spray Valve, PCV-455A **OR** PCV-455B to a demand signal of 5%. ____

2. After 5 minutes check the indicated PZR Cooldown Rate. ____

3. **IF** the cooldown rate is excessive, **THEN** throttle the Spray Valve closed to achieve the desired cooldown rate. ____

4. **IF** the cooldown rate is insufficient, **THEN** continue opening the operating RCP Loop Spray Valve, PCV-455A **OR** PCV-455B, in 5% demand signal increments until the desired cooldown rate is established. ____

NOTE:	Plant conditions may not permit the following step to be performed at this time. This procedure should be continued and this step performed when plant conditions permit.

5. **WHEN** PZR Water and Steam Space temperatures are within 50°F of RCS temperature, **THEN** perform the following:

   a. **IF** RCP "B" is running, **THEN** slowly open PCV-455A, LOOP 2 PRESSURIZER SPRAY VALVE, by placing PCV-455A, PZR SPRAY 444G, controller in Manual and slowly adjusting the demand signal to 100%. ____

   b. **IF** RCP "C" is running, **THEN** slowly open PCV-455B, LOOP 3 PRESSURIZER SPRAY VALVE, by placing PCV-455B, PZR SPRAY 444H, controller in Manual and slowly adjusting the demand signal to 100%. ____

INIT

8.2.58 **IF** this cooldown is for Refueling **OR** to open the RCS for maintenance, **THEN** contact E&C to perform the following:

– Determine if H_2O_2 is to be added to degas the RCS prior to opening the RCS.

______________________ ____
E&C Contact (Print name)

NOTE:	H_2O_2 is added as an oxidizing agent to facilitate crud removal. At least one RCP should remain in operation during the addition of H_2O_2 to ensure uniform crud removal in all RCS Loops.

CAUTION

The Demineralizer Purification Loop should be left in service during the H_2O_2 addition to prevent excess activity build up in the RC filter and Seal Injection Filters (CP-010).

8.2.59 **IF** this cooldown is for Refueling, **WHEN** the following conditions exist:

– RCS pressure is less than 350 psig

– RCS temperature is less than 250°F

– The PZR is solid

– PZR temperature is less than 250°F

THEN perform the following:

1. Verify CLOSED CVC-311, AUX PZR SPRAY ____

NOTE:	The preferred charging path is with CVC-310B open.

2. Open either CVC-310B, LOOP 2 COLD LEG CHG, **OR** CVC-310A, LOOP 1 HOT LEG CHG

   CVC-310B / CVC-310A ____
   (Circle one)

3. Verify TCV-143, VCT/DEMIN DIV, Switch is in AUTO. ____

8.2.59 (Continued) INIT

4. Add H_2O_2 IAW OP-918. ____

5. Notify the Shift Chemistry Tech and the Inside AO that H_2O_2 addition will result in increased radiation levels in the CVCS and the RC Filter may require replacement at a more frequent interval. ____

8.2.60 Verify Letdown flow at the maximum rate without exceeding 120 gpm. ____

NOTE:	Outage Management will determine the duration of the RCS purification evolution. To open the RCS, the RCS I-131 Activity shall be less than 0.01 μCi/cc **AND** H_2 concentration shall be less than 5 cc/kg. To initiate fill of the Refueling Cavity, the RCS Total Gamma Activity (excluding gases) shall be less than 0.05 μCi/cc and the Xe-133 Activity shall be less than 0.5 μCi/cc. (CR 96-00727)

8.2.61 Record the final sample results. (CR 96-00727)

– I-131 Activity ________ μCi/cc

– Hydrogen ________ cc/kg

– Total Gamma Activity (excluding gases) ________ μCi/cc

– Xe-133 Activity ________ μCi/cc

– ________________________ ____
E&C Contact (Print name)

INIT

NOTE:	The RCP(s) should remain in service to achieve the desired RCS temperature with even S/G cooling **AND** to ensure maximum purification of the RCS. RCS sample results and outage/maintenance schedules should be used to help determine whether stopping of RCP(s) is appropriate.

8.2.62 Stop RCPs as follows:

1. Sample the RCS for Boron.

   – RCS Boron Concentration ________ppm

   – ____________________________ ____
   E&C Contact (Print name)

2. **IF** RCS Boron Concentration is not greater than or equal to the required Boron Concentration recorded in 8.2.4, **THEN** borate as necessary to ensure proper RCS Boron Concentration prior to stopping RCPs. (CR 96-01500) ____

3. **IF** desired, **THEN** stop RCP "A". ____

4. **IF** desired, **THEN** stop RCP "B". ____

5. **IF** desired, **THEN** stop RCP "C". ____

NOTE:	RCS pressure shall be maintained below the LTOP setpoint of 400 psig. Auxiliary spray should be used with care when the RCS is water solid. The potential exists for RCS overpressurization with water solid conditions.

8.2.63 **WHEN** PZR temperature is less than 200°F, **THEN** continue cooling the PZR by completing the following:

1. Display ERFIS Point CHF0128A. ____

2. Open CVC-311, AUX PZR SPRAY. ____

3. Verify CLOSED CVC-310A, LOOP 1 HOT LEG CHG. ____

4. Verify CLOSED CVC-310B, LOOP 2 COLD LEG CHG. ____

5. Close CVC-312, CVC-310A BYPASS. ____

6. Adjust Charging Pump speed to obtain Charging flow greater than or equal to 45 gpm (CHF0128A). ____

8.2.63 (Continued) INIT VERI

7. Record and evaluate the forward flow test results for CVC-313, AUX SPRAY LINE CHECK VALVE. ____

	TEST DIRECTION	MEASURED VALUE	ACCEPTANCE CRITERIA	CIRCLE ONE
CVC-313	OPEN VERIFICATION	gpm	CHF0128A ≥ 45 gpm	SAT / UNSAT

8. Adjust Charging Pump Speed to obtain the desired Charging flow for current plant conditions. ____

9. Lock open CVC-312, CVC-310A BYPASS. ____ ____

10. Discontinue recording PZR Cooldown data on Attachment 10.1. ____

8.2.64 Reset the HI FLUX AT SHUTDOWN ALARM to the MODE 5 setpoint (3 times the Source Range indication with all rods in) IAW OP-002. (ACR 95-00294) ____

NOTE: While in MODE 5, ITS LCO 3.4.15 requirements for R-11 and R-12 to be capable of performing their RCS Leakage Detection function are not required. However ITS LCO 3.3.6 requires both R-11 and R-12 to be operable for CV Isolation during movement of recently irradiated fuel assemblies within the CV.

8.2.65 **IF** it is desired to reset R-11 and R-12 setpoints **AND** the conditions set forth in OMM-014 are met, **THEN** reset R-11 and R-12 setpoints. ____

NOTE: The plant is currently in MODE 5, Loops filled, since RCS pressure is above 100 psig and is covered by ITS LCO 3.4.7. When RCS pressure is reduced to below 100 psig, the RCS Loops are not considered to be filled and is covered by ITS LCO 3.4.8. Since the RCS will be depressurized in the following steps, ITS LCO 3.4.8 will be met prior to stopping RCPs.

8.2.66 Verify proper RCS Loops configuration for ITS LCO 3.4.8 as follows:

1. Verify two RHR trains are operable. ____

2. Verify one RHR train is in operation. ____

INIT

NOTE: Do not depressurize the RCS until ready to block open PORVs in GP-008. When depressurized purification flow is lost without a charging pump running. Also H_2O_2 cannot be added if needed after the charging pump is stopped.

8.2.67 Stop the RCPs as follows:

1. Verify all RCPs are STOPPED.

   – RCP "A" _____

   – RCP "B" _____

   – RCP "C" _____

NOTE: The RCP motor heaters control switch is located in the Rod Control Room. One switch controls all three RCP motor heaters.

2. Place the RCP-SPACE HEATER-SW control switch in the ON position. (ACR 93-15017) _____

CAUTION

Rapid RCS depressurization, even when done in a controlled manner, can cause non-condensable gasses, such as nitrogen and hydrogen, to come out of solution in the Pressurizer Level Instrument reference legs. These bubbles of non-condensable gas will cause indicated pressurizer level to indicate higher than actual level.
(OPEX 307324, OE27762, INPO SEN 278)

8.2.68 **IF** the applicable section in OST-254 for maintaining RHR System pressure between 100 psig and 200 psig will be performed, **THEN** slowly adjust PC-145, PRESSURE, controller to decrease RCS pressure to between 100 psig and 200 psig. _____

8.2.69 **WHEN** OST-254 is complete or as desired, **THEN** stop all Charging Pumps. _____

8.2.70 Open CVC-310B, LOOP 2 COLD LEG CHG. _____

8.2.71 Close CVC-311, AUX PZR SPRAY, _____

INIT

NOTE:	If the possibility exists for VCT pressure to exceed RCS pressure, CVC-381, SEAL WTR RTRN ISO or CVC-303A, B & C, SEAL LKOFF valves should be closed to prevent foreign material from back flushing into the RCP seals and damaging the seals. The SEAL LKOFF valves should not be used if Instrument Air will be subsequently isolated to the CV. An RCP shall not be started with CVC-381 or CVC-303A, B or C closed.

8.2.72 To ensure RCP seals are not damaged, perform the following:

1. Perform ONE of the following (N/A the step not performed):

   – Close CVC-381, SEAL WTR RTRN ISO. ____

   OR

   – Close the SEAL LKOFF Valves:

   – CVC-303A ____

   – CVC-303B ____

   – CVC-303C ____

2. Place a Caution Cap on the valve(s) closed above stating "RCS pressure is required to be greater than VCT pressure prior to opening." CT # __________ ____

8.2.73 Slowly adjust PC-145, PRESSURE, controller to 0.0. ____

8.2.74 Notify E&C to add the proper chemicals to the S/Gs to ensure Wet Layup Chemistry specifications are met.

__________________________ ____
E&C Contact (Print name)

8.2.75 **IF** the plant will be in MODE 5 greater than 48 hours **AND** the OST-702 series of OST's has not been completed within 92 days, **THEN** initiate the OST-702 series (refer to Step 8.1.5). ____

8.2.76 **IF** the plant will be in MODE 5 greater than 48 hours **AND** the OST-703 series of OST's has not been completed within 92 days, **THEN** initiate the OST-703 series (refer to Step 8.1.5). ____

INIT

8.2.77 **IF** the plant will be in MODE 5 greater than 24 hours **AND** OST-687 has not been completed within 31 days, **THEN** initiate OST-687 (refer to Step 8.1.5). ____

NOTE:	I&C personnel are required to perform actions in SPP-011.

8.2.78 Defeat Safeguards IAW SPP-011. Train "A" Defeated ____

Train "B" Defeated ____

8.2.79 **IF** purging the CV is desired, **THEN** initiate the appropriate section (normal or refueling) of OP-921. ____

NOTE:	If S/G Sludge Lancing will be performed, RCS temperature should be less than 100°F to ensure the operating limit of the sludge lance cameras will not be approached.

8.2.80 **WHEN** the RCS has been cooled to the desired temperature, **THEN** perform the following:

1. **D**iscontinue recording RCS cooldown data on Attachment 10.1. ____

2. Notify E&C to discontinue obtaining hourly RCS Boron samples.

   ______________________________ ____
   E&C Contact (Print name)

8.2.81 **WHEN** no longer needed to sustain maximum purification flow, **THEN** CLOSE CVC-309E, PCV-145 BYPASS. ____

8.2.82 Data on check valve testing of CVC-313 and CVC-351 has been reviewed.

______________________________ __________ ____
IST Review (Signature) Date

8.2.83 Verify HVH-9A and HVH-9B, CONCRETE SHIELD COOL FANS, shutdown IAW OP-921, Containment Air Handling. ____

INIT

8.2.84 Place Caution Caps on the controls for the CONCRETE SHIELD COOL FANS HVH-9A and HVH-9B, stating "Do not start the fan with personnel in the refueling cavity due to radiological concerns" (CR 96-02623).

CT # ______________ ____

8.2.85 Verify the keylock switch TURBINE BLDG SW ISOLATION DEFEAT V6-16C is in the INHIBIT position. ____

END CRITICAL STEPS

8.2.86 Perform the following:

1. Record the Reactor Cooldown Number from the Reactor Startups/Shutdowns and Trips Log book. _________ ____

2. **IF** this is cooldown number 140 or greater, **THEN** notify RESS to perform an evaluation of the transient history and design basis IAW PLP-109. (CR 98-01388) ____

3. Record the date. _________ ____

30. 004 K3.08 001

Given the following plant conditions:

- The plant is operating at 100% RTP.
- "C" Charging Pump is currently running in Manual at minimum speed.
- "A" Charging Pump is running in MANUAL at 35% demand during the performance of OP-301-1, Section 8.4.7, Charging Pump Break-In After Maintenance.
- CVC-283C, Charging Pump "A" Discharge Relief, has lifted and has not reseated.

Which ONE (1) of the following identifies the impact on Seal Injection flow(s) AND how would this malfunction be addressed by AOP-018, Reactor Coolant Pump Abnormal Conditions?

Seal Injection flows will lower to ___(1)___ and AOP-018 directs the operators to ___(2)___.

A.✓ (1) ZERO flow

(2) Isolate letdown, secure all Charging Pumps, manually isolate "A" Charging pump.

B. (1) MINIMUM flow (> 6 gpm)

(2) Isolate letdown, secure all Charging Pumps, manually isolate "A" Charging pump.

C. (1) ZERO flow

(2) Stop "A" Charging Pump and adjust the speed of "C" Charging Pump to restore normal seal injection flows.

D. (1) MINIMUM flow (> 6 gpm)

(2) Stop "A" Charging Pump and adjust the speed of "C" Charging Pump to restore normal seal injection flows.

The correct answer is A

A. Correct. The seal injection flows will lower to zero due to all charging flow being diverted through CVC-283C to the drain header. AOP-018 will direct the Operators to isolate letdown, secure all charging pumps, manually isolate "A" Charging pump. Ultimately the Operators would start an available Charging Pump, restore letdown and establish normal seal injection.

B. Incorrect. See above discussion. Plausible if the candidate incorrectly thinks that CVC-283C lifting will only affect "A" Charging Pump discharge flow or if the candidates incorrectly thinks that a check valve would prevent flow from "C" Charging Pump to CVC-283C.

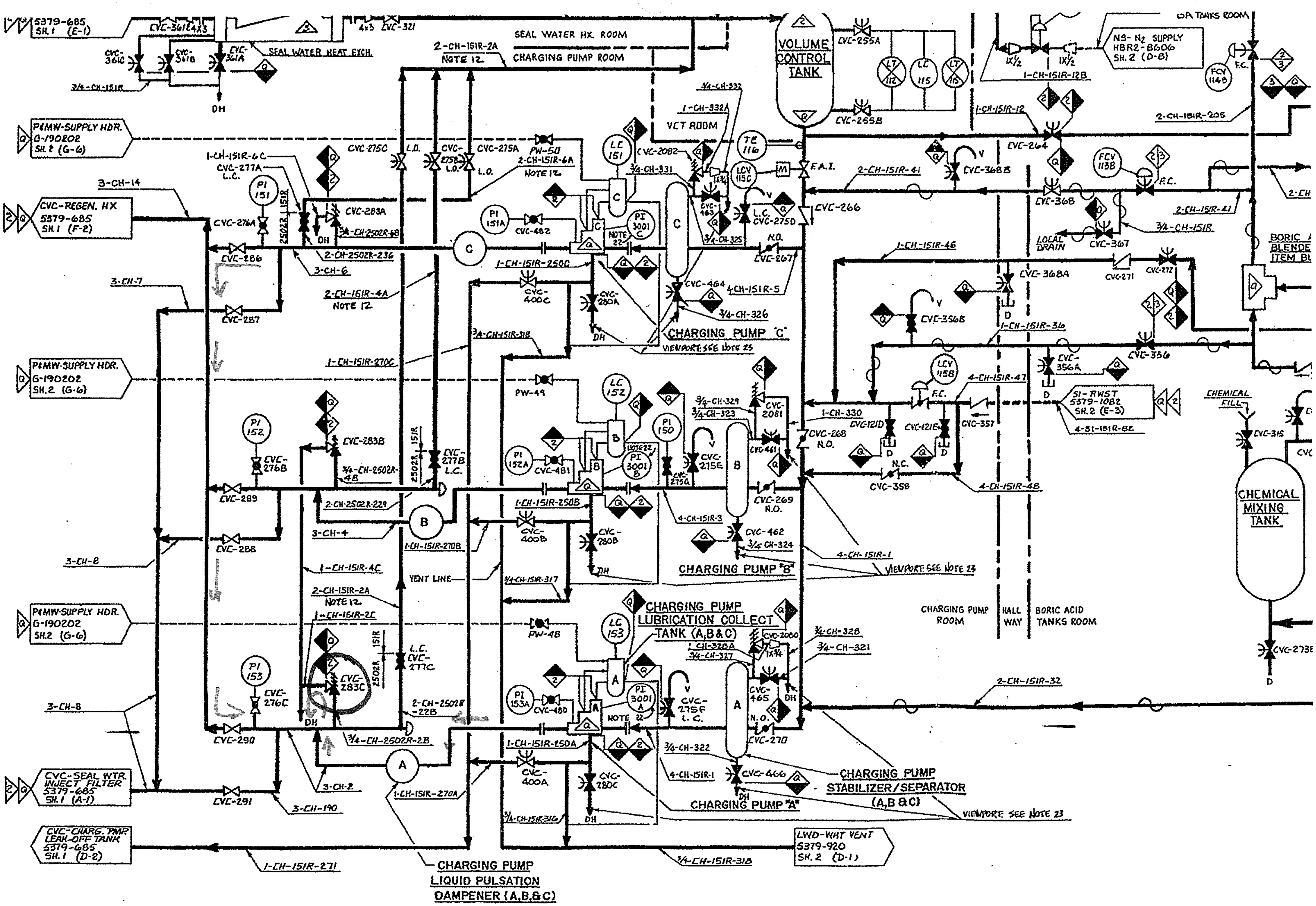
C. Incorrect. First part of distractor is correct. The second half is incorrect but plausible if the candidate incorrectly thinks that CVC-283C lifting will only affect flow from "A" and "C" Charging pump as long as "A" Charging Pump is running and a check valve is in the "A" Charging pump discharge line.

D. Incorrect. Plausible if the candidate incorrectly thinks that CVC-283C lifting will only effect "A" Charging Pump discharge flow or if the candidate incorrectly thinks that a check valve would prevent flow from "C" Charging Pump to CVC-283C.

Question 30
Tier 2 / Group 1
K/A Importance Rating - RO 3.6 SRO 3.8

Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: RCP seal injection

Reference(s) - Sim/Plant design, System Description, AOP-018, Drawing 5379-00685, Sheet 2
Proposed References to be provided to applicants during examination - None
Learning Objective - CVCS 009
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.6
Comments - K/A match because candidate must understand how a charging pump discharge relief valve failure will impact RCP seal injection.



VOLUME CONTROL TANK
CHEMICAL MIXING TANK
SEAL WATER HX. ROOM
CHARGING PUMP ROOM
SEAL WATER HEAT EXCH.
VCT ROOM
DA TANKS ROOM
N3-N2 SUPPLY HBR2-8606 SH. 2 (D-8)
P&MW-SUPPLY HDR. G-190202 SH. 2 (G-6)
CVC-REGEN. HX 5379-685 SH. 1 (F-2)
CVC-SEAL WTR. INJECT FILTER 5379-685 SH. 1 (A-1)
CVC-CHARG. PMP. LEAK-OFF TANK 5379-685 SH. 1 (D-2)
SI-RWST 5379-1082 SH. 2 (E-3)
LWD-WHT VENT 5379-920 SH. 2 (D-1)
5379-685 SH. 1 (E-1)
CHARGING PUMP "C"
CHARGING PUMP "B"
CHARGING PUMP "A"
CHARGING PUMP LUBRICATION COLLECT TANK (A,B&C)
CHARGING PUMP STABILIZER/SEPARATOR (A,B&C)
CHARGING PUMP LIQUID PULSATION DAMPENER (A,B,&C)
VIEWPORT: SEE NOTE 23
CHARGING PUMP ROOM
HALL WAY
BORIC ACID TANKS ROOM
BORIC ACID BLENDER ITEM B
CHEMICAL FILL
LOCAL DRAIN
VENT LINE
NOTE 12
NOTE 22

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-018

REACTOR COOLANT PUMP ABNORMAL CONDITIONS

REVISION 20

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides instructions to protect the Reactor Coolant Pumps after a component failure or support system malfunction.

2. ENTRY CONDITIONS

This procedure is entered for any of the following events:

- Loss of Seal Injection flow to any RCP
- High Vibration on an RCP during steady state conditions (Confirmed valid IAW APP-001-B5)
- Indication of any RCP Seal malfunction
  - RCP #1 Seal leakoff flow greater than 5 gpm
  - RCP #1 Seal leakoff flow less than 1 gpm
  - RCP #2 Seal leakoff flow greater than 0.5 gpm
  - Indication of RCP #3 Seal malfunction (repeat low standpipe level at normal pressure)
- As directed by AOP-016, Excessive Primary Plant Leakage

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Make PA Announcement For Procedure Entry.	

NOTE

- The RCP malfunctions in the Table below are listed in order of priority.
- This procedure is NOT applicable during implementation of EPP-1, Loss Of All AC OR any of its recovery procedures.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Evaluate Plant Conditions AND Go To The Appropriate Section For RCP Malfunction Not Yet Addressed:	Return to procedure and step in affect.

MALFUNCTION	SECTION
Reactor Coolant Pump Seal Failure	Section A
High Reactor Coolant Pump Vibration	Section B
Loss of Seal Injection	Section C

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 1 of 12)

* 1.	Check APP-001-D1, RCP THERM BAR COOL WTR LO FLOW alarm - ILLUMINATED	<u>IF</u> APP-001-D1 ILLUMINATES, <u>THEN</u> observe the <u>CAUTION</u> prior to Step 2 and Go To Step 2. Observe the <u>NOTE</u> prior to Step 11 and Go To Step 11.

****

<u>CAUTION</u>

<u>IF</u> more than 15 minutes elapses without RCP Seal Cooling, <u>THEN</u> Seal Cooling (CCW <u>AND</u> Seal Injection) MUST be isolated before starting CCW <u>OR</u> Charging or Seal Damage could occur.

****

* 2.	Check Elapsed Time Since All RCP Seal Cooling Was Lost - GREATER THAN 15 MINUTES	<u>IF</u> RCP Seal Cooling is <u>NOT</u> <u>OR</u> can <u>NOT</u> be restored in less than 15 minutes, <u>THEN</u> Go To Step 3. Go To Step 10.
3.	Check Plant Status - MODE 1 <u>OR</u> MODE 2	Stop the affected RCP(s) Go To Step 5.
4.	Perform The Following: a. Trip the reactor b. Trip the affected RCP(s) c. Go To Path-1 while continuing with this procedure	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 4 of 12)

10. Verify ALL Of The Following Components Needed For Thermal Barrier Cooling:

    - At least ONE CCW Pump is running.
    - CC-716A, CCW TO RCP ISO is open
    - CC-716B, CCW TO RCP ISO is open
    - FCV-626, THERM BARRIER OUTLET is open
    - CC-735, THERM BAR OUT ISO is open

NOTE

- A rupture is a leak of sufficient magnitude to require stopping the Charging Pumps or reduces Charging Pump Discharge Pressure to less than RCS Pressure.
- Charging System piping is any piping where a leak prevents the Charging Pumps from delivering flow to the Charging Line OR Seal Injection Line.

11. Determine If A Charging Pump Can Be Started:

    a. Check Charging System Piping - RUPTURED Y — RNO: a. Go To Step 12.

    b. Go To Step 17

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 6 of 12)

NOTE

- A rupture is a leak of sufficient magnitude to require stopping the Charging Pumps or reduces Charging Pump Discharge Pressure to less than RCS Pressure.
- Charging System piping is any piping where a leak prevents the Charging Pumps from delivering flow to the Charging Line OR Seal Injection Line.

16.	Check Charging System Piping - RUPTURED	Go To Step 39.
* 17.	Verify LCV-460 A&B, LTDN LINE STOP Valves - CLOSED	
* 18.	Stop The Running Charging Pump(s)	
19.	Momentarily Place RCS MAKEUP SYSTEM Switch To STOP	
* 20.	Dispatch An Operator To Locate And Isolate The Charging System Leak	

NOTE

"Charging Line" in the subsequent step is defined as the Charging Line downstream of HCV-121, CHARGING FLOW valve.

*21.	Check Location Of Break - IN CHARGING LINE	IF break is found in the Charging Line, THEN Go To Step 22. Go To Step 28.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 8 of 12)

26. Momentarily Place RCS MAKEUP SYSTEM Switch To START

27. Go To Step 41

28. Perform The Following:

a. Check leak location - IDENTIFIED

RESPONSE NOT OBTAINED:

a. Go to ONE of the below steps:

- IF the leak can NOT be readily identified, THEN assume the leak is unisolable AND Go To Step 28.b

OR

- IF the leak is identified downstream of HCV-121, CHARGING FLOW, THEN Go To Step 22.

OR

- IF the leak is identified anywhere but downstream of HCV-121, CHARGIBNG FLOW, THEN Go To Step 28.b.

b. Check Charging System Leak - ISOLABLE

RESPONSE NOT OBTAINED:

b. Perform the following:

1) Notify Operations Staff and Engineering to obtain further guidance.

2) Implement ITS 3.4.17.

3) Go To Step 44.

29. Check Charging System Leak - ISOLATED

RESPONSE NOT OBTAINED:

When the charging system leak is isolated, THEN Observe the NOTE prior to Step 30 and Go To Step 30.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 9 of 12)

NOTE

The Charging Line OR Seal Injection Lines are acceptable flow paths to the RCS.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
30.	Check Available Flow Path - WILL SUPPORT FLOW TO RCS ONCE Isolated.	Perform the following: a. Notify Operations Staff and Engineering to obtain further guidance. b. Implement ITS 3.4.17. c. Go To Step 44.
31.	Verify Flow Path - ALIGNED TO RCS	
32.	Momentarily Place RCS MAKEUP SYSTEM Switch To START	
33.	Start An Available Charging Pump	
34.	Go To Step 41	
35.	Check APP-001-A2, SEAL WTR INJ FILTER HI ΔP Alarm - ILLUMINATED	Go To Step 39.
36.	Shift Seal Injection Filters Using OP-301, Chemical And Volume Control System (CVCS)	
37.	Check Alternate Seal Injection Filter - IN SERVICE	WHEN alternate Seal Injection filter is placed in service, THEN Go To Step 41.
38.	Go To Step 41	

31. 005 K3.07 001

Given the following plant conditions:

- The plant is in Mode 6.
- RHR Train "A" is in service.
- FCV-605, RHR HX BYPASS, is set to maintain 3400 GPM.
- HCV-758, RHR HX OUTLET FLOW TO COLD LEGS, demand position set at 30%.
- The Instrument Air supply line to RHR Heat Exchanger Flow Control Valve HCV-758 becomes severed and is completely detached.
- No other air operated valves are impacted by the failure.

Which ONE (1) of the following identifies the changes in TR-604, PEN 2 RHR HX OUTLET TEMP. from the initial steady state conditions AND the RCS temperature at which refueling operations may continue IAW GP-010, Refueling?

	RHR HX OUTLET TEMP.	RCS Temp for Refueling
A.✓	Higher	135°F
B.	Higher	150°F
C.	Lower	135°F
D.	Lower	150°F

A: Correct. Total RHR flow is controlled by FCV-605, RHR HX Bypass, so it will remain constant. HCV-758 fails closed, so there is more bypass flow mixing with less HX flow, resulting in a higher temperature on the HX outlet. Per GP-008, RCS must be maintained equal to or less than 140°F to perform refueling operations. Also, TRM 1.1, Definitions, states that a MODE shall be as required by Technical Specifications, with the exception that additionally during refueling operations, Tave shall be less than or equal to 140°F.

B: Incorrect. Total RHR flow is controlled by FCV-605 and would remain at the last setpoint of approximately 3400 GPM, thus total flow would remain constant. Per GP-008, RCS must be maintained equal to or less than 140°F to perform refueling operations. During a discharge of a full core into the SFP, the temperature of the SFP water shall be maintained at or below 150°F per GP-010, Refueling and TRM 3.14. This is the basis for the use of this value as a distractor.

C: Incorrect. HCV-758 failing closed will result in no cooling flow through the RHR HX and the HX outlet temperature will rise along with the total RHR remaining constant due to the operation of FCV-605. Per GP-008, RCS must be maintained equal to or less than 140°F to perform refueling operations.

D: Incorrect. HCV-758 failing closed will result in no cooling through the RHR HX and the HX outlet temperature will rise. Per GP-008, RCS must be maintained equal to or less than 140°F to perform refueling operations. During a discharge of a full core into the SFP, the temperature of the SFP water shall be maintained at or below 150°F per GP-010, Refueling and TRM 3.14. This is the basis for the use of this value as a distractor.

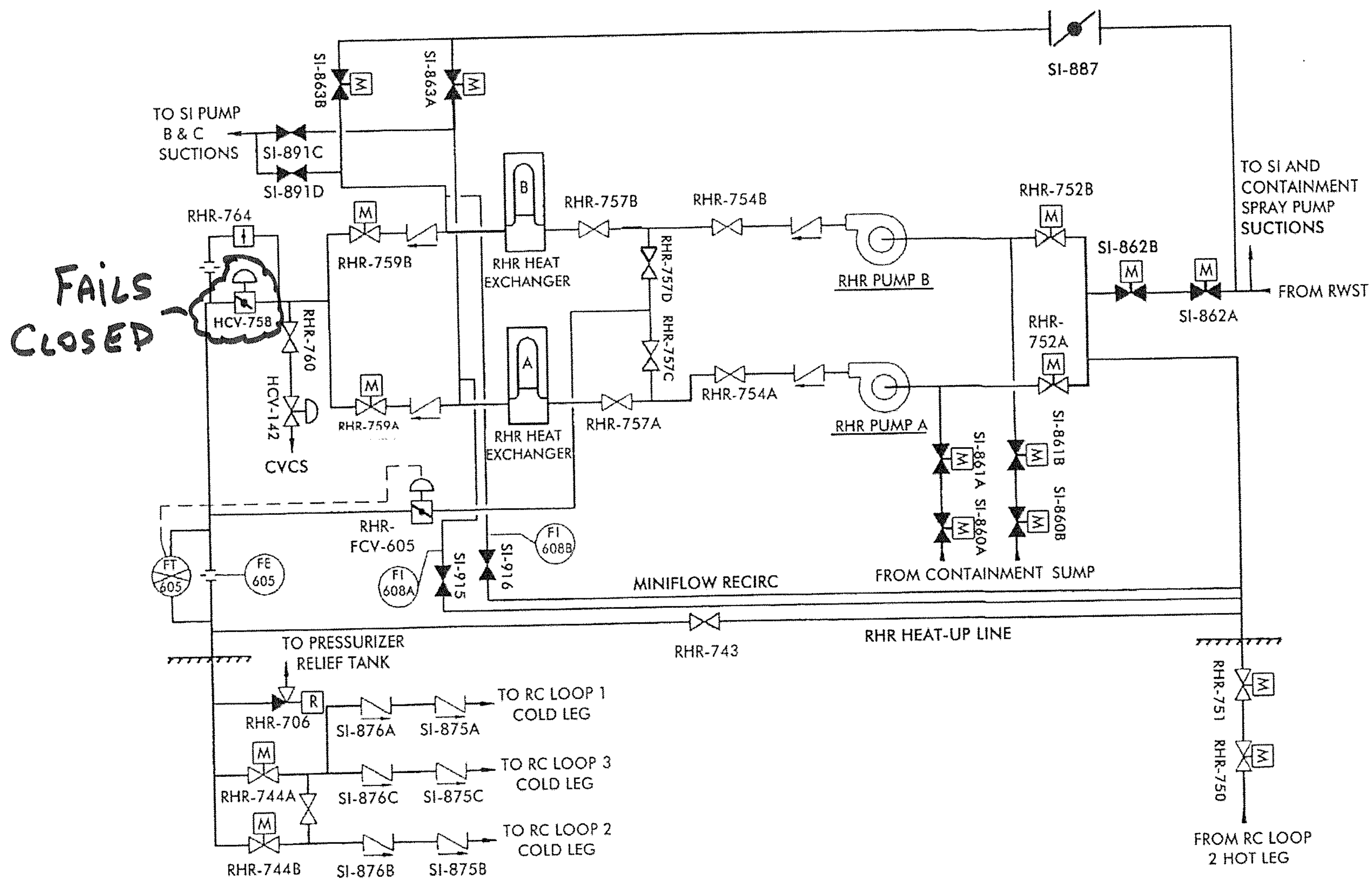
Question 31
Tier 2 / Group 1
K/A Importance Rating - RO 3.2 SRO 3.6

Knowledge of the effect of a loss or malfunction of the RHR System will have on the following: Refueling Operations

Reference(s) - Sim/Plant design, System Description, AOP-020, GP-008, TRM, GP-010
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-020 LP AOP-020-004 and RHR 009
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.6
Comments - K/A met because the candidate must analyze the effect of an instrument air line failure associated with the RHR System and how it impacts refueling operations.

RHR SYSTEM-CORE COOLING LINEUP

RHR-FIGURE-1



rhrf01

INFORMATION USE ONLY

5.2 RHR-750 and RHR-751, RHR Pump Suction from Loop 2 Hot Leg

RHR-750 and RHR-751 are interlocked with RCS narrow range pressure (PC-403). The valves can only be opened if RCS pressure is less than 445 psig. This protects against over pressure downstream of the RHR pumps and still allows RHR pump operation before securing the RCPs.

NOTE: The design RCS pressure for this interlock IAW OST-257 is ≤ 474 psig. This does not consider instrument uncertainties. The pressure setpoint of 445 psig will include the uncertainties.

RHR-750 and RHR-751 are also interlocked with SI-862 A & B and SI-863 A & B. RHR-750 and 751 will open from the RTGB only if SI-862 A & B and SI-863 A & B are closed, their breakers are closed with power available and the Normal/Defeat switches in the back of the RTGB are in Normal. If 862A/B or 863A/B are deenergized, RHR-750/751 will see these valves as being open so RHR 750/751 will not open. There is no interlock to close RHR-750/751 so under these conditions, the valves will close, but not reopen. This interlock will help to avoid depressurizing the RCS to RWST and/or over pressurizing the low pressure portions of SI system.

RHR suction from RCS Loop 2 Hot Leg passes through RHR-750, RHR-751 and CV penetration # 16. CV isolation for this penetration is said to exist when the RHR system is isolated and RHR-751 is closed.

5.3 RHR LOOP ISOL SI-862 A & B, RWST to RHR Pump Suction Isolation

Two motor operated valves are provided to isolate the RHR pump suction from the RWST. When lining up for the injection phase, they will be open. In the recirculation phase, they will be closed prior to taking suction on the CV floor. They are also closed when the RCS is being cooled by the RHR System. To prevent over pressurization of the RWST and other related low pressure piping and to prevent depressurizing the RCS to RWST, these valves are interlocked so they can't be opened unless the RHR System is less than 210 psig (862A and 863A -PC-601A, 862B and 863B -PC-600B). Keyed switches located behind the RTGB remove the control power from these valves during normal operation.

5.4 RHR-FCV-605, RHR HX Bypass Flow

FCV-605 will automatically maintain a preset flowrate through the operating RHR loop (set by operator). It is an air operated, fail closed valve. If FCV-605 did fail closed, all the flow would be directed through the RHR heat exchanger. This may result in Cooldown rate being higher than desired until valve control was obtained. This problem is addressed in AOP-020, Loss of RHR Cooling.

FCV-605 works in conjunction with hand control valve RHR-HCV-758 and FT-605. HCV-758 is adjusted to increase or decrease flow through the RHR Heat Exchangers to change the Heat up or Cooldown rate. This causes total system flow to be effected and is sensed by FT-605. The flow loop circuitry provides a control signal to FCV-605 which maintains a constant total system flow.

At power, Instrument Air is isolated to FCV-605 (Required by Tech. Specs. when >1000 psig). A portable skid mounted controller is available for use during Post Fire Repairs if FCV-605 control circuits are damaged or inoperable. These procedures would also line up to use the Nitrogen system for motive force and for valve control.

5.5 RHR-HCV-758, RHR HX Discharge Flow

HCV-758 is throttled from RTGB to control Cooldown or Heat up rate by controlling RHR flow through the heat exchanger. It is an air operated valve that fails closed.

At power, Instrument Air is isolated to HCV-758 (Required by Tech. Specs. when >1000 psig). A portable skid mounted controller is available for use during Post Fire Repairs if HCV-758 control circuits are damaged or inoperable. These procedures also allow the use of Nitrogen as a backup for motive force and for valve control.

5.6 RHR-744 A & B, RHR to RCS Cold Legs
These motor operated valves in parallel provide a redundant path to the RCS during RHR operation and safeguard actuation. The valves open automatically on a SI signal.

RHR discharge to the RCS Cold legs passes through CV penetration #17. CV isolation for this penetration is said to exist when the RHR system is isolated (outside containment) and RHR-744 A & B, RHR to RCS Cold Legs and RHR-706, RHR Relief valve are closed (inside containment).

5.7 SI-860A, 861A, 860B and 861B, CV Sump Recirc Suction

There are two suction lines which lead from the CV floor to the RHR Pumps. Each line is isolated by two of the above motor-operated valves in series. These lines are filtered by screens and have a baffle to prevent floating debris from entering the suction line.

5.8 RHR LOOP RECIRC SI-863A and 863B, RHR Loop Recirculation

These motor operated valves provide a parallel flow path back to the RWST outlet header. Two redundant manual valves (SI-891C and 891D) provide an alternate flow path to the SI Pump suction. SI-863 A & B are used to initially pump the Refueling

TRMS 1.1 Definitions (continued)

MODE (CTS 3.8.1.e)	A MODE shall be as required by Technical Specifications, with the exception that additionally during refueling operations T_{ave} shall be ≤ 140°F.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PROCESS CONTROL PROGRAM (PCP)	The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing disposal of the radioactive waste.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2339 Mwt.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

5.9 Only active SROs can supervise refueling activities. Non-active SROs can become active for SRO duties limited to fuel handling by completing one 8 hour shift under instruction of a licensed active SRO performing refueling activities. (NUREG-1262)

5.10 The Manipulator Crane Operator shall **NOT** unlatch from Fuel Assemblies which have been installed in the Core until directed to do so by the Refueling SRO.

5.11 Access to the Refueling work stations shall be limited to members of the Refueling team, and observers approved by the SM.

5.12 During the discharge of a full core into the SFP, the temperature of the SFP water shall be maintained at or below 150°F. The SFP water temperature shall be monitored once each shift when the temperature is at or below 125°F. If the temperature exceeds 125°F, it shall be monitored hourly. If the SFP temperature reaches 150°F, Fuel Assemblies will be transferred back to the CV to reduce the pool temperature below 150°F. (TRMS 3.14)

5.13 The SFP water temperature **SHALL NOT** be allowed to decrease below 68°F. The SFP structure analysis temperature limits, described in UFSAR Section 3.8 and 9.1, requires the temperature to be maintained greater than 68°F. SER 90-17 discussed potential SFP criticality impacts of failure to maintain temperature within limits, but RNP does **NOT** have temperature restrictions due to criticality assumptions used in EMF-94-113.

5.14 During Refueling operations, RCS temperature shall be maintained less than or equal to 140°F. (TRM 1.1)

5.15 Shutdown Margin in the core is analyzed over a range of 38°F to 140°F for Refueling and 38°F to 200°F for Mode 5. The lower temperature is usually the most limiting in the core. Shutdown Margin in the SFP is analyzed over a range of 68°F to 171°F. The higher temperature is usually the most limiting in the SFP.

5.16 Core Alteration is defined in ITS Section 1.0 as movement of any fuel, sources or reactivity control components, within the Reactor Vessel with the vessel head removed and fuel in the vessel. MODE 6 is defined in ITS Section 1.0 to be any time one or more Reactor Vessel Head closure bolts are less than full tensioned.

5.17 Due to the possibility of radiation streaming through handling tools with hollow sections, it is management expectation that they shall **NOT** be used unless they possess flood and drain holes. (NRC Info Notice 90-33) SA# 257269

MODE 6 CHECKLIST

INIT

NOTE:	ITS identifies RHR Pump requirements based on level above the Reactor Vessel flange. Maintaining Refueling Cavity Water level above 29 inches below the operating deck ensures level is at least 6 inches above ITS required level of 23 feet above the Reactor Vessel flange.

9. **IF** Refueling Cavity Water level is above 29 inches below the operating deck as indicated on the Refueling Cavity Level Indicator, **THEN** one RHR Train is operable **AND** in operation. (ITS LCO 3.9.4) ____

10. **IF** Refueling Cavity Water level is below 29 inches below the operating deck as indicated on the Refueling Cavity Level Indicator, **THEN** two RHR Trains are operable **AND** one RHR Train is in operation. (ITS LCO 3.9.5) ____

11. RCS Temperature is less than or equal to 140°F. (TRM 1.1) _____°F ____

12. **PERFORM** MODE 6 required SRs identified in:
    - OST-020 ____
    - OST-020-1 ____
    - OST-021 ____
    - OST-022 ____
    - OST-023 ____

	Initials	Name (Print)	Date
Performed By:	_____	______________________________	____________
	_____	______________________________	____________
	_____	______________________________	____________

Reviewed By: ______________________________ ____________
STA or CRSS Date

Approved By: ______________________________ ____________
Shift Manager Date

PERIODIC CHECK SHEET

This revision has been verified to be the latest revision available. __________ ____
Date INIT

	07-19	19-07
1. Verify the following RHR train status: (ITS LCO 3.4.8)		
– Two trains are operable	_____	_____
– One train in operation	_____	_____

PERIODIC CHECK SHEET

07-19 19-07

NOTE: AVAILABLE is defined as [SOER 09-1, REC 4]The status of a system, structure, or component (SSC) that is in service or that can be placed into service in a functional or operable state by immediate manual or automatic action. This action may be performed locally or remotely.

Equipment that is out of service (e.g. tagged out) for corrective or preventative maintenance is considered unavailable.

SSC's out of service for testing are typically considered unavailable, unless the test configuration is automatically overridden by a valid starting signal, or the function can be promptly restored either by an operator in the control room or by a dedicated operator stationed locally for that purpose.

Restoration actions must be contained in an approved work document or written procedure, must be uncomplicated (a single action or a few simple actions), and must not require diagnosis or repair. Credit for local actions can be taken only if the operator is positioned at the proper location throughout the duration of the test for the purpose of restoration of the train should a valid demand occur.

The following clarifications apply to available equipment/systems:

Automatic features may be defeated if not required in the present mode or condition.

Components such as instrumentation may be out of service provided the SSCs ability to perform its function is not degraded. Instrumentation to adequately monitor SSC performance must function.

MOVs may be deactivated in positions that support the shutdown function of the SSC.

The following equipment status verifications are performed to satisfy the requirements for Outage Risk and are **NOT** meant to replace ITS/TRM requirements. Any equipment which is required to be OPERABLE to meet ITS/TRM LCO requirements shall be operable or applicable REQUIRED ACTIONS shall be entered.

2. Verify the following equipment is AVAILABLE:

- 1 SW pump with an Emergency Power Supply

"A" / "B" / "C" / "D" ____
(Circle at least one)

"A" / "B" / "C" / "D" ____
(Circle at least one)

PERIODIC CHECK SHEET

Step 2. (Continued)		07-19	19-07
– 1 additional SW pump	"A" / "B" / "C" / "D" (Circle at least one)	_____	
	"A" / "B" / "C" / "D" (Circle at least one)		_____
1 SWBP	A" / "B" (Circle one)	_____	
	A" / "B" (Circle one)		_____
– 1 Service Water Header	North / South (Circle at least one)	_____	
	North / South (Circle at least one)		_____
– 1 CCW Heat Exchanger	"A" / "B" (Circle at least one)	_____	
	"A" / "B" (Circle at least one)		_____

PERIODIC CHECK SHEET

Step 2. (Continued)		07-19	19-07
– 1 CCW Pump with an available Emergency Power Supply	"A" / "B" / "C" (Circle one)	____	
	"A" / "B" / "C" (Circle one)		____
– 1 additional CCW Pump	"A" / "B" / "C" (Circle one)	____	
	"A" / "B" / "C" (Circle one)		____
– CV Sump Recirculation flow path to RHR Pump "A"		____	____
– CV Sump Recirculation flow path to RHR Pump "B"		____	____
– RWST with 300,000 gallons (91%) of $\geq$1950 ppm borated water		____	____
– 1 Boric Acid Storage Tank with at least 3080 gallons (42%) of Boric Acid between 20,000 ppm to 22,500 ppm Boron solution at a temperature of at least 145°F	"A" / "B" (Circle one)	____	
	"A" / "B" (Circle one)		____
– 1 SI Pump (breaker racked in)	"A" / "B" / "C" (Circle one)	____	
	"A" / "B" / "C" (Circle one)		____

PERIODIC CHECK SHEET

Step 2. (Continued) 07-19 19-07

– 1 SI injection path from the RWST

Hot Leg / Cold Leg ______
(Circle one)

Hot Leg / Cold Leg ______
(Circle one)

– 1 Charging Pump "A" / "B" / "C" ______
(Circle one)

"A" / "B" / "C" ______
(Circle one)

– 1 Charging flowpath from the BAST ______ ______

– 1 Borated Makeup source and delivery flow path components with Normal and Emergency electrical power sources available

SI-RWST / Charging-BAST ______
(Circle one)

SI-RWST / Charging-BAST ______
(Circle one)

NOTE: The secondary side of the S/G, in Containment, is **NOT** required to be intact for this requirement.

– **IF** the RCS loops are to remain intact, **THEN** 1 S/G shall be greater than or equal to 16% narrow range ______ ______

– 2 independent RCS standpipe level channels ______ ______

– 1 HVH Containment Fan Cooler HVH 1 / 2 / 3 / 4 ______
(Circle at least one)

HVH 1 / 2 / 3 / 4 ______
(Circle at least one)

PERIODIC CHECK SHEET

Step 2. (Continued) 07-19 19-07

– Both EDG "A" **AND** EDG "B" ______ ______

– Both E-1 **AND** E-2 Buses ______ ______

> **NOTE:** A DC Power train, for the purposes of Outage Risk, consists of the battery, one of two battery chargers for the battery, associated DC Buses, and AC Instrument Buses powered via inverters from the respective DC Buses.

– DC Power Train "A"

Battery "A" and Bus ______

Battery Charger "A" / "A-1" (Circle one) ______

IB 2 on Normal Power Supply ______

Battery "A" and Bus ______

Battery Charger "A" / "A-1" (Circle one) ______

IB 2 on Normal Power Supply ______

– DC Power Train "B"

Battery "B" and Bus ______

Battery Charger "B" / "B-1" (Circle one) ______

IB 3 on Normal Power Supply ______

Battery "B" and Bus ______

Battery Charger "B" / "B-1" (Circle one) ______

IB 3 on Normal Power Supply ______

PERIODIC CHECK SHEET

07-19 19-07

NOTE: The T/Cs may be read locally in the Rod Control Room IAW Attachment 10.4, Core Exit Thermocouples, if the ICCM panel is out of service.

3. Verify the following:

	07-19	19-07
– Startup Transformer is in service.	____	____
– Maintenance activities on the Power lines and Transformers which provide sole offsite power to Unit 2 are **NOT** in progress **OR** planned to occur.	____	____
– One Core Exit T/C per quadrant from ICCM "A" operable.	____	____
– One Core Exit T/C per quadrant from ICCM "B" operable.	____	____
– ERFIS Quality Codes are "OK" and temperatures are in agreement for the following:		
– RXT0001, AVG OF 5 HOTTEST THERMOCOUPLES	____	____
– RXT0050, RCS MIDLOOP TEMPERATURE TRAIN "A"	____	____
– RXT0051, RCS MIDLOOP TEMPERATURE TRAIN "B"	____	____
– 1 source range channel operable.	____	____
– CV evacuation alarm tested and audible in all areas.	____	____
– 1 CV radiation monitor operable (R-2, R-11, or R-12).	____	____
– RCS side S/G manways status (specify open or closed).		
HOT LEG A: ______ B: ______ C: ______	____	
COLD LEG A: ______ B: ______ C: ______	____	
HOT LEG A: ______ B: ______ C: ______		____
COLD LEG A: ______ B: ______ C: ______		____

PERIODIC CHECK SHEET

		07-19	19-07
4.	All active CV Closure Exception Permits (OMM-033) have been reviewed **AND** all required tools and materials specified have been inspected to verify that they are available at the respective penetrations.	_____	_____
5.	RCS temperatures maintained less than or equal to 140°F.	_____	_____

LOG EVERY SHIFT			
SHIFT	RHR LOOP BORON	RWST BORON	RWST LEVEL
07-1900			
19-0700			

	Initials	Name (Print)	Date
Performed By:	________	______________________________	__________
	________	______________________________	__________
	________	______________________________	__________
	________	______________________________	__________
	________	______________________________	__________
	________	______________________________	__________
	________	______________________________	__________

Reviewed By:

07-1900 ______________________________ ______________

19-0700 ______________________________ ______________

Shift Technical Advisor Date

Approved By: ______________________________ ______________

Shift Manager Date

3.0 **RESPONSIBILITIES**

3.1 Operations is responsible for the performance, review and approval of this procedure.

4.0 **PREREQUISITES**

4.1 The RHR System is providing core cooling **AND** RCS temperature is less than or equal to 140°F.

4.2 The reactor vessel head has been removed **AND** it is desired to flood the refueling cavity to support refueling.

5.0 **PRECAUTIONS AND LIMITATIONS**

5.1 When the Refueling Cavity water level is below the Refueling Cavity floor drain, WD-1716, REACTOR COOLANT DRAIN TANK VENT, **OR** WD-1711, REFUELING CANAL DRAIN TO RCDT, shall be CLOSED. Otherwise, the vent header could release to the CV via the RCDT.

5.2 The following precautions apply when using the SI Pumps to slow fill the Refueling Cavity:

5.2.1 SI Pump operation in the recirculation mode for more than 30 minutes shall be avoided.

5.2.2 Running pumps should be visually inspected every hour for proper operation IAW Attachment 10.2, SI Pump Inspection Sheet.

5.2.3 Only one Hot Leg flow path, SI-866A **OR** SI-866B, shall be used when operating one SI Pump.

5.2.4 SI Pump breakers shall remain racked out **OR** Control power fuses should remain removed until directed otherwise by this procedure

5.2.5 Starting limitations shall be followed IAW OP-202.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 1 of 20)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Implement the EALs	
2.	Check CV Closure Status - PENETRATIONS OPEN	Go To Step 8.
3.	Check Refueling Cavity Level - 29 INCHES OR GREATER BELOW THE OPERATING DECK	Go To Step 8.
4.	Initiate CV Closure Using OMM-033, CV Closure	
5.	Check SI Pumps - ONE SI PUMP AVAILABLE TO START FROM RTGB	Dispatch an operator to the E-1/E-2 Room to prepare to verify the breaker Racked In AND Fuses Installed for ONE SI Pump when notified by the Control Room.
* 6.	Check Core Exit T/Cs - LESS THAN 200°F	Verify ONE SI Pump breaker is Racked In AND Fuses Installed. Go To Step 44.
* 7.	Check Core Exit T/Cs - LESS THAN 175°F	Verify ONE SI Pump breaker is Racked In AND Fuses Installed while continuing with step in effect.
8.	Check Reason For Entry: • LOW FLOW OR • RHR PUMP TRIP	Go To Step 19.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 7 of 20)

NOTE

RHR Pump Starting Duty requirements are contained in ATTACHMENT 7, RHR PUMP STARTING DUTY REQUIREMENTS.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
18.	Maintain RCS Temperature as follows:	
	a. Check RCS Temperature - INCREASING	a. Perform the following: 1) IF temperature is decreasing, THEN throttle RHR-764 to stabilize temperature. 2) IF RHR flow can NOT be maintained greater than 500 gpm, THEN control RCS temperature by starting and stopping the RHR pump at desired intervals. 3) Return to procedure and step in effect.
	b. Maintain RCS Temperature using RHR-764, HIC-758 BYPASS.	
	c. Return to procedure and step in effect	
19.	Check RCS Temperature - INCREASING	Maintain RCS Temperature using HIC-758, RHR HX DISCH FLOW. Return to procedure and step in effect.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 8 of 20)

20.	Adjust HIC-758, RHR HX DISCH FLOW, To Obtain Desired Cooling FAILED CLOSED	Perform the following: a. Throttle open RHR-764, HCV-758 BYPASS to obtain desired cooling. b. IF Instrument Air has been lost to HCV-758, THEN evaluate supplying temporary motive force to HCV-758 using DSP-013, RHR Flow Control Valves Repair Procedure. c. Contact Maintenance to repair HCV-758. d. Return to procedure and step in effect.
21.	Check RCS Temperature - STABLE OR DECREASING	Go To Step 23.
22.	Return To Procedure And Step In Effect	

32. 006 A1.07 001

Given the following plant conditions:

- Plant is shutdown for refueling.
- GP-009-1, Filling the Refueling Cavity with Fuel in the Reactor Vessel, is in progress.
- Safety Injection Pumps "A" and "C" are running.
- SI-866A, LOOP 3 HOT LEG, is full open and SI-866B, LOOP 2 HOT LEG, is being opened via its Control Power Switch to increase cavity fill rate.
- Safety Injection Pump discharge pressure is at 530 psig and lowering.

Which ONE (1) of the following identifies the required actions to be taken IAW GP-009-1 to control the Safety Injection pump discharge pressure?

A. Close SI-856A **AND** SI-856B, SI Pump Recirc., valves.

B. Close SI-856A **OR** SI-856B, SI Pump Recirc., valve.

C.✓ Place SI-866B Control Power Switch to DEFEAT.

D. Place SI-866B Control Switch to CLOSE.

The correct answer is C.

A. Incorrect. Plausible because closing SI-856A AND SI-856B would have the desired effect of increasing discharge pressure. However, this is not the correct procedural direction.

B. Incorrect. Plausible because closing SI-856A OR SI-856B would have the desired effect of increasing discharge pressure. However, this is not the correct procedural direction.

C. Correct. The control power switch is taken to defeat to remove control power and stop the travel of SI-866B. This will stop the lowering of SI Pump discharge pressure and maintain it above the limit of 500 psig while maximizing SI flow rate.

D. Incorrect. Closing SI-866B with its control switch will cause the SI pump discharge pressure to rise but will limit flow to the refueling cavity since the valve will travel to the full closed position.

Question 32
Tier 2 / Group 1
K/A Importance Rating - RO 3.3 SRO 3.6

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pressure, high and low.

Reference(s) - Sim/Plant design, GP-009-1, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - GP-009-003
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 45.5
Comments - K/A met because candidate must analyze changing parameters and determine the appropriate actions to control SI pump discharge pressure.

FILLING THE CAVITY USING SI-866A

INIT

NOTE: Flow will increase rapidly as SI-866A is opened. FI-932, FI-940, ERFIS point SIF5301A, and ERFIS point SIF5303A indicate SI flow. Concise communication is needed between the operator monitoring RTGB hot leg injection flow indication and the operator manipulating the Control Power Switch for SI-866A. Placing the SI-866A Control Power Switch in DEFEAT immediately when instructed will prevent the pump from approaching a run out condition.

CAUTION

The cavity fill shall not overtake the reactor vessel head lift by the polar crane. The Reactor Cavity Water Level should not be closer than 1 foot below the Reactor Vessel Head Flange.

1. **VERIFY** the Control Power Switch for SI-866A is in NORMAL. ✓
2. **OPEN** SI-866A, LOOP 3 HOT LEG, **AND MONITOR** hot leg injection flow. ✓
3. **WHEN** flow indicates between 250 and 300 gpm as indicated on FI-932, FI-940, ERFIS point SIF5301A, or ERFIS point SIF5303A, **THEN PLACE** the Control Power Switch for SI-866A in DEFEAT.

   Indicated flow ✓ gpm FI-932/SIF5301A/FI-940/SIF5303A ✓
   Circle one
4. **INSTALL** the Control power fuses for the second SI pump **AND RECORD** pump selected.

   SI Pump ✓ ✓
5. **ESTABLISH** communications between the Control Room and the Operator at the SI Pump. ✓

CAUTION

SI Pump discharge pressure shall remain above 500 psig to prevent pump run out.

6. **MONITOR** SI Pump discharge pressure **AND MAINTAIN** pressure greater than 525 psig. ✓
7. **START** the second SI Pump **AND** record the time started.

   Time Started ✓ ✓

FILLING THE CAVITY USING SI-866A

INIT

NOTE: When the SI-866A Control Power Switch is placed in NORMAL, the valve will continue in the open direction because the open signal is still present.

8. **PLACE** the SI-866A Control Power Switch in NORMAL. ✓

9. **CHECK** SI-866A opening by observing an increase in SI flow and/or a decrease in pump discharge pressure. ✓

10. **IF** SI pump discharge pressure approaches 525 psig, **THEN PLACE** SI-866A Control Power Switch in DEFEAT. ✓

11. **IF** it is desired to use valve SI-866B to increase cavity fill rate, **THEN PERFORM** the following:

    a. **ESTABLISH** communications between the Control Room and the Operator in the SI Pump Room. ✓

    b. **MONITOR** SI Pump discharge pressure **AND MAINTAIN** pressure greater than 525 psig. ✓

    c. **PLACE** the SI-866B Control Power Switch in NORMAL. ✓

    d. **OPEN** SI-866B, LOOP 2 HOT LEG, **AND MONITOR** SI pump discharge pressure. ✓

    e. **IF** SI pump discharge pressure approaches 525 psig, **THEN PLACE** SI-866B Control Power Switch in DEFEAT. ✓

FILLING THE CAVITY USING SI-866A

INIT

12. **IF** the cavity water fill rate is overtaking the reactor vessel head lift **THEN** monitor closely **AND** perform the following as needed to maintain Reactor Cavity Water Level more than 1 foot from the Reactor Vessel Head Flange:

    a. Simultaneously **STOP** the running SI Pumps, **AND RECORD** the time stopped.

    Time Stopped _________ _____

NOTE: When SI-866A or SI-866B Control Power Switch is placed in NORMAL, the applicable valve may continue in the open direction if the open signal is still present.

    b. **VERIFY** SI-866A Control Power Switch in NORMAL. _____

    c. **VERIFY** SI-866B Control Power Switch in NORMAL. _____

    d. **VERIFY** SI-866A **AND** SI-866B valve travel has stopped. _____

    e. **VERIFY CLOSED** SI-866A, LOOP 3 HOT LEG. _____

    f. **VERIFY CLOSED** SI-866B, LOOP 2 HOT LEG. _____

    g. **IF** there is no longer a chance for the cavity water fill rate to overtake the reactor vessel head lift, **THEN COORDINATE** the restart of the cavity refill with the refuel manager **AND** proceed with the following:

        1) **VERIFY** the Control Power Switch for SI-866A is in NORMAL. _____

        2) **OPEN** SI-866A, LOOP 3 HOT LEG, **AND MONITOR** hot leg injection flow. _____

FILLING THE CAVITY USING SI-866A

12.g (Continued) INIT

3) **WHEN** flow indicates between 250 and 300 gpm as indicated on FI-932, FI-940, ERFIS point SIF5301A, or ERFIS point SIF5303A, **THEN PLACE** the Control Power Switch for SI-866A in DEFEAT.

Indicated flow ______ gpm

FI-932 / SIF5301A / FI-940 / SIF5303A (Circle one) ____

4) **INSTALL** the Control power fuses for the second SI pump **AND RECORD** pump selected **IF** required.

SI Pump ________ ____

5) **ESTABLISH** communications between the Control Room and the Operator at the SI Pump. ____

> **CAUTION**
>
> SI Pump discharge pressure shall remain above 500 psig to prevent pump run out.

6) **MONITOR** SI Pump discharge pressure **AND MAINTAIN** pressure greater than 525 psig. ____

7) **START** the second SI Pump **AND RECORD** the time started.

Time Started __________ ____

> **NOTE:** When the SI-866A Control Power Switch is placed in NORMAL, the valve will continue in the open direction because the open signal is still present.

8) **PLACE** the SI-866A Control Power Switch in NORMAL. ____

9) **CHECK** SI-866A opening by observing an increase in SI flow and/or a decrease in pump discharge pressure. ____

10) **IF** SI pump discharge pressure approaches 525 psig, **THEN PLACE** SI-866A Control Power Switch in DEFEAT. ____

FILLING THE CAVITY USING SI-866A

12.g (Continued) INIT

11) **IF** it is desired to use valve SI-866B to increase cavity fill rate, **THEN PERFORM** the following:

.a) **ESTABLISH** communications between the Control Room and the Operator in the SI Pump Room. ____

.b) **MONITOR** SI Pump discharge pressure **AND MAINTAIN** pressure greater than 525 psig. ____

.c) **PLACE** the SI-866B Control Power Switch in NORMAL. ____

.d) **OPEN** SI-866B, LOOP 2 HOT LEG, **AND MONITOR** SI pump discharge pressure. ____

.e) **IF** SI pump discharge pressure approaches 525 psig, **THEN PLACE** SI-866B Control Power Switch in DEFEAT. ____

	Initials	Name (Print)	Date
Performed By:	________	____________________	________
	________	____________________	________
	________	____________________	________
	________	____________________	________
	________	____________________	________
Approved By:	________	____________________	________
		Shift Manager	Date

If a pressure increase is desired, the respective vent can be opened and regulated nitrogen will enter the SI Accumulator. SI Accumulator nitrogen supply regulator (PCV-937) is set at 660 psig to prevent over pressurizing the lines or the SI Accumulators.

Do not open more than one vent valve at a time when the accumulator is required to be operable. This will prevent depressurizing more than one accumulator during a LOCA. If they were cross connected during a LOCA and one depressurized because of the break on that loop, then the other accumulator could also depressurize which would invalidate the LOCA analysis. (OP-202, Precaution)

5.6.4 SI Accumulator Make-Up Valves (SI-851A, 851B and 851C)

One of these air operated valves is used for each SI Accumulator. By starting a SI pump and opening valve SI-869, borated RWST water can be used to replenish a SI Accumulator. (SI-869 can be operated from the RTGB) **CAUTION**: *Filling the tank can result in rapid level changes due to the narrow level span of 14 inches.*

5.6.5 Test Valves (SI-850A, 850C and 850E)

Test valves are closed with packing tightened to prevent leakage and not presently used. (Note: Valves SI-850 A-F has Instrument Air isolated to fail the valves close.)

5.6.6 Cold Leg Injection Test (SI-850B, Loop 1; SI-850D, Loop 2; SI-850F, Loop 3)

Test valves are closed with packing tightened to prevent leakage and not presently used.

5.7 Minimum Flowpath Recirc. Valves SI-856A and B

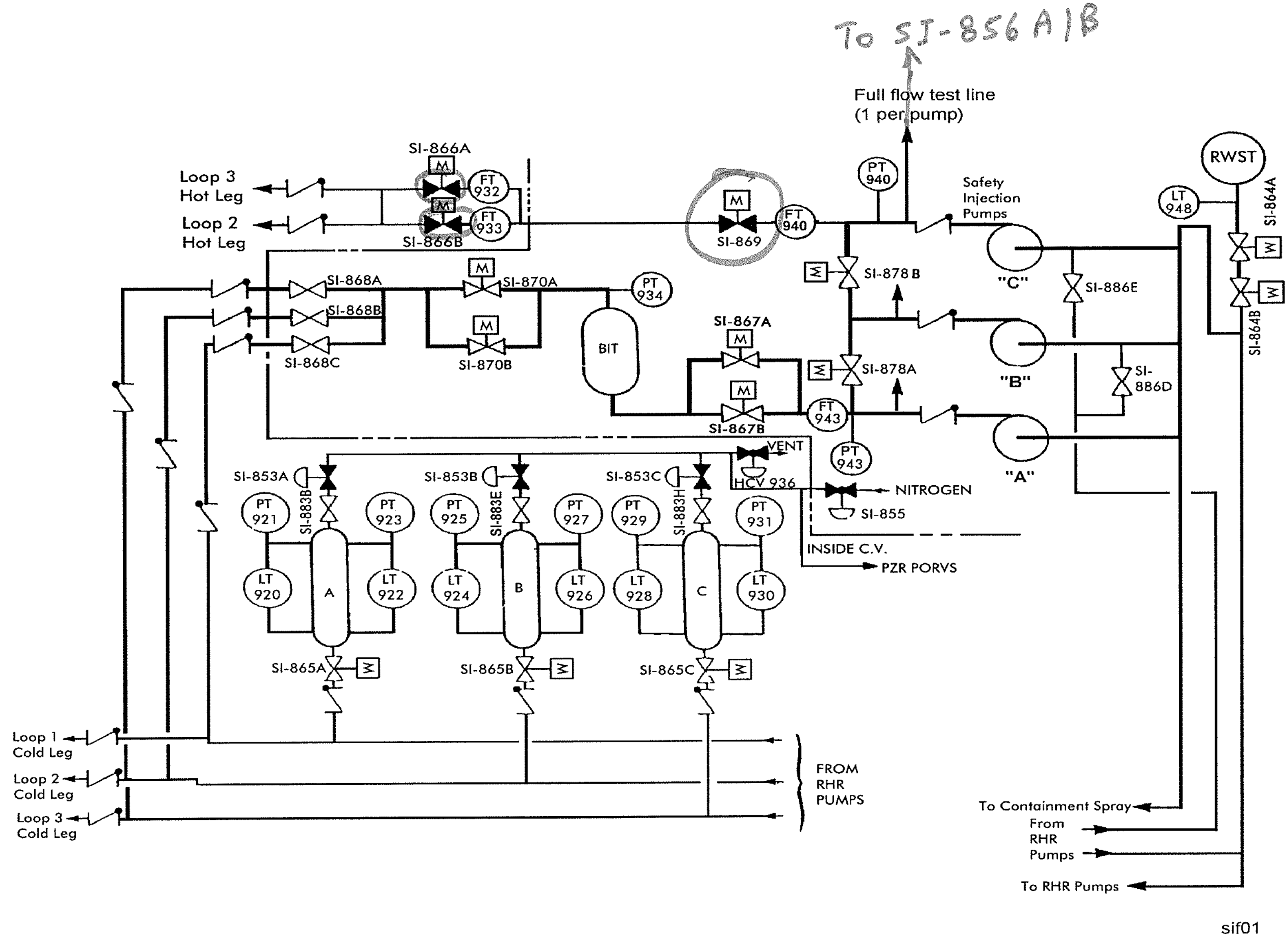
Valves SI-856A and B are normally open, air operated valves that allow flow through the minimum flow path lines 3/4-SI-1501R-100, -101, and -102 back to the RWST. To prevent damage to the SI pumps during the initial injection phase (high head, low flow), the valves are designed to fail open. The operators on valves SI-856A and B have hand wheels which allow manual closure by an operator.

5.8 SI Pump Recirculation Line Strainers, S-100A, S-100B, and S-100C

Strainers S-100A, B, and C are installed in the SI Pump recirculation lines 3/4-SI-1501R-100, -101, and -102 to prevent plugging of the restricting orifices (RO-2019A, B, and C) while having an insignificant effect on recirculation line flows. The strainers can be flushed to remove foreign matter.

INJECTION PHASE SIMPLIFIED DIAGRAM

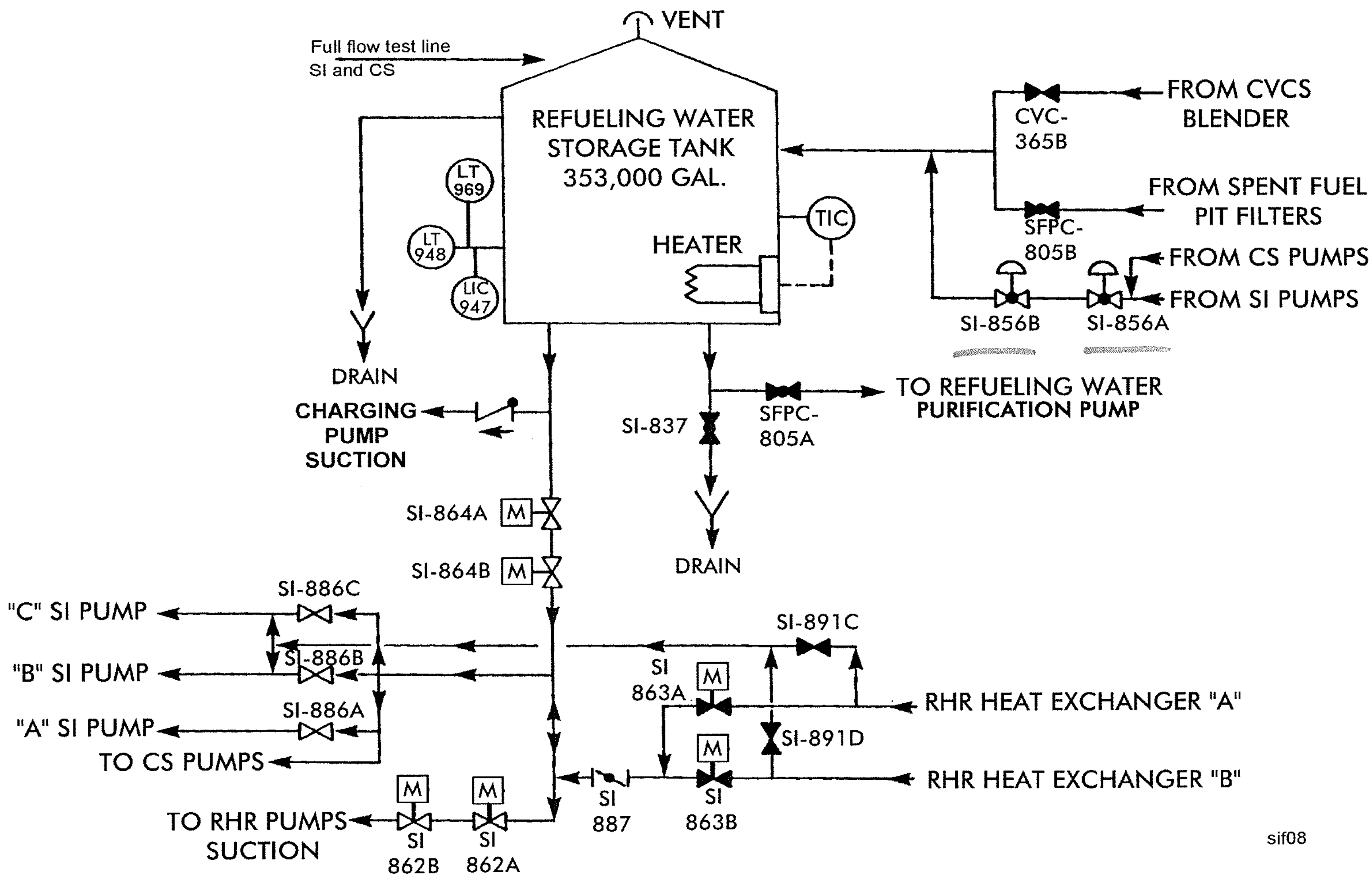
SI-FIGURE-1



INFORMATION USE ONLY

REFUELING WATER STORAGE TANK

SI-FIGURE-8



INFORMATION USE ONLY

33. 006 K6.10 001

Given the following plant conditions:

- LOCA recovery is in progress and RWST Level has lowered below 27%.
- The crew has transitioned to EPP-9, Transfer to Cold Leg Recirculation.
- While aligning the CV Sump To RHR valves, SI-860A, CV Sump to RHR, fails to open and remains in the closed position.

Which ONE (1) of the following describes the impact on the ability to provide cold leg recirculation?

A. No impact. Both RHR pumps are available with full flow capability.

B. Neither RHR Pump has the ability to use the CV Sump as a suction source.

C. Both "A" and "B" RHR Pumps are available, however, only ONE RHR Pump can be operated at a time.

D✓ "A" RHR Pump does NOT have a suction source. Only "B" RHR pump is available for cold leg recirculation.

The correct answer is D.

A. Incorrect. "A" RHR Pump will not have a suction source available. Plausible if the student incorrectly thinks that the two suction lines from the CV sump are cross connected.

B. Incorrect. Plausible if the student thinks that only one suction path is available from the sump to the RHR pumps. The student could think that the failure of SI-860A eliminates the only path.

C. Incorrect. Plausible if the student thinks that the suction lines are cross connected and a loss of one suction path eliminates the ability to operate both RHR pumps simultaneously.

D. Correct.

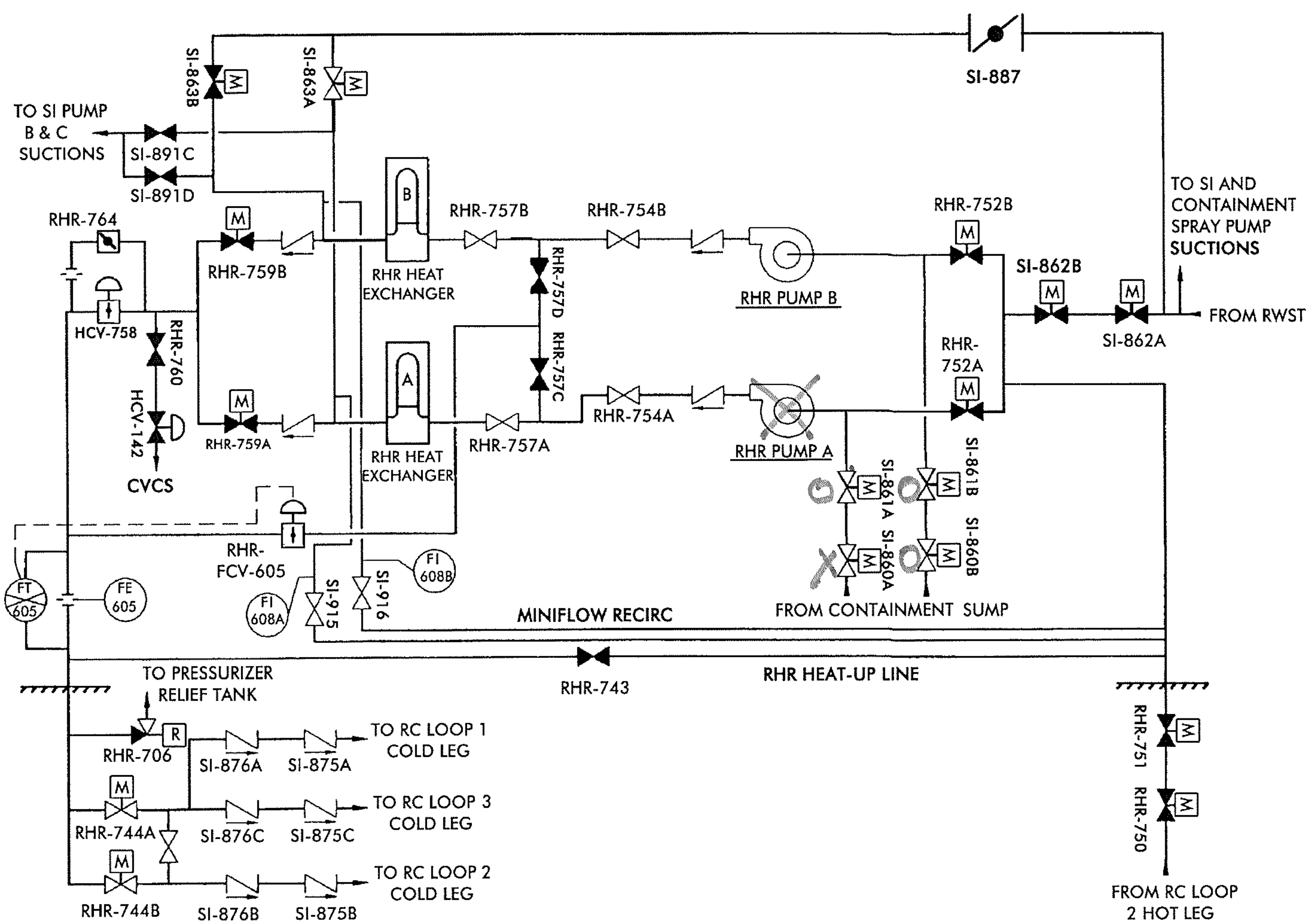
Question 33
Tier 2 / Group 1
K/A Importance Rating - RO 2.6 SRO 2.8

Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Valves

Reference(s) - Sim/Plant design, System Description, EPP-9
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-9-004
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.7
Comments - K/A match because the candidate must know how the failure of SI-860A will impact the ability to provide cold leg recirculation.

COLD LEG RECIRC - RHR FLOW<1200 GPM, RCS>125 PSIG

RHR-FIGURE-4



INFORMATION USE ONLY

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

24.	Establish Recirculation Flow As Follows:
	
	a. Verify CV SUMP TO RHR Valves for the RHR Pump to be started - OPEN:
	
	• RHR PUMP A
	  • SI-860A
	  • SI-861A
	• RHR PUMP B
	  • SI-860B
	  • SI-861B
	
	b. Verify one RHR Pump - RUNNING	

25.	Check Both The Following: • RVLIS Full Range - STABLE <u>OR</u> INCREASING <u>AND</u> • Core Exit T/Cs - STABLE <u>OR</u> DECREASING	<u>IF</u> RCS pressure is greater than 125 psig, <u>THEN</u> perform the following: a. Stop the running RHR Pump b. Go To Step 26. <u>IF</u> RCS pressure is less than 125 psig, <u>THEN</u> perform the following: a. Stop the running RHR Pump b. Start the opposite train RHR Pump.
26.	Check RWST Level - LESS THAN 9%	<u>WHEN</u> RWST level is less than 9%, <u>THEN</u> Go To Step 27.

34. 007 A1.03 001

The plant is at 100% RTP.

The following indications are reported by the RO:

- TI-471, PRT Temperature is trending up over a period of 4 hours and is now at 148°F.
- PI-472, PRT Pressure is trending up and is now at 4 psig.
- PZR PRV RC-551A, B and C Acoustic Monitor Lights are extinguished.

Which ONE (1) of the following will cause this condition?

A. ✓ PZR PORV is leaking by its seat.

B. Pressurizer Safety Valve is full open.

C. CVC-382, RCP Seal Return Line Relief, is leaking by seat.

D. CC-722A, RCP "A" Thermal Barrier Outlet Relief, is full open.

The correct answer is A.

A-Correct.

B-Incorrect. Safety valve full open would actuate the applicable PZR PRV RC-551A, B and C Acoustic Monitor Light.

C-Incorrect. CVC-382 would not cause temperature to go to 162°F. The seal return temperature is approximately 120 - 122°F.

D-Incorrect. CC-722A does not relieve to the PRT. CC-722A is plausible since it is located in Containment, however, it relieves to the CV Sump.

Question 34
Tier 2 / Group 1
K/A Importance Rating - RO 2.6 SRO 2.7

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench (PRT) tank temperature.

Reference(s) - Sim/Plant design, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - PZR 010
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 45.5
Comments - K/A met because candidate must analyze PRT parameters (pressure and temperature) and determine the cause of the given parameters.

level, which reduces the elevation head loss that spray flow must overcome. Normal spray flow is unlikely or will not occur at all when 'C' RCP is stopped and PZR level is less than 30%. Therefore, PZR pressure response may not be as expected for the above condition. (Ref. SCR 90-031)

3.4 PZR Surge Line

Nozzle Diameter	14 in.
Pipe Schedule	140
Surge Line	12 in.
Surge Line Nominal Thickness	1.125 in.
Design Pressure	2485 psig
Design Temperature	680°F

The PZR surge line, which connects the bottom of the PZR to RCS loop C hot leg, is sized such that it will pass the maximum anticipated surge flow of 20,000 gpm with a minimal pressure drop. A resistance temperature detector is installed in the surge line and provides indication and a low temperature alarm in the Control Room. Low temperature is indicative of stagnation of the PZR fluid.

The surge nozzle, located in the bottom of the vessel, is protected against thermal shock by a thermal sleeve. A retaining screen above the nozzle prevents foreign matter in the PZR from entering the RCS piping. Incoming surge flow displaces the water in the vessel as it enters the heater bundle area.

3.5 PZR Safety and Relief Valves

Three spring loaded safety valves and two PORVs provide for over pressure protection. The motive force for the PORVs is nitrogen with an IA backup.

3.5.1 Safety Valves (RC-551A, B, & C) (PZR-Figure 3 and 5)

Number	3
Capacity	293,330 lb/hr each at 3% accumulation
Set Pressure	2485 psig
Back Pressure	
Normal	3 psig
Relieving	350 psig (maximum)

The safety valves, set for the system design pressure of 2485 psig, are spring loaded, enclosed pop type, with backpressure compensation. The combined capacity of the valves is equal to, or greater than, the maximum surge rate resulting from complete loss

of load without reactor trip or any other control, except that the secondary plant safety valves are assumed to operate when steam pressure reaches their set point. A water seal is maintained below each valve seat to inhibit leakage. A resistance temperature detector (RTD) is installed in the discharge piping for each valve.

The RTD provides indication and a high temperature alarm in the control room to warn the operator of an actuated safety valve or safety valve seat leakage. Acoustic monitors are also installed on each of the three safety valves. These monitors are located in the Cable Spreading Room where they provide local indication. Control Room indication and alarm is provided to warn the operators of an actuated safety valve or safety valve seat leakage.

3.5.2 PORVs (PCV-455C & 456) (PZR-Figure 4)

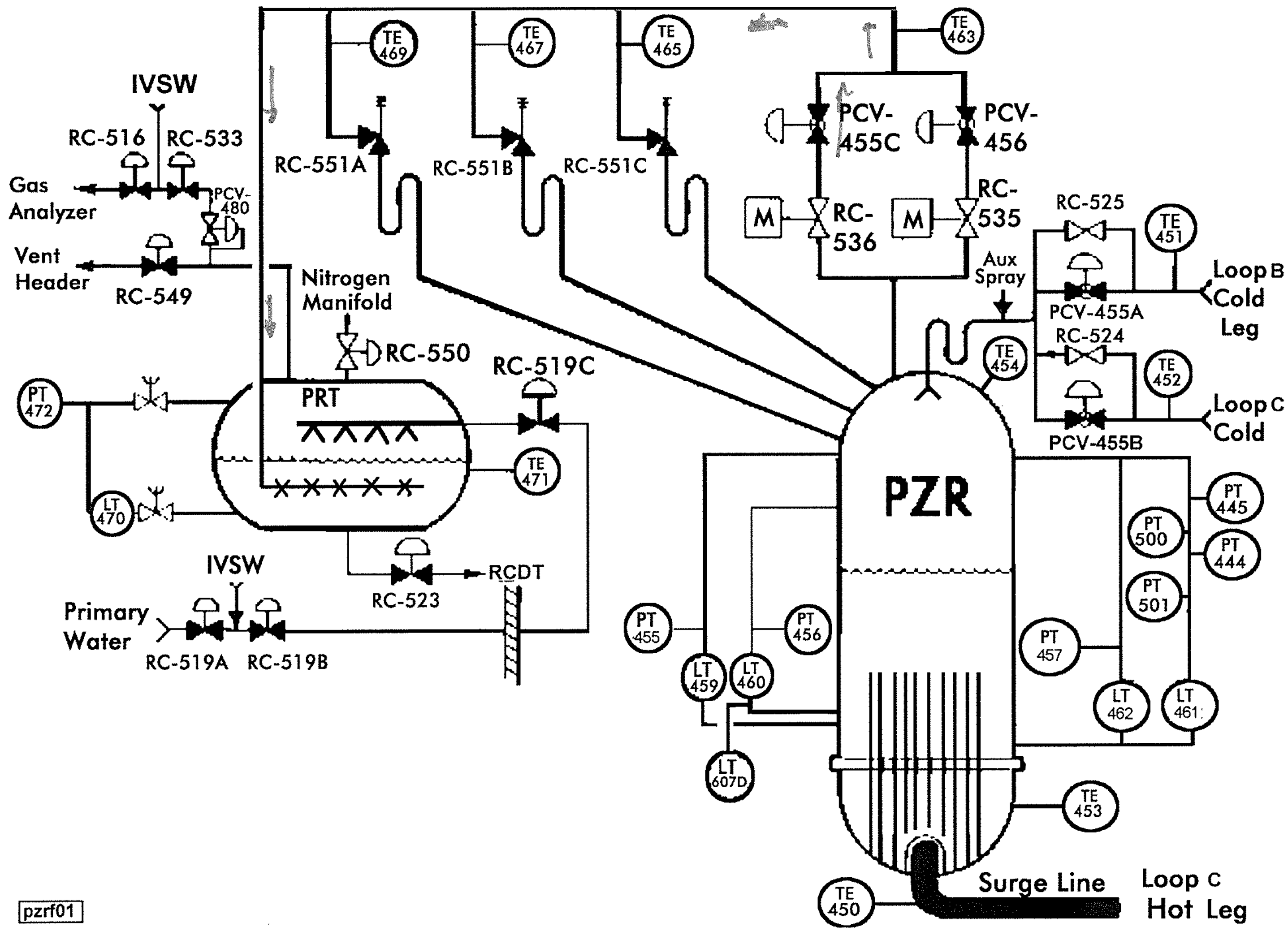
Number	2
Service	Open-Close air diaphragm
Relief capacity	210,000 lb/hr per valve
Set Pressure	2335 psig
Fluid	Saturated Steam
Relief Line Design Temperature	470°F
Relief Line Design Pressure	500 psig
Relief Line Diameter	3 inches

The two PORVs PCV-455C, powered from 125 VDC Circuit 7 on Distribution Panel "B", and PCV-456, powered from 125VDC Circuit 19 on Distribution Panel "A", have dual activating pressures. Whenever RCS temperature is above 360 F, the OVERPRESSURE PROTECTION switch is set for normal operation, and the valves will open at 2335 psig (except that PCV-455C is actuated by the variable output of a controller. That output is proportional to the error signal and reset, so that it may open at less than 2335 psig). During normal operation, the PORVs limit any pressure excursion and, thus, limit the operation of the spring-loaded PZR safety valves. An interlock with PT-455, 456, and 457 exists. This interlock will prevent the PORV's from opening unless two of three transmitters see RCS pressure greater than 2000 psig. Motor-operated Block Valves, RC-535 and RC-536, powered from 480V MCC-6, located ahead of the PORVs, are provided in order to isolate the PORVs from service should they fail to close, or leak excessively. The Block Valves may remain closed during normal operation to isolate a PORV experiencing excessive seat leakage.

Whenever the RCS temperature is between 360°F and 350°F, RCS pressure is between 375 psig and 350 psig, the OVERPRESSURE PROTECTION switch must be set for low pressure operation. The setpoint for opening is 400 psig at 360°F or less and increases as RCS temperature increases to a maximum setpoint of 2500 psig at 472°F. This

SYSTEM SIMPLIFIED DIAGRAM

PZR-FIGURE-1



INFORMATION USE ONLY

provides Low Temperature Over pressure Protection (LTOPP) for the RCS when it is water-solid, that is, at temperatures that a steam bubble cannot be maintained in the PZR. The PORVs are pneumatic valves, with nitrogen being supplied by the Plant Nitrogen System. To assure proper operation of the LTOPP System, the Instrument Air System is valved into service as a backup for the Plant Nitrogen System whenever LTOPP is in service.

For Appendix R Safe Shutdown purposes, PCV-455C and PCV-456 are classified as Hi / Lo Pressure Interface Valves. During normal operation these valves are closed except as described above. During a postulated Appendix R fire in Fire Area A5, these valves are required to be closed initially in order to maintain the RCS operating parameters (Pressurizer level, pressure and temperature) within the expected ranges.

Key operated switches are located on the Containment Fire Protection Panel in the Control Room. These are two position switches for each PORV (Normal or Isolate). The basic function of the Pressurizer PORV Normal / Isolate switches is to de-energize the electrical circuit during a postulated fire in Fire Area A5 to preclude (to the extent possible) the possibility of spurious operation of these valves. The addition of the isolation switches in conjunction with the actions taken in FP-001 ensure that the circuit is de-energized and protected against the possibility of conductor to conductor hot shorts within a cable. This will result in the Pressurizer PORVs failing closed.

3.6 PZR Relief Tank (PRT)

Design Pressure	100 psig
Design Temperature	340°F
Normal Operating Pressure	3 psig
Normal Operating Temperature	120°F
Normal Water Volume	900 ft^3
Rupture Disc Release Pressure	100 psig
Rupture Disc Relief Capacity	900,000 #/hr saturated steam
Internal Volume	1300 ft^3

The PRT is a horizontally mounted 1300 ft^3 tank inside the Containment Vessel (CV). It has a design temperature and pressure of 340°F and 100 psig respectively. It is piped to the PZR safety and PORVs by a 12" line. It is protected from over pressurization by two rupture discs that will relieve pressure to the Containment Vessel at approximately 100 psig. The rupture discs are designed to pass 900,000 lbs/hr. saturated steam.

The discharge from the High Point Vent System can also be directed to the PRT.

The PRT also collects leakage and liquid from various system relief valves located inside

the CV.

The PRT is normally filled to 70% with primary water. A 3 psig nitrogen atmosphere is maintained in the PRT to blanket the water. Primary water may be added to the tank by use of the primary water pumps and valves operated from the RTGB. Water may also be drained from the tank by utilizing either of the RCDT pumps and valves operated from the WDBRS panel.

Steam discharged to the PRT from the PZR PORVs and Safeties is directed to the sparger, a pipe containing spray nozzles, near the bottom of the PRT. This allows the high energy steam to be quenched in the water of the PRT. This will allow limited discharge of steam to the PRT before the pressure in PRT raises sufficiently to rupture the rupture discs.

4.0 INSTRUMENTATION

4.1 PZR Instrumentation

4.1.1 Temperature Instrumentation

The following temperature elements provide indication and alarm on the RTGB:

1. PZR Liquid Space (TE-453)
2. PZR Steam Space (TE-454)
3. PZR Spray Line (TE-451 and 452)
4. PZR Surge Line (TE-450)
5. Discharge of PORV's (TE-463) and each Safety Valve (TE-465, 467, and 469)

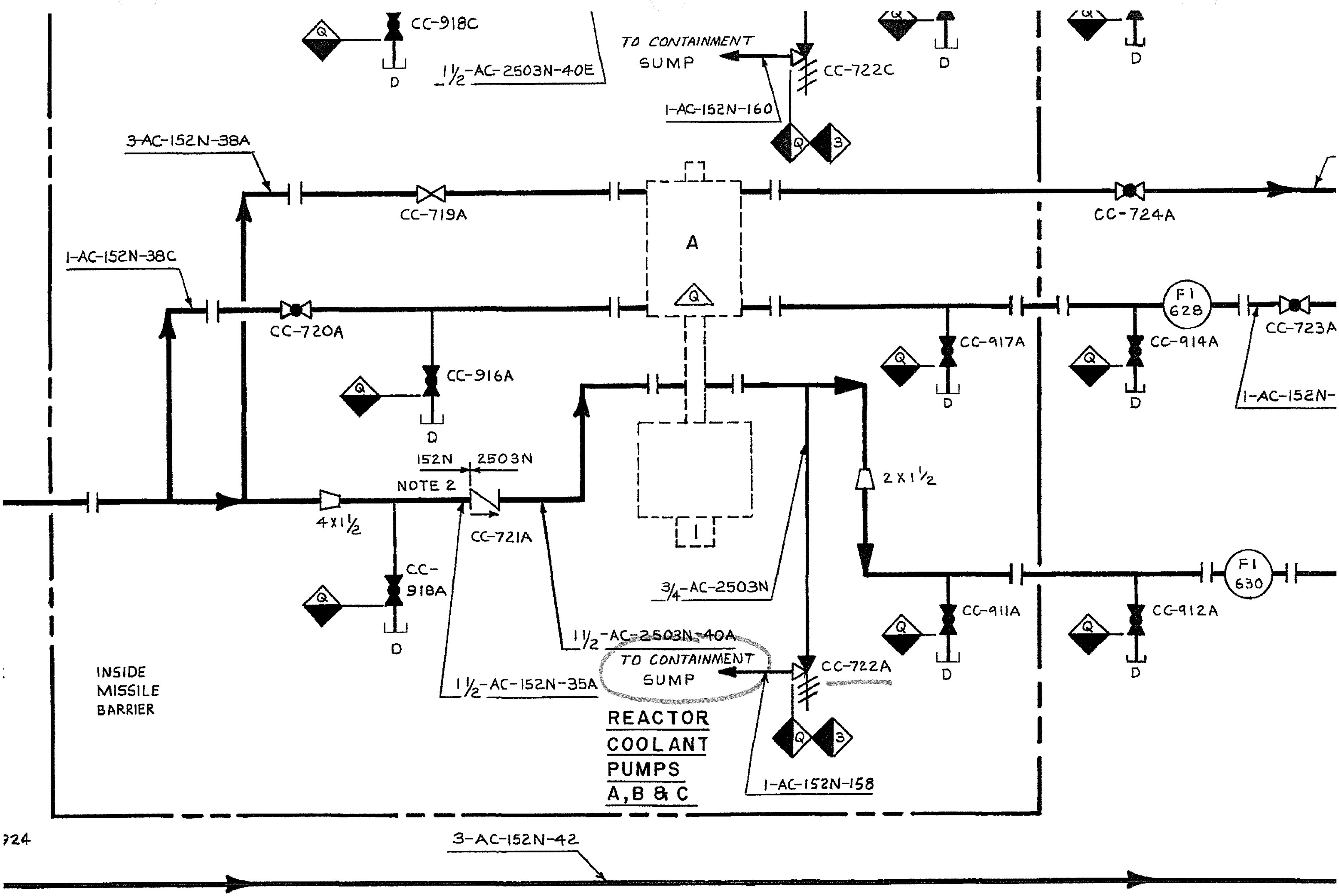
4.1.2 Level

Three PZR level transmitters, calibrated at normal operating temperatures, are used to provide signals for reactor protection (High Level Trip).

1. LT-459
2. LT-460
3. LT-461

One PZR level signal, LT-462, is provided for indication when the system is in cold condition and therefore is calibrated at cold conditions.

Channels 459, 460, and 461 are used in protection and are available for control functions by a switch on the RTGB. Normally, Channels 459 and 460 are used for control, and


CC-918C
D
1 1/2-AC-2503N-40E
TO CONTAINMENT SUMP
CC-722C
1-AC-152N-160
3-AC-152N-38A
CC-719A
A
CC-724A
1-AC-152N-38C
CC-720A
FI 628
CC-723A
CC-917A
CC-914A
CC-916A
1-AC-152N-
152N
2503N
NOTE 2
2 X 1 1/2
4 X 1 1/2
CC-721A
CC-918A
FI 630
3/4-AC-2503N
CC-911A
CC-912A
1 1/2-AC-2503N-40A
TO CONTAINMENT SUMP
CC-722A
INSIDE MISSILE BARRIER
1 1/2-AC-152N-35A
REACTOR COOLANT PUMPS A,B & C
1-AC-152N-158
3-AC-152N-42
?24

3.13 Seal Water Return Filter (Figure 10)

Specifications per CPL-HBR-M-041

Number	1
Type	Disposable cartridge
Design pressure	200 psig
Design temperature	250°F
Nominal flow rate	9 gpm
Maximum flow rate	240 gpm
Change out ΔP	20 psi
Vessel material of construction	Austenitic SS
Particulate Retention	25 micron

The seal water return filter will be in service continuously except when bypassed for cartridge replacement. Particulates from RCP No. 1 seal leakoff, RCP No. 1 seal bypass, lb/hr and flow from the excess letdown heat exchanger (when lined up to the VCT) are removed by the seal water return filter. The filter cartridge will be replaced when high radiation level or excessive differential pressure exists. This filter is located on the first floor of the Auxiliary Building. Local instrumentation is provided to determine filter differential pressure.

3.14 Seal Water Return Heat Exchanger

Number	1
Manufacturer	Westinghouse
Heat transfer rate design conditions	2.17×10^6 Btu/hr

Shell Side

Design pressure	150 psig
Design temperature	250°F
Pressure loss at design conditions	15 psi
Design flow	108,541 lb/hr
Design operating inlet temperature	105°F
Design operating outlet temperature	125°F
Fluid	Component cooling water
Material of construction	Carbon steel

Tube Side — CVCS

Design pressure	150 psig
Design temperature	250°F
Pressure loss at design conditions	10 psi

Design flow	126,756 lb/hr
Design operating inlet temperature	144°F
Design operating outlet temperature	127°F
Fluid	Borated reactor coolant
Material of Construction	Austenitic SS

The seal water return heat exchanger, located on the first floor of the Auxiliary Building, is a shell and U-tube heat exchanger. Component cooling water flows through the shell side and is throttled to maintain an outlet temperature equal to that of the VCT. The tube side has the flow from RCPs' No. 1 seal leakoffs, RCPs' No. 1 seal bypasses, and the excess letdown heat exchanger when selected to the VCT.

Higher heat load is placed on this heat exchanger when excess letdown is in service or upon a loss of seal injection to the RCP(s).

3.15 Excess Letdown Heat Exchanger

Number	1
Manufacturer	Westinghouse
Heat transfer rate design conditions	4.75 x 10^6 Btu/hr

Shell Side

Design pressure	150 psig
Design temperature	250°F
Pressure loss at design conditions	15 psi
Design flow rate	119,000 lb/hr
Normal operating inlet temperature	95°F
Normal operating outlet temperature	135°F
Fluid	Component cooling water
Material of construction	Carbon steel

Tube Side

Design pressure	2485 psig
Design temperature	650°F
Pressure loss at design conditions	15 psi
Design flow rate	12,400 lb/hr
Normal operating inlet temperature	547°F
Normal operating outlet temperature	195°F (maximum)
Fluid	Borated reactor coolant
Material of construction	Austenitic SS

35. 007 A2.01 001

Given the following plant conditions:

- Plant is in MODE 5.
- Residual Heat Removal (RHR) system is in service providing core cooling.
- The plant is in the process of taking the Pressurizer solid IAW GP-007, Plant Cooldown From Hot Shutdown To Cold Shutdown.

While swapping RHR Pumps, RHR-706, RHR System Relief, lifts and does not reseat.

Which ONE of the following completes the statements below?

The rupture disc will actuate when pressure reaches __(1)__ psig in the PRT.
IAW AOP-020, Loss of Residual Heat Removal (Shutdown Cooling), __(2)__.

A. (1) 50

(2) secure the RHR pumps and gag the relief valve.

B. (1) 50

(2) gag the relief valve ONLY

C.✓ (1) 100

(2) secure the RHR pumps and gag the relief valve.

D. (1) 100

(2) gag the relief valve ONLY.

The correct answer is C.

A. Incorrect. The setpoint for the rupture disc is 100 psig. The second half of the response is correct.

B. Incorrect. The setpoint for the rupture disc is 100 psig. The second half of the response is incorrect. The relief valve will be gagged, however, the RHR pumps must be secured to stop or limit the RCS leakage to the CV.

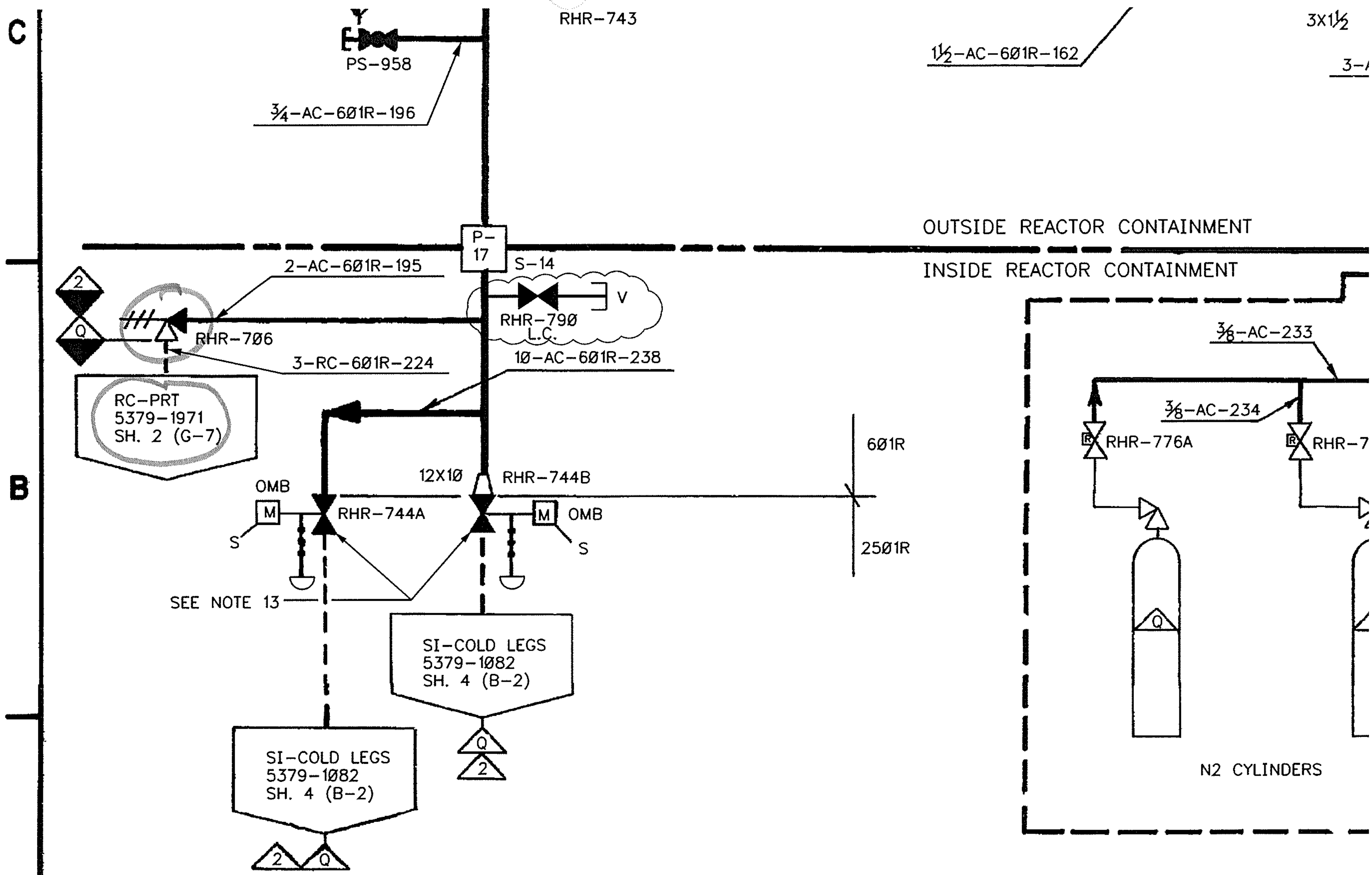
C. Correct. The rupture disc relieves at 100 psig. AOP-020 will direct the operators to secure both RHR pumps due to lowering RCS level.

D. Incorrect. The setpoint for the rupture disc is correct. The second half of the response is incorrect. The relief valve will be gagged, however, the RHR pumps must be secured to stop or limit the RCS leakage to the CV.

Question 35
Tier 2 / Group 1
K/A Importance Rating - RO 3.9 SRO 4.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck-open PORV or code safety.

Reference(s) - Sim/Plant design, System Description, AOP-020, AOP-033
Proposed References to be provided to applicants during examination - None
Learning Objective - PZR 010
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A met because candidate must know the impact of the safety valve lifting on PRT pressure and rupture disc and the mitigative strategy to address the failure.



REVISIONS

34	CR: 98-113 SR: 97ØØØ4Ø Y: R.MACDONALD HK: C. L. OLIVER PPR: C. L. OLIVER 3/27/98
35	CR: 99-395 Y: E. S. PERKINS HK: J. L. WALTERS PPR: C. L. OLIVER 7/29/99
36	CR: ØØ-462 SR ØØ-196, ØØ-198 Y: E. S. PERKINS HK: J.L. WALTERS PPR: C.L. OLIVER 11/8/ØØ
37	CR: ØØ-598 R: 25543 Y: F.W. GRANTHAM HK: D.M. EDWARDS PPR: C.L. OLIVER 12/29/ØØ
38	CR: Ø2-483 R: 54566 C/A #1 Y: E.S. PERKINS HK: J.L. WALTERS PPR: C.L. OLIVER 8/5/Ø2
39	C 5Ø8Ø5 Y: E.S. PERKINS HK: J.L. WALTERS PPR: C.L. OLIVER 11/4/Ø2
4Ø	C 51299 Y: J. L. WALTERS HK: E.S. PERKINS PPR: C.L. OLIVER 5/4/Ø4
41	C 71464 LECTRONICALLY IGNED

provides Low Temperature Over pressure Protection (LTOPP) for the RCS when it is water-solid, that is, at temperatures that a steam bubble cannot be maintained in the PZR. The PORVs are pneumatic valves, with nitrogen being supplied by the Plant Nitrogen System. To assure proper operation of the LTOPP System, the Instrument Air System is valved into service as a backup for the Plant Nitrogen System whenever LTOPP is in service.

For Appendix R Safe Shutdown purposes, PCV-455C and PCV-456 are classified as Hi / Lo Pressure Interface Valves. During normal operation these valves are closed except as described above. During a postulated Appendix R fire in Fire Area A5, these valves are required to be closed initially in order to maintain the RCS operating parameters (Pressurizer level, pressure and temperature) within the expected ranges.

Key operated switches are located on the Containment Fire Protection Panel in the Control Room. These are two position switches for each PORV (Normal or Isolate). The basic function of the Pressurizer PORV Normal / Isolate switches is to de-energize the electrical circuit during a postulated fire in Fire Area A5 to preclude (to the extent possible) the possibility of spurious operation of these valves. The addition of the isolation switches in conjunction with the actions taken in FP-001 ensure that the circuit is de-energized and protected against the possibility of conductor to conductor hot shorts within a cable. This will result in the Pressurizer PORVs failing closed.

3.6 PZR Relief Tank (PRT)

Design Pressure	100 psig
Design Temperature	340°F
Normal Operating Pressure	3 psig
Normal Operating Temperature	120°F
Normal Water Volume	900 ft^3
Rupture Disc Release Pressure	100 psig
Rupture Disc Relief Capacity	900,000 #/hr saturated steam
Internal Volume	1300 ft^3

The PRT is a horizontally mounted 1300 ft^3 tank inside the Containment Vessel (CV). It has a design temperature and pressure of 340°F and 100 psig respectively. It is piped to the PZR safety and PORVs by a 12" line. It is protected from over pressurization by two rupture discs that will relieve pressure to the Containment Vessel at approximately 100 psig. The rupture discs are designed to pass 900,000 lbs/hr. saturated steam.

The discharge from the High Point Vent System can also be directed to the PRT.

The PRT also collects leakage and liquid from various system relief valves located inside

the CV.

The PRT is normally filled to 70% with primary water. A 3 psig nitrogen atmosphere is maintained in the PRT to blanket the water. Primary water may be added to the tank by use of the primary water pumps and valves operated from the RTGB. Water may also be drained from the tank by utilizing either of the RCDT pumps and valves operated from the WDBRS panel.

Steam discharged to the PRT from the PZR PORVs and Safeties is directed to the sparger, a pipe containing spray nozzles, near the bottom of the PRT. This allows the high energy steam to be quenched in the water of the PRT. This will allow limited discharge of steam to the PRT before the pressure in PRT raises sufficiently to rupture the rupture discs.

4.0 INSTRUMENTATION

4.1 PZR Instrumentation

4.1.1 Temperature Instrumentation

The following temperature elements provide indication and alarm on the RTGB:

1. PZR Liquid Space (TE-453)
2. PZR Steam Space (TE-454)
3. PZR Spray Line (TE-451 and 452)
4. PZR Surge Line (TE-450)
5. Discharge of PORV's (TE-463) and each Safety Valve (TE-465, 467, and 469)

4.1.2 Level

Three PZR level transmitters, calibrated at normal operating temperatures, are used to provide signals for reactor protection (High Level Trip).

1. LT-459
2. LT-460
3. LT-461

One PZR level signal, LT-462, is provided for indication when the system is in cold condition and therefore is calibrated at cold conditions.

Channels 459, 460, and 461 are used in protection and are available for control functions by a switch on the RTGB. Normally, Channels 459 and 460 are used for control, and

CONTINUOUS USE

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AOP-020

LOSS OF RESIDUAL HEAT REMOVAL (SHUTDOWN COOLING)

REVISION 31

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides the instructions necessary to mitigate the loss of RHR in all conditions for which RHR can be aligned to provide shutdown cooling. This includes loss of RHR cooling for reasons such as RCS leakage, loss of power, loss of Service Water or Component Cooling Water, RHR pump cavitation, and inadequate RHR flow or abnormal reductions in RHR cooling.

This procedure is applicable in Modes 4, 5, and 6 when fuel is in the vessel.

2. ENTRY CONDITIONS

Direct entry from any condition resulting in a loss of RHR pump(s), RHR pump cavitation, abnormal RHR flow or temperature control, or excessive loss of RCS inventory while RHR is aligned for shutdown cooling.

As directed by the following other procedures:

- AOP-005, Radiation Monitoring System, when a low level in the SFP exists due to an RCS leak with the SFP GATE VALVE open.
- AOP-014, Component Cooling Water System Malfunction, resulting in stopping of the RHR Pumps while in CSD.
- AOP-016, Excessive Primary Plant Leakage, if less than 200°F and leakage exceeds Charging Capacity.
- AOP-017, Loss Of Instrument Air, if the loss of Instrument Air has affected core cooling while on RHR.

\- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Check RCS Level - LESS THAN -72 INCHES (69% FULL RANGE RVLIS)	<u>IF</u> RCS Level becomes less than -72 inches (69% FULL RANGE RVLIS), <u>THEN</u> verify BOTH RHR Pumps stopped. Go To Step 3.
2.	Verify BOTH RHR Pumps - STOPPED	
3.	Make PA Announcement For Procedure Entry	

<u>NOTE</u>

FRP-S.1 is <u>NOT</u> applicable for this event unless directed by the CSFSTs.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	From The RTGB, Verify Reactor Tripped As Follows: • REACTOR TRIP MAIN <u>AND</u> BYP - OPEN • Rod Position indication - ZERO • Rod Bottom lights - ILLUMINATED	<u>IF</u> the reactor does <u>NOT</u> trip, <u>THEN</u> dispatch an Operator to the Rod Drive MG Set Room to Open REACTOR TRIP BREAKERS A <u>AND</u> B.
5.	Check RCS Level - DECREASING: • Pressurizer level <u>OR</u> • RCS loop standpipe level <u>OR</u> • RVLIS <u>OR</u> • Refueling Cavity Watch report	<u>IF</u> either PZR PORV is failed open due to loss of input from PT-500 <u>OR</u> PT-501, <u>THEN</u> place the associated LTOPP Arming Switch to the NORMAL position. <u>IF</u> the event does <u>NOT</u> involve a loss of Inventory, <u>THEN</u> Go To Section E, Loss Of RHR Flow Or Temperature Control. <u>IF</u> RHR Pumps have been stopped due to loss of Inventory, <u>THEN</u> Go To Step 6.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Verify All Letdown Flowpaths Isolated As Follows: • CVC-460A & B, LTDN LINE STOP Valves - CLOSED • HIC-142, PURIFICATION FLOW Controller - ADJUSTED TO 0% • HIC-137, EXCESS LTDN FLOW Controller - ADJUSTED TO 0% • CVC-387, EXCESS LTDN STOP - CLOSED	
7.	Check Charging Pump Status - ALL STOPPED	Increase speed on the running Charging Pump to maximum. Go To Step 12.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Establish Charging Flow As Follows:	
	a. Check VCT Level - GREATER THAN 12.5 INCHES	a. Perform the following: 1) Verify OPEN LCV-115B, EMERG MU TO CHG SUCT. 2) Verify CLOSED LCV-115C, VCT OUTLET. 3) Go To Step 8.e.
	b. Verify RCS makeup concentration set to value greater than current RCS boron.	
	c. Verify LCV-115C, VCT OUTLET - OPEN	
	d. Verify LCV-115B, EMERG MU TO CHG SUCT - CLOSED	
	e. Verify HIC-121, CHARGING FLOW Controller - ADJUSTED TO 0% (OPEN)	
	f. Verify CVC-310B, LOOP 2 COLD LEG CHG - OPEN	f. Verify OPEN CVC-310A, LOOP 1 HOT LEG CHG.
9.	Start One Charging Pump	
10.	Observe charging flow on FI-122A	
11.	Increase Speed On The Running Charging Pump To Maximum	
12.	Check RCS Level - DECREASING	<u>IF</u> the RHR System is still in service, <u>THEN</u> Go To AOP-016, Excessive Primary Plant Leakage. <u>IF</u> the operating RHR Pump has been stopped, <u>THEN</u> Go To Step 19.
13.	Start One Charging Pump	<u>IF</u> no other Charging Pumps are available, <u>THEN</u> Go To Step 19.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
14.	Increase Speed On The Running Charging Pump To Maximum	
15.	Check RCS Level - DECREASING	IF the RHR System is still in service, THEN Go To AOP-016, Excessive Primary Plant Leakage. IF the operating RHR Pump has been stopped, THEN Go To Step 19.
16.	Start One Charging Pump	IF no other Charging Pumps are available, THEN Go To Step 19.
17.	Increase Speed On The Running Charging Pump To Maximum	
18.	Check RCS Level - DECREASING	IF the RHR System is still in service, THEN Go To AOP-016, Excessive Primary Plant Leakage.
19.	Check RCS Temperature Prior To Event Start - LESS THAN OR EQUAL TO 200°F	Go To AOP-033, Shutdown LOCA.
20.	Stop RHR Pumps	
21.	Isolate RHR By Closing The Following Valves: • RHR-750, RHR LOOP SUPPLY • RHR-751, RHR LOOP SUPPLY • RHR-744A, RHR COLD LEG INJ • RHR-744B, RHR COLD LEG INJ	
22.	Verify All RCPs - STOPPED	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*23.	Check Charging Pump Suction - ALIGNED TO VCT	IF RWST level decreases to 9%, THEN perform the following: a. Reduce charging flow to within the capacity of the RCS Makeup System. b. Verify OPEN LCV-115C, VCT OUTLET. c. Verify CLOSED LCV-115B, EMERG MU TO CHG SUCT, AND CVC-358, RWST TO CHARGING PUMP SUCTION. Observe NOTE prior to Step 26 and Go To Step 26.
*24.	Check VCT Level - LESS THAN 12.5 INCHES	IF VCT level decreases to less than 12.5 inches, THEN perform Step 25. Observe NOTE prior to Step 26 and Go To Step 26.
25.	Align Charging Pump Suction From The RWST As Follows:	
	a. Check RWST level - GREATER THAN 9%	a. Perform the following: 1) Reduce charging flow to within the capacity of the RCS Makeup System. 2) Observe NOTE prior to Step 26 and Go To Step 26.
	b. At the RTGB, verify OPEN LCV-115B, EMERG MU TO CHG SUCT	b. Verify OPEN CVC-358, RWST TO CHARGING PUMP SUCTION, prior to continuing.
	c. Verify CLOSED LCV-115C, VCT OUTLET	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

The intent of this procedure is to maintain the CV Purge in service if the Equipment Hatch is not installed.

26. Initiate CV Closure Using OMM-033, CV Closure

27. Dispatch An Operator To Open The Breakers For Containment Sump Pumps A and B:
    - CV SUMP PUMP A - MCC-2
    - CV SUMP PUMP B - MCC-1

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

The RCS Loops are considered filled if the RCS is capable of being pressurized such that a secondary heat sink can be established through natural circulation.

****

CAUTION

Changes in RCS pressure may result in inaccuracies in RCS Loop Standpipe indications.

****

28.	Check RCS Level Prior To Event Start - BELOW -36 INCHES	Perform the following: a. Implement the EALs. b. Notify the SSO OR STA that Attachment 9, Potential Technical Specifications, is available for reference. c. IF the Reactor Vessel Head is removed, THEN Go To Section B, Loss Of RHR Inventory - Vessel Head Off. d. IF the Reactor Vessel Head is installed AND the RCS Loops NOT filled, THEN Go To Section C, Loss Of RHR Inventory - Vessel Head On. e. IF the RCS Loops are filled, THEN Go To Section D, Loss Of RHR Inventory - Level Stable Or Increasing.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section D

Loss Of RHR Inventory - Level Stable Or Increasing

(Page 1 of 16)

1. Evacuate Non-essential Personnel From Containment As Follows:

   a. Place the VLC Switch in the EMERG position

   b. Depress and hold CV EVACUATION HORN Pushbutton for 15 seconds

   c. Announce The Following Over Plant PA System:

      "ALL NON-ESSENTIAL PERSONNEL EVACUATE CV UNTIL FURTHER NOTICE"

   d. Depress and hold CV EVACUATION HORN Pushbutton for 15 seconds

   e. Repeat PA announcement

NOTE

- The most likely sources of high capacity loss are in the discharge side of the RHR flow path such as the RHR relief and inadvertent opening of flow paths back to the RWST.
- RHR-706, RHR SYSTEM RELIEF, is located on the first level of the Containment in the mechanical penetration area between the B & C RCP Pump Bay door and the outer wall approximately 25 feet off the floor directly above V12-11.
- A gagging device for the RHR relief is located in the AOP-020 tool bag.

2.	Check RHR Relief Valve - FAILED	Go To Step 4.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section D

Loss Of RHR Inventory - Level Stable Or Increasing

(Page 2 of 16)

3.	Perform The Following: a. Dispatch personnel to Containment to gag RHR-706, RHR SYSTEM RELIEF b. Go To Step 5	
4.	Locate And Isolate Leakage <u>AND</u> Known Drain Paths • SI-863A <u>AND</u> B, RHR LOOP RECIRC • SI-891C <u>AND</u> D, RHR PUMP DISCHARGE TO SI PUMPS "B" & "C" (Located in Pipe Alley overhead just inside door) • ANY other path that may have been opened via outage activities	
5.	Verify All Available CV AIR RECIRC COOLERs - RUNNING: • HVH-1 • HVH-2 • HVH-3 • HVH-4	
6.	Check CV Closure Status - PENETRATION OPEN	Observe the <u>NOTE</u> prior to Step 14 and Go To Step 14.
7.	Verify At Least One SI Pump - AVAILABLE	
8.	Warn Personnel In Containment That The RCS Will Be Refilled And To Stand Clear RCP Pump Bays	

CONTINUOUS USE

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AOP-033

SHUTDOWN LOCA

REVISION 15

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides the instructions necessary to mitigate a LOCA during a cooldown, after the SI Accumulators are isolated and before the RCS reaches 200°F. This procedure incorporates the guidance of ARG-2.

2. ENTRY CONDITIONS

   Entry into this procedure is made for RCS leak rates greater than makeup capability during the interval following SI Accumulator isolation, but prior to reaching 200°F via three mechanisms:

   - Direct entry after diagnosing above.
   - On direction of AOP-016, Excessive Primary Plant Leakage.
   - On direction of AOP-020, Loss Of Residual Heat Removal (Shutdown Cooling).

- END -

AOP-033 Not Appl. in MODE 5

36. 008 K4.09 001

Given the following plant conditions:

- MODE 1 at 100% RTP
- "B" CCW pump is in service
- The normal supply breaker to 480V Bus E-1 Trips Open.

Which ONE (1) of the following identifies the status of the CCW pumps TWO (2) minutes after the event?

A. All CCW Pumps started on Low Pressure

B. ONLY "A" CCW Pump started on Low Pressure

"B" and "C" CCW Pumps started on Blackout Sequencer

C. ONLY "C" CCW Pump started on Low Pressure

"B" CCW Pump started on Blackout Sequencer

D✓ "A" and "C" CCW Pumps started on Low Pressure

"B" CCW Pump started on Blackout Sequencer

The correct answer is D.

A - Incorrect. "A" and "C" CCW Pumps will start on low pressure. "B" CCW Pump will start on the Blackout Sequencer.

B - Incorrect. "A" and "C" CCW Pumps will start on low pressure. "B" CCW Pump will start on the Blackout Sequencer.

C- Incorrect. "A" and "C" CCW Pumps will start on low pressure. "B" CCW Pump will start on the Blackout Sequencer.

D - Correct. When the EDG output breaker is closed as would be the case when EDG "A" energizes 480V Bus E-1, the low pressure start for the affected CCW Pump ("B") will be inhibited due to the blackout sequencer starting the pump.

Question 36
Tier 2 / Group 1
K/A Importance Rating - RO 2.7 SRO 2.9

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The "standby" feature for the CCW pumps

Reference(s) - Sim/Plant design, System Description, OP-306
Proposed References to be provided to applicants during examination - None
Learning Objective - CCW 005
Question Source - RNP Bank - Modified
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7
Comments - K/A met because the candidate must analyze the given conditions and determine which pumps will start on designed Low Pressure setpoint and which pumps will start from the Blackout Sequencer.

intermediate system between the reactor coolant and the SW cooling system. This double barrier arrangement reduces the probability of leakage of high pressure radioactive coolant to the SW System.

The CCW Radioactive Liquid Monitor (R-17) samples the CCW at the inlet of the CCW pumps and will alarm if excess radioactivity is detected as may occur following a leak into the CCW System (i.e., non-regenerative heat exchanger or RCP thermal barrier leak).

Most of the CCW System piping within the containment building is located outside the concrete shield wall. This location provides radiation shielding which allows for maintenance and inspections to be performed during power operation.

The surge tank accommodates expansion, contraction and in-leakage of water, and ensures a continuous CCW supply until a leaking cooling line can be isolated. The tank is vented to the Waste Holdup Tank.

The surge tank relief valve is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a RCP thermal barrier cooling coil. The relief valve discharges to the Waste Holdup Tank.

In event of leakage or failure of the RCP thermal barrier cooling coil, the relief valves downstream are designed to maintain the RCS pressure boundary with closure of the associated isolation valves.

The relief valves on the cooling water line downstream from the waste gas compressor, boric acid evaporator, excess letdown, seal water return, non-regenerative, spent fuel pit and RHR heat exchangers are sized to relieve the volumetric expansion which would occur if the heat exchanger were isolated.

Makeup to the CCW surge tank is available from the primary water pumps (normal) and from the demineralized water system (backup).

3.0 COMPONENT DESCRIPTION

3.1 Component Cooling Water Pumps

Quantity	3
Type	Horizontal Centrifugal
Rated Capacity	6000 gpm
Rated Head	180 ft H_2O
Motor Horsepower	350 HP
Casing Material	Cast Iron
Design Temperature	200°F
Power Supply	

Pump "A"	480V DS Bus, Westinghouse type DS switchgear & ACB
Pump "B"	480V Bus E-1, Westinghouse type DB switchgear & ACB
Pump "C"	480V Bus E-2, Westinghouse type DB switchgear & ACB

The CCW pumps are located in the CCW Heat Exchanger Room, first level of the Auxiliary Building. The three single stage centrifugal pumps have mechanical seals on both sides of the casings. The pumps are driven by a 480 volt 350 HP, 3 phase AC motors. Minimum flow for CCW pumps during continuous operation should be greater than 2200 gpm per pump to minimize the potential for pump cavitation and excessive vibration. The CCW pump motors are provided with overcurrent and undervoltage protection by its breakers tripping open. All three CCW pumps are provided with auto start features on a CCW low pressure signal. On a loss of offsite power, only the CCW pumps supplied from the emergency buses receive power such that they can start on demand. Refer to Section 5.1, CCW Pump Controls, for more details on the above features. Starting limitations for the CCW pump motors are included in appropriate Operating Procedures.

3.2 Component Cooling Water Heat Exchangers

Quantity 2
Type Shell and Straight Tube

Heat Transferred (Shutdown condition)	29.35 x 10^6 BTU/hr
Shell Side (CCW)	
Inlet Temperature	115°F
Outlet Temperature	108°F
Design Flow Rate	4.46 x 10^6 lb/hr
Design Temperature	200°F
Design Pressure	150 psig
Material	Carbon Steel
Tube Side (SW)	
Inlet Temperature	95°F (99°F after ESR 98-362, summer '99)
Outlet Temperature	101°F
Design Flow Rate	4.96 x 10^6 lb/hr
Design Pressure	150 psig
Design Temperature	200°F
Material	90/10 Copper Nickel

The CCW heat exchangers are arranged in parallel. They are single pass shell and tube heat exchangers. Component cooling water flows on the shell side while SW flows

APP-002-E5 SI PMP COOL WTR LO FLOW, alarms at 50 gpm via FIC-658 to indicate low CCW flow to the SI pump coolers.

APP-036-D8 PROCESS MONITOR HI RAD, will alarm when R-17 reaches setpoint.

APP-036-J9 CCW PMP A LOCAL CONTROL, will alarm when CCW Pump "A" Transfer Switch in Local control on the Dedicated Shutdown Charging Pump Room Control Panel.

4.7 Relief Valve Setpoints

Refer to Attachment 10.4 for a complete listing of the CCW relief valves and setpoints.

4.8 Instrument Setpoints

Refer to Attachment 10.5 for a complete listing of the CCW Instrument setpoints.

5.0 CONTROLS AND PROTECTION

5.1 CCW Pump Controls (CWD-B-1980628 Sh00201, Sh00205, Sh00209)

5.1.1 RTGB Control Switches

a. Normal Operation
CCW pump control switch is a Three Position Switch (STOP/AUTO/START) spring return to AUTO, located on the RTGB. Placing the Control Switch to the START position will start the associated CCW pump. Placing the switch to the STOP position will stop the pump. The AUTO position is not labeled on the RTGB.

b. Auto Functions (Low pressure during normal operation)

The standby pump(s) auto starts when the CCW Pump Discharge Header pressure drops to 78 psig. (This auto start may be blocked from other events which are discussed later).

5.1.2 Dedicated Shutdown Control Switches

CCW Pump "A" has a two position control switch (STOP/START) spring return to center, located on the Dedicated Shutdown Charging Pump Room Control Panel. This switch allows operation of the pump during certain postulated App. "R" fire scenarios and/or a station blackout event.

A transfer switch with two positions (LOCAL/REMOTE) for CCW Pump "A" is located on the Dedicated Shutdown Panel. This switch is placed in the LOCAL position to allow operation of the CCW Pump "A" from the Dedicated Shutdown Panel Switch. Placing

this switch to LOCAL will actuate the CCW PMP A LOCAL CONTROL alarm on APP-036-J9.

When the transfer switch is in LOCAL, placing the CCW pump "A" switch to the START position will start the CCW pump and placing the switch to the STOP position will stop the pump.

5.1.3 Accident / Design Basis Event Operation

- Loss of Offsite Power (EDGs available)

  CCW Pump "A" - manual restart, if power available to the DS bus. The pump will auto start on low CCW pressure, with power available on the bus.

  CCW Pump "B" - auto restart after 30 seconds via the safeguards sequencer blackout (LOOP) logic after Emergency Diesel Generator "A" output breaker closes.

  CCW Pump "C" - auto restart after 30 seconds via the safeguards sequencer blackout (LOOP) logic after Emergency Diesel Generator "B" output breaker closes.
  During a Loss of Offsite Power, all three CCW pumps are load shed from their respective buses. CCW pumps B & C which are supplied from emergency bus E1 and E2 will trip on bus undervoltage via the 27 Bus Undervoltage Relays. CCW pump A will trip when the undervoltage trip device of the feeder breaker senses the loss of voltage on the DS bus.

- Safety Injection (SI)/LOCA

  CCW pumps will NOT auto start on the SI sequencer. If an SI signal is present with voltage on respective bus, the operating CCW pump(s) will continue to operate.

  CCW pumps "A", "B" & "C" , if running prior to SI, will continue running as long as power is available.

5.1.4 Auto Or Manual Start Blocks

- LOCA with Loss of Offsite Power

  CCW pumps "B" & "C" will not auto restart on low pressure. CCW pumps "B" and "C" will be stripped from their respective buses and must be manually started.

  CCW pump "A" will auto restart on low pressure if DS Bus energized.

8.1.1.2 (Continued)

NOTE:	When no CCW Pumps are running, there exists an auto start signal to all pumps due to low discharge header pressure. Auto start on low pressure may be locked out by placing the control switch in the STOP position with the low pressure alarm locked in. After one pump is running, this signal will **NOT** exist if sufficient pressure is being delivered by the running pump.

b. **PERFORM** the following applicable steps **ONLY** for the **FIRST** CCW Pump started **AND INITIAL** in the blanks provided (N/A remaining steps):

		CCW PMP A	CCW PMP B	CCW PMP C
1a)	**VERIFY** Attachment 10.3, CCW Pump "A" Prestart Checklist is complete.			
1b)	**VERIFY** Attachment 10.4, CCW Pump "B" Prestart Checklist is complete.			
1c)	**VERIFY** Attachment 10.5, CCW Pump "C" Prestart Checklist is complete.			
2a)	**RACK IN** breaker for CCW PUMP "B" Bus E-1, 52/22C.			
2b)	**RACK IN** breaker for CCW PUMP "C" Bus E-2, 52/26C.			

ALARM

CCW PMP LO PRESS

AUTOMATIC ACTIONS

1. Standby Component Cooling Pump starts

CAUSE

1. Trip of Running CCW Pump
2. All CCW Pumps Stopped
3. Insufficient number of CCW Pumps Running for system requirements
4. Leak in CCW System
5. Transfer of CCW Pumps (expected alarm)

OBSERVATIONS

1. CCW Pump status lights
2. Component Cooling Water Flow (FI-613)
3. CCW Surge Tank Level (LI-614B)

ACTIONS CK (✓)

NOTE: Receiving alarm APP-001-F5 when the handswitch for a non-operating (idle) CCW Pump is being held in the STOP position **MAY** require resetting the applicable CCW Pump(s) lockout IAW OP-306 to allow for auto start on low CCW pressure. It is possible to receive an alarm and not receive a pump lockout. Initiation of the lockout depends on the amount of time system pressure is below the low pressure set-point. EC 73622

1. **IF** running CCW Pump has tripped, **THEN VERIFY** Standby Pump starts. _____
2. **IF** a CCW Pump can **NOT** be started, **THEN REFER TO** AOP-014. _____
3. **IF** system load is high, **THEN DISPATCH** an Operator to check CCW Pressure. (local gauge) _____
4. **IF** a CCW rupture is in progress, **THEN REFER TO** AOP-014. _____

DEVICE/SETPOINTS

1. PC-611 / 78 psig

POSSIBLE PLANT EFFECTS

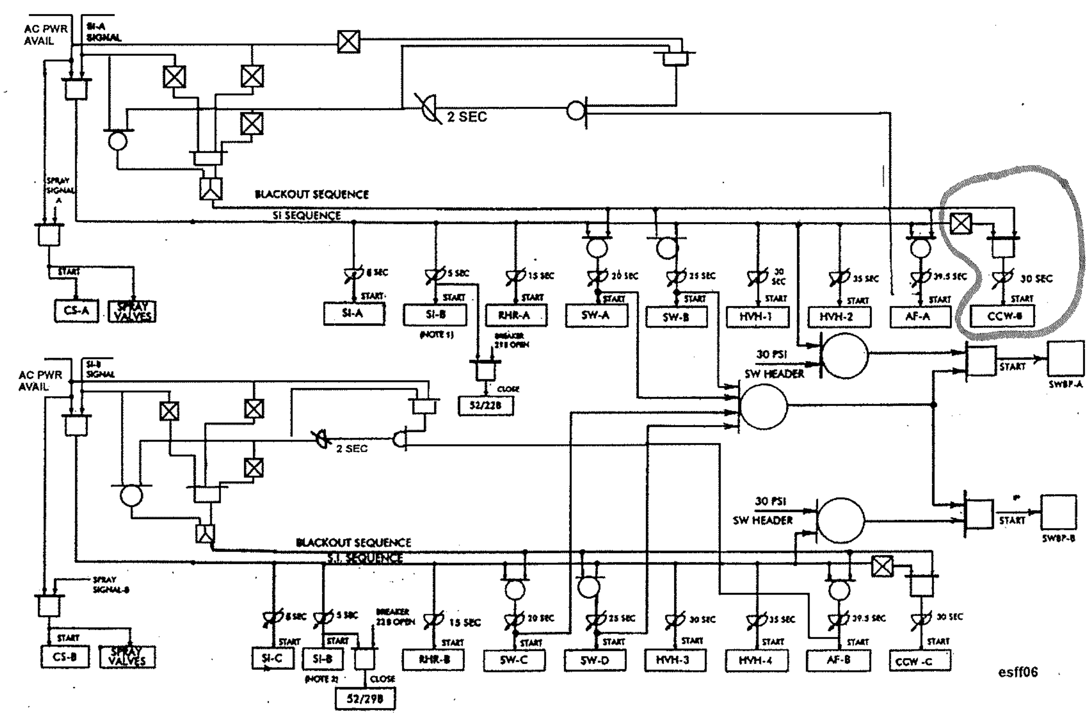
1. Loss of CCW

REFERENCES

1. ITS LCO 3.7.6
2. AOP-014, Component Cooling Water System Malfunction
3. OP-306, Component Cooling Water Pump
4. CWD B-190628, Sheet 595, Cable B

LOGIC DIAGRAM - SAFEGUARDS SEQUENCING LOGIC

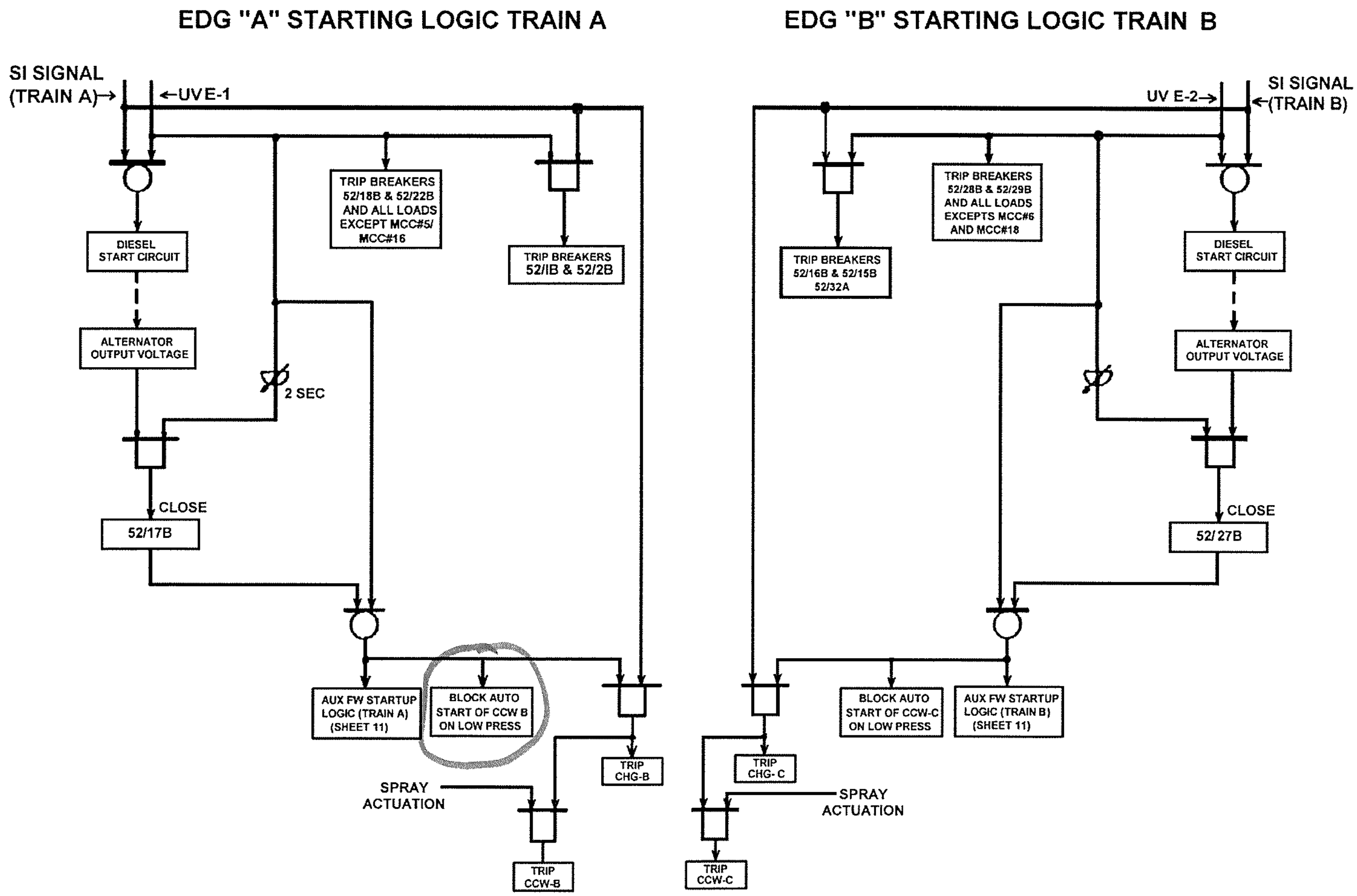
ESF-FIGURE-6



INFORMATION USE ONLY

EDG STARTUP LOGIC

ESF-FIGURE-11



INFORMATION USE ONLY

37. 010 K5.02 001

A pressurizer code safety valve has indications of leakage.

The following indications exist:

- Pressurizer pressure is 2225 psig and stable.
- Safety Valve tailpipe temperature indicates 231°F and rising.
- PRT pressure is 6 psig and rising 1 psi every 10 minutes

Which ONE (1) of the following completes the statement below?

The temperature downstream of the safety valves corresponds to ....

A. the PRT saturation pressure because of the loss of enthalpy of a throttling process.

B.✓ the PRT saturation pressure because of the constant enthalpy of a throttling process.

C. the saturation pressure of the presssurizer steam space because of the loss of enthalpy of a throttling process.

D. the saturation pressure of the presssurizer steam space because of the constant enthalpy of a throttling process.

The correct answer is B.

A. Incorrect. There is not a loss of enthalpy in the throttling process.

B. Correct.

C. Incorrect. Plausible if student thinks that the temperature will be equal to the PZR Steam temperature. There is not a loss of enthalpy in the throttling process.

D. Incorrect. Plausible if student thinks that the temperature will be equal to the PZR Steam Temperature. The constant enthalpy portion of the answer is correct.

A constant enthalpy process means same BTU/LBM of fluid. Using mollier diagram, go to the new pressure(PRT) directly to the right on a constant enthalpy line. Follow the pressure line up to the saturation curve to determine the temperature of the fluid.

Question 37
Tier 2 / Group 1
K/A Importance Rating - RO 2.6 SRO 3.0

Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Constant enthalpy expansion through a valve.

Reference(s) - Sim/Plant design, System Description, Steam Tables, Mollier Diagram, PWR Generic Fundamentals - Thermodynamics
Proposed References to be provided to applicants during examination - None
Learning Objective - PZR 004
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 45.7
Comments - K/A met because the candidate must know that temperature downstream of the leaking valve will correspond to saturation pressure of the PRT tank. Candidate is directly asked the basis for this thermodynamic process.

Example 4-6

THROTTLING

A throttling process is one in which the fluid is made to flow through a restriction (for example, a partially opened valve, or orifice, although the concept also applies to a pipe break) causing a considerable drop in the pressure of the fluid. The main effect is a significant pressure drop without any work interactions or changes in kinetic or potential energy. This may also result in a small increase in the internal energy.

The General Energy Equation for a throttling event can be simplified by making reasonable assumptions.

1. The elevation difference is negligible, so the potential energy terms can be ignored.

2. For a typical throttling event where steady flow conditions hold, the velocity in is approximately equal to the velocity out. Therefore, the kinetic energy terms cancel.

3. Since the fluids are flowing through the valve (or crack), the fluids perform no work on the valve, and because valve does no work on the fluid, the work term can be ignored.

4. In most steady flow applications, the throttling device is insulated, so the heat transfer is insignificant.

For a typical throttling event, the General Energy Equation reduces to:

$$P_1V_1 + U_1 = P_2V_2 + U_2$$

Equation 4-37

This equation can be reduced even further by replacing the PV + U terms on both sides with enthalpy terms and then factoring out the mass. Making this substitution and rearranging will yield the following, so the General Energy

Equation for a throttling process for specific energy terms can be reduced to:

$$P_1\upsilon_1 + u_1 = P_2\upsilon_2 + u_2$$

or

$$h_{in} = h_{out}$$

Equation 4-38

The points chosen in analyzing the process must be selected at sufficient distance from the point of throttling to ensure stable, uniform flow. The sudden expansion and pressure drop causes turbulence downstream of the flow obstruction.

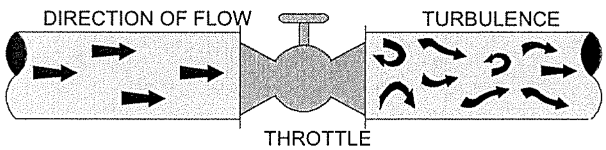


Figure 4-17 Flow Throttling

Throttling is essentially a constant enthalpy process, as shown in the h-s and T-s diagrams in Figure 4-18. Enthalpy remains constant while entropy increases. No work is done and no heat is added. The result is a pressure drop and a slight increase in velocity.

The change in specific volume (Equation 4-38) depends on the fluid being processed. Throttling is inherently irreversible with no ideal equivalent. For this reason, point 2′ is used in Figure 4-18. As the pressure decreases from P_1 to $P_{2'}$, enthalpy is constant and entropy increases. If the pressure decreases to a value found above or to the right of the saturated vapor line (point 3), then the process fluid (steam) will be in the superheated region. The value of throttling lies in the ability to provide a simple means of controlling equipment and processes. System efficiency is sacrificed to some degree to achieve simplification in the manipulation of equipment. Efficiency is lost due to an increase in entropy.

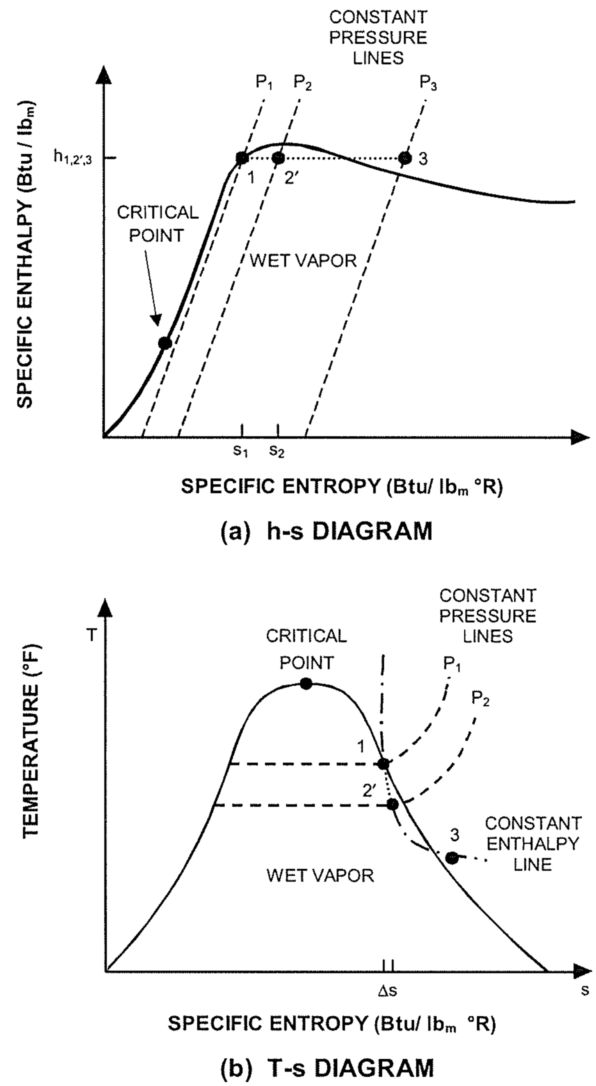


Figure 4-18 Property Diagrams of a Steam Throttling Process

A steam safety relief valve at 2250 psia is leaking a small amount. The relief tank is at 20 psia. Essentially the entire pressure drop occurs across the valve. Using a Mollier diagram, the quality of the steam exiting the valve is 66.5%.

When a throttling process is analyzed, a small opening of the valve is assumed. When the valve is nearly shut, the entire pressure drop is across the valve, and the process is considered isenthalpic. The more open the valve is, the smaller the pressure drop across the valve. A

greater percentage of the pressure drop becomes a function of the head loss in the piping. Pressure reduction due to frictional head loss is not isenthalpic.

The temperature-enthalpy diagram indicates an interesting behavior of water with respect to enthalpy. The specific enthalpy of saturated water vapor increases with temperature until approximately 450°F. Above 450°F, the specific enthalpy decreases with increasing temperature up to the critical point.

This phenomenon is particularly important during rapid pressure drops occurring during plant operations. At initially high temperatures (i.e., 650°F), an isenthalpic drop to atmospheric pressure results in high quality steam at 212°F. At initially lower temperatures (i.e., 400°F), an isenthalpic drop to atmospheric pressure results in superheated steam at approximately 320°F. For example, if a relief valve in the main steam line fails partially open, steam is throttled through the valve. The steam exiting is initially at a temperature of approximately 545°F. The steam experiencing an isenthalpic pressure drop to atmospheric pressure exits the throttled valve as superheated steam at a temperature of 290°F. On the other hand, steam is throttled through a valve at a temperature of 653°F and enters the piping to the relief tank (20psia) as a saturated mixture at 230°F. Because the processes are isenthalpic, an originally higher steam temperature exits the throttle valve at a lower temperature.

A Safety Relief Valve lifts when the saturated steam pressure reaches 1,110 psia. Calculate the resultant steam temperature when pressure drops to atmospheric. Is the steam still saturated?

Example 4-7

38. 010 K6.01 001

The plant is at 100% RTP and the following occurs:

- The PZR pressure master controller (PC-444J) is failing HIGH.
- The PZR spray valves are fully OPEN.
- The PZR PORVs are CLOSED.

Which ONE (1) of the following identifies the value of the controller output (PC-444J) for this condition?

A. 80%

B.✓ 70%

C. 60%

D. 50%

The correct answer is B.

A. Incorrect. At 80% output the PORV would be open.

B. Correct.

C. Incorrect. At 60% output the spray valves would only be partially open. They do not start to open until 56.25% demand signal and are fully open at 68.75%. The spray valves would be approximately 30% open.

D. Incorrect. At 50% output the spray valves would be fully closed. The spray valves do not start to open until 56.25%.

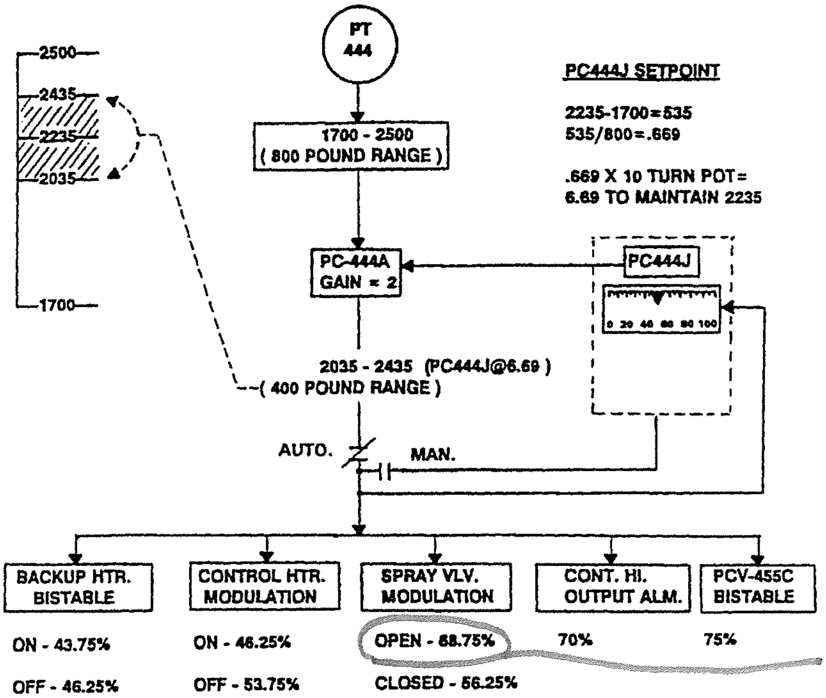
Question 38
Tier 2 / Group 1
K/A Importance Rating - RO 2.7 SRO 3.1

Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems

Reference(s) - Sim/Plant design, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - PZR 006
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.7
Comments - K/A met because candidate must analyze a set of conditions given for PZR pressure components and know what the pressurizer pressure controller output should be for these given conditions.

PC-444A CONTROLLER

PZR-FIGURE-7



68.75% (SPRAY FULL OPEN)

DETERMINATION OF EXPECTED CONTROLLER OUTPUT

1. B/U HEATERS - ON = 2210

2210-2035 = 175

175/400 = .4375 OR 43.75%

2. SPRAY VALVE OPENING = 2260

2260-2035 = 225

225/400 = .5625 OR 56.25%

56.25% (SPRAY START TO OPEN)

3. PCV-455C OPENS = 2335 (PORV)

2335-2035 = 300

300/400 = .75 OR 75%

pzrf09

INFORMATION USE ONLY

39. 012 A4.04 001

The plant is at 100% RTP when "A" Condensate Pump trips due to electrical fault.

Which ONE (1) of the following completes the statement below?

The FIRST reactor trip signal will be generated as soon as bistable lights for..........
(Assume no operator action.)

A. "SG NO. 2 LO-LO Level LC484A1" **AND** "SG NO. 2 LO-LO Level LC485A1" Illuminate.

B. "SG NO. 2 LO-LO Level LC484A1" **OR** "SG NO. 2 LO-LO Level LC485A1" Illuminate.

C. "SG NO. 2 LO Level LC484B1" **AND** "SG NO. 2 LO Level LC485B1" Illuminate.

D✓ "SG NO. 2 LO Level LC484B1" **OR** "SG NO. 2 LO Level LC485B1" Illuminate.

The correct answer is D.

A. Incorrect. This will create a reactor trip, but with steam flow / feed flow mismatch the reactor will trip at 30% S/G level.

B. Incorrect. This will not cause a trip since the logic for Low-Low Water Level Trip is 2 out of 3 Narrow Range Level Transmitters on 1 out of 3 S/Gs.

C. Incorrect. This will cause a trip but it is not necessary to have both LT-484 AND LT-485 reach the setpoint of 30%.

D. Correct.

Question 39
Tier 2 / Group 1
K/A Importance Rating - RO 3.3 SRO 3.3

Ability to manually operate and/or monitor in the control room: Bistable, trips, reset and test switches

Reference(s) - Sim/Plant design, System Description, Logic Diagrams
Proposed References to be provided to applicants during examination - None
Learning Objective - RPS 006
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5 to 45.8
Comments - K/A is met because candidate must understand the effect of a failure / trip of one condensate pump and know which bistables will be illuminated to give the first reactor trip signal.

necessary range of protection afforded by the OTΔT. This trip occurs when 2 out of 3 PZR Pressure Signals decreases below the trip setpoint. This trip is automatically blocked below 10% (P-7).

This trip is dynamically compensated based on the rate of change in pressure.

b. Setpoint - PC-455C, PC-456C, PC-457C
PM-455A, PM-456A, PM-457A

Trip Setpoint	1844 psig
Lead Time Constant	10 sec.
Lag Time Constant	1 sec.

4.1.5.13 High Pressurizer (PZR) Water Level Trip (Figure 31)

a. The High PZR Water Level Trip provides a back-up to the High PZR Pressure Trip and prevents the PZR Safety and Relief Valves from relieving water for credible accident conditions. This trip occurs when 2 out of 3 PZR Water Level Signals exceeds the trip setpoint. This trip is automatically blocked below 10% (P-7).

b. Setpoint - LC-459A, LC-460A, LC-461A/91% of span

4.1.5.14 Steam/Feedwater Flow Mismatch Trip (Figure 32)

a. The Steam/Feedwater Flow Mismatch Trip provides protection for the Reactor against an anticipated Loss of Heat Sink. This trip occurs when 1 out of 2 flow elements sense that Feedwater Flow is < Steam Flow and 1 out of 2 Steam Generator(S/G) Level Elements decrease below the setpoint in any S/G.

b. Setpoint - FC-478A, FC-478B/0.64 x 10^6 lbs/Hr
FC-488A, FC-488B/0.64 x 10^6 lbs/Hr
FC-498A, FC-498B/0.64 x 10^6 lbs/Hr
AND
LC-474B, LC-475B/30% of Span
LC-484B, LC-485B/30% of Span
LC-494B, LC-495B/30% of Span

4.1.5.15 S/G Low-Low Water Level Trip (Figure 33)

a. The S/G Low-Low Water Level Trip provides protection for the Reactor by preventing operation without adequate heat removal capability in the event of a sustained Steam/Feedwater Flow mismatch which is sufficiently small not to be sensed by the Steam/Feedwater Flow Mismatch Trip. This trip occurs when 2 out

of 3 S/Gs Narrow Range Level Elements on 1 out of 3 S/G_s decrease below the setpoint.

b. Setpoint - LC-474A, LC-475A, LC-476A/16% of Span
LC-484A, LC-485A, LC-486A/16% of Span
LC-494A, LC-495A, LC-496A/16% of Span

4.1.5.16 Safeguards Signal Trip

a. The Engineered Safeguards Signal Trips ensure that the Reactor will be shut down during a severe accident. This trip is initiated if the Engineered Safety Features Actuation System is automatically or manually actuated.

b. Setpoint - Refer to SD-006, Engineered Safety Features System

4.1.5.17 Turbine Trip/Reactor Trip (Figure 34)

a. The Turbine Trip/Reactor Trip provides overpressure or overtemperature protection for the RCS on a Loss of Load. This trip occurs when 2 out of 3 Auto-Stop Oil Pressure Signals decreases below the setpoint < 45 psig or when both Main Turbine Stop Valves are closed. This trip is active above 10% (P-7) and automatically blocked below 10% (P-7).

4.1.5.18 Manual Trip

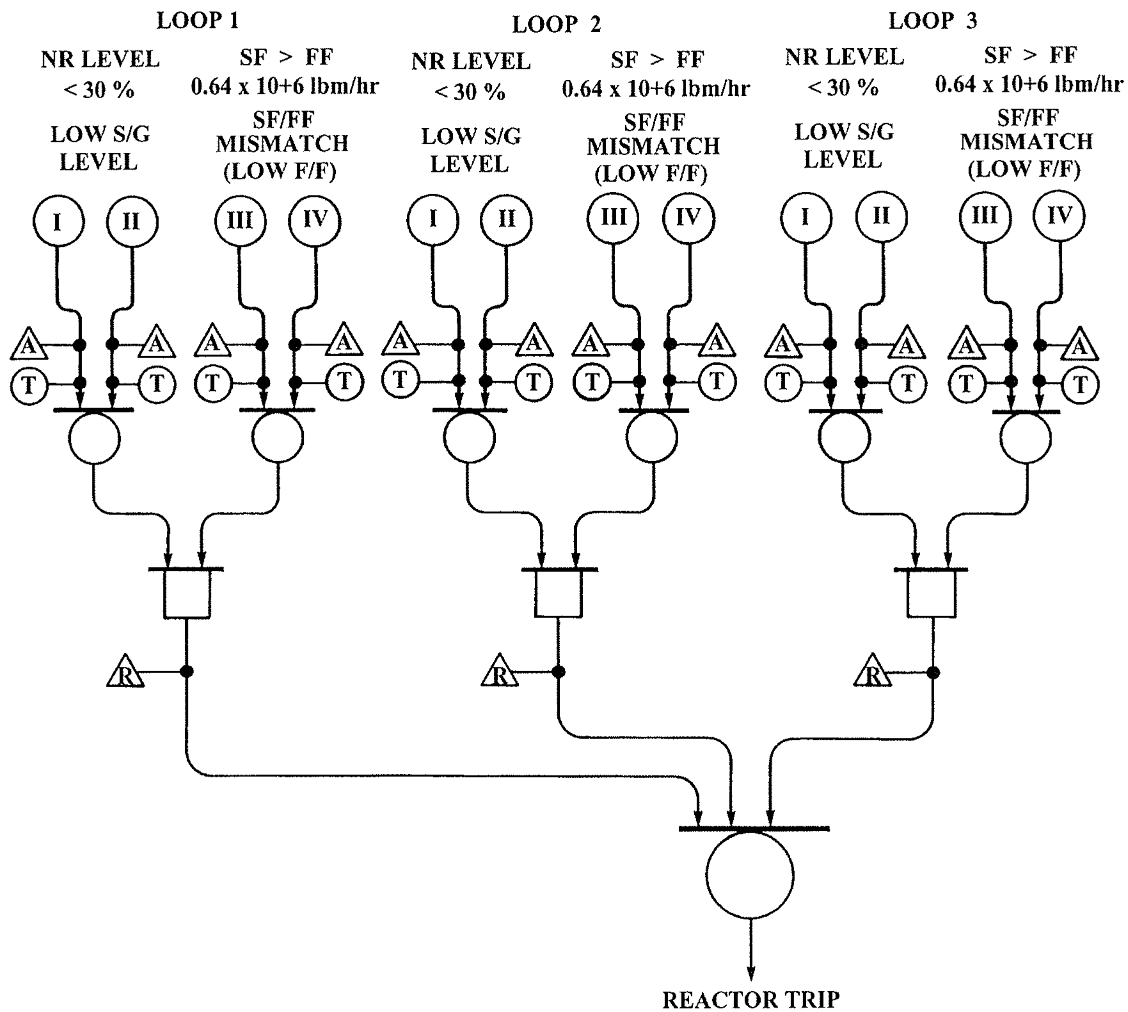
a. The Reactor can be tripped, manually, from either of the two Reactor Trip Pushbuttons located on the RTGB.

4.2 Permissive Circuits (See Attachment 10.1)(Figures 11, 12 & 13)

STM/FW FLOW MISMATCH WITH LOW S/G LEVEL TRIP LOGIC

RPS-FIGURE-32

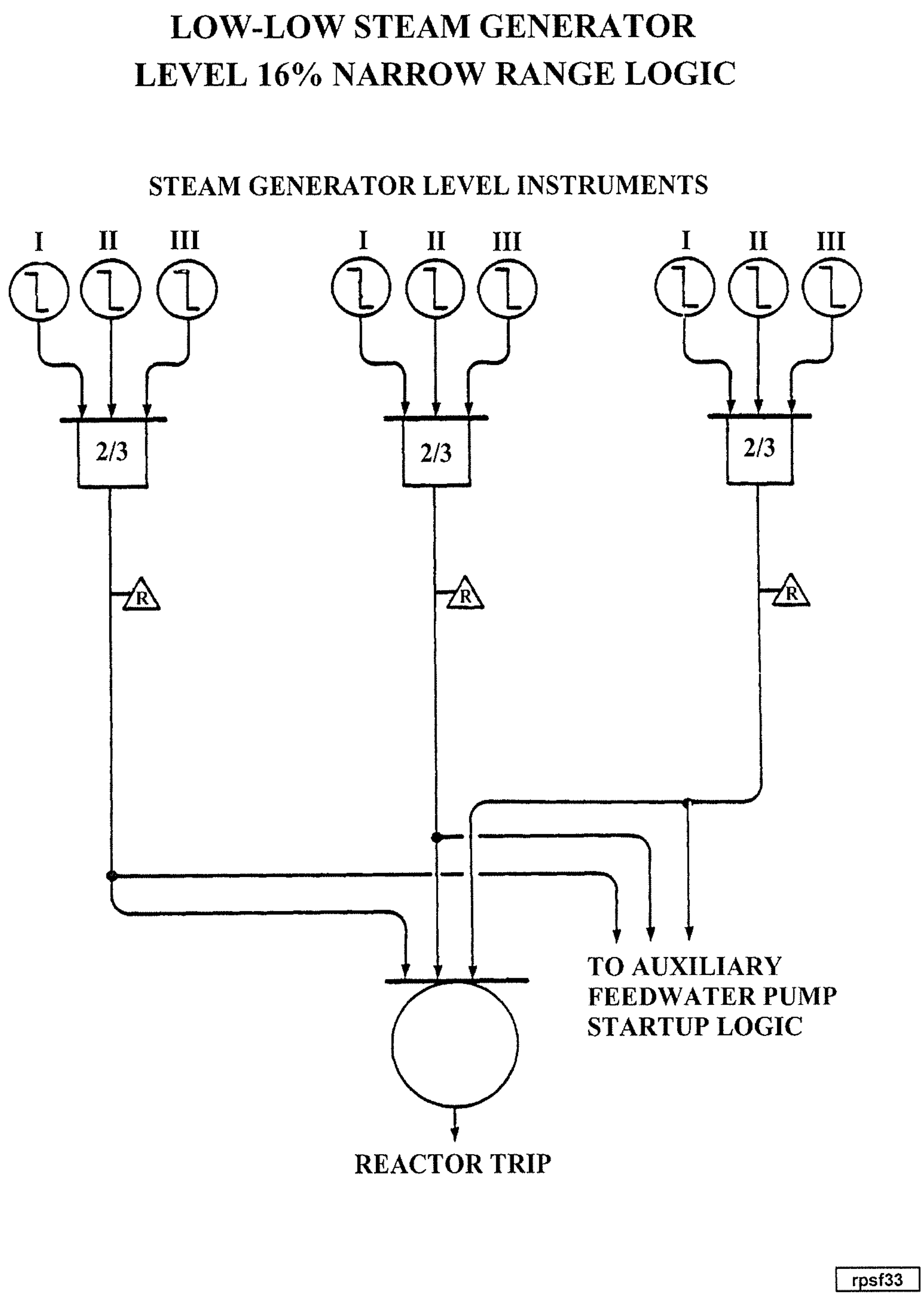
STM/FW FLOW MISMATCH WITH LOW S/G LEVEL TRIP LOGIC

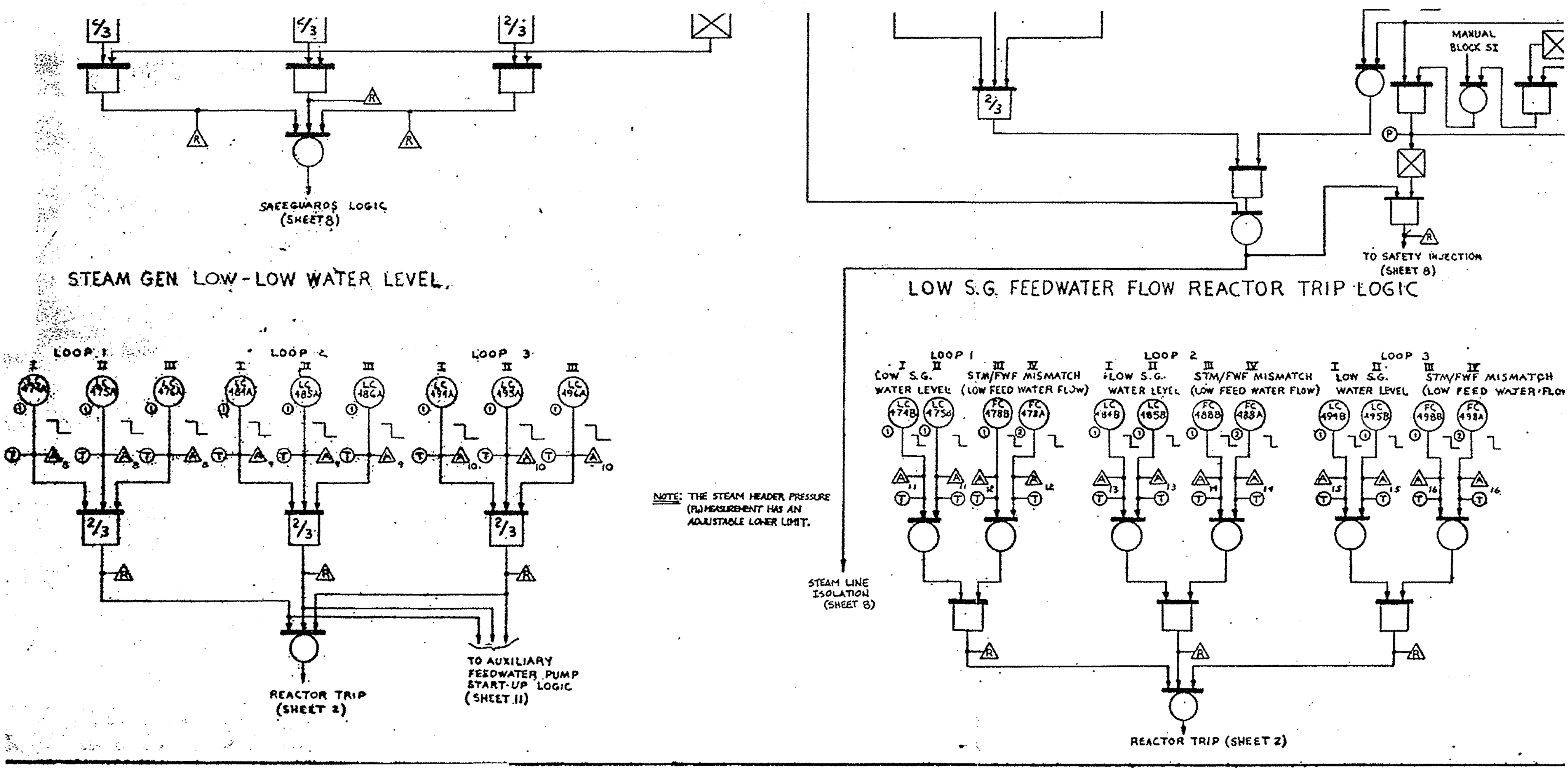


rpsf32

LOW-LOW STEAM GENERATOR LEVEL 16% NARROW RANGE LOGIC

RPS-FIGURE-33

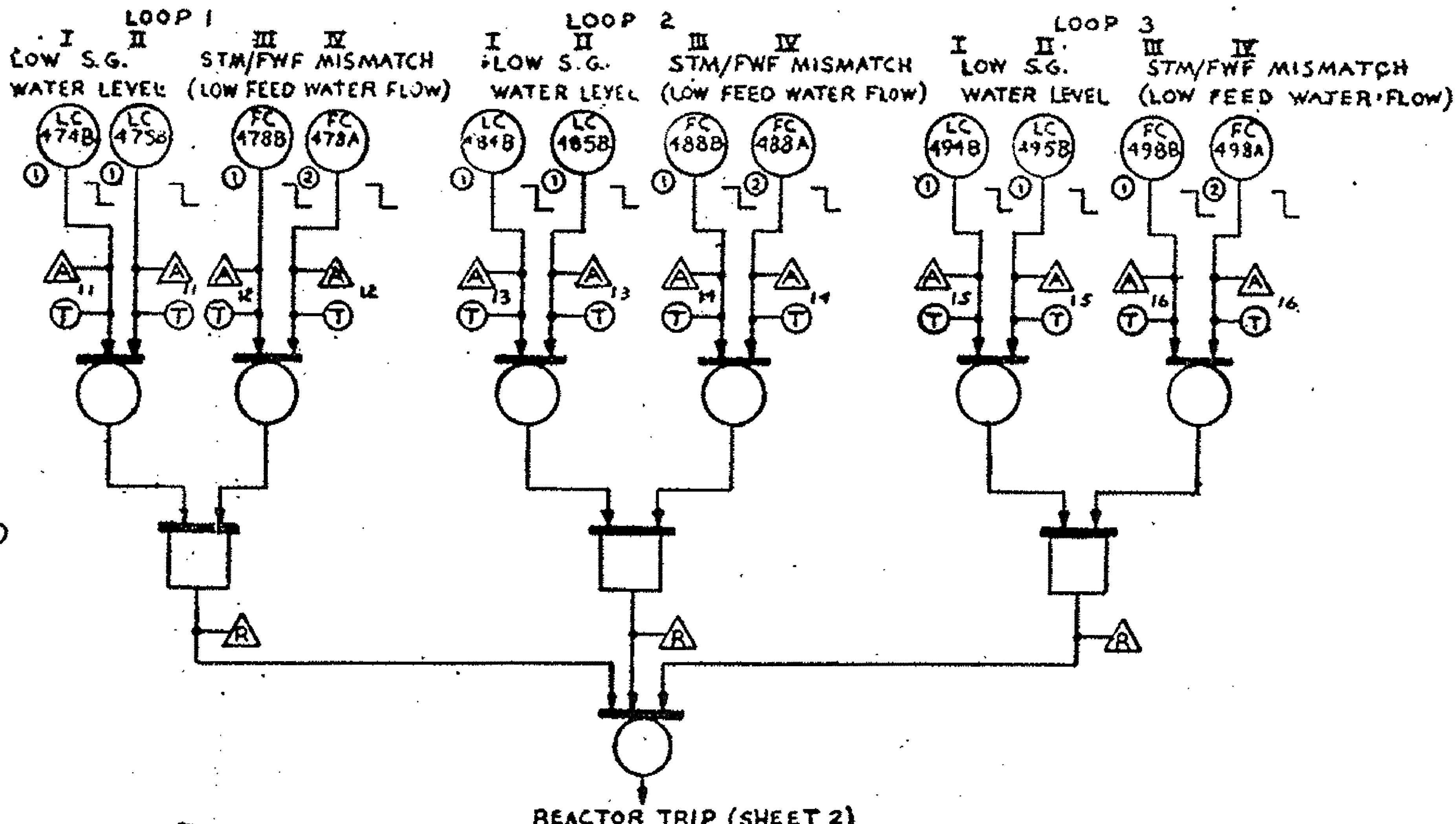




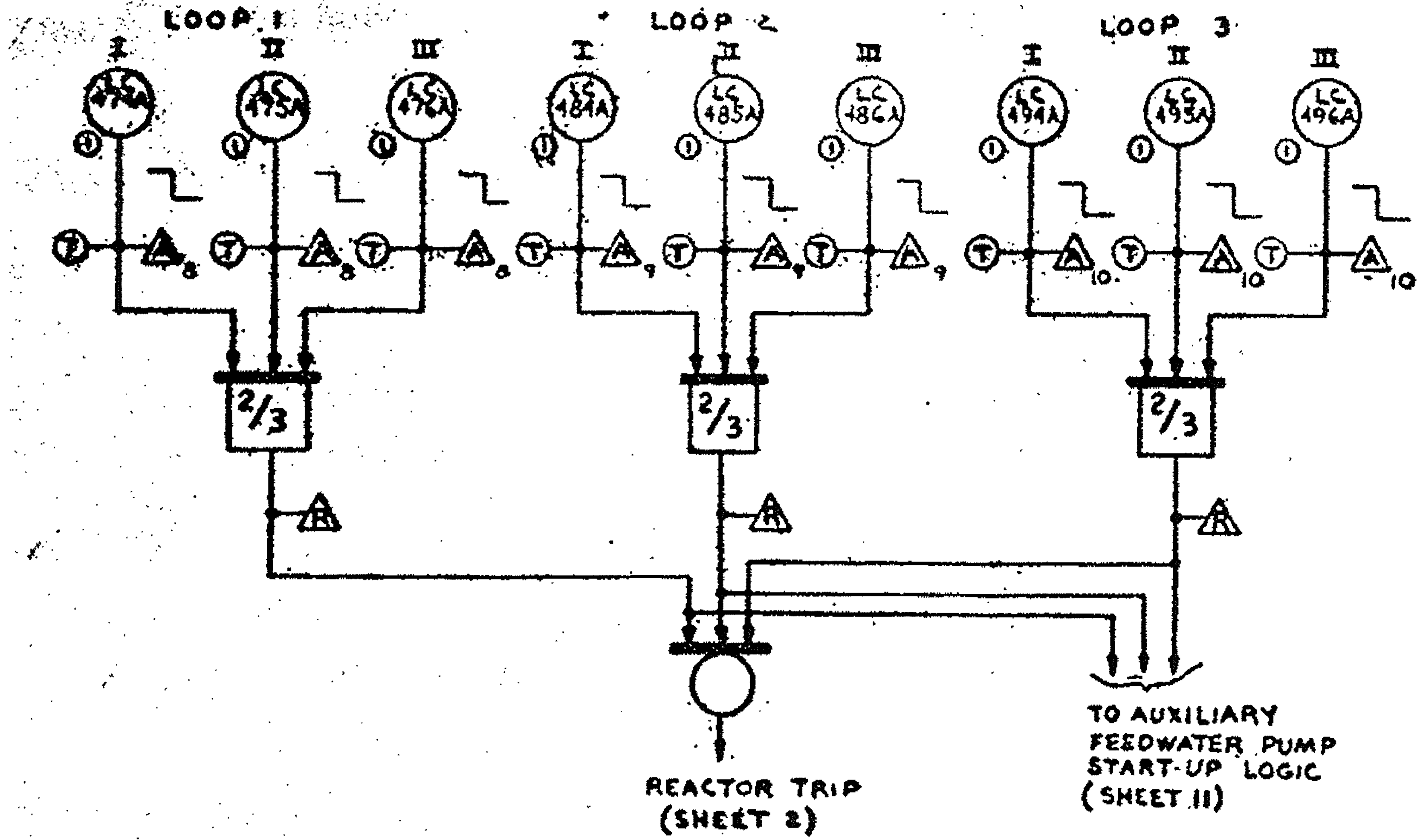
STEAM GEN LOW-LOW WATER LEVEL
SAFEGUARDS LOGIC (SHEET 8)
LOOP 1
LOOP 2
LOOP 3
REACTOR TRIP (SHEET 2)
TO AUXILIARY FEEDWATER PUMP START-UP LOGIC (SHEET 11)
NOTE: THE STEAM HEADER PRESSURE (P_H) MEASUREMENT HAS AN ADJUSTABLE LOWER LIMIT.
LOW S.G. FEEDWATER FLOW REACTOR TRIP LOGIC
MANUAL BLOCK SI
TO SAFETY INJECTION (SHEET 8)
STEAM LINE ISOLATION (SHEET 8)
LOW S.G. WATER LEVEL
STM/FWF MISMATCH (LOW FEED WATER FLOW)
REACTOR TRIP (SHEET 2)

(SHEET 8)

LOW S.G. FEEDWATER FLOW REACTOR TRIP LOGIC



STEAM GEN LOW-LOW WATER LEVEL



40. 013 K5.01 001

Containment Spray Actuation requires __(1)__ channels in __(2)__ sets (or trains) of logic.

	1	2
A.	2 of 3	1 of 2
B.✓	2 of 3	2 of 2
C.	2 of 4	1 of 2
D.	2 of 4	2 of 2

The correct answer is B.

A Train is defined by 3 detectors for Spray Actuation. 2 sets of logic (6 detectors total) will actuate spray.

A. Incorrect. Need 2 of 2 sets (trains) of logic.

B. Correct.

C. Incorrect. There are 3 CV Pressure detectors per train for HI-HI Pressure. There are 4 CV Pressure detectors, one of which is used for extended CV pressure range.

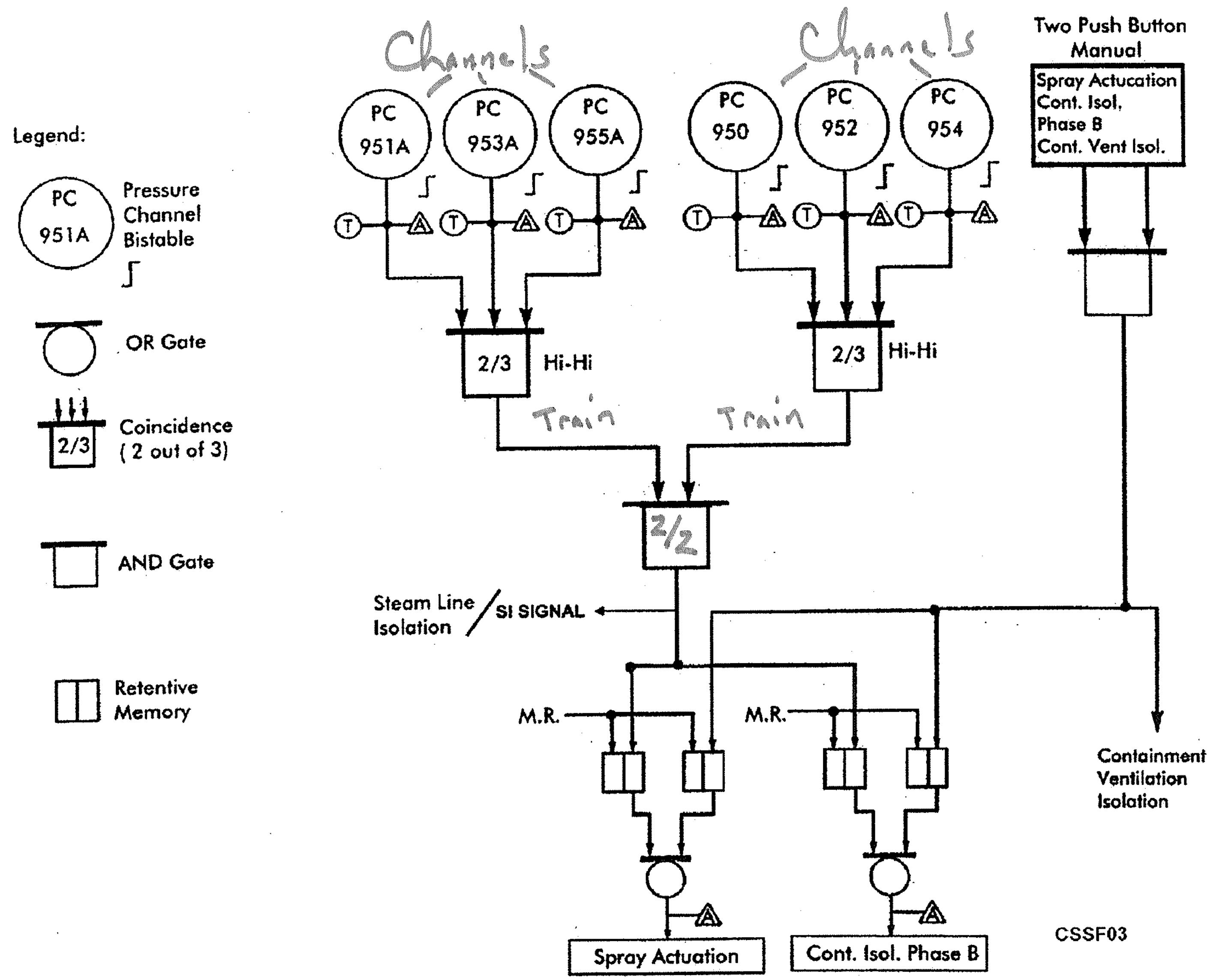
D. Incorrect. There are 3 CV Pressure detectors per train for HI-HI Pressure.

Question 40
Tier 2 / Group 1
K/A Importance Rating - RO 2.8 SRO 3.2

Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Definitions of safety train and ESF channel.

Reference(s) - Sim/Plant design, System Description, Logic Diagrams, TS 3.3.2
Proposed References to be provided to applicants during examination - None
Learning Objective - ESF 004
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 45.7
Comments - K/A met because candidate must know the coincidence (channels and trains) associated with Containment Pressure instruments that are required for an Automatic Containment Spray Actuation.

LOGIC DIAGRAM ON SPRAY ACTUATION
CSS-FIGURE-3



INFORMATION USE ONLY

4.3.2 Containment Spray Automatic Signal Actions

The actions caused by a Containment Spray Automatic signal are listed below:
1. Spray actuation
2. Phase "B" containment isolation
3. Steam line isolation

4.3.3 Containment Spray Manual Signal Actions

The actions caused by a Containment Spray Manual Signal are listed below:
1. Spray actuation
2. Phase "B" containment isolation
3. C.V. ventilation isolation

4.4 Safety Injection and Containment Spray Setpoints

4.4.1 Safety Injection Signal

1. High steam line flow (1/2 per line and 2/3 lines) differential pressure corresponding to 37.25% flow at no load to 20% load and increases linearly to 109% at full load. Load is a function of turbine first stage pressure.
   Low steam line pressure (2/3 lines) 614 psig
   Low T_{avg} - (2/3 loops) 543°F

2. High steam line differential pressure. (2/3 per line, 1/3 Lines)
   (P_{Header} - P_{Line}) 100 psid
   P_{Header} has a low limit pressure setpoint which prevents this signal from decreasing below 585 psig.

3. Low pressurizer pressure (2/3 pressures)
   Pressurizer pressure 1715 psig

4. Containment high pressure (2/3) 4 psig

5. Manual (1/2) either pushbutton

6. Containment Hi-Hi Pressure (2/3 on both trains) 10 psig

4.4.2 Containment Spray Signal

1. Containment Hi-Hi Pressure (2/3 on both trains) 10 psig

4.2.2 Spray Header Flow "A", FT-958A and "B", FT-958B

Spray Header "A" and "B" RTGB flow indication with a range of 0-1500 gpm.

4.3 CV Pressure

NOTE: The CV pressure transmitters are not part of the Spray system but are listed for information. (See SD-006, Engineered Safety Features)

There are nine (9) transmitters located in the Aux. Bldg. near the IVSW tank area. Three are used for the (2 out of 3) HI pressure SI signal at 4 psig (PC-951B, 953B, & 955B). Six supply the HI-HI pressure signal actuation at 10 psig (PC-950, 951A, 952, 953A, 954 & 955A). (CSS-Figure-3) (**NOTE:** 2 groups with 3 transmitters each, 2/3 transmitters from 2/2 groups generates the HI-HI signal.) Places to sense pressure in the CV include the High, Mid and Low points. (Reference P&ID HBR2-6490).

There is one narrow range pressure transmitter that is used for RTGB indication and alarm, PI-950B. There are two Wide Range Accident channels that are used for indication, PI-956 & 957 and are located on the Core Cooling Monitor Panels.

4.4 Local Instrumentation

There are local pressure indicators on the discharge of Spray Pumps "A" and "B". There is also a local Spray Pump test line flow indicator.

4.5 Alarms (CSS-Figure-5)

APP-002-C1 FEEDWATER ISO/CV SPRAY OVRD/RESET, This alarm is caused when the Feedwater Isolation or CV Spray Reset switch is placed in the OVRD/RESET position. Loss of automatic functions (manual available)

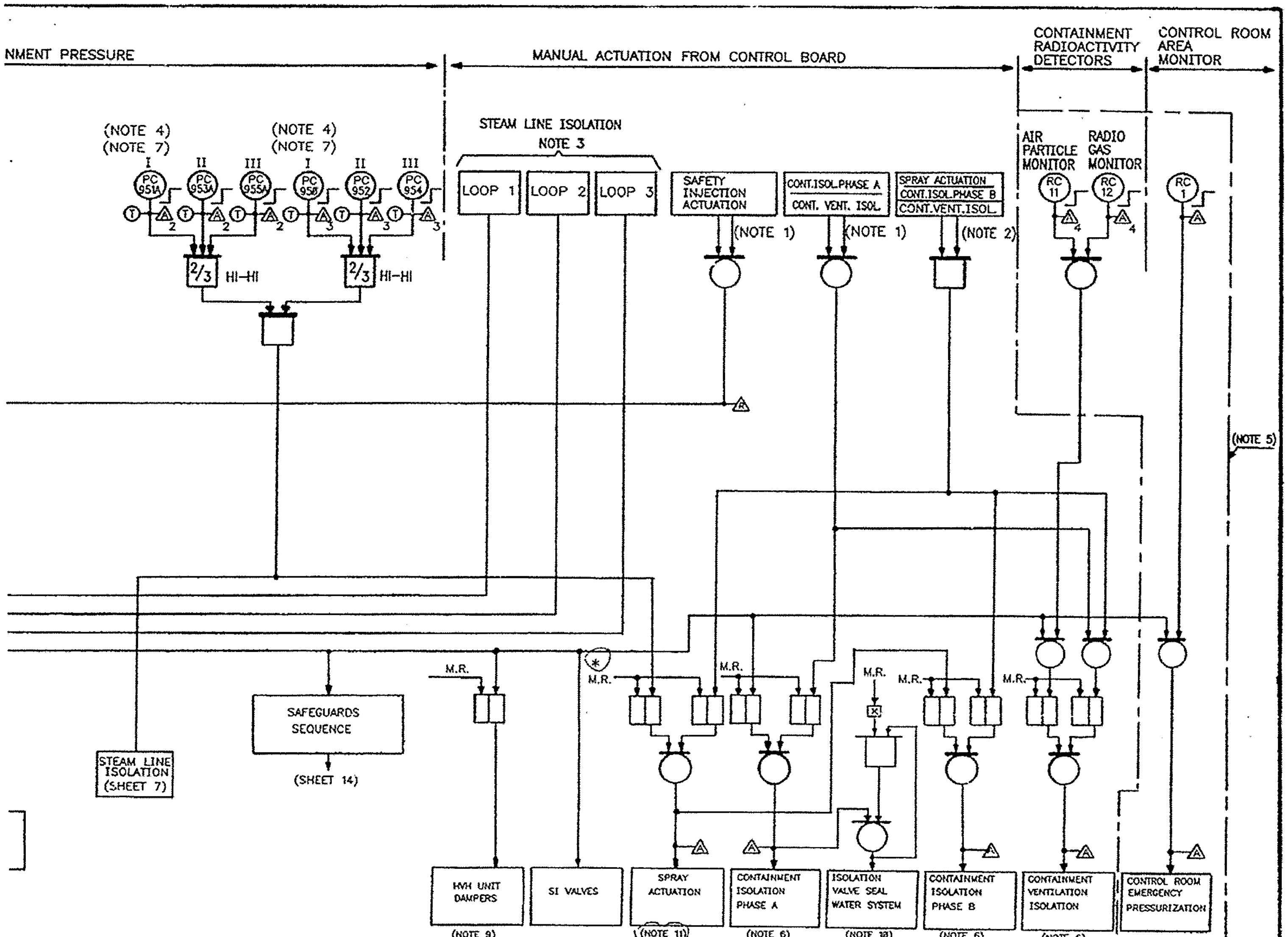
APP-002-D1 SPRAY ACTUATION and **APP-002-D2** CV ISOL PHASE B, Both will alarm at **10 psig** from PC-950, PC-951A, PC-952, PC-953A, PC-954, PC-955A. These alarms come in if 2/3 Hi-Hi Containment Pressure Bistable on both channels or if manual initiation has been actuated by depressing 2 pushbuttons simultaneously.

APP-002-E1 CV SPY PMP COOL WTR LO FLOW, Alarms at **30 gpm** from FIC-657. This alarm is caused by loss of component cooling to the pump (s).

APP-002-F2 SPRAY ADD TANK LO LEVEL, Alarm at **36%** from LC-949. ITS limit is 32% (2505 gallons). The tank should be filled to normal level.

ENGINEERING SAFETY FEATURES

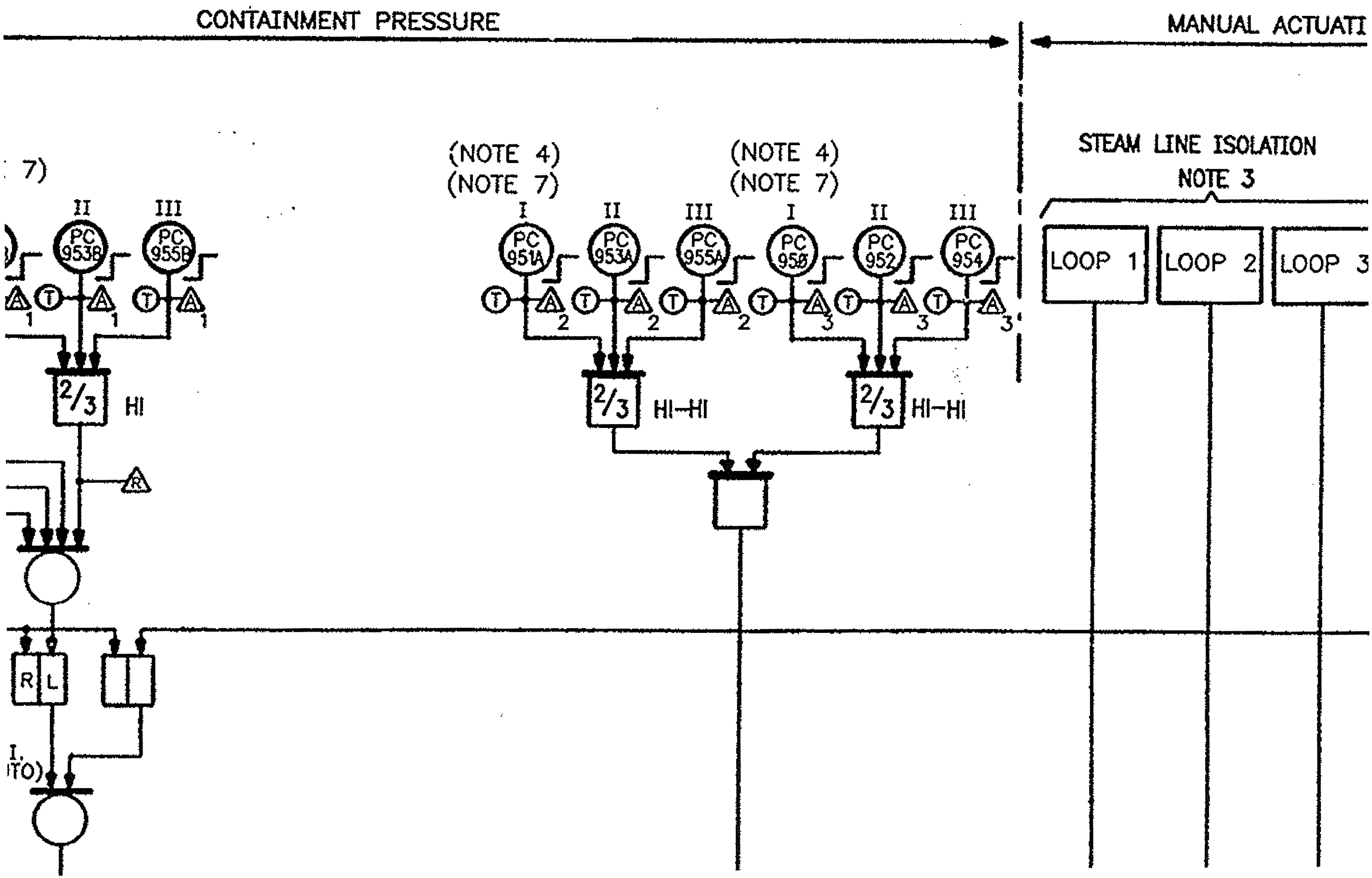
ESF-FIGURE-4



TED BY CONTAINMENT ISOLATION SIGNAL (PHASE A&B) AND
'ED BY CONTAINMENT VENTILATION ISOLATION ARE ALL
D IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL
ESE COMPONENTS TO RETURN TO THE POSITION HELD
NT OF THE ACTUATION SIGNAL.
k PC-951B ARE FED BY THE DIESEL AUTOMATICALLY

9. COMPONENTS ACTUATED BY SI SIGNAL ARE INDIVIDUALLY SEALED IN (LATCHED) SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE POSITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.

10. COMPONENTS ACTUATED BY SI SIGNAL, CONT. VENT. ISOL. AND


CONTAINMENT PRESSURE
MANUAL ACTUATI
STEAM LINE ISOLATION
NOTE 3
(NOTE 4)
(NOTE 7)
(NOTE 4)
(NOTE 7)
7)
PC 953B
PC 955B
PC 951A
PC 953A
PC 955A
PC 950
PC 952
PC 954
2/3
HI
HI–HI
LOOP 1
LOOP 2
LOOP 3
R
L

41. 022 A4.05 001

Given the following Containment temperature indications:

Time	TI-950B	ERFIS Point CVT0001
1100	118.0°F	118.3°F
1200	118.5°F	118.7°F
1300	119.0°F	119.2°F
1400	120.5°F	119.8°F
1500	121.0°F	120.2°F
1600	121.5°F	120.6°F

(RTGB) TI-950B - Containment Average Temperature
(ERFIS) CVT0001 - Containment Average Air Temperature

Which ONE (1) of the following identifies when ITS LCO 3.6.5, Containment Air Temperature, must be entered IAW PLP-118, Hot Weather Operations?

A. 1200

B. 1300

C. 1400

D. ✓ 1500

The correct answer is D.

PLP-118 states that Containment average air temperature on ERFIS (CVT0001) uses the volumetric weighted average developed by Engineering. This value should be used as the "official" temperature. If ERFIS is OOS then the CV average temperature is computed IAW SPP-035, Containment Bulk Average Temperature Measurement. Only if ERFIS is OOS and ITS SR 3.6.5.1 frequency will be exceeded prior to completing SPP-035 would the RTGB meter TI-950B be utilized.

A. Incorrect. At 1200, CVT0001 exceeds 118.5°F. This is the temperature at which deepwell water injection will be aligned to the HVH Service Water IAW SPP-038.

B. Incorrect. At 1300, both temperatures are greater than or equal to 119°F. This is the temperature at which PLP-118 directs the operators to maintain CV Pressure less than 0.1 psig to allow a containment purge to be started in as short a time as possible to provide for additional cooling.

C. Incorrect. At 1400, TI-950B exceed the ITS 3.6.5 CV Temperature limit of 120°F. However, TI-950B is not the official CV Average temperature indication IAW PLP-118.

D. Correct. This is the earliest time in which CVT0001 exceeds 120°F.

Question 41
Tier 2 / Group 1
K/A Importance Rating - RO 3.8 SRO 3.8

Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system.

Reference(s) - Sim/Plant design, System Description, OP-921, APP-002
Proposed References to be provided to applicants during examination - None
Learning Objective - CVHVAC 007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5 to 45.8
Comments - K/A met because candidate is given a set of Containment Temperature readings and must determine when the ITS limit has been exceeded.

Progress Energy

I
Information Use

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 1

PART 2

PLP-118

HOT WEATHER OPERATIONS

REVISION 11

1.0 **PURPOSE**

1.1 Provide guidance concerning the various activities needed to prepare the plant for hot weather.

1.2 Provide general guidance when the area is experiencing drought conditions. Drought conditions that may affect the plant are most likely to occur during hot weather conditions due to the high lake, river, and stream evaporation rates. However, these actions may be implemented at any time of the year.

2.0 **REFERENCES**

2.1 Improved Technical Specification LCO 3.6.5, LCO 3.6.6, LCO 3.7.8 and 5.5.8

2.2 Technical Requirements Manual 5.5.8

2.3 AP-020, Heat Stress Procedure

2.4 AP-010, Housekeeping Instructions

2.5 MMM-006, Calibration Program

2.6 MMM-019, Safety Related Instrument Isolation Valve Line-up Procedure

2.7 OP-101, Reactor Coolant System And Reactor Coolant Pump Startup And Operation

2.8 OP-903, Service Water System

2.9 OP-921, Containment Air Handling

2.10 OMP-003, Shutdown Safety Function Guidelines

2.11 PLP-037, Conduct of Infrequently Performed Tests or Evolutions and Pre-Job Briefs

2.12 SPP-035, Containment Bulk Average Temperature Measurement

2.13 SPP-038, Installation and Removal of Supplemental Cooling for HVH-1, 2, 3, & 4

2.14 OPS-RFPC-00014, Robinson Fossil Plant Impoundment Operation

2.15 AR 91082, RNP Drought Mitigation Plan

2.16 SOER-02-1, Severe Weather

2.17 OP-301-2, Chemical and Volume Control System (Demineralizer Operations)

3.0 RESPONSIBILITIES

3.1 Plant General Manager is responsible for:

- Ensuring adequate resources are allocated for the testing, maintenance, and operation of Unit 2 equipment, components and systems
- Ensuring the implementation of this procedure during warm weather conditions each year. (Discretionary implementation of this procedure is allowed.)
- Ensuring implementation of drought mitigation activities when necessary

3.2 Operations is responsible for ensuring the requirements of this procedure are met.

4.0 PREREQUISITES

N/A

5.0 PRECAUTIONS AND LIMITATIONS

5.1 Two Containment Fan Coolers are required when Reactor Coolant Pump(s) are running with RCS temperature greater than 140°F. The intent of this requirement is to maintain RCP motor winding temperature less than 248°F.

5.2 **IF** containment temperature exceeds 120°F, **THEN** the REQUIRED ACTIONS of Improved Technical Specification LCO 3.6.5 shall be entered.

5.3 In accordance with OMP-003, during Modes 5 **AND** Mode 6 with cavity level less than 23 feet - 6 inches (<23'-6") and fuel in the CV, an HVH unit **AND** a Service Water Booster Pump with normal and emergency power must be AVAILABLE during the following plant conditions:

- CV Equipment Hatch removed **OR** in the process of being removed /installed
- RCS at reduced inventory

8.1.5 The following is applicable to Containment temperature:

NOTE:	The Containment average air temperature on ERFIS (CVT0001) uses the volumetric weighted average developed by Engineering (RNP-M/HVAC-1061). This value should be used as the "official" temperature. If ERFIS is out of service and ITS SR 3.6.5.1 frequency will be exceeded, then containment temperature should be obtained in the following order of preference and used as the "official" temperature: - Perform SPP-035 - Use the RTGB meter - Make a containment entry to obtain temperature readings

1. **IF** containment temperature exceeds 115 degrees, **THEN EVALUATE** the need to initiate the installation of supplemental cooling for HVH-1, 2, 3, & 4 per SPP-038.

NOTE:	The following action places Deep Well Pump A in an unavailable status, so due consideration of plant risk is necessary.

2. **IF** containment temperature is greater than or equal to 118.5°F, **THEN COORDINATE** with maintenance and OSU to begin deepwell water injection into the HVH Service Water IAW SPP-038.

NOTE:	A release permit for a containment purge should be ready to be issued in the event containment temperature exceeds 120°F and is **NOT** expected to decrease below 120°F in the next 8 hours.

3. **IF** containment temperature is greater than or equal to 119°F, **THEN ATTEMPT** to maintain containment pressure less than 0.1 psig by performing pressure releases as necessary to allow a containment purge to be started in as short a time as possible to provide for additional cooling.

8.1.5 (Continued)

4. **IF** containment temperature is greater than or equal to 119.8°F, **THEN NOTIFY** the Manager - Operations, the On-call Manager, **AND** the NRC resident inspector of plant status.

X ~~5.~~ **IF** containment temperature exceeds 120°F, **THEN ENTER** REQUIRED ACTIONS of Improved Technical Specification LCO 3.6.5.

NOTE:	The following step requires R-11 **AND** R-12 to be operable. Use of the "Refuel" position as directed by OP-921 may help to improve containment cooling by removing hot air from the area near the top of the refueling cavity.

a. **IF** containment temperature is **NOT** expected to be restored to less than 120°F within 8 hours, **THEN INITIATE** a containment purge to attempt to maintain temperature at 120°F or less.

NOTE:	ERFIS point CVT0001 shall **NOT** be used as the "official" temperature until one hour after the containment purge has been secured to allow temperatures to stabilize.

b. **IF** a containment purge was initiated, **THEN STOP** the containment purge in time to allow for a one hour stabilization period for a temperature check prior to reaching the end of the 8 hour action statement.

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	24 hours

42. 026 K2.01 001

Given the following plant conditions:

- Plant is at 100% RTP.
- Breaker 52/17, START-UP TRANSFORMER TO 4KV BUS 3, trips.
- A Large Break LOCA occurs 20 seconds later.

Which ONE (1) of the following identifies the configuration of the power supplies to CV Spray Pumps "A" and "B"?

CV Spray Pump "A" powered from __(1)__ and CV Spray Pump "B" powered from __(2)__.

A.✔ (1) SUT

(2) "B" EDG

B. (1) SUT

(2) SUT

C. (1) "A" EDG

(2) "B" EDG

D. (1) "A" EDG

(2) SUT

The correct answer is A.

A. Correct. When breaker 52/17 trips open, 4KV Bus 3 is de-energized which de-energizes 480V Bus E-2, "B" EDG starts from the 480V Bus E-2 undervoltage and re-energizes the bus. The Large Break LOCA results in a reactor trip, safety injection and generator lockout. Since no failure occurred with breakers 52/12 or 52/7 and the SUT, 480V Bus E-1 is supplied from offsite power via the Startup Transformer.

B. Incorrect. First part of answer is correct. Second half of answer is incorrect since breaker 52/17 tripped prior to the plant trip and LOCA.

C. Incorrect. "A" EDG will receive a start signal due to the safety injection signal. However, since the SUT is supplying off-site power to 4 KV Bus 2 the power to 480V Bus E-1 will not be provided by "A" EDG. "A" EDG will be operating unloaded. The second half of the answer is correct.

D. Incorrect. Both parts of the answer are incorrect. The answer is plausible if the candidate is confused and thinks that 4 KV BUS 3 provides power to 480V Bus E-1.

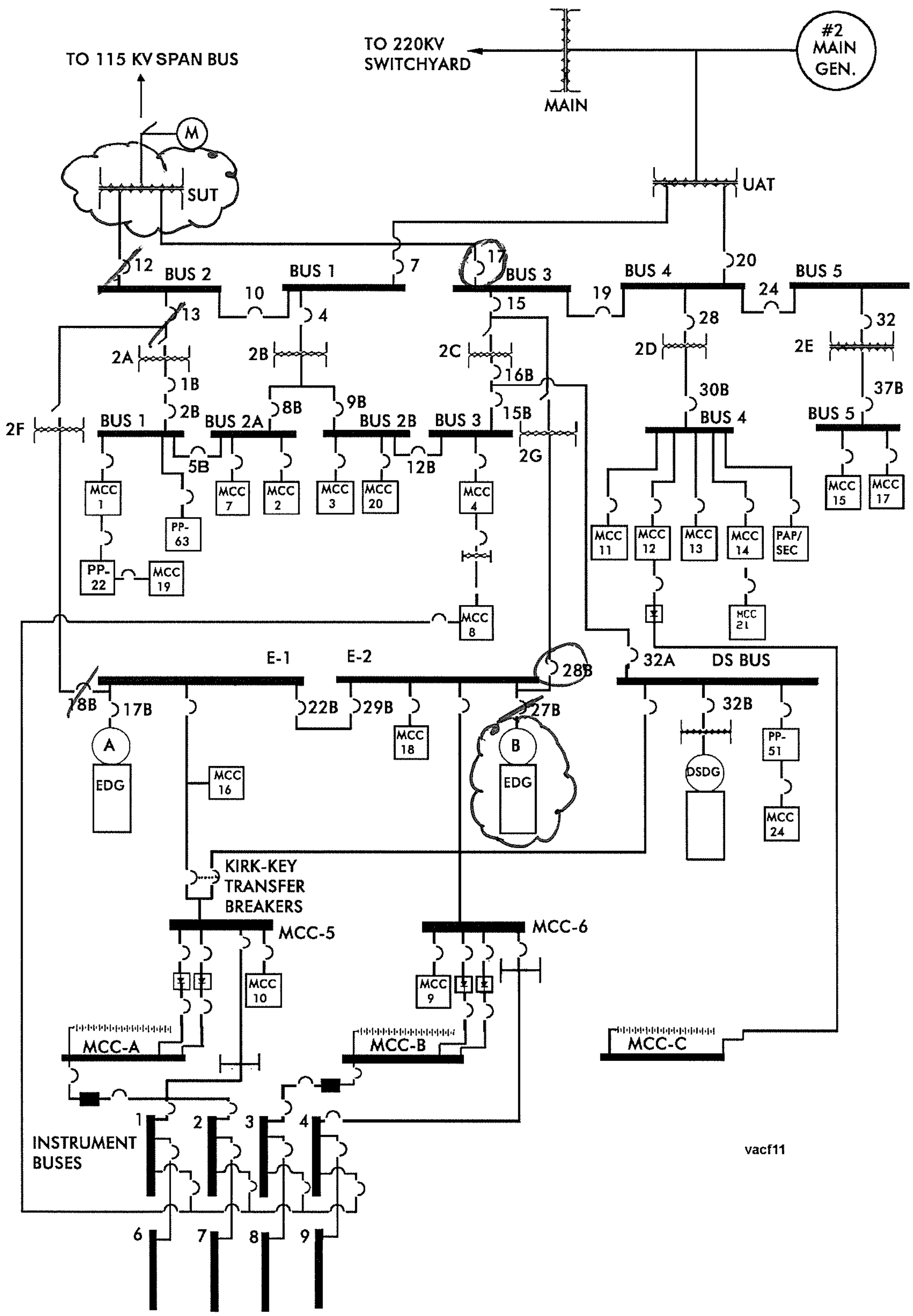
Question 42
Tier 2 / Group 1
K/A Importance Rating - RO 3.4 SRO 3.6

Knowledge of bus power supplies to the following: Containment spray pumps

Reference(s) - Sim/Plant design, System Description, EDP-002
Proposed References to be provided to applicants during examination - None
Learning Objective - CSS 006
Question Source - RNP Bank
Question History - RNP ILC-09 NRC Exam - Modified
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7
Comments - K/A is met because candidate must analyze a given electrical plant transient and LB LOCA and identify the power supplies aligned to provide power.

PLANT AC DISTRIBUTION

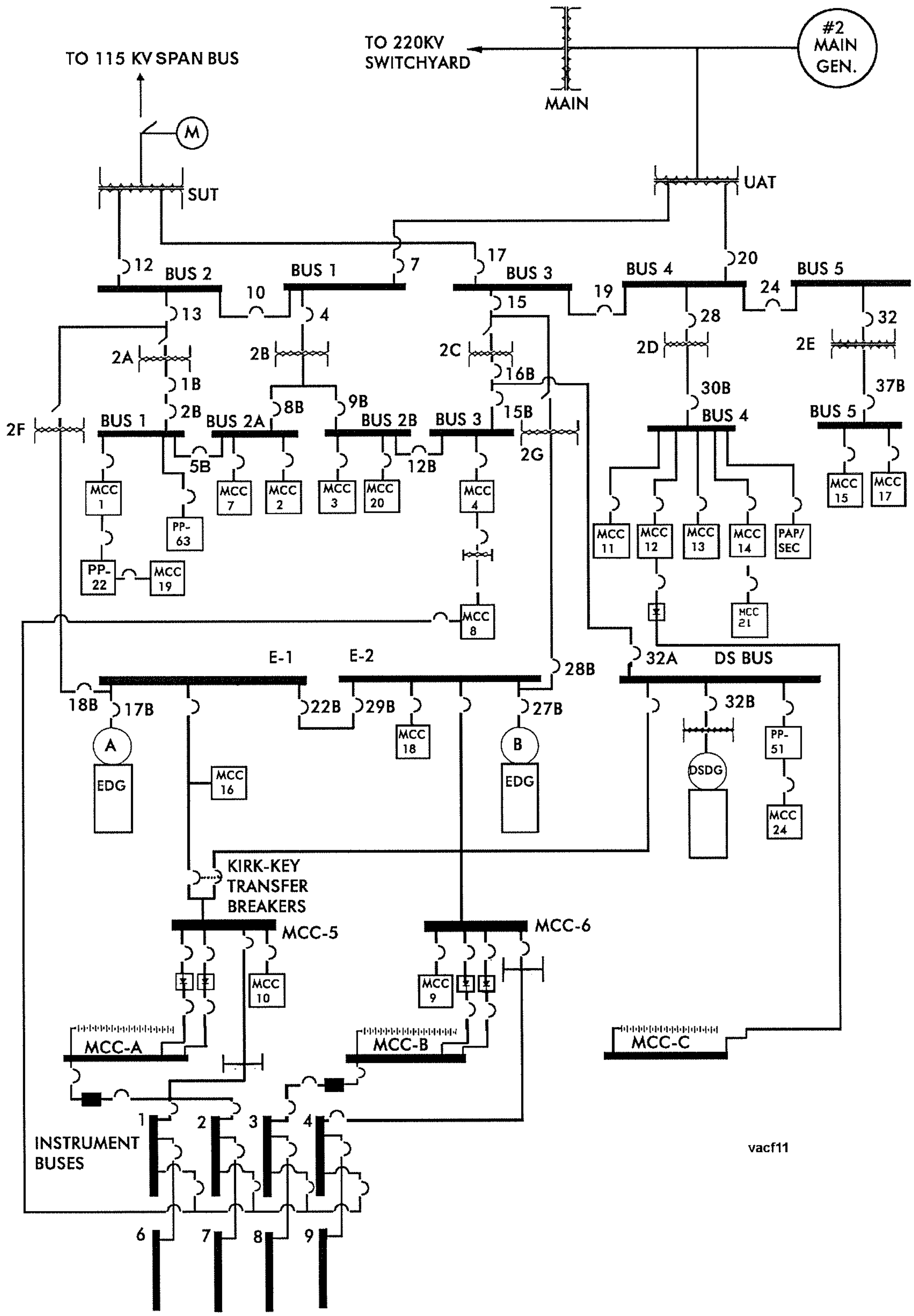
VAC-FIGURE-11



INFORMATION USE ONLY

PLANT AC DISTRIBUTION

VAC-FIGURE-11



INFORMATION USE ONLY

8.0 480V-E1

CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
480V-E1 POWER SUPPLY: NORMAL - 4160V BUS 2 (52/13)LOCATION: E-1/E-2 ROOM			
17A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
17B	**EMERGENCY DIESEL GENERATOR A TO 480V BUS E-1** 480V-E1	890	52/17B
18A	**PT'S & METERING EQUIPMENT (**)** N/A	N/A	N/A
18B	**STATION SERVICE TRANSFORMER 2F TO 480V BUS E-1** 480V-E1	892	52/18B
19A	**CV SPRAY PUMP A** CV-SPRAY-PMP-A	287	52/19A
19B	**CV RECIRC FAN, HVH-1** HVH-1	511	52/19B
19C	**SERVICE WATER PUMP B** SW-PMP-B	832	52/19C
20A	**AUX FEEDWATER PUMP A** AFW-PMP-A	651	52/20A
20B	**SERVICE WATER PUMP A** SW-PMP-A	831	52/20B
20C	**CV RECIRC FAN, HVH-2** HVH-2	512	52/20C
21A	**FEED TO MCC-5 (NORM POWER) & MCC-16** MCC-5, MCC-16	1187	52/21A
21B	**CHARGING PUMP B** CHG-PMP-B	162B	52/21B
21C	**SAFETY INJECTION PUMP A** SI-PMP-A	237	52/21C
22A	**RESIDUAL HEAT REMOVAL PUMP A** RHR-PMP-A	214	52/22A
22B	**480V BUS E-1 SUPPLY TO SI PUMP B** 480V-E1, E2	891	52/22B
22C	**COMPONENT COOLING WATER PUMP B** CCW-PMP-B	205	52/22C

* Compartment 17A also contains two amp meters, two amp meter switches, one volt meter, one volt meter switch, two undervoltage relays, four overcurrent relays, and two auxiliary relays.

** Compartment 18A also contains eight run time meters, three degraded grid relays, one degraded grid trip signal, three test switches, and one degraded grid voltage switch.

9.0 480V-E2

480V-E2 POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: E-1/E-2 ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
23A	**CHARGING PUMP C** CHG-PMP-C	163B	52/23A
23B	**SAFETY INJECTION PUMP C** SI-PMP-C	239	52/23B
23C	**FEED TO MCC-6** MCC-6	1188	52/23C
24A	**SERVICE WATER PUMP C** SW-PMP-C	833	52/24A
24B	**CV RECIRC FAN, HVH-4** HVH-4	514	52/24B
24C	**AUX FEEDWATER PUMP B** AFW-PMP-B	655	52/24C
25A	**CV RECIRC FAN, HVH-3** HVH-3	513	52/25A
25B	**SERVICE WATER PUMP D (NORMAL SUPPLY)** SW-PMP-D	834B	52/25B
25C	**CV SPRAY PUMP B** CV-SPRAY-PMP-B	290	52/25C
26A	**FEED TO MCC-18** MCC-18	1189	52/26A
26B	**RESIDUAL HEAT REMOVAL PUMP B** RHR-PMP-B	216	52/26B
26C	**COMPONENT COOLING WATER PUMP C** CCW-PMP-C	209	52/26C
27A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
27B	**EMERGENCY DIESEL GENERATOR B TO 480V BUS E-2** 480V-E2	895	52/27B

43. 039 G2.1.7 001

Given the following plant conditions:

- The plant is at 100% RTP.
- The OAO reports a steam leak near the Main Steam Isolation Valves.
- The following indications are noted in the Control Room:
  - Power Range Nuclear Instruments are rising.
  - T_{avg} is lowering.
  - Steam flow and feed flow have risen.
  - Power Limit Warning alarm on ERFIS has been received.
  - Reactor power is 100.3% and slowly rising.

Which ONE(1) of the following identifies both:

(1) the time in core life that will result in the largest reactivity excursion
AND
(2) the required operator actions IAW OMM-001-2, Shift Routines and Operating Practices?

A.✓ (1) EOL

(2) reduce power by using the Valve Position Limiter.

B. (1) EOL

(2) trip the reactor and enter PATH-1.

C. (1) BOL

(2) reduce power by using the Valve Position Limiter.

D. (1) BOL

(2) trip the reactor and enter PATH-1.

The correct answer is A.

A: Correct. The largest reactivity change would occur at EOL due to the moderator temperature coefficient (MTC) being much larger than at BOL. CALO program will provide an audible alarm (Power Warning Limit Alarm) on the ERFIS computer when the program calculates reactor thermal power greater than 2339 MW_t. OMM-001-2 directs the operator to reduce power using the Valve Position Limiter to ensure that thermal power limits are not exceeded. Tave decreasing shows that there is excessive steam flow which causes reactor power to increase due to the MTC.

B: Incorrect. The largest reactivity change would occur at EOL due to the moderator temperature coefficient (MTC) being much larger than at BOL. The immediate operator response would to be to reduce steam demand by using the Valve Position Limiter. If power cannot be controlled then the crew may ultimately decide to trip the reactor and enter PATH-1.

C: Incorrect. At BOL the moderator temperature coefficient (MTC) is smaller (or less negative) than at EOL, and at times it could even be a positive value. The action listed is correct.

D: Incorrect. Both options are incorrect. See discussion above.

Question 43
Tier 2 / Group 1
K/A Importance Rating - RO 4.4 SRO 4.7

Main and Reheat Steam : Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Reference(s) - Sim/Plant design, System Description, OMM-001-2
Proposed References to be provided to applicants during examination - None
Learning Objective - MSS 013, OMM-001-2 LP Objective # 2
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 43.5 / 45.12 / 45.13
Comments - This question matches the K/A by having the candidate determine at which point in core life a steam leak would have the largest operational impact on the plant. The question also has the candidate identify the appropriate actions for the given malfunction.

8.3 Continuous Calorimetric Program

8.3.1 The Continuous Calorimetric Program (CCP) is used to monitor core thermal power during periods of **STEADY STATE** operation. The CCP does not account for changing volumes or changes in stored energy. In addition, averaging routines built into the CCP will require as much as 5 minutes before a change in an input value is fully reflected in the display.

8.3.2 ERFIS simultaneously calculates calorimetrics using FWUFM, Feedwater Flow and Steam Flow. Logic built into the Continuous Calorimetric program (CALO) will cause the calculation to halt and the result to be "0" and labeled NCAL for most situations not allowed by the uncertainty calculations.

- FWUFM CALO should be used when it is available.
- If FWUFM CALO is **NOT** available **AND** a power reduction is **NOT** required IAW TRM 3.25, **THEN** the following are required until FWUFM is restored:
  - Use NIN0001, Average Power Range Percent Power.
  - If NIN0001, Average Power Range Percent Power is **NOT** available, **THEN** use the highest individual one minute average NI Power on ERFIS (NIN0041M, NIN0042M, NIN0043M, **OR** NIN0044M).
  - Maintain less than or equal to 100% reactor power using indication described above.
  - **NO** positive reactivity changes are permitted except for power reductions (adds positive reactivity due to power defect).

8.3.3 Diverse indications of Reactor power; such as NIs, Loop Delta T, Unit load, and Turbine 1st Stage Pressure, should be monitored at all times to ensure the Reactor is operated at less than or equal to 2339 MW thermal when the FWUFM System is in service and 2300 MW thermal when FWUFM is not in service as required by TRM 3.25 limitation. As specified in EST-155, at 100% RTP (2339 MW thermal) the allowed range for ΔT is 57°F to 58°F and the allowed range for Turbine 1st Stage Pressure is 654.33 psi to 668.73 psi. Diverse indications of Reactor Power during transient conditions may be affected by changes in RCS temperature resulting in NI shielding and changes in secondary efficiency affecting Unit Load and Turbine 1st Stage Pressure. Loop Delta T indications are the least affected by RCS temperature changes and secondary efficiency changes. Due to the TR-412 trend display scale and normal fluctuation in the ΔT digital displays of ΔT and ΔT power it may be difficult to identify small changes in power. Steam Flow Calorimetric is an effective tool to validate LEFM response, especially during transient conditions. LEFM increases that are solely a result of increases in feed flow would not be seen as a power increase using the Steam Flow Calorimetric. [CAPR 237401]

8.3.4 During Steady State Operations, reactor power may indicate greater than the allowed RTP for brief periods of time with no operator action required. It is **NOT** intended that this flexibility be used to make up for lost generation when the Period Average falls significantly below Target Power level for extended periods. These periods should be limited as follows:

1. The allowed thermal power with FWUFM in service is 2339MWth. With FWUFM out of service for longer than allowed by TRM 3.25, thermal power will be limited to 2300 MWth. The setpoints at which a power limit warning will be received are as follows: (Ref – RNP2 6004-CALO-SRS-001

   a. The 8 hour average power is >2339 MWt and less than 1 hour remains in the 8 hour period

   b. The 1 minute average power is >2339.35 MWt (100.015%) for 290 minutes continuously.

   c. The 1 minute average power is >2340.75 MWt (100.075%) for 50 minutes continuously.

   d. The 1 minute average power is >2342.50 MWt (100.150%) for 20 minutes continuously.

8.3.4.1 Continued

e. The 1 minute average power is >2346.00 MWt (100.299%) for 5 minutes continuously

f. The 4 minute average power is >2346.00 MWt (100.299%) instantaneously

2. The 4 minute average thermal power indication should not be allowed to consistently exceed the allowed RTP without taking operator action to reduce power. Small fluctuations above and below the allowed thermal power limit are expected for short durations and are part of steady state operations. The magnitude, trend and average of the fluctuations must be evaluated to determine if a power reduction is required.

3. The one hour average thermal power indication should not be allowed to exceed the allowed thermal power. It is recognized that the one hour average thermal power could exceed the allowed thermal power at the beginning of a new 8 hour period. It is not intended that a power reduction be initiated immediately in this condition. The trend should be evaluated and power reductions initiated if required.

8.3.5 During Planned Evolutions, operators should consider reducing power in advance of planned plant evolutions that have a potentially high likelihood of causing power to increase above the licensed power limit. If during the evolution, power increases above the licensed power limit, operators should take action to restore power to or below the licensed limit.

NOTE:	Power level is limited to a maximum of 100.3% (2346 MWth) per LDCR 02-0012, Appendix K Power Uprate.

8.3.6 Reactor power may indicate greater than the 2339 MWth (100%) for brief periods with no operator action required. It is **NOT** intended that this flexibility be used to make up for lost generation when the Period Average falls significantly below Target Power level for extended periods.

8.3.7 Upon receiving any valid calorimetric Power Alarm, Reactor power shall be reduced to less than 2339 MWth, or until the calorimetric Power Alarm is cleared using the Valve Position Limiter. A time/power target will be given. Power must be reduced to less than the target power by the time given to prevent exceeding the limits. The cause of the Time / Power alarm may be determined by clicking on the ALARM INFORMATION button. [CAPR 237401]

8.3.8 No information will be displayed for periods in which values cannot be calculated. For example, If ERFIS is OOS for greater than thirty minutes, no Period Average, Projected Power, or Recommended Power will be calculated. For periods in which ERFIS data is unavailable for less than thirty minutes **AND** the power change for the period ERFIS was OOS is less than 1%, the CCP will generate missing data using linear interpolation. If it is anticipated that the Period Average will **NOT** be calculated, the Reactor should be operated at less than 2339 MW Thermal. When the CCP is OOS, Reactor power shall be maintained less than or equal to 100% as indicated by other diverse indications to ensure Time / Power limits are not exceeded.

8.3.9 Other conditions may exist which effect the validity of the CCP calculation. When these conditions exist, other indications of Reactor power shall be used. Conditions which affect the validity of the CCP calculation include:

- Operation at less than 15% power will cause some inputs to the CCP calculation to be outside reasonability limits.
- If Charging temperature or Letdown flow data is unavailable, the CCP calculation will default to data for operation with only the 60 gpm orifice in service. Operation with greater than 60 gpm letdown flow will cause the calculation to be non-conservative.
- Feeding any S/G with a MDAFW Pump will result in non-conservative results.

44. 039 G2.4.20 001

Given the following plant conditions:

- A Steam Generator Tube Rupture (SGTR) is in progress.
- The crew is in PATH-2 and are commencing the RCS cooldown to target temperature.
- The procedure has directed to "Dump Steam to the condenser from intact S/Gs at the maximum rate".

Which ONE (1) of the following describes the term "Maximum Rate" as specified in the NOTE in PATH-2?

As fast as possible ...

A. while not exceeding the High Steam Line ΔP limit.

B. but less than the limits of the Integrity Critical Safety Function Status Tree limit.

C✓ while keeping steam flow less than the High Steam Flow Main Steamline Isolation setpoint.

D. but less than the cooldown limits of Curve 3.4, Reactor Coolant System Pressure - Temperature Limitations for Cooldown.

The correct answer is C.

Rapid RCS cooldown is expected during this event, cooldown rates will be exceeded. The note provides a warning that the Main Steam Line Isolation actuation will still occur if High Steam Line flow setpoints are exceeded during the RCS rapid cooldown.

A. Incorrect. SI initiation will occur from High Steam Line Delta P signal during normal conditions. During this evolution in PATH-2, SI has already been blocked by the Low PZR Pressure / High Steam Line Delta P manual block switch. Additionally, MSIV isolation does NOT occur from a High Steam Line Delta P signal.

B. Incorrect. The cooldown will be performed at a maximum rate with an attempt to ensure that a MSIV Isolation signal is not received. The cooldown is a high priority.

C. Correct. MSIVs will isolate if the setpoint is exceeded, potentially stopping the cooldown if the crew is using Condenser Steam Dumps.

D. Incorrect. The cooldown will be performed at a maximum rate with an attempt to ensure that a MSIV Isolation signal is not received. If the MSIVs are closed then steam will be dumped at maximum rate using the S/G PORVs at maximum rate with no regard to normal cooldown rates.

Question 44
Tier 2 / Group 1
K/A Importance Rating - RO 3.8 SRO 4.3

Main and Reheat Steam : Knowledge of the operational implications of EOP warnings, cautions, and notes.

Reference(s) - Sim/Plant design, System Description, PATH-2, Grid locations D-2 and D-3; PATH-2 BD, Pages 69-70.
Proposed References to be provided to applicants during examination - None
Learning Objective - PATH-2-003
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13
Comments - K/A is met because the candidate must know the basis for the note in PATH-2 that provides a warning that the Main Steam Line Isolation actuation will still occur if High Steam Line flow setpoints are exceeded during the RCS rapid cooldown.

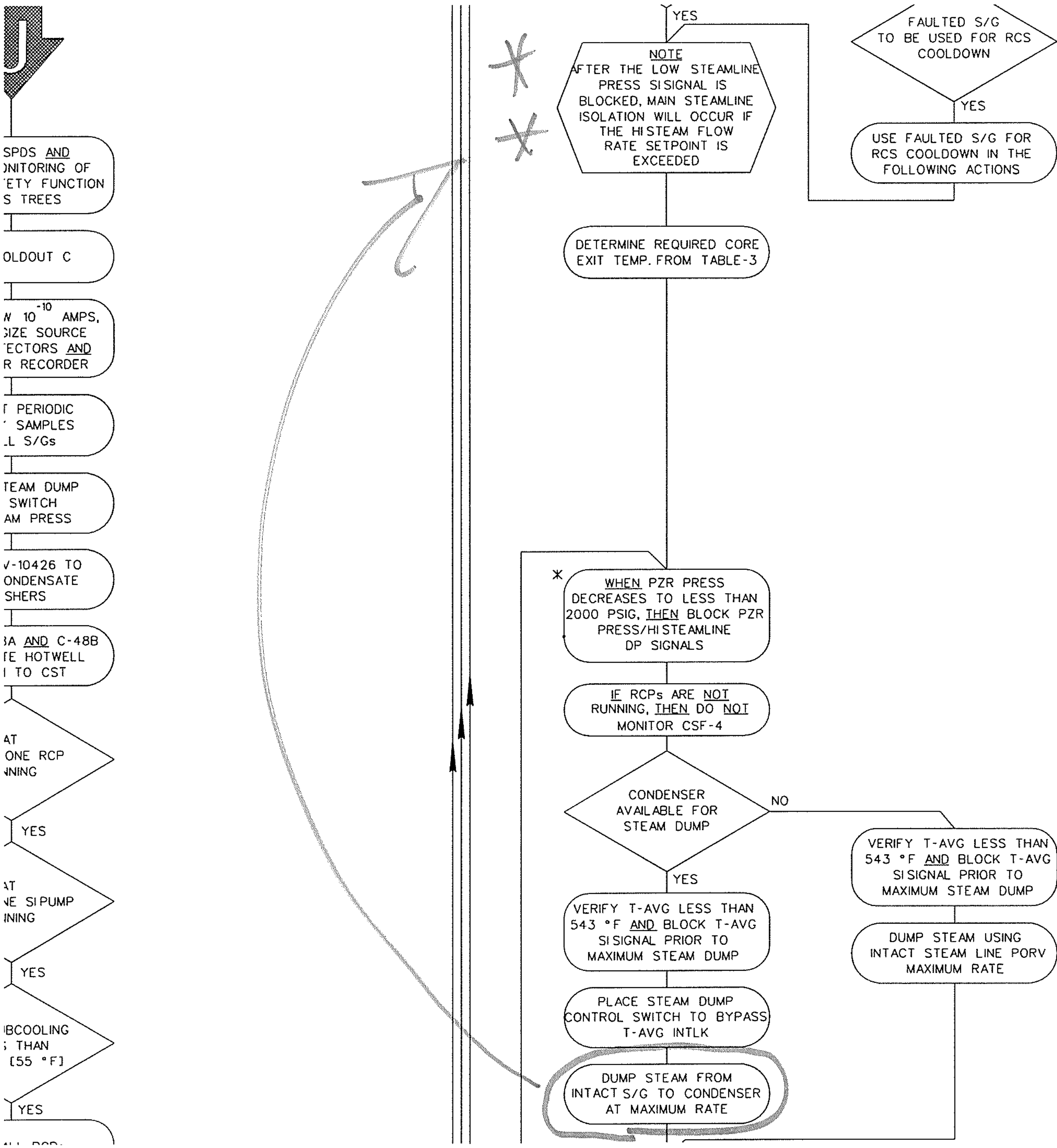


TABLE-3

RUPTURED S/G PRESS (PSIG)	REQUIRED CORE EXIT TEMP (°F)
GREATER THAN 1000	490 [470]
900 - 1000	480 [460]
800 - 899	465 [445]
700 - 799	450 [430]
600 - 699	435 [415]
500 - 599	415 [395]
400 - 499	395 [375]
300 - 399	365 [345]
250 - 299	340 [320]

RNP STEP | WOG STEP | BASIS/DIFFERENCES

WOG BASIS (continued)

KNOWLEDGE:

- It is not intended for the operator to reevaluate the required core exit temperature or precisely interpolate between values listed in the table.
- When the required core exit temperature is reached, the intact steam generator pressure (or feed flow to a faulted steam generator) should be controlled to maintain that temperature.
- Cooldown of the RCS should be completed before continuing in the guideline.
- Natural circulation flow in the ruptured loops may stagnate during this cooldown. The hot leg temperature in that loop may remain significantly greater than the intact loops. In addition, safety injection flow into the cold leg may cause the cold leg fluid temperature to decrease rapidly in that same loop. Steps to depressurize the RCS and terminate SI should be performed as quickly as possible after the cooldown has been completed to minimize possible pressurized-thermal shock of the reactor vessel.
- RCS cooldown should proceed as quickly as possible and should not be limited by the 100°F/hr Technical Specification limit. Integrity limits should not be exceeded since the final temperature will remain above 350°F.
- The RCP trip criteria (Step 1) does not apply after a controlled cooldown is initiated.
- If more than one steam generator is ruptured, the lowest ruptured steam generator pressure should be used to determine the required core exit temperature. If cooldown to a target core exit temperature is already in progress when a subsequent SGTR is diagnosed the operator should stop the cooldown until the subsequent ruptured steam generator is isolated since continuing the cooldown would lower the pressure in the newest ruptured steam generator and result in unnecessary releases prior to its isolation from the intact steam generators. The target core exit temperature should be reexamined to determine if the temperature should be reduced based on the subsequent ruptured steam generator pressure. If a RCS depressurization is in progress, although it does not impact the pressure in the newest ruptured steam generator, for the sake of simplicity it should be stopped and the plant stabilized by the operator until the newest ruptured steam generator is isolated.

RNP DIFFERENCES/REASONS

There are no significant differences.

SSD DETERMINATION

This is not an SSD.

D2 | 2N6 | RNP STEP

NOTE: AFTER THE LOW STEAMLINE PRESS SI SIGNAL IS BLOCKED, MAIN STEAM ISOLATION WILL OCCUR IF THE HI STEAM FLOW RATE SETPOINT IS EXCEEDED

RNP STEP	WOG STEP	BASIS/DIFFERENCES

WOG BASIS

PURPOSE: To alert the operator to the potential for inadvertent steamline isolation during the subsequent steam generator depressurization.

BASIS:

An automatic protection feature is provided to close the main steamline isolation valves when the steam pressure rate signal is exceeded. In the following step, the operator is instructed to dump steam from the intact steam generators which may result in exceeding the rate setpoint. Therefore, this note is intended to alert the operator of this possibility.

KNOWLEDGE:

The rapid cooldown should be continued using the atmospheric steam dumps if MSIV closure occurs.

RNP DIFFERENCES/REASONS

The RNP note is worded in the form of a high flow instead of a pressure rate of decrease to reflect the RNP signal. There are no significant differences.

SSD DETERMINATION

This is not an SSD.

D2 6 RNP STEP

DETERMINE THE REQUIRED CORE EXIT TEMP

WOG BASIS

See above

RNP DIFFERENCES/REASONS

There are no significant differences.

SSD DETERMINATION

This is not an SSD.

D3 1N6 RNP STEP

WHEN PZR PRESS DECREASES TO LESS THAN 2000 PSIG, THEN BLOCK PZR PRESS/HI STEAMLINE DP SIGNALS

45. 059 A3.02 001

Given the following plant conditions:

- The plant is at 60% RTP.

Which ONE (1) of the following describes the expected S/G Narrow Range Levels and the basis for that level?

The S/Gs will indicate ____(1)____ and the basis for this level is to ____(2)____.

A. (1) 45%

(2) maintain sufficient level to prevent a reactor trip from a 50% load rejection and is low enough to minimize moisture carryover.

B. (1) 45%

(2) maintain the mass in the S/G great enough to stay above the reactor trip setpoints and minimizes the consequences of a steam break accident.

C.✓ (1) 52%

(2) maintain sufficient level to prevent a reactor trip from a 50% load rejection and is low enough to minimize moisture carryover.

D. (1) 52%

(2) maintain the mass in the S/G great enough to stay above the reactor trip setpoints and minimizes the consequences of a steam break accident.

Correct answer is C.

A. Incorrect. The level should be 52%. S/G levels are controlled 39 to 52% from 0 to 20% power and 52% from 20% to 100%. 45% level would be correct if the S/G levels were controlled from 39 to 52% from 20% to 100%. The basis for the level at this power level is correct.

B. Incorrect. The level should be 52%. S/G levels are controlled 39 to 52% from 0 to 20% power and 52% from 20% to 100%. 45% level would be correct if the S/G levels were controlled from 39 to 52% from 20% to 100%. The basis given in this answer is for the no load steam generator level of 39%.

C. Correct.

D. Incorrect. The level is correct. The basis given in this answer is for the no load steam generator level of 39%.

Question 45
Tier 2 / Group 1
K/A Importance Rating - RO 2.9 SRO 3.1

Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G

Reference(s) - Sim/Plant design, System Description,
Proposed References to be provided to applicants during examination - None
Learning Objective - FW 007
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5
Comments - K/A met because candidate know the expected S/G programmed level for a given power level and the basis for this level.

The auto/manual selector switch on the side of the Bailey positioner at the feedwater regulating valves cannot be used for local pneumatic valve control. If manual is selected, the valve will fail closed.

A wide-range level channel, calibrated for no-load conditions, aids manual level control from hot shutdown to cold shutdown. This channel consists of a recorder, high and low level alarms (only function when steam generator pressure is less than 614 psig), and indicators. Automatic pressure-temperature compensation is not necessary.

Besides the main feedwater regulating valve (FCV-478, -488, or -498), for each steam generator there is a bypass valve (FCV-479, -489, or -499) which is intended to provide manual feedwater flowrate control at low loads. During normal "at power" operation of the plant the bypass valve is closed.

The bypass valve operation is controlled from the RTGB controller station which provides manual positioning of the valve. The opening of the bypass valve is prevented in the presence of either safety injection or high steam generator water level.

The auto/manual selector switch on the side of the Bailey positioner at the feedwater regulating bypass valves can be used for local pneumatic control.

The main feedwater regulating valves and bypasses are operated by utilizing the instrument air system pressure through controllers to properly position the valves.

The main feedwater regulating valves and the bypass regulating valves have OPEN and CLOSED light indications. These position indication lights are located on the RTGB adjacent to their respective controllers.

The main feedwater regulating valves rely on motor operated block valves if needed to isolate them from the main feedwater pump discharge. These valves are controlled from the RTGB and supplied power from: FW-V2-6A for "A" steam generator from MCC-5, FW-V2-6B for "B" steam generator from MCC-6, and FW-V2-6C for "C" steam generator from MCC-6.

5.1.2 Steam Generator Water Level Control

Steam generator level is programmed for operation. The normal no load level is 39% and is programmed from 39% to 52% from 0% to 20% power. From 20% to 100% power the level remains at 52%.

The three-element feedwater control system compares actual steam generator level to the program level derived from turbine first stage pressure (for power level, selected from either PT-446 or PT-447) and any difference between the signals is the level error.

The pressure compensated steam flow signal and the feedwater flow signal are compared

and produce a flow error signal. The two error signals are then used to control the position of the main feedwater regulating valves.

For normal operation, the dominate signal for controlling the valves is the level error. For large transients the flow error will be most significant.

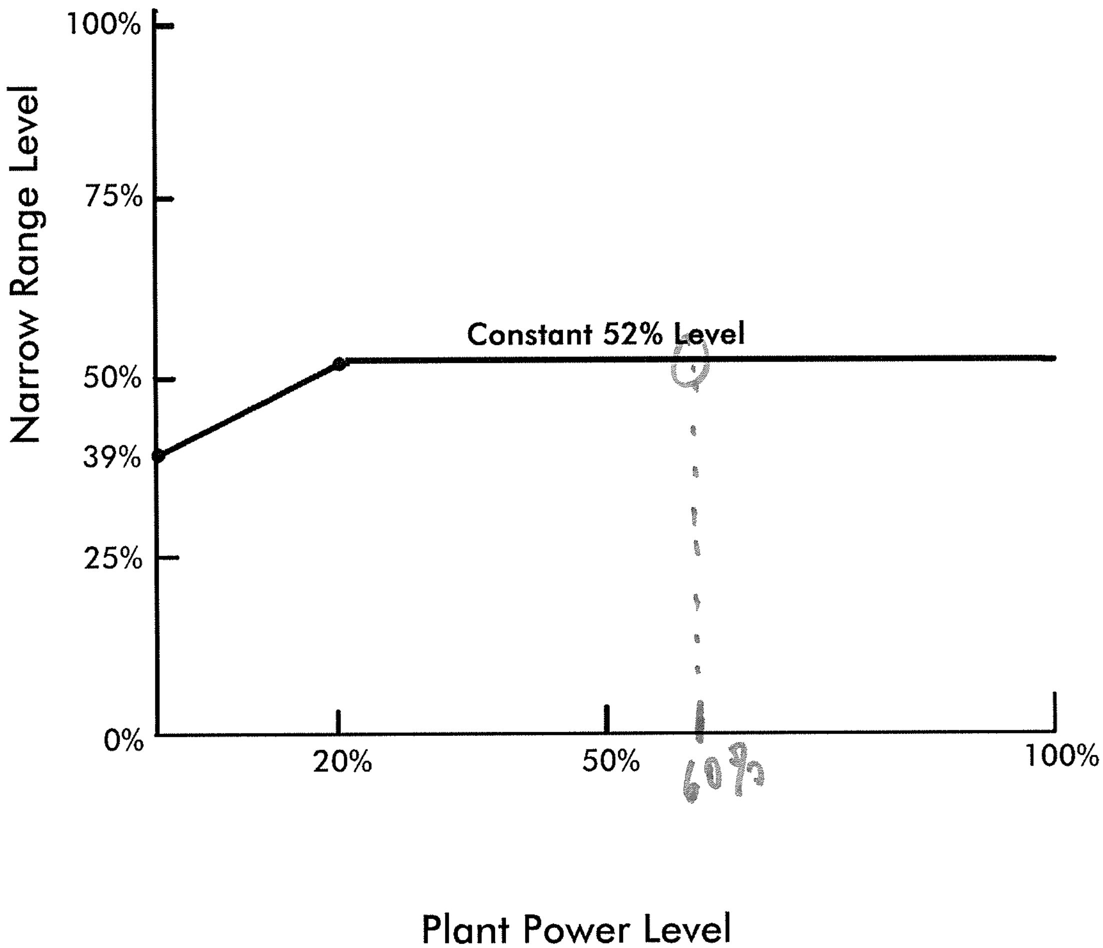
The feedwater control system can only control the valves when in automatic. If the valves are placed in manual, then the manual pushbuttons are used in controlling valve position. These controllers are located on the RTGB. Also located on the RTGB are the controls for bypass valves. These are operated manually from potentiometers that indicate from 0% to 100%.

The no load steam generator level (39%) is to maintain the mass in the steam generator great enough to stay above the reactor trip setpoints and minimizes the consequences of a steam break accident.

The normal at power (greater than 20%) steam generator level (52%) is provided to maintain sufficient level to prevent a reactor trip from a 50% load rejection and is low enough to minimize moisture carryover.

RAMPED STEAM GENERATOR LEVEL PROGRAM

SG-FIGURE-12



sgf12

46. 061 K5.01 001

"B" MFP is OOS for Maintenance and the following occurs:

- The Reactor was manually Tripped while operating at 20% RTP due to a trip of "A" MFP
- PZR Level is 20% and slowly lowering.
- Tave is 545°F and lowering.
- RCS Pressure is 2050 psig and lowering.
- Steam Generator Blowdown is Isolated.
- S/G levels are as follows:
    - "A" S/G Narrow Range level is 40% and rising.
    - "B" S/G Narrow Range level is 41% and rising.
    - "C" S/G Narrow Range level is 45% and rising.

Which ONE (1) of the following provides the actions that are required to be taken next IAW EPP-4, Reactor Trip Response?

A. Initiate Safety Injection.

B. Borate to Cold Shutdown Boron.

C.✓ Reduce Auxiliary Feedwater Flow.

D. Close the MSIVs and MSIV Bypasses.

The correct answer is C.

A. Incorrect - Safety Injection initiation criteria have not been met. PZR is greater than 20%. PZR is lowering due to the RCS cooldown.

B. Incorrect - EPP-4 will only direct borating to CSD Boron if RCS temperature decreases to less than 530°F.

C. Correct - EPP-4 will direct the operator to reduce total feed flow to stop cooldown since S/Gs are greater than 8%, S/G Blowdown is isolated and Tave is less than 547°F and lowering.

d. Incorrect - Closure of the MSIVs and MSIV Bypasses is only required if the reduction in feed flow does not stop the cooldown.

Question 46
Tier 2 / Group 1
K/A Importance Rating - RO 3.6 SRO 3.9

Knowledge of the operational implications of the following concepts
as the apply to the AFW: Relationship between AFW flow and RCS heat transfer

Reference(s) - Sim/Plant design, System Description, EPP-4, EPP-Foldout A
Proposed References to be provided to applicants during examination - None
Learning Objective - AFW 010
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 45.7
Comments - K/A met because candidate must analyze plant conditions given following a plant due to a loss of main feedwater. Based on this analysis the candidate must identify the effect of AFW flow due to RCS temperature lowering and determine the appropriate actions of reducing AFW flow.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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PART 4

END PATH PROCEDURE

EPP-4

REACTOR TRIP RESPONSE

REVISION 25

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides the necessary instructions to stabilize and control the plant following a Reactor trip without a Safety Injection.

2. ENTRY CONDITIONS

   Path-1 when a Reactor trip has occurred and SI is not initiated or required.

\- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Determine If Procedure Exit Is Warranted:	
	a. Check Attack on RNP Site - IN PROGRESS	a. Go To Step 2
	b. Check either of the below events - IN PROGRESS • Total Loss Of SW OR • Loss Of Lake Robinson Dam integrity	b. IF a total loss SW OR a loss of Lake Robinson Dam integrity occurs due to hostile action, THEN Go To EPP-28, Loss of Ultimate Heat Sink. Go To Step 2.
	c. Go To EPP-28, Loss Of Ultimate Heat Sink	
* 2.	Check SI Signal - INITIATED	IF SI initiation occurs during this procedure, THEN Go To Path-1, Entry Point A. Go To Step 4.
3.	Go To Path-1, Entry Point A	
4.	Perform The Following: a. Reset SPDS b. Initiate monitoring of Critical Safety Function Status Trees	
5.	Open Foldout A	
* 6.	Check RCS Temperature - STABLE AT OR TRENDING TO 547°F	IF RCS temperature is NOT stable at OR trending to 547°F, THEN Go To Step 8.
7.	Go To Step 9	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Control RCS Temperature As Follows:	
	a. Check RCS temperature - LESS THAN 547°F AND DECREASING	a. Go To Step 8.g.
	b. Stop dumping steam	
	c. Verify S/G blowdown isolation valves - CLOSED • FCV-1930A&B • FCV-1931A&B • FCV-1932A&B	
	d. Check S/G levels - ANY GREATER THAN 8%	d. Perform the following: • Establish FW bypass flow greater than $0.2x10^6$ pph until level in at least one S/G is greater than 8%. OR • Establish AFW flow greater than 300 gpm until level in at least one S/G is greater than 8%. Go To Step 8.f.
	e. Reduce total feed flow, as necessary, to stop cooldown	
	f. Check RCS cooldown - STOPPED	f. Close MSIVs AND MSIV BYPs. IF RCS Temperature decreases below 530°F, THEN borate the RCS to CSD boron concentration.
	g. Check RCS temperature - GREATER THAN 547°F AND INCREASING	g. Go To Step 9.

(CONTINUED NEXT PAGE)

CONTINUOUS USE

FOLDOUT A

(Page 1 of 7)

1. RCP TRIP CRITERIA

   a. IF BOTH conditions below are met, THEN stop all RCPs:

   - SI Pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW TO THE CORE
   - RCS Subcooling - LESS THAN 35°F [55°F]

   b. IF the PHASE B Isolation Valves are Closed, THEN stop all RCPs.

2. SI ACTUATION CRITERIA

   IF EITHER condition below occurs, THEN Actuate SI and Go To PATH-1, Entry Point A:

   - RCS Subcooling - LESS THAN 35°F [55°F]
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [32%]

   NIETHER ARE MET.

3. MSR ISOLATION CRITERIA

   Perform the following to isolate the MSRs:

   a. IF ANY Purge OR Shutoff Valve does not indicate fully closed, THEN place the associated RTGB Switch to CLOSE.

   b. IF ANY Purge OR Shutoff Valve can NOT be closed from the RTGB AND RCS temperature is less than 540°F and lowering, THEN close the MSIVs AND MSIV BYPs.

   c. IF a loss of power prevents isolation of the MSRs, THEN close the MSIVs AND MSIV BYPs.

47. 062 K2.01 001

Which ONE (1) of the following describes the 4160 Volt power supplies to the "A and "B" RCPs?

	RCP "A"	RCP "B"
A.	Bus 1	Bus 2
B.	Bus 2	Bus 4
C.✓	Bus 1	Bus 4
D.	Bus 2	Bus 1

The correct answer is C.

A. Incorrect. All busses listed are 4160 Volt busses. RCPs are not supplied by like designated busses.

B: Incorrect. All busses listed are 4160 Volt busses.

C: Correct. RCPs are not supplied by like-designated busses.

D: Incorrect. All busses listed are 4160 Volt busses.

Question 47
Tier 2 / Group 1
K/A Importance Rating - RO 3.3 SRO 3.4

Knowledge of bus power supplies to the following: Major system loads.

Reference(s) - Sim/Plant design, System Description, EDP-001
Proposed References to be provided to applicants during examination - None
Learning Objective - RCS 005
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7
Comments - K/A met because candidate must know the power supplies to the RCPs.

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PART 12

EDP-001

4160V AC BUSSES

REVISION 6

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1.0 **4160V AC Buss No. 1** 4

2.0 **4160V BUSS NO. 2** 5

3.0 **4160V AC BUSS NO. 3** 6

4.0 **4160V AC BUSS NO. 4** 7

5.0 **4160V AC BUSS NO. 5** 8

1.0 **4160V AC Buss No. 1**

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
Reactor Coolant Pump "A"	1	52/1	109
Circulating Water Pump "A"	2	52/2	811
Feedwater Pump "A"	3	52/3	615
Station Service Transformer 2B	4	52/4	933
Heater Drain Pump "A"	5	52/5	625
Condensate Pump "A'	6	52/6	605
Unit Aux to 4KV Bus 1	7	52/7	926
PTs and Fan Equipment	8	N/A	948
PTs and Fan Equipment and Metering	9	N/A	948
4KV Bus 1 - 2 Tie	10	52/10	928

2.0 4160V BUSS NO. 2

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
PTs and Fan Equipment	11	N/A	948
Start-Up to 4KV Bus 2	12	52/12	927
Station Service Transformers 2A and 2F	13	52/13	932
Reactor Coolant Pump "C"	14	52/14	105

3.0 4160V AC BUSS NO. 3

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
Station Service Transformer 2C and 2G	15	52/15	934
PTs and Fan Equipment	16	N/A	949
Start-Up Transformer to 4KV Bus 3	17	52/17	929B
PTs and Fan Equipment	18	N/A	949

4.0 **4160V AC BUSS NO. 4**

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
4KV Bus 3 - 4 Tie	19	52/19	931
Unit Aux to 4KV Bus 4	20	52/20	930
PTs and Fan Equipment	21	N/A	949
Condensate Pump "B"	22	52/22	606
Circulating Water Pump "B"	23	52/23	813
Feed to 4KV Bus 5	24	52/24	1344
Heater Drain Pump "B"	25	52/25	626
Feedwater Pump "B"	26	52/26	620
Reactor Coolant Pump "B"	27	52/27	101
Station Service Transformer 2D	28	52/28	1041

5.0 **4160V AC BUSS NO. 5**

Location: Turbine Bldg., 1st Level,

Grid Location 3B

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
4KV Bus 4 to 4KV Bus 5	29	N/A	1344
PTs and Control Power Transformer	30	N/A	N/A
SPARE	31	52/31	N/A
Station Service Transformer 2E	32	52/32	1399
Circulating Water Pump "C"	33	52/33	815
SPARE	34	52/34	N/A

48. 063 A2.01 001

Given the following plant conditions:

- The plant is at 100% RTP.
- APP-036-D1, BATT CHARGER A/A-1 TROUBLE, annunciated.
- OAO reports the following indications from Battery Charger "A-1":
    - +40 Volts on the Ground Detection Voltmeter.
    - 135 Volts DC on Charger Voltage

What are the adverse impacts of the reported indications and the actions required to be taken IAW APP-036-D1?

A. Normally non-energized portions of switchgear may be energized.

Place both "A" and "A-1" Battery Chargers in Standby.

B.✓ Normally non-energized portions of switchgear may be energized.

Swap Battery Chargers to place Battery Charger "A" In-service.

C. Over-voltage condition could damage "A" Batteries.

Place both "A" and "A-1" Battery Chargers in Standby

D. Over-voltage condition could damage "A" Batteries.

Swap Battery Chargers to place Battery Charger "A" In-service.

The correct answer is B.

A. Incorrect. Placing both "A" and "A-1" Battery Chargers in Standby would require input from Engineering along with Work Order Instructions or Procedure Direction. The first half of the answer is correct.

B. Correct.

C. Incorrect. An Over-voltage condition does not exist. 135 Volts DC is at the upper end of the voltage range that the charger will be set when placing it in service. Placing both "A" and "A-1" Battery Chargers in Standby would require input from Engineering along with Work Order Instructions or Procedure Direction.

D. Incorrect. An Over-voltage condition does not exist. 135 Volts DC is at the upper end of the voltage range that the charger will be set when placing it in service. Swapping battery chargers is correct.

Question 48
Tier 2 / Group 1
K/A Importance Rating - RO 3.3 SRO 3.4

Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds

Reference(s) - Sim/Plant design, System Description, OMM-035, OP-601, APP-036
Proposed References to be provided to applicants during examination - None
Learning Objective - DC 007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A met because the candidate must predict the adverse impacts of a DC ground and determine the required actions to mitigate.

Progress Energy

I
INFORMATION
USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 1

OMM-035

GROUND ISOLATION

REVISION 12

8.0 INSTRUCTIONS

NOTES: Attachment 10.1, Ground Isolation Sequence Log, should be used to document all notifications, component starting, and component stopping during the ground isolation process. This will provide valuable information during trouble shooting and when restoring the electrical system to a normal lineup.

Attaching a copy of Attachment 10.1 to the shift turnover sheet will provide the following shifts a list of components which have been checked for a ground.

CAUTION

When a ground is present, normally non-energized portions of electrical components such as breaker cubicles and motor housings may be energized. Therefore, extreme caution should be exercised and protective measures should be used to prevent personnel injury from electrical shock when attempting to isolate grounds.

8.1 Annunciator

8.1.1 **IF** Annunciator APP-005-C6, ANNUNCIATOR SYSTEM DC GROUND, is illuminated **OR** there are any other indications of an annunciator system dc ground, **THEN GO TO** 8.2.

8.1.2 **IF** Annunciator APP-009-C4, 4160V SWITCHGEAR GROUND, is illuminated **OR** there are any other indications of a 4160V switchgear ground, **THEN GO TO** 8.3.

8.1.3 **IF** Annunciator APP-009-E7, 480V GRD FAULT, is illuminated **OR** there are any other indications of a 480V ground, **THEN GO TO** 8.4.

8.1.4 **IF** Annunciator APP-036-G8, DS Bus OVER VOLT/GROUND, is illuminated **AND** the DS Bus is being powered from the DS Diesel separated from SST-2C, **THEN GO TO** 8.5.

8.1.5 **IF** Annunciator APP-036-D1, BATTERY A/A-1 TROUBLE, is illuminated **OR** there are any other indications of a DC system ground associated with "A" train, **THEN GO TO** 8.6.

8.1.6 **IF** Annunciator APP-036-D2, BATTERY B/B-1 TROUBLE, is illuminated **OR** there are any other indications of a DC system ground associated with "B" train, **THEN GO TO** 8.7.

5.0 PRECAUTIONS AND LIMITATIONS

5.1 A review of PLP-037 criteria indicates that no additional management involvement is required beyond that routinely provided by first line Supervision. The Shift Manager **OR** Senior Control Operator should review this procedure with their shift prior to implementation.

5.2 The Busses with ground annunciators are:

- 4160V-1, 2, 3, 4 (5 alarms through 4)
- 480V-1, 2B (2A alarms through 2B), 3 (DS alarms through 3), 4, 5, E-1, E-2
- All of the DC system

5.3 Busses are not transferred to alternate power supplies in this procedure. To transfer a Bus in most cases would require jumpers and that is beyond the scope of this procedure.

5.4 Grounds can be on cyclic loads and therefore an attempt to reset the ground relays should be done:

- while each component's breaker is opened and documented on Attachment 10.1, Ground Isolation Sequence Log.
- on a frequent basis when not changing components and documented on Attachment 10.1.

5.5 Train separation is a good method to use when ground isolating. If a component on the opposite train can be started, it will minimize impact on the plant.

5.6 When a ground is identified, the breaker for that component should be opened as soon as possible. It is not the intent of this procedure to direct operations of the plant. Normal or Emergency Operating Procedures as applicable should be used.

8.6 APP-036-D1 - BATT CHARGER A/A-1 TROUBLE (CWD 955 Cable C)

8.6.1 Record Battery Charger ground readings on Attachment 10.1, Ground Isolation Sequence Log, every 4 hours.

8.6.2 Check hourly to see if Annunciator APP-036-D1 has reset, **AND** record the results on Attachment 10.1.

8.6.3 Place the alternate Battery Charger in service IAW OP-601, DC Supply System.

8.6.4 **IF** the Annunciator resets, **THEN** the ground is clear, write a WR on the Battery Charger removed from service.

8.6.5 Cycle the following breakers on Distribution Panel "A" in an attempt to isolate the ground, observe the annunciator after each breaker is opened. If the ground has cleared, the annunciator will extinguish when reset.

- CKT 3, HYDROGEN CONTROL PANEL
- CKT 5, LIGHTING PANEL LP-33
- CKT 6, 125V DC POWER PANEL "A-1"

NOTE:	If possible, Inverter "C" should be shutdown prior to opening its breaker.

- CKT 9, INVERTER "C"
- CKT 14, GAS STRIPPER CONTROL CABINET "A"
- CKT 17, MAIN AND AUXILIARY TRANSFORMER ANNUNCIATORS (APP-009-C5 and APP-009-C6)
- CKT 23, STARTUP TRANSFORMER ANNUNCIATOR (APP-009-C7)

8.6.6 Based on plant conditions, cycle any breakers that are mutually agreed upon by Engineering and Operations.

8.6.7 Contact I&C Planning to force generate a trouble shooting work order from PMID 17644 RQ 07.

8.6.8 **WHEN** the ground is identified, **THEN** establish Plant conditions necessary to isolate the component with the ground.

8.6.9 **IF** the ground cannot be isolated, **THEN** establish safety barriers and warning signs around the grounded component.

ALARM

BATT CHARGER A/A-1 TROUBLE

AUTOMATIC ACTIONS/CAUSE

NOTE: Following a Loss of Off-Site Power (LOOP), the In Service Battery Charger will automatically restart. The annunciator MAY alert to momentary DC overvoltage condition during the battery charger start.

CAUSES	CHARGER	AUTO ACTION
DC Overvoltage	A and A-1	Charger trips
AC Input Failure	A and A-1	**IF** the battery charger is "In Standby", **THEN** it will trip. **IF** the battery charger is "In Service", **THEN** it should restart.
Fuse blown	A	Charger trips
DC Output Breaker open	A-1	Charger trips
Ground Test	A	None
DC Ground	A and A-1	None
DC Undervoltage	A-1	None
Fan Failure	A	None (See NOTE and ACTION 9)
High temperature	A	None

OBSERVATIONS

1. ERFIS Points APV3022A, DC MCC-A VOLT, and APV3024A, DC MCC-A CURRENT

ACTIONS CK (✓)

1. **IF** a loss of MCC-5 has occurred, **THEN REFER** to AOP-024. _____
2. **IF** required, **THEN DISPATCH** an operator to check the Battery Charger. _____
3. **IF** the in service charger is determined to be inoperable **AND** the standby charger is available, **THEN PLACE** the standby charger in service IAW OP-601. _____
4. **IF** cause is **NOT** due to loss of power supply **OR** Charger failure, **THEN CHECK** for DC ground using OMM-035. _____
5. **IF** a ground is indicated, **THEN IDENTIFY AND ISOLATE** the grounded equipment. _____
6. **IF** a ground is suspected to exist on the battery, **THEN CONTACT** Engineering to evaluate the possibility of isolating the battery with Battery Charger A **OR** A-1 carrying the bus load (either battery charger can be used to support this step). _____
7. **IF** Battery "A" discharged to less than 110 Volts **OR** overcharged to greater than 150 Volts, **THEN CONTACT** I&C to perform the applicable sections of MST-903 within 24 hours as required by ITS SR 3.8.6.2. _____
8. **IF** Battery "A" discharged to less than 109 Volts **OR** overcharged to greater than 150 Volts, **THEN** consideration should be given to switching APP-036 power supply to Distribution Panel C in accordance with OP-601. _____

ACTIONS (Continued) CK (✓)

NOTE:	This Note and Step 9 applies to Battery Charger A only. Battery Charger A is **NOT** inoperable with one failed fan. The receipt of a high temperature alarm **OR** a failure of the remaining fan would render the battery charger inoperable.

9. **IF** a fan failure occurs, **THEN PERFORM** the following as applicable:

   a. **IF** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601 (this will reduce the frequency of the required compensatory action to check the local temperature alarm and remove the load from the charger with only one fan.) _____

   b. **IF** a standby battery charger is **NOT** available, **THEN CHECK** the local alarm display **AND** operating cooling fan on the In Service battery charger at least every two hours until the condition is resolved **OR** the other battery charger becomes available and is placed in service. _____

   c. **IF** the In Service battery chargers were transferred in Step 9.a above, **THEN CHECK** the local alarm display **AND** operating cooling fan on the standby battery charger for proper operation at least every two hours until the condition is resolved **OR** the battery charger is removed from service for Maintenance. _____

   d. **IF** a local high temperature alarm is observed **OR** the remaining operating fan fails on the In Service battery charger without a standby battery charger available, **THEN REFER TO** ITS 3.8.4 or 3.8.5. _____

   e. **IF** a local high temperature alarm is observed **OR** the remaining operating fan fails on the standby battery charger, **THEN SHUTDOWN** the standby battery charger per OP-601. _____

   f. **INITIATE** actions to have the failed fan repaired. _____

10. **IF** high temperature indication is present **AND** both fans are operating, **THEN PERFORM** the following as applicable:

    a. **IF** air flow under and over the In Service charger is blocked **THEN RESTORE** air flow under and over the In Service charger. _____

    b. **IF** there is no air obstruction, **THEN PLACE** the standby battery charger in service per OP-601 **AND SHUTDOWN** the charger with high temperature per OP-601. _____

11. **IF** blown fuse indication is present, **THEN PLACE** the standby battery charger in service per OP-601 **AND SHUTDOWN** the charger with blown fuse indication per OP-601. (The unit with the blown fuse indication is to be considered inoperable.) _____

ACTIONS (Continued) CK (✓)

12. **IF** low DC output voltage indication is present, **THEN PERFORM** the following as applicable:

    a. **IF** charger in float, **THEN ADJUST** float adjustment until output voltage is 133-135 V. _____

    b. **IF** charger in equalize, **THEN ADJUST** equlize adjustment until output voltage is 138-139 V. _____

    c. **IF** voltage can **NOT** be satisfactorily adjusted **AND** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601. _____

13. **IF** high or low voltage shutdown indication **AND** high DC output voltage is illuminated, **THEN PLACE** the standby battery charger in service per OP-601 if available. _____

14. **IF** low AC input voltage is illuminated **AND** a standby battery charger is available, **THEN PLACE** the standby battery charger in service per OP-601. (When unit with low ac input voltage is placed in standby, it WILL trip.) _____

DEVICE/SETPOINTS

NOTE:	This note is applicable to Battery Charger A-1 and **NOT** for Battery Charger A. Ground alarm circuit cards are not calibrated to a certain pickup voltage. No set point or tolerances are specified during testing, only that the ground relay energizes and de-energizes by adjusting a 10 k ohm resistor on the test board. Per Power Conversion Products, the ground will actuate when a system ground of 13 k ohms is sensed.

1. Battery Charger A- X314/X311 (X305, X306, X307, X308, X309, X310, X312, X313) / Variable
2. Battery Charger A-1 - K1, K2, K4, K5, K7 / Variable

POSSIBLE PLANT EFFECTS

1. Low Battery voltage

REFERENCES

1. ITS LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6
2. OMM-035, Ground Isolation
3. OP-601, DC Supply System
4. MST-903, Station Battery Charge
5. AOP-024, Loss of Instrument Bus
6. CWD, B-190628, Sheet 955, Cable C
7. VTM 762-209-181

49. 063 K4.02 001

A fire has occurred in DC Distribution Panel 'B'. The panel has been de-energized.

Which ONE (1) of the following identifies the effect on the Electrical Distribution System?

Control Power lost to 4160 V Busses....

A. 3 & 4 and cannot be restored.

B.✓ 3 & 4. Control Power can be restored using a safety switch in the 4160 V Switchgear Room.

C. 1 & 2 and cannot be restored.

D. 1 & 2. Control Power can be restored using a safety switch in the 4160 V Switchgear Room.

The correct answer is B.

A-Incorrect. Control Power can be restored using the safety switch. Breakers will not trip open on a loss of control power. Breakers will remain in current position and cannot be remotely cycled.

B-Correct. EPP-27, Attachment 2, Locally Restoring Deenergized AC Busses, step 6. Breakers will remain As-Is on a loss of control power.

C-Incorrect. Control Power lost to 4160V Busses 3 & 4. Breakers will not trip open on a loss of control power.

D-Incorrect. Control Power lost to 4160V Busses 3 & 4. Second half of answer is correct. The breakers will remain As-Is on a loss of control power.

Question 49
Tier 2 / Group 1
K/A Importance Rating - RO 2.9 SRO 3.2

Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties.

Reference(s) - Sim/Plant design, System Description, EDP-004, EPP-27
Proposed References to be provided to applicants during examination - None
Learning Objective - DC 010
Question Source - Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7
Comments - K/A is met because candidate must know the impact of a loss of DC Control power on the Electrical Distribution System and whether or not a backup is available.

2.0 DISTRIBUTION PANEL A

DISTRIBUTION PANEL "A"		
Power Supply: 125V DC MCC "A" Drawings:B-190627, SH DPA	Location: On 125V DC MCC "A"	
CKT#	LOAD	DWG
1	480V Switchgear No. E-1	CWD 955
2	4160V Switchgear Busses 1 & 2	CWDs 955, 1340, 1821
3	Hydrogen Control Panel	CWD 875
4	480V Switchgear Busses 1 & 2A	CWD 955
5	Lighting Panel LP-33 (Alt Pwr via Auto Transfer Switch)	
6	125V DC Power Panel "A-1"	
7	Startup Transformer Motor Operated Disconnects	CWD 925
8	Diesel Generator "A" Exciter	CWD 880
9	Inverter "C"	
10	Reactor Trip Breaker "A" and Reactor trip Bypass Breaker "B"	CWD 45
11	Inverter "A"	
12	Rod Drive M-G Set "A"	CWD 71
13	Main Generator Exciter Field Breaker	CWD 862
14	Gas Stripper Control Cabinet "A"	CWD 173
15	Generator Lockout Relay 86P	CWD 912
16	Aux. Panel "D-C" Fuse Panel	CWD 955
17	Spare	
18	Reactor Protection Train "A"	CWD 955
19	PCV-456 Pressurizer PORV	CWD 119

DISTRIBUTION PANEL "B"		
Power Supply: 125V DC MCC "B"	Location: On 125V DC MCC "B"	
Drawings: B-190627 Sh DPB		
CKT#	LOAD	DWG
1	480V Switchgear No. E-2	CWD 956
2	4160V Switchgear Busses 3 & 4	CWDs 956, 1341, 1340
3	4160V Breaker Test Panel	5379-1702
4	480V Switchgear Busses 2B & 3	CWDs 956, 1341, 1340
5	125V DC MCC "B-A"	
6	Sample Panel	CWD 88
7	PCV-455C Pressurizer PORV	CWD 120A
8	Diesel Generator "B" Exciter	CWD 881
9	Reactor Trip Breaker "B" & Reactor Trip Bypass Breaker "A"	CWD 46
10	Annunciator Panel (RTGB)	CWD 956
11	Waste Disposal Panel	CWD 351
12	Diesel Generator "B" Control Power	CWD 950
13	Turbine Emergency Trip	CWD 711
14	Gas Stripper Panel "B"	CWD 174
15	Gas Analyzer Panel	CWD 319
16	Aux . Panel "G-C" Fuse Panel	CWD 956
17	Generator Lockout Relay 86 BU	CWD 913
18	Reactor Protection Train "B"	CWD 956
19	Reverse Current Valves	CWD 740

INFORMATION USE

ATTACHMENT 1

MAJOR EFFECTS / LOAD LIST

(Page 1 of 4)

Major Effects:

Reactor	Will trip due to loss of power to 52/RTB undervoltage coil.
Turbine	Will trip via 20/AST from Rx Trip (20/ET has lost power).
Generator	Will receive lockout signal. However, 86P cannot open OCB 52/8 & 52/9 due to the loss of their control power. This causes a Breaker Failure scheme which trips OCB 52/3, 52/6, 52/7, 52/12, 52/14 and the downstream breakers on the Darlington SCPSA line. The Exciter Field Breaker will open.
4KV Busses 1 & 2	If initially on SUT, nothing will happen. If initially on UAT, the busses will auto-transfer due to the Rx Trip. In either case, 4KV busses 1 and 2 and all downstream busses and equipment will remain energized.
4KV Bus 3	Will remain energized on the SUT. 4KV Bus 3 and 480V Bus 3 will lose DC Control Power (including a loss of protective relaying).
4KV Busses 4 & 5	4KV Bus 4 will try to auto-transfer to Bus 3 but cannot due to the loss of DC Control Power. Thus, 4KV Busses 4 & 5 and all downstream busses and equipment will deenergize. 4KV Bus 4 and 480V Bus 4 will lose DC Control Power (including a loss of protective relaying). Control Power (and protective relaying) will remain for 4KV Bus 5 and 480V Bus 5.
Emergency Bus E-1	Will remain energized. SST 2F will lose cooling fans.
Emergency Bus E-2	Will remain energized on the SUT but will lose DC Control Power (including a loss of protective relaying). SST 2G will lose cooling fans.
DS Bus	Will remain energized with Control Power available.
EDG A	Remains available, if needed.
EDG B	Auto-starts due to loss of power to air start solenoids but will not field flash and output breaker will not close.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

LOCALLY RESTORING DEENERGIZED AC BUSSES

(Page 1 of 6)

NOTE

- The following step will transfer Control Power to 480V Busses 2B and 3 via the DS System DC Supply.
- DS Distribution Panel "A" (DS)is located on the east wall of the 4160V Switchgear Room.
- The DS System Safety Switches are located on Stanchion R32 in the center of the 4160V Switchgear Room.
- Attachment 3 contains instructions that may be used for local operation of 480V breakers.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Perform The Following To Align Control power To 480V BUSSES 2B & 3: a. In DS Distribution Panel A (DS), Close breaker DP-A(DS)-4. b. Place SAFETY SWITCH 4, 480V BUSSES 2B & 3 in the EMERGENCY Position	
2.	Inform Control Room That DC Control Power Has Been Restored To 480V BUSSES 2B & 3	
3.	At Cubicle 21, Check 4160V BUS 4 - DEENERGIZED	Observe the <u>NOTE</u> prior to Step 6 <u>AND</u> Go To Step 6.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

LOCALLY RESTORING DEENERGIZED AC BUSSES

(Page 2 of 6)

NOTE

- When DC control power is lost, breakers will remain as they were before 4160V power was lost.
- The front panel control switch and indicators will not be operable. Breakers must be manually tripped (inside the cubicle) by lifting the trip lever at the bottom of the breaker.

4. Trip The Following Breakers On 4160V Bus 4:
   - 52/19, 4KV BUS 3-4 TIE
   - 52/20, UNIT AUX TO 4KV BUS 4
   - 52/22, CONDENSATE PUMP B
   - 52/23, CIRCULATING WATER PUMP B
   - 52/24, FEED TO 4KV BUS 5
   - 52/25, HEATER DRAIN PUMP B
   - 52/26, FEEDWATER PUMP B
   - 52/27, REACTOR COOLANT PUMP B
   - 52/28, STATION SERVICE TRANSFORMER 2D
5. Inform The Control Room That 4160V Bus 4 Will Auto Transfer To The SUT During The Next Step

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

LOCALLY RESTORING DEENERGIZED AC BUSSES

(Page 3 of 6)

NOTE

- The following step will place Control Power on 4160V Busses 3 and 4 from the DS System DC Supply.
- This step assumes that the Lock Relays have NOT been reset.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Place SAFETY SWITCH 3, 4160V BUSSES 3 & 4, To EMERGENCY Position	
7.	Contact Control Room For The Following:	
	a. Notify Control Room that DC control power has been restored to 4160V BUSSES 3 AND 4	
	b. Check 4160V BUS 4 - ENERGIZED	b. WHEN 4160V Bus 4 is energized, THEN Go To Step 8.
8.	Verify The Following Breakers On 480V BUS 4 - TRIPPED: • STATION SERVICE TRANSFORMER 2D TO 480V BUS 4 (CMPT-30B) • FEED TO MCC-11 (CMPT-30C) • FEED TO MCC-13 (CMPT-31A) • FEED TO MCC-12 (CMPT-31B) • PAP WEST BUILDING AND SECURITY POWER SUPPLY (CMPT-31C) • FEED TO MCC-14 (CMPT-31D)	

10.0 DISTRIBUTION PANEL A (DS)

DISTRIBUTION PANEL "A" (DS)		
Power Supply: "A" Power Supply (DS)	Location: 4160V Switchgear Room	
CKT#	LOAD	DWG
1	5KVA Inverter Battery	CWD 1340
2	5 KVA Inverter	CWD 1340
3	SAFETY-SW-1 (4KV Bus 1 and 2 DC Switch)	CWDs 1340, 1821
4	SAFETY-SW-3 (4KV Busses 3 and 4) and SAFETY-SW-4 (480V Busses 2B and 3)	CWDs 1340, 1341
5	DS Bus Indicating Lights and Control Power for 52/32B Trip Circuit, Control Power for DS Bus Undervoltage and Overvoltage Alarm Circuits	CWDs 1340, 1016
6	Pressurizer PORV post-fire emergency control station for operation PCV-456	CWD 128

50. 064 A2.14 001

Given the following plant conditions:

- Plant is at 100% RTP.
- An Electrician accidently opens breaker 52/18B, Normal Supply Breaker for E-1.
- "A" EDG starts on UV and energizes Emergency Bus E-1.

Which ONE (1) of the following identifies the effect of stopping "A" EDG from the RTGB while it is loaded on the isolated Emergency Bus E-1?

After taking the "A" EDG Control switch on the RTGB to STOP "A" EDG will ....

A. trip and must be manually reset prior to placing back in service.

B. maintain its current speed and voltage due to Shutdown Control Logic having a run priority.

C.✓ attempt to shutdown but will restart due to the Shutdown Control Logic having a run priority.

D. maintain its current speed and voltage due to the App. R Isolation Switches being in the ISOLATE position.

The correct answer is C.

A. Incorrect. Correlating this to the effect of having a loss of off-site power while the EDG is operating in parallel with site power IAW OP-604 Section 8.4. Section 8.4 has a large caution statement that state that the EDG may trip during a loss of off-site power while EDG is operating in parallel. The section has steps for the IAO to locally reset the EDG should it trip in this scenario.

B. Incorrect. Candidate may confuse the run priority to mean that the EDG will not stop is carrying an isolated bus since it is providing the only source of power.

C. Correct.

D. Incorrect. Candidate may not understand the purpose of the App. R Isolation Switches. The switches are a recent modification that removes control power from the local control panel and eliminates concerns with a fire.

Question 50
Tier 2 / Group 1
K/A Importance Rating - RO 2.7 SRO 2.9

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects (verification) of stopping ED/G under load on isolated bus.

Reference(s) - Sim/Plant design, System Description, OP-604
Proposed References to be provided to applicants during examination - None
Learning Objective - EDG 007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A is met because candidate must know the operations impact of attempting to stop a EDG under load while on an isolated bus.

Also located on the panel are switches and pushbuttons (these controls have a protective cover to prevent inadvertent misoperation):

a. Regulator Switch
b. Parallel Switch - Normally left in the ISOLATE POSITION, except when running in parallel with another power source.
c. Manual Voltage Control
d. Auto Voltage Control (with potentiometer adjustment lock)
e. Reset Push-button - Will reinstate voltage on the generator if it has been removed with the diesel running.
f. Voltage Shutdown Push-button - Removes voltage from the generator if the diesel is to remain running without putting the generator on the line.
g. Ammeter Switch - Selects the phase indicated on the generator ammeter.
h. Voltage Switch - Selects the phase indicated on the generator voltmeter.
i. Remote Engine START-STOP Switch.
j. Speed Control Switch - Used for loading the generator and can be used to change diesel engine speed when unparalleled.
k. Normal Bus Synchroscope - Used for observing synchronism for closure of the Normal Bus Supply ACB when the Diesel Generator is carrying its Emergency Bus.
l. Normal Bus Supply ACB Control Switch - Used for closing the Normal Bus Supply breaker, 52/18B or 52/28B (Requires RTGB APP R NORMAL/ISOLATE switch to be in NORMAL).
m. Generator Synchroscope - Used for observing synchronism for closure of the EDG Output BKR when the Emergency Bus is already being powered from its normal source. This switch must be ON in order to close the EDG Output BKR locally, regardless of whether the emergency bus is energized or not.
n. EDG Output BKR Control Switch - Used for closing the Diesel Generator Output breaker, 52/17B or 52/27B. In order to close the EDG Output Breaker locally at the Generator Control Panel, the Generator Synchroscope must be selected to ON and the RTGB APP R switch to be in NORMAL.

Also included is a generator hour meter that shows hours of generator operation. Located at the bottom of the panel are relays that provide generator electrical protection, and are described in section 5.2.2 of this SD.

4.7 Shutdown Control Logic

The shutdown control logic of the diesel generators is a run priority. Therefore, if a shutdown sequence has been initiated with the units in remote control and a start signal is received during this sequence, the units will return to speed and be ready for service as needed.

5.0 CONTROLS AND PROTECTION

5.1 RTGB Controls

5.1.1 Generator Breaker Controls

EMERG DG A TO BUS E1(2) BKR 52/1(2)7B	Positions - TRIP / neutral / CLOSE
480V BUS E1(2) MAIN BKR 52/1(2)8B	Positions - TRIP / neutral / CLOSE
EMERG DG A(B) BUS E1(2) 17B(27B) APP R ISOLATION	Positions - NORMAL/ISOLATE

Interlocks for the EDG A Output breaker, 52/17B:

- Breaker can close if 52/22B or 52/29B Open.
- Breaker can close if 52/22B and 52/29B Closed, <u>AND</u> 52/27B and 52/28B Open.
- Breaker will Auto-Close on an Undervoltage or Degraded Voltage on E-1 <u>AND</u> voltage at the EDG, <u>IF</u> 52/22B or 52/29B Open.
- Breaker will Auto-Close on an Undervoltage or Degraded Voltage on E-1 <u>AND</u> voltage at the EDG, <u>IF</u> 52/22B and 52/29B Closed, <u>AND</u> 52/27B and 52/28B Open.
- Breaker's associated synchroscope on the Generator Control Panel must be ON in order to <u>close</u> breaker from Generator Control Panel. Breaker can be closed from RTGB regardless of synchroscope position.
- RTGB APP R switch must be in NORMAL (not ISOLATE)
-

Interlocks for the EDG B Output breaker, 52/27B:

- Breaker can close if 52/22B or 52/29B Open.
- Breaker can close if 52/22B and 52/29B Closed, <u>AND</u> 52/17B and 52/18B Open.
- Breaker will Auto-Close on an Undervoltage or Degraded Voltage on E-2 <u>AND</u> voltage at the EDG, <u>IF</u> 52/22B or 52/29B Open.
- Breaker will Auto-Close on an Undervoltage or Degraded Voltage on E-2 <u>AND</u> voltage at the EDG, <u>IF</u> 52/22B and 52/29B Closed, <u>AND</u> 52/17B and 52/18B Open.
- Breaker's associated synchroscope on the Generator Control Panel must be ON in order to <u>close</u> breaker from Generator Control Panel. Breaker can be closed from RTGB regardless of synchroscope position.
- RTGB APP R switch must be in NORMAL (not ISOLATE)

Interlocks for the Emergency Bus E-1 Normal Supply Breaker, 52/18B:

- Breaker can close if 52/17B or 52/13 Open,
- Breaker will Auto-Open on an Undervoltage, or Degraded Voltage for 10 seconds, on E-1,
- Breaker cannot be closed from the RTGB while 52/13 is racked out. The 52/13 "b" contact that supplies the 52/18B permissive interlocks is not connected to the interlock circuits when racked out. 52/18B must be closed from either the EDG Generator Control Panel, or manually. 52/18B can still be opened from the RTGB when 52/13 is racked out,
- Breaker's associated synchroscope on the Generator Control Panel must be ON in order to close breaker from Generator Control Panel. Breaker can be closed from RTGB regardless of synchroscope position.
- RTGB APP R switch must be in NORMAL (not ISOLATE)

Interlocks for the Emergency Bus E-2 Normal Supply Breaker, 52/28B:

- Breaker can close if 52/27B or 52/15 Open,
- Breaker will Auto-Open on an Undervoltage, or Degraded Voltage for 10 seconds, on E-2,
- Breaker cannot be closed from the RTGB while 52/15 is racked out. The 52/15 "b" contact that supplies the 52/28B permissive interlocks is not connected to the interlock circuits when racked out. 52/28B must be closed from either the EDG Generator Control Panel, or manually. 52/28B can still be opened from the RTGB when 52/15 is racked out,
- Breaker's associated synchroscope on the Generator Control Panel must be ON in order to close breaker from Generator Control Panel. Breaker can be closed from RTGB regardless of synchroscope position.
- RTGB APP R switch must be in NORMAL (not ISOLATE)

5.1.2 Diesel Generator Ventilation Control

5.1.2.1 DG RM A Supply Fan HVS-6, MCC-5

Three Position Switch - STOP / neutral / START- spring return to unmarked neutral position

Off "Green" and ON "Red" indicating lights for each fan are mounted above the switch to provide status indication. The switch can be used to START or STOP the associated fan. Fans will automatically START when the DIESEL GENERATOR is started (in any mode).

5.9 When the Diesel is running and the Trips Defeat Key Switch is in the TRIPS DEFEATED position; the Diesel should be manually tripped if a condition exists that would automatically trip the Diesel. These conditions are: (Rail 92R0044)

- Coolant Temperature High 205°F
- Crankcase Pressure + 0.5 inches H_2O
- Lube Oil Low Pressure 18 psig
- Coolant Low Pressure consistently less than 25 psig with engine at full speed

5.10 Synchroscopes will be left OFF unless in use for synchronizing to prevent damaging by inadvertent energizing of two synchroscopes.

5.11 The Diesel Generator shall **NOT** be operated at speeds below 900 rpm with the Field Excitation in service. To take Field Excitation out-of-service, the Diesel Generator shall be above a speed of 200 rpm and below a speed of 900 rpm and the field flashed. With the field flashed, momentarily depressing the VOLTAGE SHUTDOWN pushbutton on Generator Control Panel, will remove Field Excitation from service. When the VOLTAGE SHUTDOWN pusbutton is released Generator voltage should be verified to drop to 0. To reinstate Field Excitation, Diesel Generator speed should be between 890 and 910 rpm and the RESET pushbutton on the Generator Control Panel depressed. If Field Excitation was taken out-of-service and Diesel Generator speed was dropped below 200 rpm or Diesel Generator was stopped, Field Excitation will reset automatically and is required to be taken out-of-service if Diesel Generator speeds stays below 900 rpm.

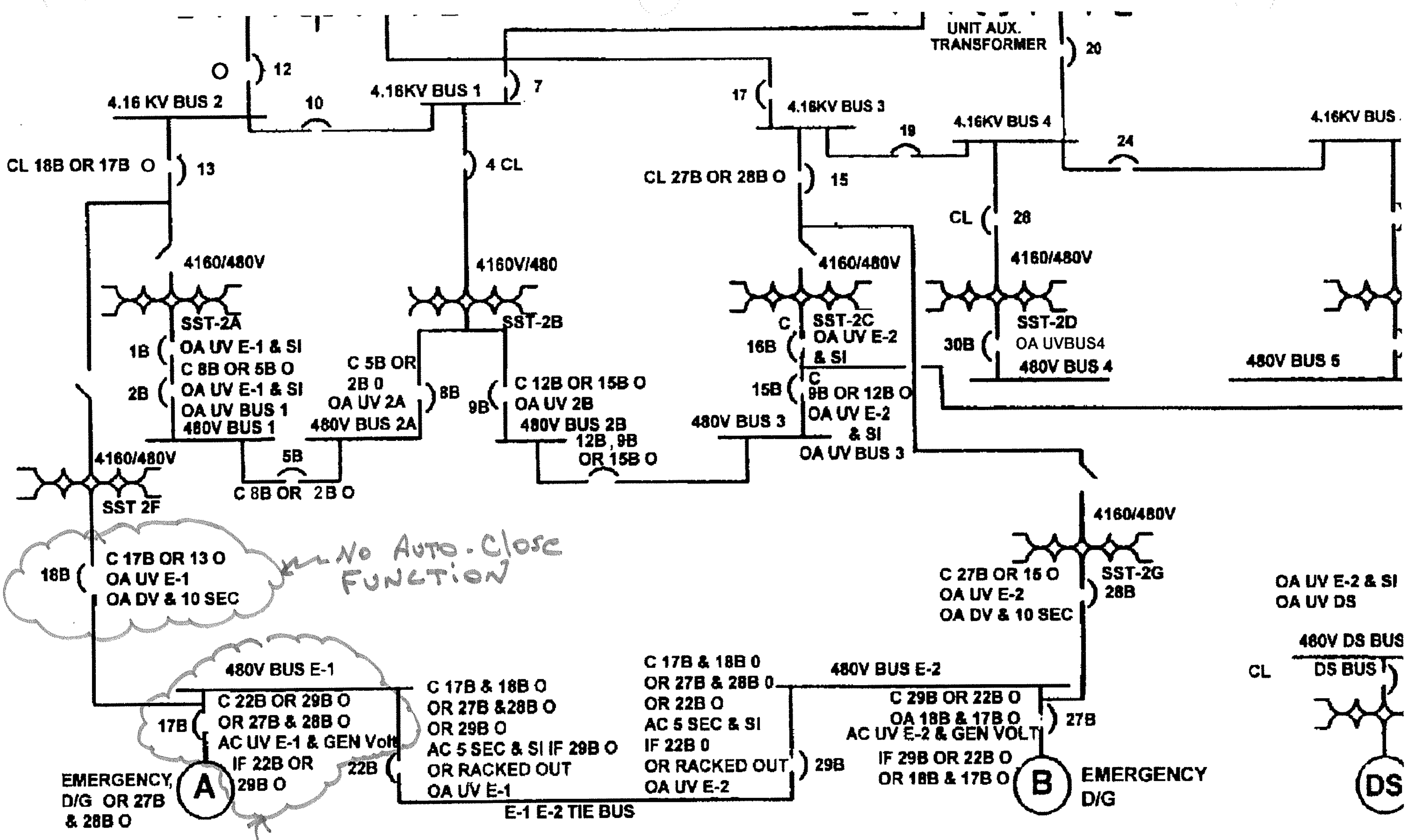
5.12 The Parallel Switch will be in the PARALLEL position when synchronizing manually to an Energized Bus. The Parallel Switch will be in the ISOL position when the Diesel is lined up for an Auto Start, because it will be energizing a dead bus.

5.13 Except during Overspeed Trip Device testing, bearing checks, OST-163, or as requested by Engineering, the EDG shall be loaded and run for 25 to 30 minutes at between 1000 to 1250 KW after each start. This will prevent the buildup of Lube Oil in the Exhaust System, which could cause a fire. (ESR 95-00254)

5.14 The Diesel Generators should be run unloaded for 3 to 5 minutes to allow the Diesel Generators to cool down following a loaded run.

5.15 The Diesel Generators have a shutdown control logic that is run priority. Therefore, if the units are being shutdown with the control mode in remote and a start signal is received during this sequence, the units will return to speed and be ready for service as needed.

5.16 The diesel operator concurrently provides continuous fire watch while the diesel is running.



No Open-Auto Function

LEGEND

O - OPEN
C - MANUAL CLOSE RTGB
CL - MANUAL CLOSE LOCAL
AC - AUTO CLOSE
OA - AUTO OPEN

UV - UNDERVOLTAGE
DE - DE-ENERGIZE
DV - DEGRADED VOLTAGE
SI - SAFETY INJECTION

51. 073 G2.4.18 001

Which one of the following describes the basis for initiating a Containment Ventilation Isolation signal during the performance of FRP-S.1, Response to Nuclear Power Generation / ATWS?

To isolate......

A✓ non-essential containment ventilation penetrations to prevent potential release of radioactive materials from containment.

B. R-11 and R-12, CV Air and Plant Stack Monitors, to prevent potential release of radioactive materials from containment.

C. non-essential containment ventilation penetrations due to delayed response time of R-11 and R-12 during an ATWS.

D. R-11 and R-12, CV Air and Plant Stack Monitors, due to delayed response time of R-11 and R-12 during an ATWS.

The correct answer is A.

A. Correct.

B. Incorrect. Exert from FRP-S.1 Basis Document: *Non-essential ventilation penetrations are isolated to prevent potential release of radioactive materials from containment.* Isolation of R-11 and R-12 will not occur as part of the containment ventilation isolation signal. R-11 and R-12 are isolated from a Phase A signal.

C. Incorrect. The half of the answer is correct. The second half of answer is an incorrect reason for verifying containment ventilation. During an ATWS an Automatic Containment Ventilation Signal does not normally occur and is performed via the RNO step in FRP-S.1. The student may think the basis for these actions is equipment related vice taking a prudent action to isolate CV Ventilation Penetrations.

D. Incorrect. Exert from FRP-S.1 Basis Document: *Non-essential ventilation penetrations are isolated to prevent potential release of radioactive materials from containment.* Isolation of R-11 and R-12 will not occur as part of the containment ventilation isolation signal. R-11 and R-12 are isolated from a Phase A signal.

Question 51
Tier 2 / Group 1
K/A Importance Rating - RO 3.3 SRO 4.0

Knowledge of the specific bases of EOPs : Process Radiation Monitoring System.

Reference(s) - Sim/Plant design, System Description, FRP-S.1 Basis Document
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-S.1-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 4110 / 43.1 / 45.13
Comments - K/A is met because the candidate must know the basis for taking R-11 or R-12 to HV OFF during FPR-S.1 to generate a Containment Ventilation Isolation Signal.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

4 4 WOG BASIS

PURPOSE: To add negative reactivity to bring the reactor core subcritical

BASIS:

After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. The intended boration path here is the most direct one available, not requiring SI initiation, but using normal charging pump(s).

Several plant specific means are usually available for rapid boration and should be specified here in order of preference. Methods of rapid boration include emergency boration, normal boration with maximum boric acid flow, and safety injection actuation. If these methods of rapid boration are not available, a slower RCS boration can be performed by using the RWST. To allow for maximum flow during a boration using positive displacement charging pumps, it may be necessary to increase the speed controller of these pumps. It should also be noted that SI actuation will trip the main feedwater pumps. If this is undesirable, the operator can manually align the system for safety injection and manually start the high-head SI pumps.

The check on RCS pressure is intended to alert the operator to a condition which would reduce charging or SI pump injection into the RCS and, therefore, boration. The PRZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient. The contingent action is a rapid depressurization to a pressure which would allow increased injection flow. When primary pressure drops 200 psi below the PORV pressure setpoint, the PORVs should be closed. The operator must verify successful closure of the PORVs, closing the isolation valves, if necessary.

RNP DIFFERENCES/REASONS

The RNP procedure provides detail, as requested by the ERG. The RNP procedure checks the flow path alignment prior to verifying two pumps running since this is a more logical alignment for positive displacement pumps. There is no deviation from the intent of the step.

ERG Step 4d is not included in the RNP procedure due to use of high head positive displacement charging pumps. This step served the purpose of giving guidance to plants with centrifugal charging pumps. Centrifugal charging pumps did not have the high head capability to discharge Borated water into the RCS at system pressures above 2335 psig. HBR uses High Head Positive Displacement Pumps that can deliver sufficient Borated water into the RCS at this pressure.

SSD DETERMINATION

This is an SSD per criterion 4.

5 5 WOG BASIS

STEP: Verify Containment Ventilation Isolation

PURPOSE: To ensure non-essential containment ventilation penetrations are isolated

BASIS:

Non-essential ventilation penetrations are isolated to prevent potential release of radioactive materials from containment.

This step is addressed in FR-S.1 in accordance with the ATWS analytical case of the "Accidental Depressurization of the RCS Without Reactor Trip" (See Section 2.0, page 48) which results in the most releases of mass and energy into the containment. As a result, verification of containment ventilation should conservatively always be performed independent of the RCS pressure.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

RNP DIFFERENCES/REASONS

RNP has revised this step to be a Check step with an RNO providing specific guidance for the action to initiate of Containment Ventilation Isolation. The RNP step performs the same actions that Verify would accomplish in the WOG step.

SSD DETERMINATION

This is not an SSD.

6-7 C6 WOG BASIS

PURPOSE: To alert the operator that he should verify proper actuation of all SIS actuated equipment

BASIS:

It is possible to make a transition to this guideline without having performed the verification of automatic SI actions in E-0. This caution specifically instructs the operator to perform the verification. This verification is started after Steps 1 through 5 of FR-S.1 since the first five steps deal directly with ATWS mitigation while the E-0 actions deal with system alignment for design basis events.

KNOWLEDGE:

Verification of automatic SI actions should be initiated and performed in parallel with the subsequent steps of this guideline as manpower and time permit.

RNP DIFFERENCES/REASONS

The ERG caution has been placed in two actions steps, formatted as a continuous action in order to remove the action from cautions and notes. The steps for verification have been placed in a supplement.

SSD DETERMINATION

This is an SSD per criterion 11.

8 6 WOG BASIS

PURPOSE: To determine if earlier control room actions were successful in producing reactor and turbine trips and, if not, to initiate local actions

BASIS:

Reactor trip is the fastest mechanism for adding negative reactivity to the reactor core. Turbine trip removes a large source of positive reactivity addition (heat removal by steaming), and will conserve SG inventory for the limiting ATWS event. If any of these actions have not been successfully achieved when attempted from the control room, an operator should be dispatched to perform the actions locally. Local actions were delayed until now because they will be more time consuming to initiate and complete, but may still be effective. Local reactor trip actions are performed first since the sooner a trip is obtained the less severe the ATWS transient will be.

RNP DIFFERENCES/REASONS

The local actions have been moved to step 1 in order to expedite tripping the reactor (see step 1 above).

SSD DETERMINATION

This is not an SSD.

52. 076 K1.01 001

SW-739, CCW HEAT EXCHANGER "A" RETURN, and SW-740, CCW HEAT EXCHANGER "B" RETURN, are throttled in the closed direction.

Which ONE(1) of the following completes the statement below?

This will cause Service Water Header Pressure to __(1)__ and CCW Heat Exchanger Outlet Temperature to __(2)__.

A. (1) lower

(2) lower

B. (1) lower

(2) rise

C. (1) rise

(2) lower

D.✓ (1) rise

(2) rise

The correct answer is D.

A. Incorrect - Throttling SW-739 / SW-740 in the closed direction will cause a rise in SW Header Pressure along with a reduction of SW flow through the CCW Heat Exchanger. The reduction in SW flow through the Heat Exchanger will cause a rise in CCW Heat Exchanger Outlet temperature.

B. Incorrect - Throttling SW-739 / SW-740 in the closed direction will cause a rise in SW Header Pressure along with a reduction of SW flow through the CCW Heat Exchanger. The reduction in SW flow through the Heat Exchanger will cause a rise in CCW Heat Exchanger Outlet temperature.

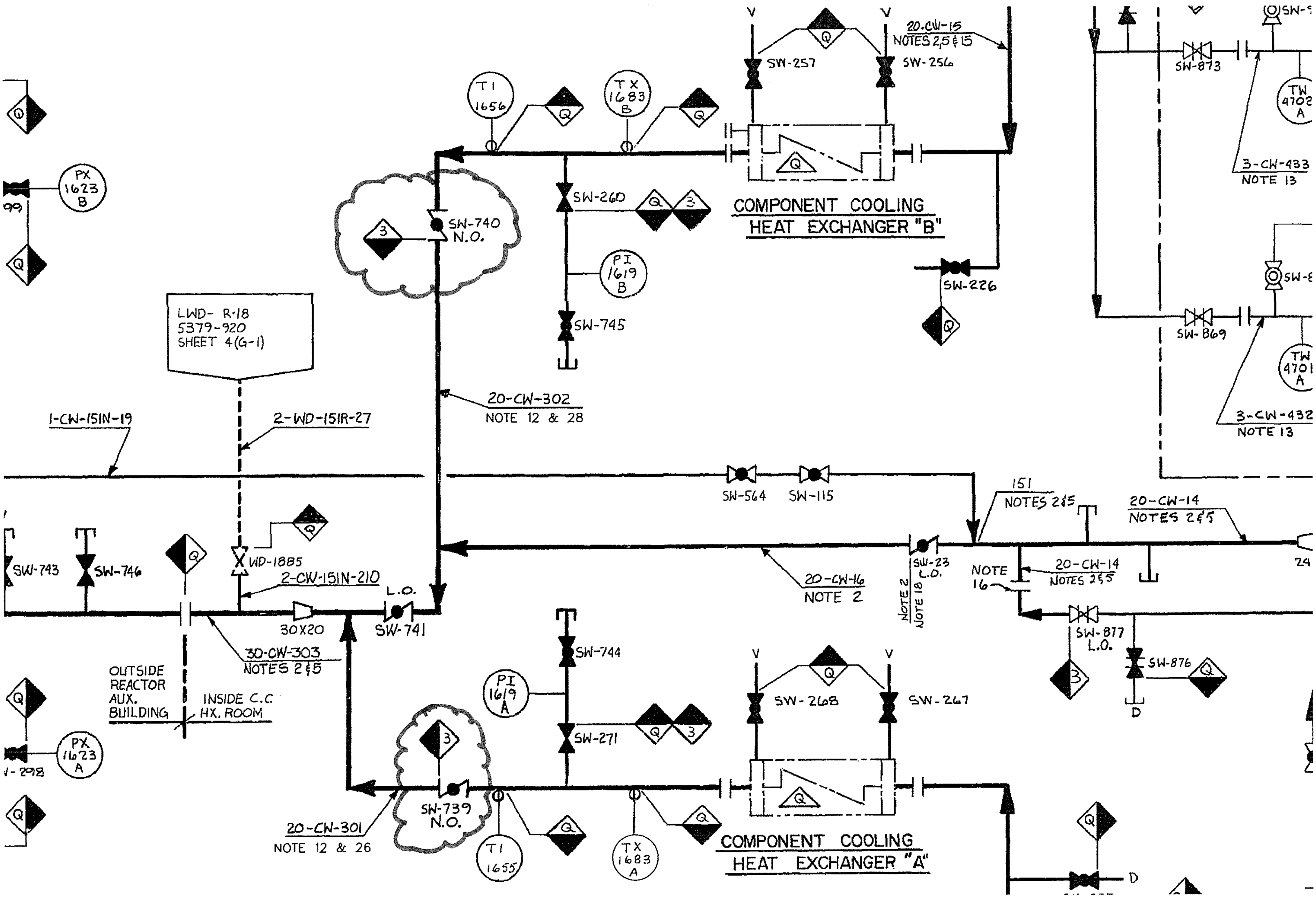
C. Incorrect - Throttling SW-739 / SW-740 in the closed direction will cause a rise in SW Header Pressure along with a reduction of SW flow through the CCW Heat Exchanger. The reduction in SW flow through the Heat Exchanger will cause a rise in CCW Heat Exchanger Outlet temperature.

D. Correct - Throttling SW-739 / SW-740 in the closed direction will cause a rise in SW Header Pressure along with a reduction of SW flow through the CCW Heat Exchanger. The reduction in SW flow through the Heat Exchanger will cause a rise in CCW Heat Exchanger Outlet temperature.

Question 52
Tier 2 / Group 1
K/A Importance Rating - RO 3.4 SRO 3.3

Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: CCW system

Reference(s) - Sim/Plant design, System Description, OP-903, Drawing G-190199 Sheet 9
Proposed References to be provided to applicants during examination - None
Learning Objective - SW 007
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.2 to 41.9 / 45.7 to 45.8
Comments - K/A is met because the candidate must know how throttling SW return valves will effect SW Header Pressure and CCW HX Outlet temperature.


COMPONENT COOLING HEAT EXCHANGER "B"
COMPONENT COOLING HEAT EXCHANGER "A"
20-CW-15 NOTES 2,5 & 15
SW-257
SW-256
SW-226
T I 1656
T X 1683 B
SW-260
P I 1619 B
SW-745
SW-740 N.O.
LWD- R-18 5379-920 SHEET 4(G-1)
20-CW-302 NOTE 12 & 28
1-CW-151N-19
2-WD-151R-27
SW-564
SW-115
151 NOTES 2&5
20-CW-14 NOTES 2&5
SW-23 L.O.
NOTE 2 NOTE 18
NOTE 16
20-CW-16 NOTE 2
SW-877 L.O.
SW-876
D
SW-873
SW-869
3-CW-433 NOTE 13
3-CW-432 NOTE 13
TW 4702 A
TW 4701 A
PX 1623 B
PX 1623 A
SW-743
SW-746
WD-1885
2-CW-151N-210
L.O.
SW-741
30X20
30-CW-303 NOTES 2&5
OUTSIDE REACTOR AUX. BUILDING
INSIDE C.C HX. ROOM
SW-744
P I 1619 A
SW-271
SW-268
SW-267
SW-739 N.O.
20-CW-301 NOTE 12 & 26
T I 1655
T X 1683 A

5.14 Valves V6-16A, V6-16B, and V6-16C have automatic closure features to isolate the Turbine Building.

- Valve V6-16A will close if PSL-1616A reaches 31 psig decreasing for 60 seconds with a 20ET Turbine Trip signal present.
- Valve V6-16B will close if PSL-1684A reaches 31 psig decreasing for 60 seconds with a 20AST Turbine Trip signal present.
- Valve V6-16C will close if PSL-1616B **OR** PSL-1684B reaches 31 psig decreasing for 60 seconds with a 20ET **OR** 20AST Turbine Trip signal present.

Keylock switches are provided to inhibit the auto-closure feature for V6-16A, V6-16B, or V6-16C for maintenance, testing, or when the unit is in cold shutdown. (MODES 5 or 6)

NOTE: The most recently performed SP-1479 will provide the number of turns corresponding to POS 1 and POS 2. This is information is being provided should it be necessary to replace the POS 1 / POS 2 labels.

This information is provided locally by Instructional Aid 96-OP-15.

5.15 Throttling of SW-739, CCW HEAT EXCHANGER "A" RETURN, and SW-740, CCW HEAT EXCHANGER "B" RETURN, shall be limited to position 1 (POS 1) with 2 CCW Heat Exchangers in service, and position 2 (POS 2) with one CCW Heat Exchanger in service. This is to prevent exceeding the capability of 2 SW pumps during a 2 SW pump accident response.
These valve positions will permit approximately 5000 gpm flow with 2 heat exchangers in operation (POS 1) and 10,000 gpm flow with 1 heat exchanger in operation (POS 2).

8.2.2 Adjusting Service Water Pressure Using SW-739/SW-740 INIT VERI

1. This revision has been verified to be the latest revision available. ______ ____
Date

CAUTION

With Service Water supplied to a single CCW heat exchanger, running Service Water pump(s) may need to be limited to less than or equal to two running pumps to maintain desired pressures and to avoid excessive piping vibration. Higher heat loads such as during Summer conditions may warrant operation of three SW pumps to maintain system pressure at the discretion of the SM.

2. **IF** only CCW Heat Exchanger "A" is in service, **THEN THROTTLE** SW-739, CCW HEAT EXCHANGER "A" RETURN, as necessary to establish the following conditions:

   - SW-739 throttled open less than or equal to POS 2 ____ ____
   - SW header pressure 40 to 50 psig ____
   - CCW Heat Exchanger outlet temperature less than or equal to 105°F ____

3. **IF** only CCW Heat Exchanger "B" is in service, **THEN THROTTLE** SW-740, CCW HEAT EXCHANGER "B" RETURN, as necessary to establish the following conditions:

   - SW-740 throttled open less than or equal to POS 2 ____ ____
   - SW header pressure 40 to 50 psig ____
   - CCW Heat Exchanger outlet temperature less than or equal to 105°F ____

8.2.2 (Continued)

		INIT	VERI
4.	**IF** both CCW Heat Exchangers are in service, **THEN THROTTLE** SW-739 **AND** SW-740 as necessary to establish the following conditions:		
–	SW-739 throttled open less than or equal to POS 1	____	____
–	SW-740 throttled open less than or equal to POS 1	____	____
–	SW header pressure 40 to 50 psig	____	
–	CCW Heat Exchangers outlet temperature less than or equal to 105°F	____	

	Initials	Name (Print)	Date
Performed By:	______	______________________	____
	______	______________________	____
	______	______________________	____
Approved By:		______________________ Shift Manager	____ Date

5.28 The Service Water Header pressures, as indicated on PI-1616 and PI-1684, should be maintained within the following normal ranges:

- 40 to 50 psig with less than four pumps operating.
- 40 to 55 psig with four pumps operating.

Service Water Header pressure is approximately 3 psig less than Service Water Pumps discharge pressure.

Service Water Header pressures of up to 57 psig are acceptable, provided none of the Service Water Pumps are operated continuously in the unstable flow range of 60 to 68 psig pump discharge pressure.

The Service Water System should **NOT** be operated with header pressures of greater than 57 psig. The Service Water Pumps should **NOT** be operated in the unstable flow region of 60 to 68 psig pump discharge pressure. If operation in the unstable flow region is necessary, the time of operation should be minimized and shall **NOT** exceed 30 minutes. If a Service Water Pump is operated in the unstable region of 60 to 68 psig discharge pressure, the pump is **NOT** inoperable, however, Engineering should be contacted to determine if additional pump testing is needed.

Operating with header pressures greater than the normal ranges is **NOT** desirable and should be avoided when possible. Service Water pressure can be lowered by initiating or increasing Service Water flow through the Turbine Lube Oil Coolers, Hydrogen Coolers or the Exciter Air Coolers.

Prior to isolating Service Water to a CCW heat exchanger, running Service Water pump(s) may need to be reduced to less than or equal to two running pumps to maintain desired pressures and to avoid excessive piping vibration. Higher heat loads such as during Summer conditions may warrant operation of three SW pumps to maintain system pressure at the discretion of the SM.

When Service Water is isolated to a CCW Heat Exchanger, sufficient Service Water flow paths should be maintained to support two Service Water Pump operation and **NOT** result in excessive piping vibration. During significant Maintenance periods such as refueling outages, this can be accomplished by maintaining Service Water flow through a minimum of two containment HVH units or other methods such as continuous Service Water strainer backwash. If sufficient system flow paths to minimize vibration can **NOT** be maintained, Engineering should be consulted and consideration given to reducing supply to one Service Water pump. (NCR 00301890)

53. 078 K3.02 001

Which ONE (1) of the following identifies the fail position of TCV-144, NON-REG HX OUTLET TEMP CONTROL, valve on a loss of Instrument Air AND the effect this will have on reactivity?

TCV-144 will fail __(1)__ AND this will insert __(2)__ reactivity.

A.✓ (1) OPEN

(2) Positive

B. (1) OPEN

(2) Negative

C. (1) CLOSED

(2) Positive

D. (1) CLOSED

(2) Negative

The correct answer is A.

A. Correct. Valve will fail open and increase cooling of letdown through the NRHX. This cooler letdown water will cause the Demins to have a larger affinity for Boron. Boron will be removed from the letdown flow and cause a positive reactivity insertion due to the decrease in boron concentration.

B. Incorrect. Valve will fail open and increase cooling of letdown through the NRHX. This cooler letdown water will cause the Demins to have a larger affinity for Boron. Boron will be removed from the letdown flow and cause a positive reactivity insertion due to the decrease in boron concentration. Since TCV-144 failing open results in a lower letdown temperature the candidate could correlate this to a lowering of the affinity of Boron in the Demin. which would mean that boron would be released. The release of boron would result in a negative reactivity insertion.

C. Incorrect. TCV-144 will fail open to maintain cooling to the letdown and protect the Demins. Several valves in the CVCS system fail in the closed direction and the candidate may not understand the basis for having this valve fail open. If TCV-144 failed closed then letdown temperature would increase resulting in the Demins releasing boron and thus a negative reactivity insertion would occur. The candidate may correlate the increase of letdown temperature to an increase in the affinity of boron in the Demin. which would mean that boron concentration would lower. The lower boron concentration would equate to a positive reactivity insertion.

D. Incorrect. TCV-144 will fail open to maintain cooling to the letdown and protect the Demins. Several valves in the CVCS system fail in the closed direction and the candidate may not understand the basis for having this valve fail open. If TCV-144 failed closed then letdown temperature would increase resulting in the Demins releasing boron and thus a negative reactivity insertion would occur. The candidate may correlate the increase of letdown temperature to an increase in RCS temperature which, due to the negative MTC, would result in a negative reactivity insertion.

Question 53
Tier 2 / Group 1
K/A Importance Rating - RO 3.4 SRO 3.6

Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Systems having pneumatic valves and controls

Reference(s) - Sim/Plant design, System Description, AOP-017
Proposed References to be provided to applicants during examination - None
Learning Objective - AIR 009, CVCS
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.6
Comments - This meets the K/A by having the candidate identify the failed position of a CVCS air operated valve and identify the effect this failure will have on RCS boron concentration and thus reactivity.

ATTACHMENT 1

MAJOR COMPONENTS AFFECTED BY LOSS OF IA

(Page 2 of 5)

1. (CONTINUED)

   v. LCV-115B, EMERG MU TO CHG SUCTION - NO FAILURE (Back-Up Air)

   w. LCV-460 A & B, LTDN LINE STOPS - CLOSED

   x. PCV-145, LETDOWN PRESSURE PCV - OPEN

   y. TCV-143, VCT/DEMIN DIV - FAILS TO VCT

   z. TCV-144, NON-REG HX OUTLET TEMP CONTROL - OPEN

2. Component Cooling Water System Components FAIL POSITION

   a. CC-739, CCW FROM EXCESS LTDN HX - CLOSED

3. Containment Ventilation System Components FAIL POSITION

   a. CV VENTILATION ISOLATION VALVES - CLOSED

4. Feedwater and Condensate System Components FAIL POSITION

   a. FEED REG VALVES - CLOSED

   b. FEED REG BYPASS VALVES - CLOSED

   c. HCV-1459, LP HEATERS BYP - OPEN

   d. LCV-1417A, HOTWELL LEVEL CONTROL VALVE - OPEN

   e. LCV-1530A, HEATER DRAIN TANK LEVEL CONTROL VALVE - AS IS

   f. LCV-1530B, HEATER DRAIN PUMPS SUCTION DUMP TO CONDENSER - OPEN

   g. FCV-1596, COND QUENCH TO HDT "A" REG - OPEN

   h. FCV-1597, COND QUENCH TO HDT "B" & DRAIN PUMPS SUCT REG - OPEN

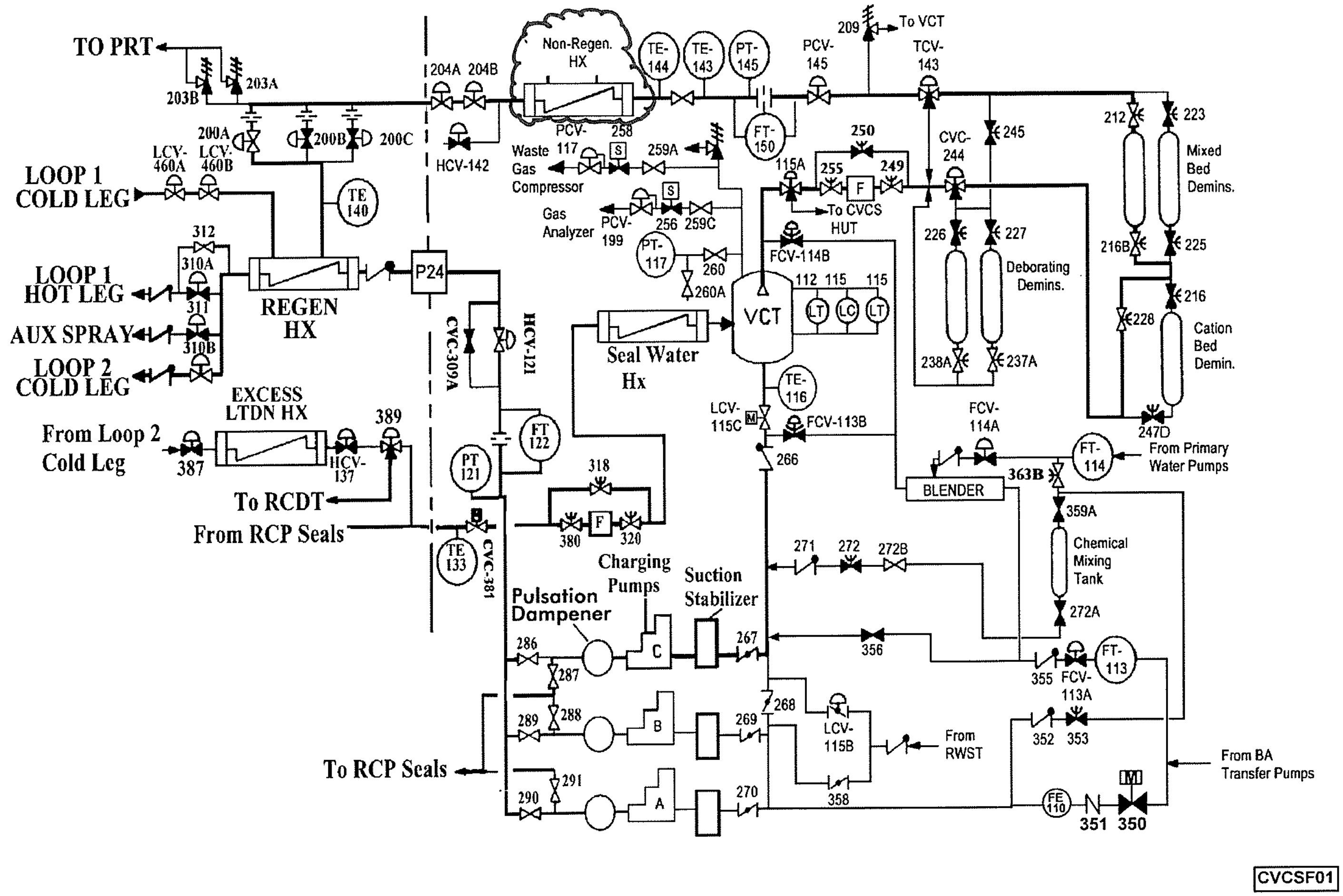
   i. QCV-10426, COND POL SEC BYP - OPEN

5. Instrument Air System Components FAIL POSITION

   a. PCV-1716, INSTRUMENT AIR ISO TO CV - CLOSED

CVCS FLOW DIAGRAM

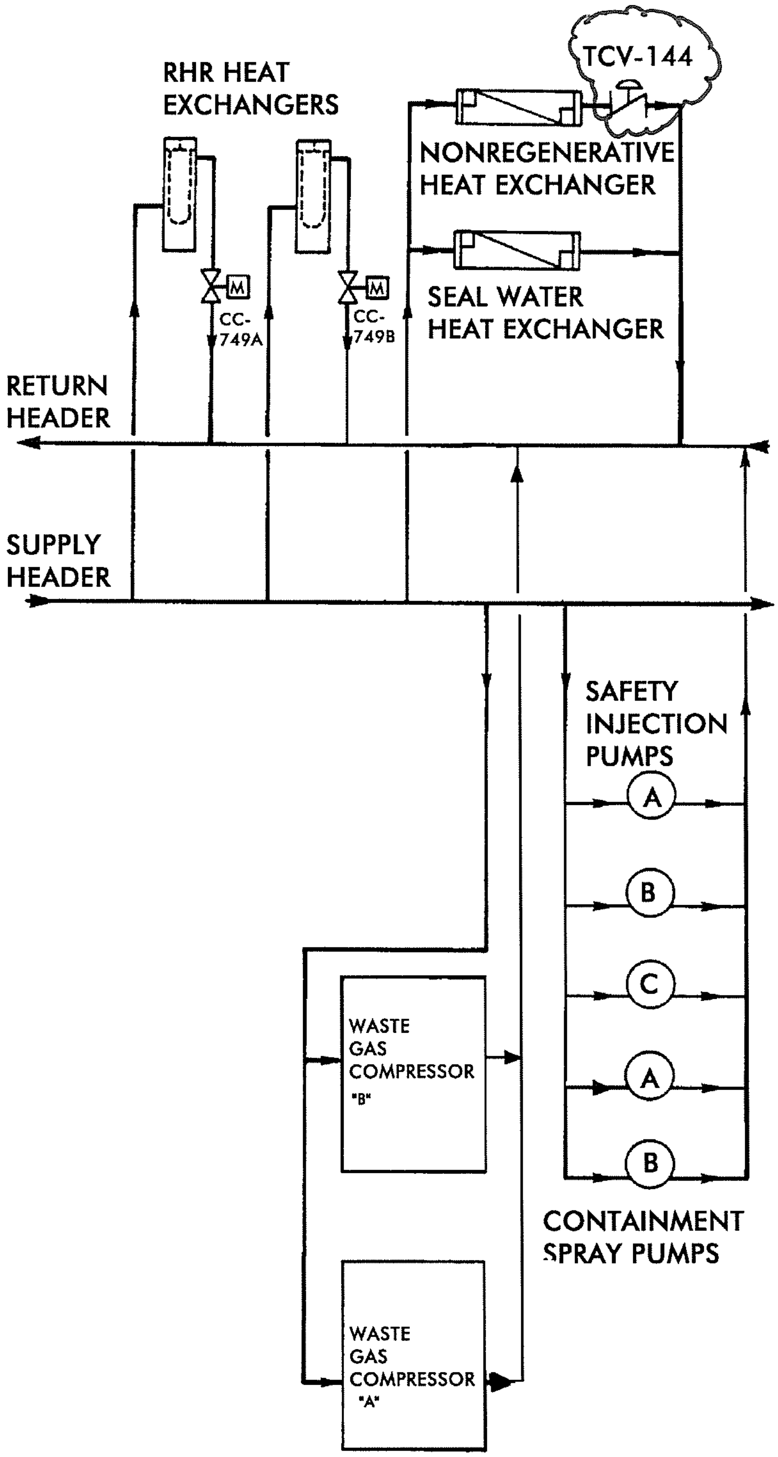
CVCS-FIGURE-1



INFORMATION USE ONLY

COMPONENT COOLING WATER SYSTEM

CCW-FIGURE-2



8.4.19 Adjusting TCV-144

INIT

1. This revision has been verified to be the latest revision available. ________ Date ____

2. Charging **AND** Letdown are in service. ____

3. The pertinent Reactivity Management requirements of OPS-NGGC-1306 have been referenced **AND** discussed during a pre-job brief for this activity. ____

BEGIN CRITICAL STEPS

4. TCV-143, VCT/DEMIN DIVERSION VALVE, in AUTO. ____

5. **IF** more than one letdown orifice is in service **AND** it is desired to reduce Letdown flow, **THEN PERFORM** the following:

   a. **NOTIFY** RC that Letdown flow will be changed therefore the affected areas should be monitored for changing radiological conditions. ____

      RC person notified: ____________________

   b. **VERIFY ONE** letdown orifice in service. ____

   - CVC-200A, LTDN ORIFICE (45 gpm) OPEN / CLOSED (Circle one) ____
   - CVC-200B, LTDN ORIFICE (60 gpm) OPEN / CLOSED (Circle one) ____
   - CVC-200C, LTDN ORIFICE (60 gpm) OPEN / CLOSED (Circle one) ____

   c. **VERIFY** a corresponding reduction in charging flow to match the reduced letdown flow. ____

   d. **WHEN** letdown temperature has stabilized, THEN CONTINUE with this procedure. ____

8.4.19.5 (Continued) INIT

NOTE: Seal inlet temperatures may vary seal leak-off flow. Increasing temperature will decrease the fluid viscosity which should provide slightly more flow at steady state conditions. Likewise lowering seal inlet temperature should provide slightly less flow at steady state conditions.

Letdown temperature can be monitored using TI-144 on the RTGB **OR** ERFIS point CHT0145A displayed on a terminal in the Control Room.

Setting TC-144, temperature potentiometer to between 3.33 and 4.33 will normally control Letdown temperature between 100 degrees F and 115 degrees F as indicated on TI-144, NON REG HXOUTLET TEMP.

Setting TC-144 potentiometer to the low end of the control band will result in improved RCP seal performance.

CAUTION

Small temperature changes should be applied, increments of approximately 1-2°F with a 1 to 2 hour stabilization period. Closely monitor the results.

During normal plant operation RCP seals injection temperature limited to 60°F minimum and 130°F maximum. Exceeding these values could cause damage to the RCP seals.

The temperature of the reactor coolant letdown, downstream of the Non-Regenerative Heat Exchanger should be maintained less than 127ºF and must not exceed 140ºF.

e. **ADJUST** the potentiometer on TC-144 to establish the desired temperature. ____

6. **VERIFY** TCV-144 is in the desired position. ____

8.4.19 (Continued) INIT

7. **ADJUST** VCT temperature as required to maintain the limits within the requirements in the table below. ____

PARAMETER	NORMAL	MINIMUM	MAXIMUM
No. 1 Seal Leakoff Flow (1)	1-3.2 gpm	0.8 gpm	6.0 gpm

8. **VERIFY** during temperature adjustments that the RCP operational limits (eg. Bearing temperatures, etc.) **DO NOT** significantly change. ____

(1) If No. 1 Seal leakoff flow is less than 1 gpm **OR** greater than 5 gpm, but within minimum and maximum values, RCP and seal parameters should be monitored and RESS should be notified to contact Westinghouse. If No. 1 Seal leakoff flow is greater than 3.2 gpm, refer to the DS/SBO limits listed in OP-101, section 5.2, Step 5.2.1.25.

END CRITICAL STEPS

	Initials	Name(Print)	Date
Performed By:	________	____________________	________
	________	____________________	________
	________	____________________	________
	________	____________________	________

Approved By: ______________________________ ________
Shift Manager Date

54. 103 A4.01 001

Given the following plant conditions:

- The plant is at 100% RTP.
- APP-002-B7, CV NAR RANGE HI/LO PRESS illuminates.
- CV Pressure indicates -0.4 psig, lowering (more negative).

Which ONE (1) of the following describes the action necessary to clear the alarm IAW OP-921, Containment Air Handling?

Open the Containment ....

A. Pressure Relief Valves V12-10 and V12-11 ONLY.

B. Pressure Relief Valves V12-10 and V12-11 AND lower Service Water flow to the Containment HVH units.

C.✓ Vacuum Relief Valves V12-12 and V12-13 ONLY.

D. Vacuum Relief Valves V12-12 and V12-13 AND lower Service Water flow to the Containment HVH units.

The correct answer is C.

A-Incorrect. Wrong valves. Appropriate for a pressure relief.

B-Incorrect. Wrong valves. Appropriate for a pressure relief. Lowering SW flow to the CV HVH units would result in an increase in CV temperature and pressure. The candidate may think that this is an appropriate act.

C-Correct. Negative pressure requires vacuum relief.

D-Incorrect. First part of question is correct. Lowering SW flow to the CV HVH units would result in an increase in CV temperature and pressure. The candidate may think that this is an appropriate act.

Question 54
Tier 2 / Group 1
K/A Importance Rating - RO 3.2 SRO 3.3

Ability to manually operate and/or monitor in the control room: Flow control, pressure control, and temperature control valves, including pneumatic valve controller

Reference(s) - Sim/Plant design, System Description, APP-002, OP-921
Proposed References to be provided to applicants during examination - None
Learning Objective - CV 007, CVHVAC Obj. 3
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5 to 45.8
Comments - K/A is met because the candidate must analyze the given CV pressure indication and determine the proper action to adjust the parameter into the desired range.

ALARM

CV NAR RANGE HI/LO PRESS

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

High

1. Instrument Air **OR** Nitrogen Leakage in the CV
2. Secondary leakage in the CV
3. RCS leakage in the CV

Low

1. Cooldown of CV atmosphere following purge **OR** pressure relief.

OBSERVATIONS

1. Containment Vessel Pressure Indicators PI-950B, PI-951, PI-953, PI-955, PI-950A, PI-952, and PI-954.

ACTIONS		CK (✓)
1.	**IF** excessive RCS leakage is indicated, **THEN REFER TO** AOP-016.	______
2.	**IF** excessive Secondary, Instrument Air, **OR** Nitrogen leakage is indicated, **THEN INITIATE** efforts to repair, as appropriate.	______
3.	**IF** Instrument Air pressure can **NOT** be maintained, **THEN REFER TO** AOP-017.	______
4.	**IF** required, **THEN PERFORM** a CV pressure relief using OP-921.	______
5.	**IF** required, **THEN PERFORM** a CV vacuum relief using OP-921.	______
6.	**REFER TO** ITS LCO 3.6.4.	______

DEVICE/SETPOINTS

1. PC-950B / +0.9 psig
2. PC-950B / -0.4 psig

POSSIBLE PLANT EFFECTS

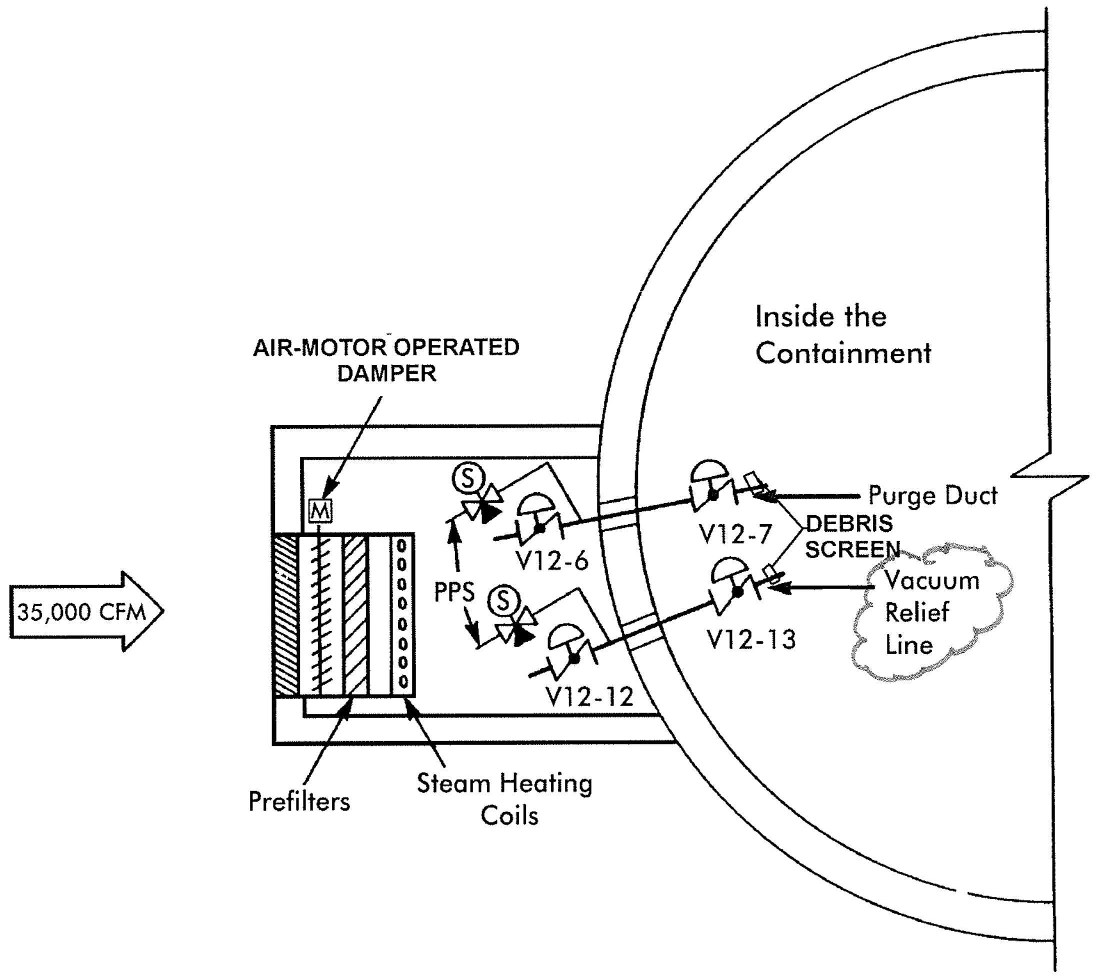
1. TECH SPEC LCO

REFERENCES

1. ITS LCO 3.6.4
2. AOP-017, Loss of Instrument Air
3. AOP-016, Excessive RCS Leakage
4. OP-921, Containment Air Handling
5. Hagan Drawing, H5957D73 (5379-3505)
6. CWD B-190628 Sh 496

OUTDOOR AIR MAKEUP SYSTEM AND VACUUM RELIEF SYSTEM

CVHVAC-FIGURE-1



cvhvaf01

INFORMATION USE ONLY

5.0 PRECAUTIONS AND LIMITATIONS

5.1 The maximum ΔP across the absolute filters is 2" W.C.

5.2 The Containment Recirculation Fans, HVH-1 through 4, shall meet the operability requirements of Improved Technical Specifications LCO 3.6.6 when applicable.

5.3 The Containment Purge System shall meet the operability requirements of ITS LCO 3.9.7 when applicable.

5.4 If the Containment Building internal pressure exceeds +1.0 psig internal pressure or -0.8 psig internal vacuum, then the REQUIRED ACTIONS of Improved Technical Specifications LCO 3.6.4 shall be entered.

5.5 If HVH-1, 2, 3, or 4 CV Recirc Fan is determined to have reverse rotation, the affected Fan Cooler Unit shall be started and the On-Call Operations Manager shall be notified.

5.6 When Safeguards has **NOT** been defeated in accordance with SPP-011, HVH-2 and HVH-4 will be considered the lead CV Recirc Fans and shall be started first and stopped last. When Safeguards has **NOT** been defeated in accordance with SPP-011 **AND** HVH-2 **OR** HVH-4 is **NOT** operating, then the associated breaker(s) control power fuses shall be removed and the CV Recirc Fan declared inoperable. Removing the breakers control power fuses renders the HVH units inoperable and requires entry into an LCO REQUIRED ACTION when TECH SPECS requires four CV Recirc Fans to be operable (MODES 1, 2, 3, and 4). This system configuration is to reduce the chances of a degraded voltage condition when the right circumstances are present and the plant experiences a "LOCA with offsite power available". (NRC Inspection Report No. 50-261/89-25, ESR 94-00616)

5.7 Repositioning the Containment HVH Unit Local/Remote Selector **OR** any activity which results in a loss of control power to the Containment HVH Unit breaker will result in the associated Containment HVH Unit Normal Intake Damper failing closed. The Normal Intake Damper will remain closed after the control power is restored until manually repositioned with the HVH UNIT DAMPER SWITCH. (CR 99-01218, CR 99-01751)

CONTINUOUS USE

INIT

8.4.2 Containment Vacuum Relief When Containment Integrity is Required

1. Initial Conditions

a. This revision has been verified to be the latest revision available. ______ ____
Date

NOTE:	The CV NAR RANGE HI/LO PRESS annunciator (LO setpoint) alarms at -0.4 psig. (APP-002-B7)

b. Electrical Distribution is in service IAW OP-603. ____

c. Instrument and Station Air System is in service IAW OP-905. ____

d. **VERIFY** Containment Ventilation Isolation is **RESET**. ____

e. **VERIFY** the Containment Purge Supply and Exhaust valves **CLOSED**: (ITS LCO 3.6.3)

V12-6 ____

V12-7 ____

V12-8 ____

V12-9 ____

8.4.2 (Continued) INIT VERI

2. Instructions

a. **PLACE** CV VAC RELIEF V12-12 & V12-13 Control Switch to the open position. ____

b. **CHECK**, by position indicating lights, the following are **OPEN**.

V12-12	____	
V12-13	____	
CV Intake Damper	____	

c. **WHEN** Containment pressure reaches between -0.025 psig and 0.0 psig on PI-950B, **THEN PERFORM** the following:

1) **PLACE** the CV VAC RELIEF V12-12 & V12-13 Control Switch to the close position. ____

2) **VERIFY**, by position indicating lights, the following are **CLOSED**.

V12-12	____	____
V12-13	____	____
CV Intake Damper	____	

	Initials	Name (Print)	Date
Performed By:	________	______________________	__________
	________	______________________	__________
	________	______________________	__________

Approved By: ______________________________ __________
Shift Manager Date

55. 103 K1.02 001

Given the following plant conditions:

- The RCS is at 275°F
- All RCPs are running.
- CVC-381, RCP Seal Return Isolation, has been declared inoperable due to improperly set torque switches.

Which ONE (1) of the following statements is correct IAW the RNP ITS?

A. Close CVC-381 ONLY.

B.✓ Close and deactivate CVC-381.

C. CVC-381 operability is NOT required for the given plant conditions.

D. Close CVC-380, Seal Water Return Filter Inlet, AND station a Dedicated Operator IAW OP-923, Containment Integrity.

The correct answer is B

UFSAR Section 6.2 shows CVC-381 as a single containment isolation valve with the CVCS as a closed system. This scenario would put you in LCO 3.6.3.C and not A or B.

A. Incorrect - ITS 3.6.3 required that the Containment Isolation Valve be isolated by use of at least one closed and de-activated automatic valve, closed manual valve or blind flange. CVC-381 is an motor operated valve that must be de-activated.

B. Correct

C. Incorrect - The given RCS temperature of 275°F places the plant in Mode 4. Containment Isolation Valves (ITS 3.6.3) are required to be Operable in Modes 1, 2, 3, and 4.

D. Incorrect - CVC-380 is the first manual isolation valve downstream of CVC-381, however, this valve cannot be credited as providing containment integrity. The filter bypass line taps in upstream of CVC-380 and contains a pressure indicator and two isolation valves that would need to be closed, if allowed by procedure and ITS. There is an option in OP-923, Containment Integrity, to assign a Dedicated Operator to close any "normally closed" Containment Isolation Valve to support approved procedures for short durations using administrative controls. The stem of the question does not support use of this option since the valve has just been declared inoperable.

Question 55
Tier 2 / Group 1
K/A Importance Rating - RO 3.2 SRO 3.3

Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems: Containment isolation/containment integrity

Reference(s) - Sim/Plant design, System Description, UFSAR Section 6.2, ITS 3.6.3, OP-923
Proposed References to be provided to applicants during examination - None
Learning Objective - CV 008
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.2 to 41.9 / 45.7 to 45.8
Comments - This meets the K/A by testing the candidates knowledge of the physical layout of the seal return line and associated containment isolation valve. The question also tests the candidates knowledge of what actions are required to meet ITS 3.6.3.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 2

OP-923

CONTAINMENT INTEGRITY

REVISION 41

5.0 PRECAUTIONS AND LIMITATIONS

5.1 During Modes 1, 2, 3, and 4, the Containment, Containment Air Lock **AND** each Containment Isolation Valve shall be OPERABLE.

5.2 The CRS / SM shall identify a Dedicated Operator if any of the valves in Attachment 10.2, Containment Isolation Valve Checklist - Main Steam System are to be opened with containment integrity set IAW Attachment 10.1, Containment Integrity Valve Checklist. Attachment 10.2 will be used to document "normally closed" Main Steam System Containment isolation valves that are opened when required for Reactor Coolant System temperature control / steam line warm-up.

5.3 A Dedicated Operator for operating valves on Attachment 10.1 **OR** Attachment 10.2 **SHALL** meet the following criteria:

- **NOT** a member of the minimum shift compliment required by OMM-001-2, with the exception of the BOP Operator as identified below.
- With the exception of the Main Steam Line PORV's (RV-1, RV-2, RV-3) and the MSIV Bypass Valves (MS-353A, B, and C) which require the dedicated operator to be available to locally close the valves, the BOP Operator may be designated for components operated on the RTGB. (CR 98-01039)
- For components operated on the RTGB, will be a licensed RO.
- For components operated in the field, will be qualified for the applicable watchstation **OR** qualified to perform valve operations IAW OMM-001-5 **AND** thoroughly briefed on the task by the CRS/SM.
- May perform other duties but shall remain available to immediately close the affected valve(s) to establish Containment Integrity when notified. Individual shall remain in the vicinity of the valves **OR** in the Control Room.
- Will maintain constant communications with the Control Room via radio or other similar device when outside the Control Room.

5.4 The Containment Integrity Valve Checklist, Attachment 10.1, concentrates on valves, flanges, plugs, and caps, and when completed, provides assurance that the requirements of containment integrity are being met provided the systems are intact. The checklists do **NOT** identify all containment barriers, such as certain pressure transmitters or closed system components outside containment. If questions arise regarding containment integrity, the UFSAR, Reactor Containment Isolation Generic Issue Document, and Engineering should be consulted.

5.5 During Modes 1, 2, 3, and 4, components on Attachment 10.1 **NOT** in the required position may require entry into an Improved Technical Specification LCO 3.6.3 Required Action Statement. Components on Attachment 10.1 may be operated to support approved procedures for short durations using administrative controls and designation of a dedicated operator by the CRS/SM as outlined in Precaution and Limitation 5.3 without entering the LCO Action Statement. The component manipulations shall be properly documented including the dedicated operator identification (in the procedure being performed, Attachment 10.2 of this procedure if applicable, autolog, or OPS-NGGC-1303, Attachment 5). (NCR 00297818)

5.6 Reactor Containment penetration design provides for two redundant barriers at each penetration. Redundant means that either barrier is capable of isolating the penetration regardless of the state of operation or failure of the other. For HBR, these barriers, or containment boundaries, are comprised of:

- Closed system inside containment
- Closed system outside containment
- Isolation valve(s)
- Pipe cap(s)
- Blind flange(s)

Containment Isolation Valves (CIV's) are a subset of all the available containment barriers. The UFSAR and Reactor Containment Isolation Generic Issue Document identify CIV's uniquely.

5.7 This procedure has been screened IAW PLP-037 criteria. The Case determination is defined at the beginning of the section(s).

5.8 The principles of **ALARA** shall be followed in planning and performing work and operations in the Radiation Control Area.

PIPE ALLEY (Continued)

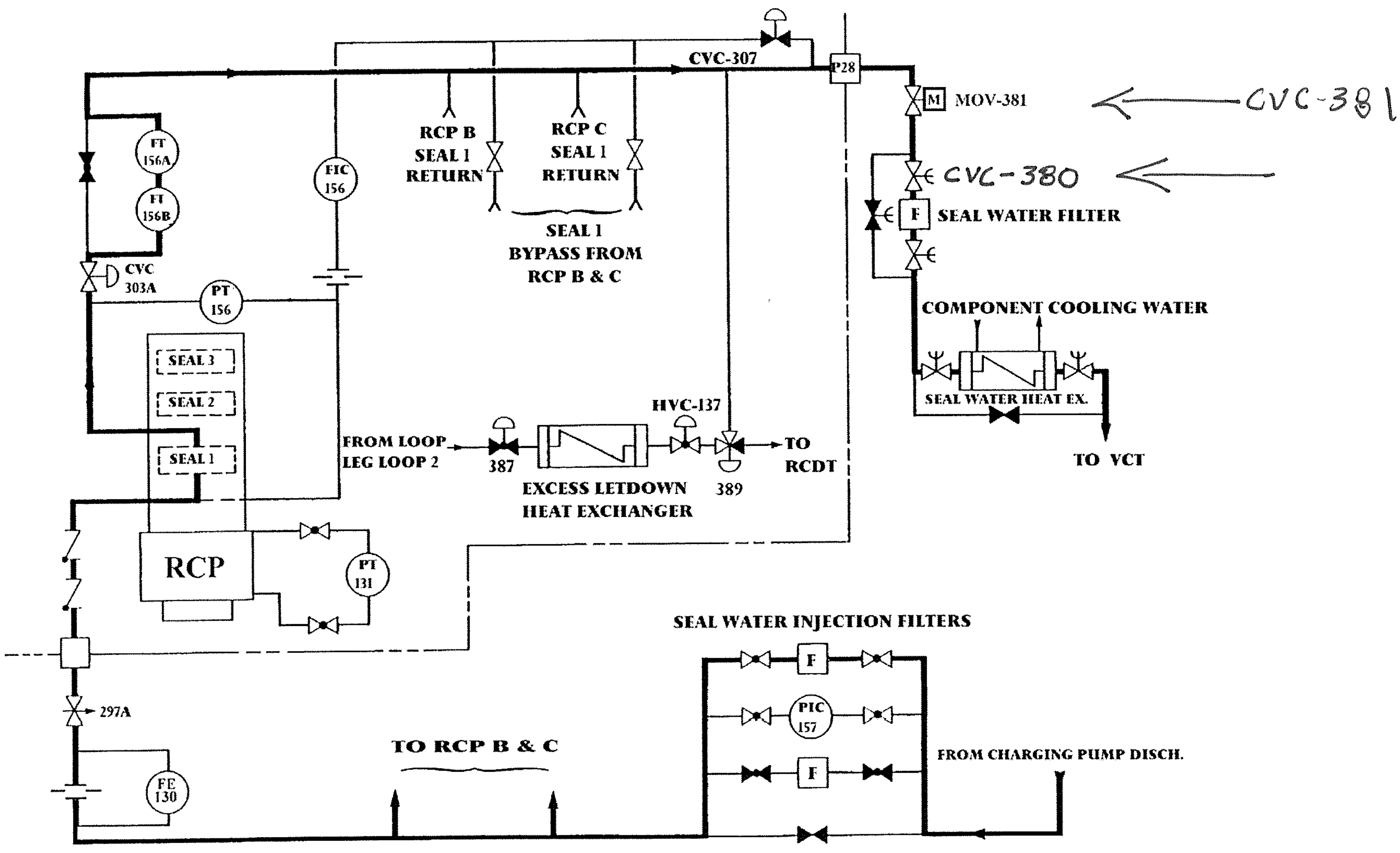
VALVE NO.	VALVE DESCRIPTION	POSITION	INIT	TAG ATTD	VERI
* CC-716B	CLG WTR INLET ISOL VALVE	OPERABLE		(1)	
* CC-730	BRG CLG WTR OUTLET ISOL VALVE	OPERABLE		(1)	
* FCV-626	THERM BARRIER OUTLET ISOL VALVE	OPERABLE		(1)	
* CC-932	FCV-626 AND CC-735 BYPASS ISOLATION	LOCKED CLOSED		(1) (2)	
CC-735	THERM BARRIER OUTLET ISOL VALVE	OPERABLE			
*CC-938	HIGH SIDE DRAIN (FIC-626A)	LOCKED CLOSED			
		CAP INSTALLED			
*CC-939	LOW SIDE DRAIN (FIC-626A)	LOCKED CLOSED			
		CAP INSTALLED			
* CC-739	EXCESS LTDN HX OUTLET VALVE	OPERABLE		(1)	
* CVC-204A	LETDOWN LINE ISOL	OPERABLE		(1)	
* CVC-204B	LETDOWN LINE ISOL	OPERABLE		(1) (2)	
CVC-299R	RCP "A" SEAL INJECTION LINE VENT	LOCKED CLOSED			
		CAP INSTALLED			
* CVC-381	RCP SEAL WTR RTRN ISOL	OPERABLE		(1) (2)	

* Containment Isolation Valve (CIV)

(1) Verify CV Isolation (Orange) Tag attached

(2) Verify EOP/DSP (Blue or Green) Tag attached

SEAL INJECTION
CVCS-FIGURE-10



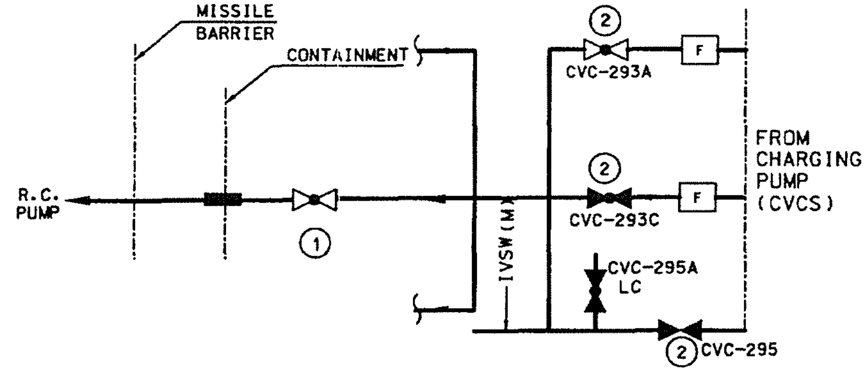
CVCSF10

INFORMATION USE ONLY

TABLE 6.2.4-1 (Continued)

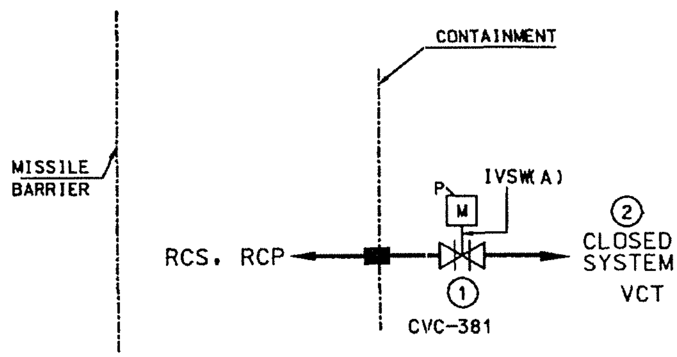
PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4.	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
22	Excess Letdown Heat Exchanger Cooling Water Out	-7	CC-739 C.S.	Gate -	Air -	Yes -	Open -	Open -	Closed -	FC -	T -	- -	No -	W -	NE
23	Letdown Line	-7	CVC-204A CVC-204B	Globe Globe	Air Air	Yes Yes	Open Open	Closed Closed	Closed Closed	FC FC	T T	A A	No No	W W	NE
24	Charging Line	-7	CVC-282 CVC-202A CVC-309A	Globe Gate Globe	Man. Man. Man.	No No No	Open Open Closed	Open Open Closed	Closed* Closed* Closed*	- - -	No No No	M M M	No* No* No*	W W W	*May be used for High Pressure Safety Injection. Line is isolated after charging is shutdown to provide long term recovery.E
25	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297C CVC-293C CVC-293A CVC-295	Globe Globe Globe Gate	Man. Man. Man. Man.	No No No No	Throt. Closed Open Closed	Open Closed Open Closed	Closed* Closed* Closed* Closed	- - - -	No No No No	M M M M	Yes* Yes* Yes* Yes*	W W W W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is in secvice. Line is isolated after RCP is shutdown.to provide long term recovery. E
26	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297B CVC-293C CVC-293A CVC-295	Globe Globe Globe Gate	Man. Man. Man. Man.	No No No No	Throt. Closed Open Closed	Open Closed Open Closed	Closed* Closed* Closed* Closed	- - - -	No No No No	M M M M	Yes* Yes* Yes* Yes*	W W W W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is in secvice. Line is isolated after RCP is shutdown.to provide long term recovery. E
27	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297A CVC-293C CVC-293A CVC-295	Globe Globe Globe Gate	Man. Man. Man. Man.	No No No No	Throt. Closed Open Closed	Open Closed Open Closed	Closed* Closed* Closed* Closed	- - - -	No No No No	M M M M	Yes* Yes* Yes* Yes*	W W W W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is in secvice.Line is isolated after RCP is shutdown.to provide long term recovery. E
28	Reactor Coolant Pump Seal Water Return Line	-8	CVC-381 C.S.	DOV -	Mot. -	Yes -	Open -	Open -	Closed* -	As is -	P -	A -	No -	W -	*Closes on P Signal.NE

PENETRATIONS NO. 25, 26, 27 - REACTOR COOLANT PUMP SEAL WATER SUPPLY LINE

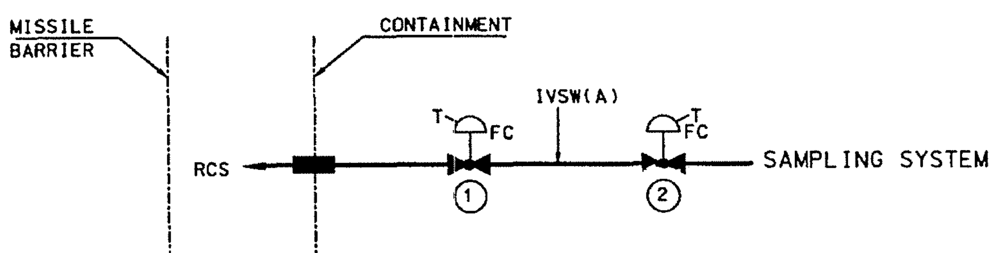


BARRIER	P-25	P-26	P-27
1	CVC-297C	CVC-297B	CVC-297A

PENETRATION NO. 28 - REACTOR COOLANT PUMP SEAL WATER RETURN LINE



PENETRATIONS NO. 29, 30, 31 - REACTOR COOLANT SYSTEM SAMPLE LINE



BARRIER	P-29	P-30	P-31
1	PS-956A	PS-956C	PS-956E
2	PS-956B	PS-956D	PS-956F

Revision No. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-25, P-26, P-27, P-28, P-29, P-3Ø, P-31	FIGURE 6.2.4-8

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.
5. Enter applicable Conditions and Required Actions of LCO 3.6.8, "Isolation Valve Seal Water (IVSW) System," when required IVSW supply to a penetration flowpath is isolated.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. --------------------- One or more penetration flow paths with one containment isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTE-------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------- Verify the affected penetration flow path is isolated.	 Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. --------------------- One or more penetration flow paths with two containment isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. --------------------- One or more Penetration flow paths with one containment isolation valve inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	<u>AND</u>		
	C.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

(continued)

56. 001 K3.01 001

Plant is at 100% RTP.

Which ONE (1) of the following describes an INITIAL effect of one control rod dropping fully into the core?

This will cause the Charging Pump Speed to initially __(1)__ due to the Pressurizer Reference Level __(2)__.

A.✓ (1) lower

(2) lowering

B. (1) rise

(2) lowering

C. (1) lower

(2) rising

D. (1) rise

(2) rising

The correct answer is A.

A. Correct. Tave will lower resulting in a lower PZR Reference Level sending a signal to the Charging Pump Controls in Auto to slow down. This will result in a reduction in Charging Flow.

B. Incorrect. Tave will lower resulting in a lower PZR Reference Level sending a signal to the Charging Pump Controls in Auto to slow down. This will result in a reduction in Charging Flow. Eventually, the reduction in Tave will cause actual PZR level to lower below the Pressurizer Reference Level and result in a rise in charging pump speed.

C. Incorrect. Tave will lower resulting in a lower PZR Reference Level sending a signal to the Charging Pump Controls in Auto to slow down. This will result in a reduction in Charging Flow. Eventually, the reduction in Tave will cause actual PZR level to lower below the new Pressurizer Reference Level and result in a rise in charging pump speed.

D. Incorrect. Tave will lower resulting in a lower PZR Reference Level sending a signal to the Charging Pump Controls in Auto to slow down. This will result in a reduction in Charging Flow. Eventually, the reduction in Tave will cause actual PZR level to lower below the Pressurizer Reference Level and result in a rise in charging pump speed. This answer is plausible since a rise in PZR Reference Level will typically cause a rise in charging pump speed, depending on actual PZR level at the beginning of the event.

Question 56
Tier 2 / Group 2
K/A Importance Rating - RO 2.9 SRO 3.0

Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: CVCS

Reference(s) - Sim/Plant design, System Description, GFES, AOP-001-BD
Proposed References to be provided to applicants during examination - None
Learning Objective - RDCNT 009
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.6
Comments - K/A met because candidate must understand the effect a dropped rod will have and Tave and thus Pressurizer Reference Level and Charging Pump speed.

Main Body (Continued)

Step	Description
6	This step provides transition to Section C if movement of the rod selector switch was successful in stopping the uncontrolled rod movement. If movement has stopped then operation may continue.
7	This step directs making a PA announcement for procedure entry if unexpected or continuous rod motion was not the reason for procedure entry. Refer to previous discussion for making the PA announcement for additional discussion.
8-10	These steps provide diagnosis for multiple rod drops. Multiple rod drops are a serious challenge and warrant prompt action. Since the possibility of a large reactivity transient is present for this event, the reactor is tripped instead of attempting to control the transient. The step is worded as an analyze action to allow for the operator to view the symptoms and make a determination. The reactor trip is called for in Modes 1 and 2. If in Mode 3 the step transitions to Section A.
11	This step will commence actions to restore Tavg to program Tref. A dropped rod will insert negative reactivity which will initially cause Tavg to be reduced by an amount in reactivity equal to the worth of the dropped rod. If left unattended Tavg will eventually partially recover, with a delta-T (from initial) equivalent to the reactivity loss from the doppler. The intent of the procedure is to manually control the Turbine to establish Tavg to Tref. Attachment 1, Turbine Load Adjustment, provides actions to quickly reduce turbine load to restore Tavg to program Tref. At this point no attempt will be made to move rods in order to assure that an inadvertent movement of a dropped rod will not occur.
12-13	This step provides the diagnostics for a dropped rod. With the current core configuration at RNP there should be no location that will not produce a prompt drop or 5% drop in the NIS when operating at elevated power levels. The key indications for a dropped rod are those that involve an actual flux change. The other indications are derived from IRPI and therefore do not differentiate between an IRPI failure and an actual dropped rod. The step is worded to allow any combination of symptoms. The bulleted list of symptoms is not intended to be any particular logic combination. The intent of the step is to review the symptoms, and based on operator judgment, determine if a dropped rod exists. After diagnosing a dropped rod the operator will transition to the appropriate section of the procedure for a dropped rod. If there is any doubt between dropped rod and IRPI failure, the operator should assume that the rod is dropped. Subsequent steps will verify the rod dropped. Additional items to look at for determination of a QPTR were added based upon a LOCT comment. This will provide consistency in RO/BOP response.

LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would de-energize and any backup heater in manual would remain energized.

5.1.6 PZR Level Control Setpoints (PZR Figure 9)

1. Level program as function of Median Select T_{avg} (TM-459)

for T_{avg} 547°F	22.2% of level span
for T_{avg} 575.9°F	53.3% of level span
(Program is linear from 547°F to 575.9°F)	
Low limit	22.2% of level span
High limit	53.3% of level span

2. Low-Low Level Heater Cutout (LC-459C, LC-460C) — 14.4% of level span

3. Level Controller (LC-459F)

Proportional gain	10% charging pump speed/% level deviation
Reset time constant	430 seconds

4. Letdown Valve Isolation — 14.4% of level span

5. Back-up Heaters on — +5% of programmed level

6.0 SYSTEM OPERATION

6.1 Normal Operation

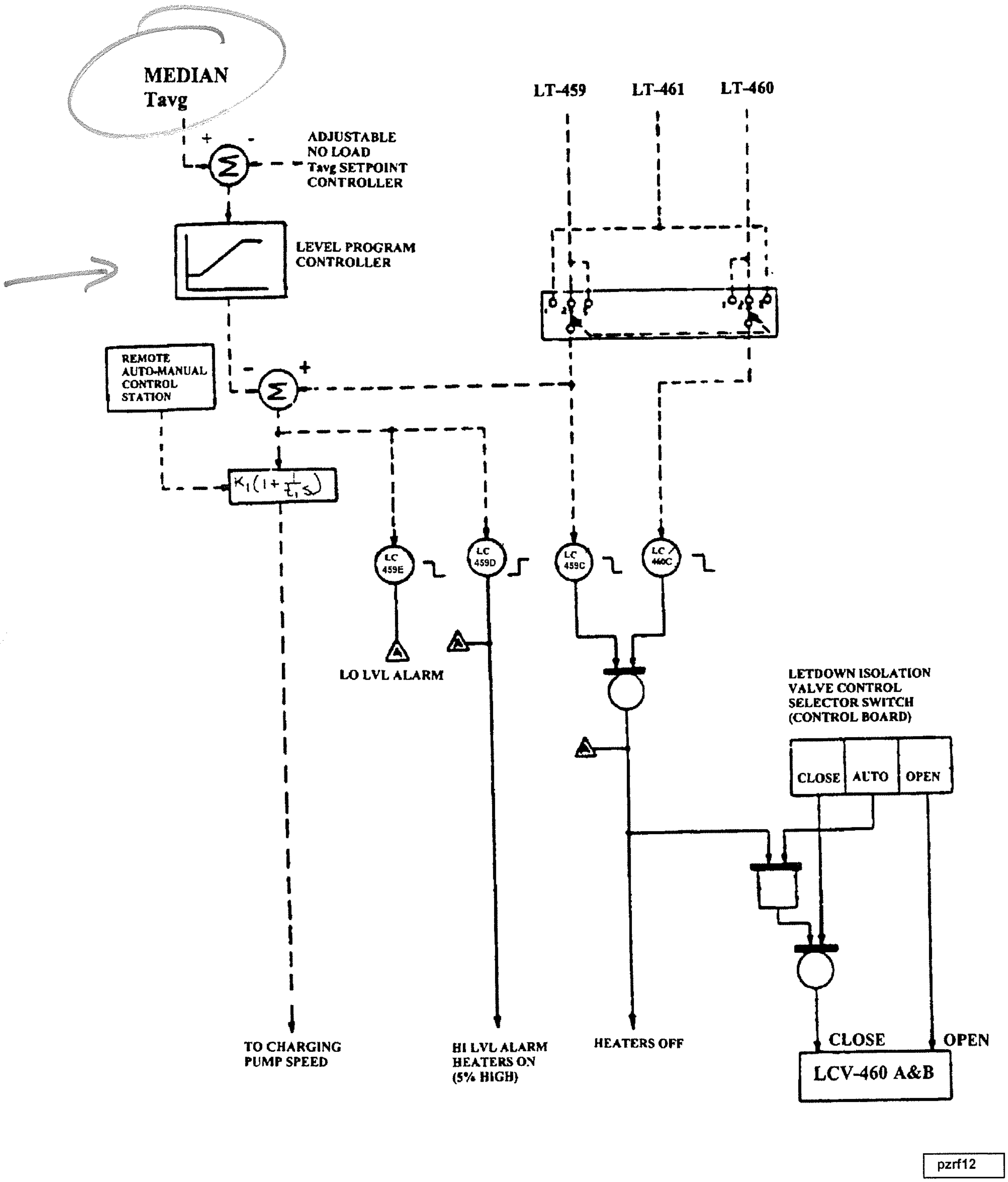
Insurge of RCS Coolant - produced by increase in T_{avg}. An insurge of coolant will reduce volume of the steam bubble causing an increase in the temperature and pressure of the steam. The steam space or bubble becomes superheated and some minor condensation occurs at surface and on walls.

The increased pressure causes the spray valve to open which cools and condenses a part of the steam bubble, thereby reducing pressure.

The increase in level will energize backup heaters if the level increases to 5% above

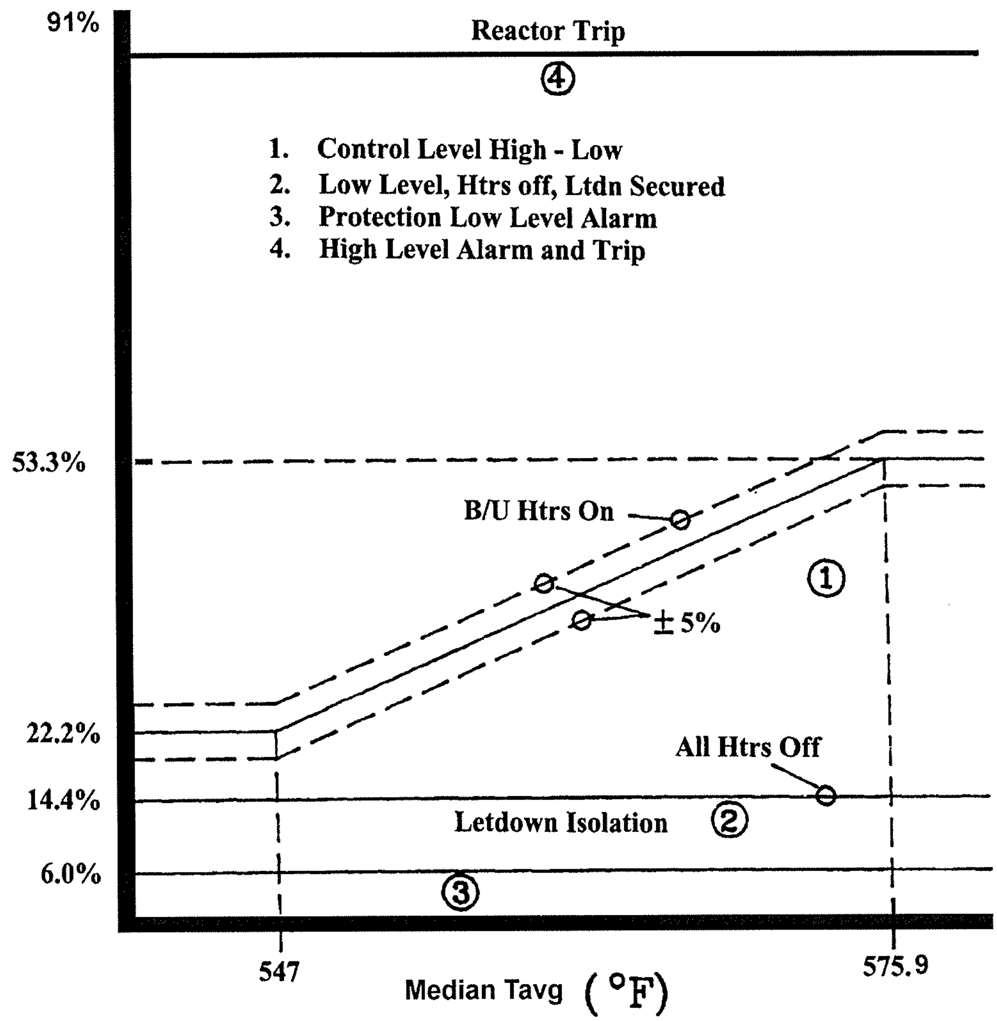
LEVEL CONTROLLER

PZR-FIGURE-10



INFORMATION USE ONLY

PROGRAM LEVEL
LEVEL SETPOINTS
PZR-FIGURE-9



pzrf11

INFORMATION USE ONLY

57. 015 K2.01 001

Given the following plant conditions:

- The plant is at 100% RTP.
- A loss of Inverter "A" occurs.

Which ONE (1) of the following identifies ALL the excore Nuclear Instrument channels that have lost power?

A. N-41 and N-51

B. N-43 and N-52

C. ✓ N-32, N-36 and N-42

D. N-31, N-35 and N-41

The correct answer is C.

A. Incorrect. N-41 is part of NIS Cabinet A that receives power from Instrument Bus 1. Instrument Bus 1 is powered from MCC-5. N-51 (Normal Supply) is from Instrument Bus 1.

B. Incorrect. N-43 is part of NIS Cabinet C that receives power from Instrument Bus 3. Instrument Bus 3 is powered from Inverter "B". N-52 receives power from Instrument Bus 8. Instrument Bus 8 receives power from Instrument Bus 3.

C. Correct. N-32, N-36 and N-42 are part of NIS Cabinet B that receives power from Instrument Bus 2. Instrument Bus 2 is powered from Inverter "A".

D. Incorrect. N-31, N-35 and N-41 are part of NIS Cabinet A that receives power from Instrument Bus 1. Instrument Bus 1 is powered from MCC-5. "A" Inverter can confused with NIS Cabinet "A" which may lead to a wrong assumption.

Question 57
Tier 2 / Group 2
K/A Importance Rating - RO 3.3 SRO 3.7

Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

Reference(s) - Sim/Plant design, System Description, EDP-008
Proposed References to be provided to applicants during examination - None
Learning Objective - NIS 005
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7
Comments - K/A met because candidate must know the power supplies to the NIS cabinets as well as the normal power supply to Instrument Bus 2.

LP 33 - Exit LP, transfer from LP 29 to 125Vdc MCC-A.

3.8.1.2 Photo Cell Actuation

LP 22 - Turbine Area Mezzanine Floor
LP 26, circuits 3 & 19 - Outside Control Room and Aux. Bldg. East stairwell
LP 34 - 230KV switchyard
LP 35 - Intake Structure
With each breaker in each LP labeled as to what lights it energizes or de-energizes, the breakers desired to be put in-service can be placed in the ON position and those breakers desired to be out of service placed in the OFF position.

3.9 Power Panels

These smaller electrical panels are located in various areas remote to their source of supply. These panels provide power to equipment located in the vicinity of the panel. An Electrical Distribution Procedure, EDP-007, describes their location, power supply, and the equipment that they provide power to.

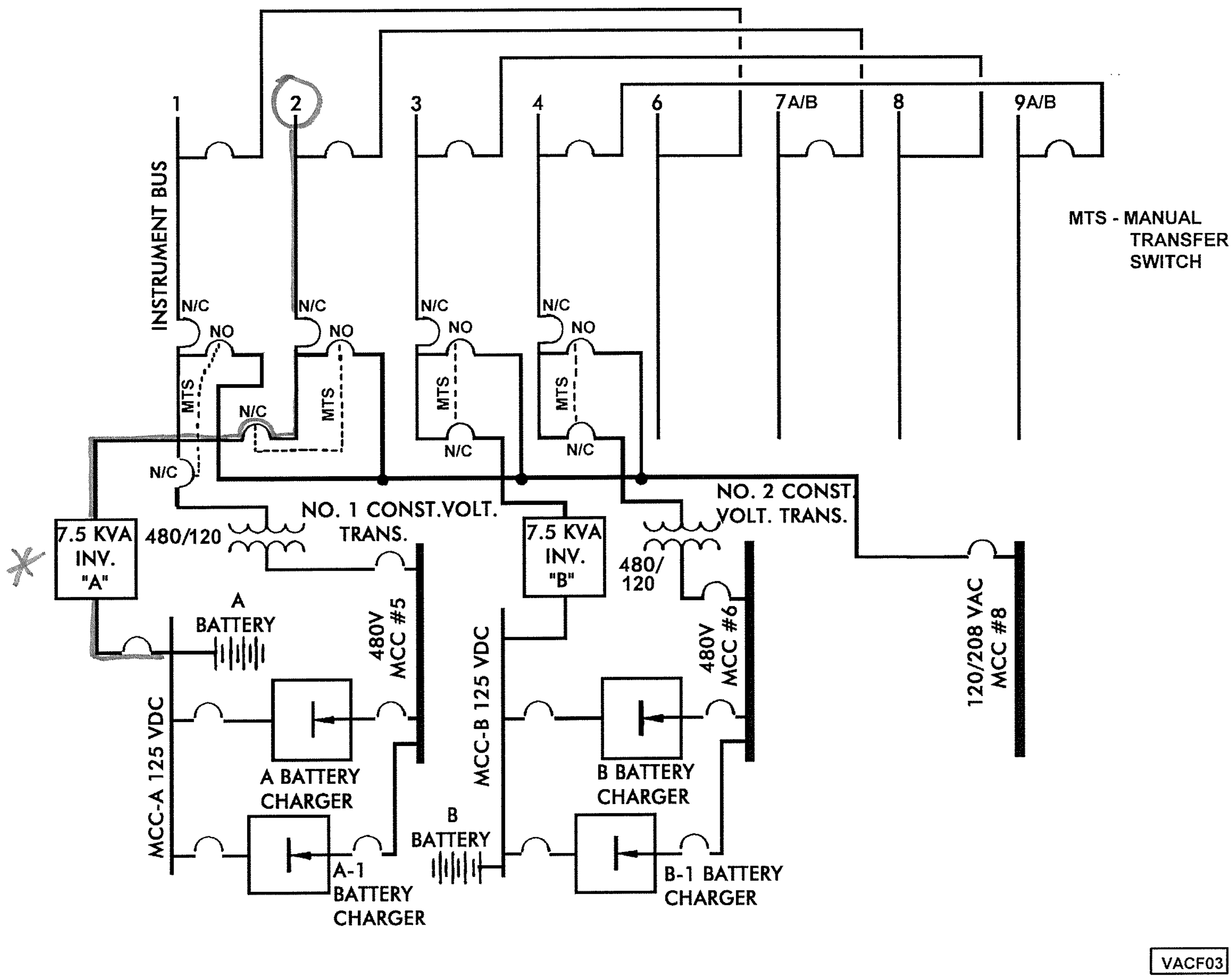
3.10 Instrument Bus Equipment (See Figure 3)

The instrument power is provided from a reliable source to supply vital plant instrumentation during all plant conditions. The loads supplied by each IB can be found in the Electrical Distribution Procedure, EDP-008.

The instrument buses are normally fed from:
IB No. 1 from MCC-5
IB No. 2 from Inverter A
IB No. 3 from Inverter B
IB No. 4 from MCC-6
IB No. 6 from IB No. 1
IB No. 7 from IB No. 2
IB No. 8 from IB No. 3
IB No. 9 from IB No. 4

An alternate power supply from MCC-8 is provided for IBs 1 through 4 but only one IB should be supplied by MCC-8, to maintain train separation. The breakers that supply normal and alternate power are located in a cabinet below the IB Panel they supply. The breakers for normal & alternate power are mechanically interlocked so that both cannot be closed at the same time. These breakers are a break-before-make setup and should

120 VAC INSTRUMENT POWER SUPPLY
VAC-FIGURE-3



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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 12

EDP-008

INSTRUMENT BUSSES

REVISION 23

INSTRUMENT BUS NO. 1
Location: Safeguards Room, East Wall
Normal Power: MCC-5 (4BL) / Alternate Power: MCC-8 (2GL)

	SPARE		Instrument Bus No.1 Power Supply (From INST BUS 1 PWR XFER SW)
1	NIS Cabinet "A" (CWD-442)	2	NIS Cabinet "A" (CWD-442)
3	Hagan Rack 1 (CWD-409)	4	SPARE
5	Hagan Rack 3 (CWD-415)	6	Hagan Rack 5 (CWD-461)
7	Hagan Rack 6 (CWD-415)	8	Hagan Rack 10 (CWD-475)
9	RTGB "A": TB-SM 21,22; "B": TB-TH 54,55; "C":TB-UE 48,49 ; (NOTE 1) ; NR-45 rec, NR-46 rec, TR-410 rec (CWD 462, 963); TR- 412 rec (CWD 412, 963); LR-477 rec (CWD 415, 963); FR-478 rec (CWD 964); FR-154A rec (CWD 477, 963); FR-154B rec (CWD 478, 963)	10	Instrument Bus No. 6
11	Safeguards Rack 51 (CWD-420)	12	SPARE
13	Aux. Panel "ME" 1 and 2 Status Lights (CWD-068)	14	TIC-651A, TIC-651B (CWD-488)
15	Misc. flow, level, and temp. controls FIC-629 (CWD-488)	16	TIC-100, TIC-627, FIC-637, FIC-658, TIC-625(CWD-488)
17	Aux. Panel "CA" TB 29 and 30; Aux. Panel DE (LM-1454A) (CWD 515, 601)	18	**(NOTE: See EDP-005, Aux Panel DD for individual loads from Q-1 & Q-2)** PNL-Q-1-45V, PNL-Q-2-45V (CWD-965) **(NOTE: See EDP-005, 65VDC Dist. Panel "A" for individual loads from Q-3)** PNL-Q-3-65V; PI-1702, PI-1705 (CWD 590, 965); PI-1420, PI-1458 (CWD 602, 966); PI-1421B (CWD 602, 966); LI-1417A (CWD 602, 965); LI-1454A (CWD 601, 965); PI-4004 (CWD 769, 965); PI-1301, PI-1304, PI-1310, PI-1311, PI-1312, PI-1313 (CWD 709, 965); PI-2096A (CWD 726, 966); PI-1616 (CWD 841, 965); PI-1684 (CWD 840, 965); LI-1912 (CWD 594, 965)
19	SPARE	20	Instrument Cabinet "A" (CWD-135)
21	(3 Pole Bkr)	22	Excore Neutron Flux Detector System Channel N-51, NI-51A, NI-51B, NR-53 pens 1&2 (CWD-450B normal power);
23		24	Hagan Rack 2 (CWD-463) Isolator Rack 29, Channel 1
25	SPARE	26	SPARE
27	(3 Pole Bkr)	28	(3 Pole Bkr)
29		30	

NOTE 1: CWD's 412, 415, 432, 441, 462, 477, 478, 595, 963, 964

INSTRUMENT BUS NO. 2
Location: Safeguards Room, East Wall
Normal Power: Inverter "A" / Alternate Power: MCC-8 (2GL)

SPARE		**Instrument Bus No.2 Power Supply (From INST BUS 2 PWR XFER SW)**	
1	NIS Cabinet "B" (CWD-444)	2	NIS Cabinet "B" (CWD-444)
3	Hagan Rack 4 (CWD-476) Isolator Rack 29, Channel 2	4	Hagan Rack 7 (CWD-416)
5	Hagan Rack 9 (CWD-475)	6	Hagan Rack 11 (CWD-460)
7	Hagan Rack 12 (CWD-456); FR-113 rec (CWD 481, 964, 5379-3484);	8	Hagan Rack 13 (CWD-418)
9	RTGB A:TB-SM 27 and 28 RTGB C:TB-UE 45 and 46 (NOTE 1) TR-413 rec (CWD 468, 963, 5379-3502); NR- 47 rec (CWD 443, 963); FR-488 rec (CWD 964)	10	Quenching Valve Control
11	Instrument Bus No.7A and 7B	12	RMS console #1 receptacle (powers generator temp recorder) (CWD 877)
13	FIC-632, (CWD-488)	14	Lundell AC (Note 2)
15	Turning gear automatic control (CWD-731)	16	Waste Disposal System Panel Misc. Relays (CWD 367, 368)
17	MDAFW Pump "A" Flow control valve FCV-1424, FIC-1424 (CWD-657)	18	**(See EDP-005, 65VDC Dist Panel "B for individual loads from PNL Q-6)** PNL-Q-6-45V, (CWD-966) **(See EDP-005, Aux Panel GD for individual loads from PNL Q-4)** PNL-Q-4-65V (CWD-966); PI-1421A (CWD 602, 966); LI-1454B (CWD 601A, 966); LI-1941 (CWD 601, 966); PI-1911(CWD 594, 966)

NOTE 1: CWD's 433, 443, 468, 481, 963, and 964.

NOTE 2: FIC-154, FIC-155, and FIC-156 are powered by Lundell AC power.

INSTRUMENT BUS NO. 3

Location: Safeguards Room, East Wall

Normal Power: Inverter "B" / Alternate Power: MCC-8 (2GL)

SPARE		Instrument Bus No.3 Power Supply (From INST BUS 3 PWR XFER SW)	
1	SPARE	2	Hagan Rack 8 (CWD 417)
3	Hagan Rack 14 Isolator Rack 30, Channel 3	4	Hagan Rack 15 (CWD 457)
5	Hagan Rack 16 (CWD 418)	6	Hagan Rack 17 (CWD 421); LI-970 (CWD 495); LI-969 (CWD 494A)
7	Hagan Rack 18 (CWD 422)	8	RTGB "C" TB-UE 42 and 43 RTGB "A" TB-SM 41 and 42 (CWD's 434, 963 and 964); FR-498 rec (CWD 964)
9	AFW Pump "B" Flow Control Valve FCV-1425, FIC-1425 (CWD 658)	10	NIS Cabinet "C", CWD 446
11	Safeguards Rack 63 (CWD 420)	12	Safeguards Rack Status Lights (CWD 397)
13	FIC-635, (CWD-489)	14	FIC-638, FIC-657 (CWD-489)
15	Exhaust hood spray valves (CWD 761)	16	Turbine Supervisory Recorder, Net Generation Recorder, Turbine MSR Temperature Recorder (CWD 791, 792, 793)
17	SPARE	18	Load, Frequency control panel (CWD 722)
19	NIS Cabinet "C" (2 Pole Bkr) (CWD 446)	20	Safeguards sequencing relays Train "B"
21		22	BLANK
23	Power Panel No. 26 Alternate Supply	24	BLANK
25	Instrument Bus No. 8 (3-pole breaker)	26	SPARE (3 Pole Bkr)
27		28	
29		30	

INSTRUMENT BUS NO. 4

Location: Safeguards Room, South Wall

Normal Power: MCC-6 (2FL) / Alternate Power: MCC-8 (2GL)

SPARE		**Instrument Bus No.4 Power Supply (From INST BUS 4 PWR XFER SW)**	
1	NIS Cabinet "D" (CWD 448)	2	NIS Cabinet "D" (CWD 448)
3	Hagan Rack 19 (CWD 702) Isolator Rack 30 Channel 4	4	Hagan Rack 20 (CWD 486);
5	Hagan Rack 21 (CWD 402)	6	Hagan Rack 22 (CWD 401)
7	Hagan Rack 23 (CWD 476)	8	Hagan Rack 24 (CWD 421)
9	Hagan Rack 25 (CWD 422)	10	Hagan Rack 28 (CWD 471)
11	RTGB "A" TB-SM 55 and 56, (NOTE 1) TR-604 rec (CWD 487, 963, 5379-3506); PR-444 rec (CWD 455B, 963); LR-459 rec (CWD 458, 963); TR-408 rec (CWD 405, 963)	12	Instrument Bus No.9A and 9B
13	SPARE	14	Unit 2 Protection Relay Panel, 115KV Yard Voltmeter (CWD 265)
15	Aux. Panel "BC"-TB 49 and 50 Steam Dump System (CWD 701)	16	**(CAUTION 1)** Misc. Relay Rack 50 Power Terminal Block (HBR2-11336 sheets 9-16)
17	SPARE	18	**(CAUTION 2)** Misc. Relay Rack 50 Power Terminal Block (HBR2-11336 sheets 1-8)

CAUTION 1: Opening Ckt Bkr #16 will impact multiple relays in the rear of Rack 50 – evaluate drawings HBR2-11336 sheets 9-16 prior to opening Breaker #16.

CAUTION 2: Opening Ckt Bkr #18 will impact multiple relays in the front of Rack 50 – evaluate drawings HBR2-11336 sheets 1-8 prior to opening Breaker #18.

NOTE 1: CWD's 405, 447, 455B, 458, 487, and 963.

INSTRUMENT BUS NO. 8 Location: Safeguards Room, East Wall Power Supply: Instrument Bus No. 3, Ckt 25 (fused panel)			
CKT	FUSE SIZE	FUSE TYPE	LOAD
10			FT-110E/I Boric Acid Bypass Flow, FI-110 (CWD 474)
11			Pressurizer Spray Valve PCV-455B position lights (CWD 470)
12			TI-580, LI-802, PI-957, PI-8111-2 (CWD 533)
13	N/A	N/A	SPARE
14			FQ-958B CV Spray Flow, FI-958B (CWD 494B)
15	N/A	N/A	SPARE
16			V2 Safeguard relay (Rack 63) (CWD 397)
17	N/A	N/A	SPARE
18			Channel II CET/CCM Signal Processor Cabinet TM-578 (CWD 1700); TI-433 & pen 3 on TR-413 (5379-3502)
19	N/A	N/A	SPARE
20			FI-1425A, FI-1426B (AFW) (CWD 623A, 623B)
21	N/A	N/A	SPARE
22			Excore Neutron Flux Detector System Channel N-52, NI-52A, NI-52B, NR-53 pens 3&4(CWD450C&D)
23	N/A	N/A	SPARE
24	N/A	N/A	SPARE
25			Boric acid heat trace Local Annunciator No 3 (CWD180C);
26			Boric acid heat trace recorder No. 3 (CWD 180B)
27			CV Avg Temp Channel TT-950B, TQ-950B, TI-950B (CWD 044)
28			Aux. panel "GB" TB 5 and 6 (turbine auto stop trip) (CWD 711)
29			Reactor Protection Rack 55 (CWD 438)
30			Reactor Protection Rack 60 (CWD 438)
31			Turbine oil temperature TT-2097, TI-2097A (CWD 726)
32			PQ 4005 Turbine main oil pump discharge (CWD 726)
33			Hydrogen control cabinet electronics , PI-1900 Gen H2 press, AI-1900 Gen H2 Purity (CWD-876)
34	N/A	N/A	SPARE
35	N/A	N/A	SPARE
36	N/A	N/A	SPARE

- REACTOR TRIP BLOCK (<P-8)

  Provided by PR channels with a setpoint of <40% and a coincidence of 3/4 < setpoint. The permissive light on the RTGB will light when < setpoint. The permissive's function is to prevent a Rx trip from a loss of flow or RCP breaker open in a single loop or Turbine trip. The trip auto reinstates above setpoint when 2/4 PR channels are >40%.

- POWER ABOVE P-10

  Provided by PR channels with a setpoint of 10% and a coincidence of 2/4 > setpoint. Allows manual blocking of IR rod stop (20% setpoint), high flux reactor trip (low setpoint @ 24%), and IR reactor trip (25% current equivalent). Also will auto deenergize SR by removing detector high voltage.

  When P-10 is actuated it also provides an input into the permissive REACTOR TRIP BLOCK P-7. This will re-instate the "at power" trips, which had been blocked. These trips are: high pressurizer level, low reactor coolant flow, and low pressurizer pressure.

4.6.2 Refer to Attachment 10.1 for a complete listing of the NIS Instrument Setpoints.

4.6.3 Refer to Attachment 10.2 for a complete listing of the NIS Monitor Lights.

4.6.4 Refer to Attachment 10.3 for a complete listing of the NIS Status Lights

4.7 Power Supplies

NIS Cabinets A, B, C, and D receive power from the Instrument Buses 1, 2, 3, and 4 respectively.

R.G. 1.97 Excore Neutron Flux Monitoring System Channel N-51 receives power through Kirk-Key lock transfer switches powered from Instrument Bus No. 1 and DSS 120Vac panel.

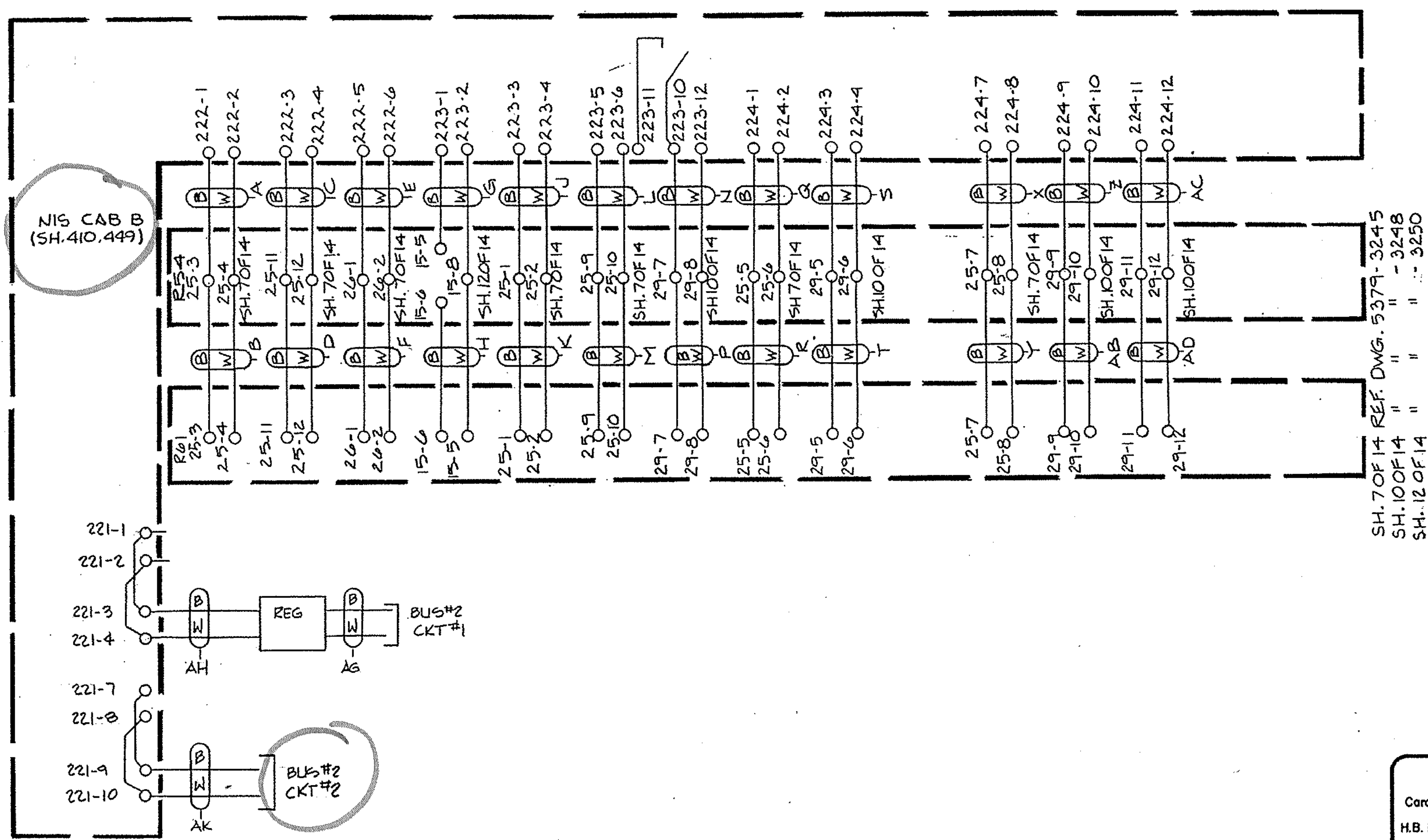
R.G. 1.97 Excore Neutron Flux Monitoring System Channel N-52 receives power from Instrument Bus No. 8.

4.8 Monitor Lights

4.8.1 SR

- SOURCE RANGE TRIP BLOCKED
- NIS TRIP BYPASS NI 31 (in bypass)
- NIS TRIP BYPASS NI 32 (in bypass)

444



REF DWG. 5379-3638

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
				6	3/18/81	[illegible]	RSM [illegible]
				5	[illegible]	[illegible]	HEL JUC
				4	[illegible]	TW	CHC JS
				3	[illegible]	[illegible]	FLD JS
				2	[illegible]	FLD	JS
7	[illegible]	JW	[illegible]	1	[illegible]	[illegible]	GHM JSKB

NOTES	NOTES

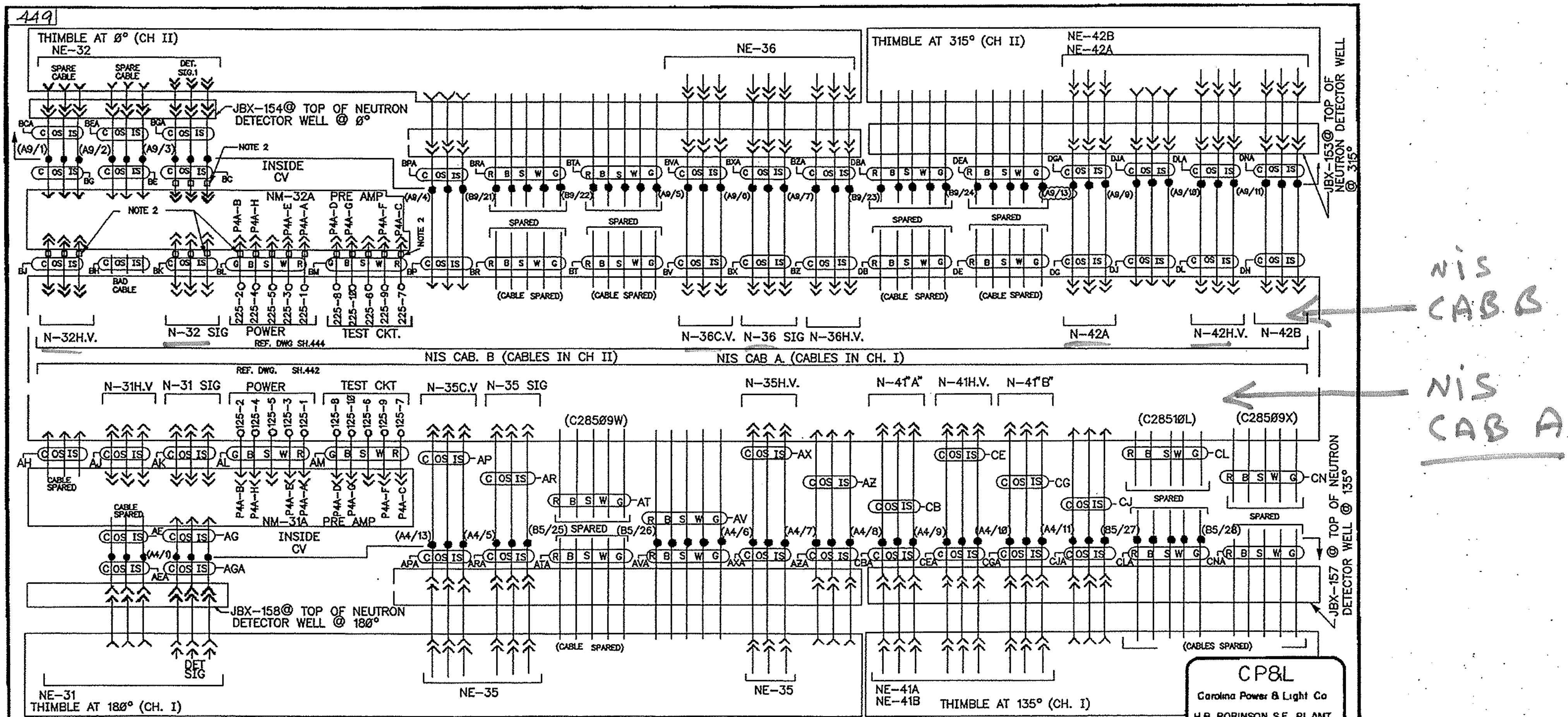
NUCLEAR INSTRUMENTATION SYSTEM SH4

VENDOR DRAWING NO: 500B452

CP&L
Carolina Power & Light Co.
H.B. ROBINSON S.E. PLANT
UNIT- NO. 2
Hartsville, South Carolina

Dsgn. (W)	Scale: NONE
Drn EDS/MH	Date: 3-16-82
Ck. RSM	
Appr. [illegible]	

DRAWING TITLE CONTROL WIRING DIAGRAM
DRAWING NO: B-190628
SHEET NO: 444 Rev. 7



NOTES: 1. ALL CABLES FROM JUNCTION BOXES TO THIMBLES SUPPLIED TO FIT MFR.

—●— RAYCHEM SPLICE

2. ½" FERRITE BEADS —□— - REF. EE 94-073

REF. DWG. 5379-4862(5651D45), AND B-190628 SH.442, 444

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED	NO	NOTES	NO	NOTES
12	9/1/98	CO	CLO	6	3/4/90	JW	CLO	9	DCR: 94-1186	11	DCR 98-463
11	11/7/98	JW	CLO	5	2/4/90	JW	CAL	8	DCR: 94-410	12	DCR 98-780, ESR 9800506
10	11/7/97	DE	CLO	4	5/29/90	JW	CAL	7	DCR: 90-2069; MOD: 1016	13	EC 54800
9	[illegible]	JS	CLO	3	[illegible]	[illegible]	RSM	6	DCR: 90-200		
8	[illegible]	JB	CLO	2	[illegible]	AB	GHM	5	DCR: 89-249		
7	[illegible]	[illegible]	CLO	13	4/5/03	EP	[illegible]	4	DCR: 88-616		

NUCLEAR INSTRUMENTATION SYSTEM DETECTOR CONNECTIONS CH. I & II

VENDOR DRAWING NO 5008452

Dsgn | Scale NONE
Drn ED/MW | Date 3/1/[illegible]
Ck RSM
Appv

CP&L
Carolina Power & Light Co
H B ROBINSON S E PLANT
UNIT- NO 2
Hartsville, South Carolina

DRAWING TITLE CONTROL WIRING DIAGRAM
DRAWING NO B-190628
SHEET NO 449 | Rev 13

58. 016 K1.12 001

The plant is at 50% RTP when the steam pressure input into the "A" S/G Level Control System fails high slowly.

Which ONE (1) of the following actions will occur?

Initially, feedwater flow to the "A" S/G will ....

A. lower due to lower indicated steam flow.
The level error signal will NOT result in a rise in feedwater flow.

B. lower due to lower indicated steam flow.
Eventually, the level error signal will result in a rise in feedwater flow.

C. rise due to higher indicated steam flow.
The level error signal will NOT result in a reduction of feedwater flow.

D.✓ rise due to higher indicated steam flow.
Eventually, the level error signal will result in a reduction of feedwater flow.

The correct answer is D.

A. and B. Incorrect. The rise in steam pressure input to steam flow causes a rise in indicated steam flow. Steam flow is density compensated. A rise in pressure is proportional to an increase in density. The steam flow instruments will indicate a higher steam flow due to the false higher density input.

C. Incorrect. The feedwater flow will rise due to higher indicated steam flow. Since the feedflow rise is greater than the actual steam flow the steam generator level will rise. The level rise will create a level error signal that will reduce the total error signal causing the FRV to close.

D. Correct.

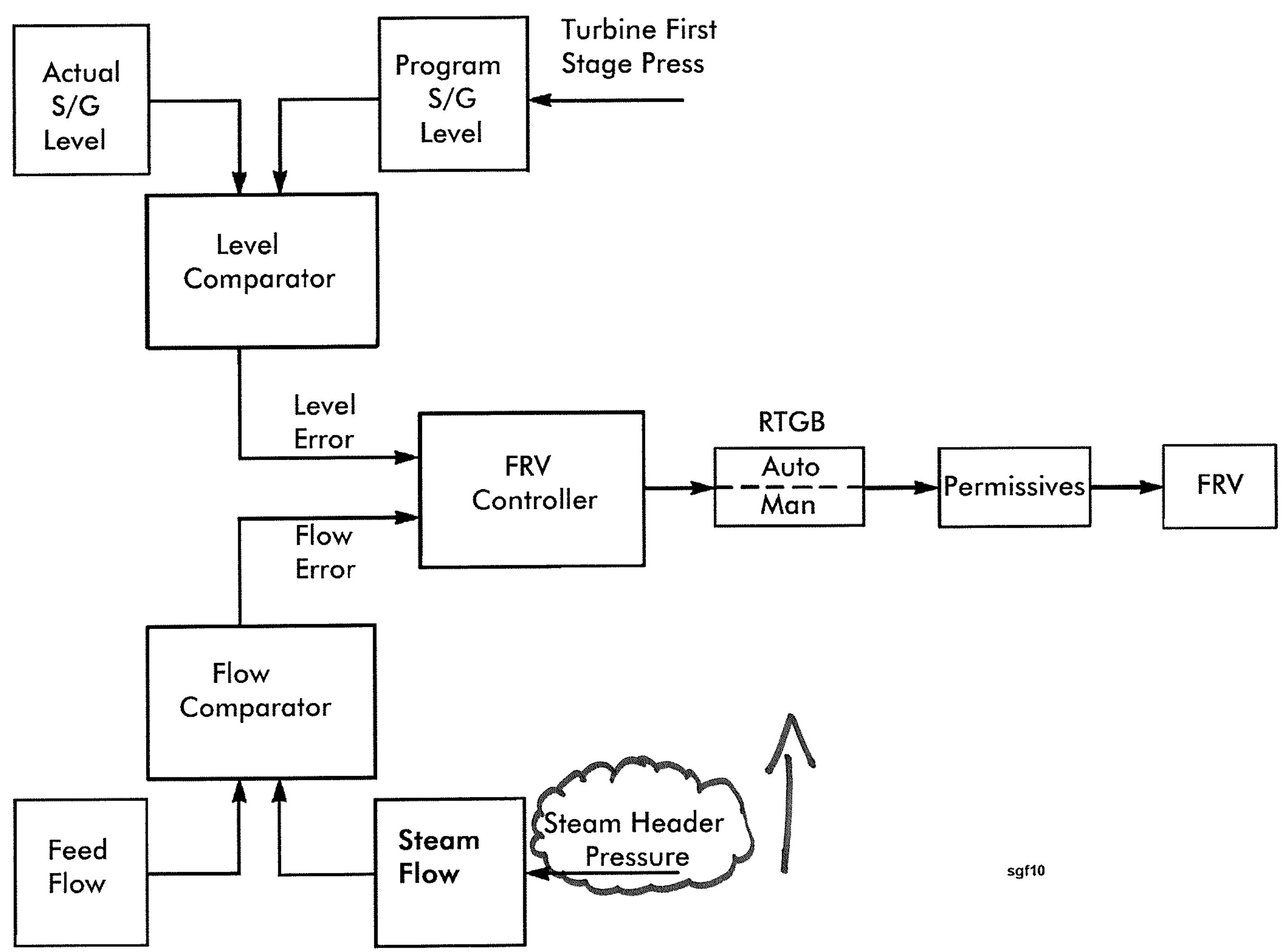
Question 58
Tier 2 / Group 2
K/A Importance Rating - RO 3.5 SRO 3.5

Knowledge of the physical connections and/or cause / effect relationships between the NNIS and the following systems: S/G

Reference(s) - Sim/Plant design, System Description,
Proposed References to be provided to applicants during examination - None
Learning Objective - SG 010
Question Source - Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.2 to 41.9 / 45.7 to 45.8
Comments - K/A is met because candidate must know the effect of the failure of the steam pressure input to steam flow and its relationship with the S/G Level Control System.

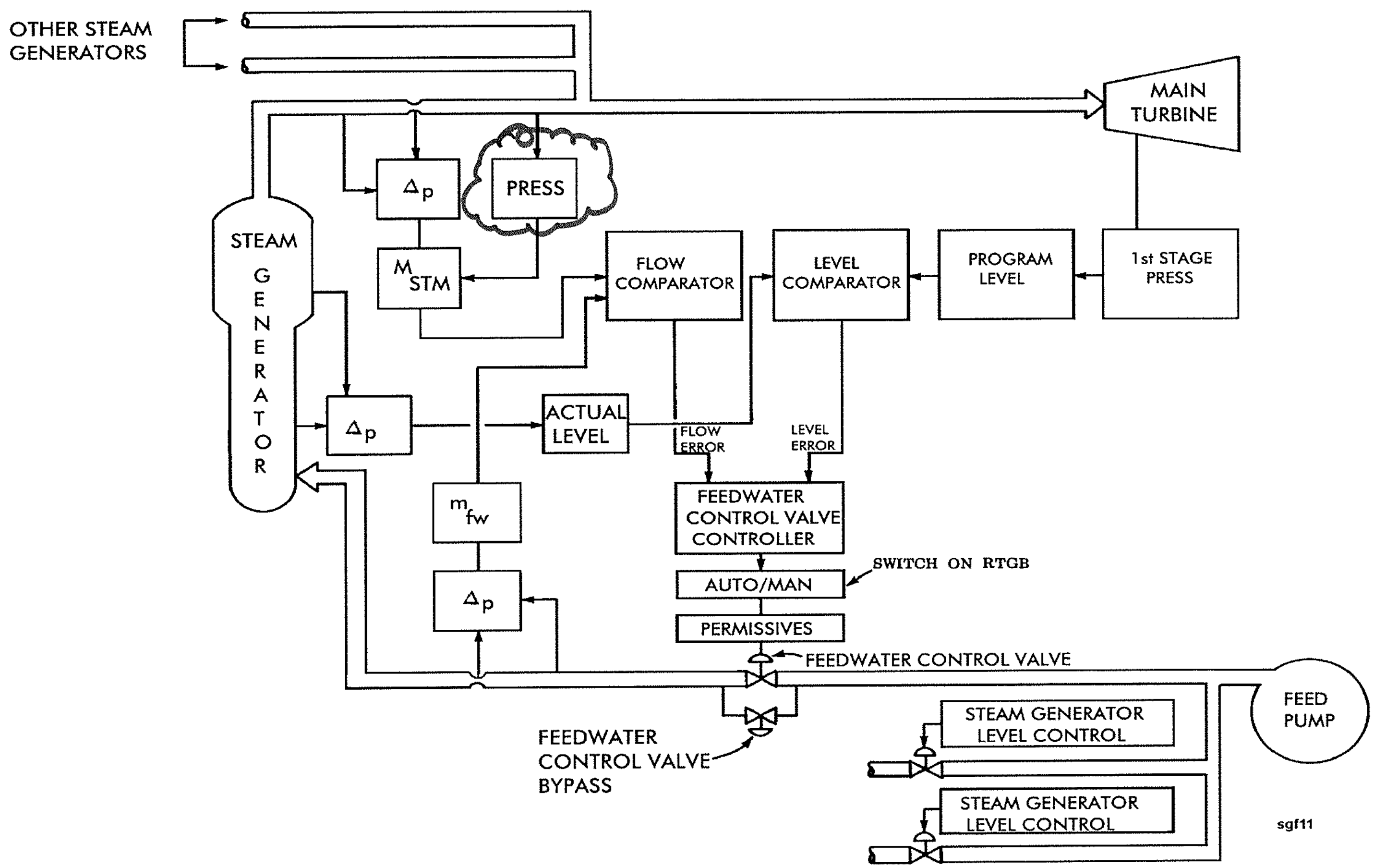
BASIC STEAM GENERATOR LEVEL CONTROL SYSTEM

SG-FIGURE-10



STEAM GENERATOR WATER LEVEL CONTROL SYSTEM

SG-FIGURE-11



5.0 CONTROLS AND PROTECTION

5.1 Control Systems

5.1.1 Main Feedwater Control

The steam generator level contrl ensures the proper water inventory for various operational and accident conditions. The control is achieved by variations in the feedwater flowrate.

In normal power operation, the position of the feedwater control valve is controlled by the three-element controller (feedwater flow, steam flow, and level deviation), with the programmed level setpoint input from selected turbine first stage pressure. Both the steam flow and feed flow signals used for control can be supplied by either of two transmitters as selected by a control board mounted selector switch.

The setpoint of the level controller is a function of load (selectable turbine first stage pressure from PT-446 or PT-447). The level setpoint is programmed to linearly rise from 39 to 52% as load increases from 0% to 20% turbine load and then remains constant as load continues to increase. The output of the level controller is then fed into the feedwater control valve controller. The level input is only from LT-476, LT-486, and LT-496 associated with steam generators A, B, and C, respectively.

Failure of a steam flow or feedwater flow transmitter causes a flow mismatch alarm to annunciate. The steam flow, feed flow, and level signals used for control are recorded on a three pen recorder.

To prevent excessive moisture carryover caused by high steam generator water level, a high water level overrides all other control signals and closes the feedwater control valve of the affected steam generator. This signal also trips both main feedwater pumps, closes the feedwater control bypass valve, and trips the turbine. The signal is derived from coincidence of two out of three level channels. This feedwater isolation must be reset on the RTGB after the water level drops below the trip point.

The feedwater control valve signals are listed below in the order of increasing priority:

1. Three-element control system
2. Manual control (RTGB)

3a. Low T_{AVG} and reactor trip (closes main feed regulating valves only)

3b. High steam generator water level (closes main feed regulating valve and bypass valve of affected steam generator, trips feedwater pumps, trips turbine)

3c. Safety Injection (closes main feed regulating valves, bypass valves, feedwater block valves, trips feedwater pumps, trips turbine)

The auto/manual selector switch on the side of the Bailey positioner at the feedwater regulating valves cannot be used for local pneumatic valve control. If manual is selected, the valve will fail closed.

A wide-range level channel, calibrated for no-load conditions, aids manual level control from hot shutdown to cold shutdown. This channel consists of a recorder, high and low level alarms (only function when steam generator pressure is less than 614 psig), and indicators. Automatic pressure-temperature compensation is not necessary.

Besides the main feedwater regulating valve (FCV-478, -488, or -498), for each steam generator there is a bypass valve (FCV-479, -489, or -499) which is intended to provide manual feedwater flowrate control at low loads. During normal "at power" operation of the plant the bypass valve is closed.

The bypass valve operation is controlled from the RTGB controller station which provides manual positioning of the valve. The opening of the bypass valve is prevented in the presence of either safety injection or high steam generator water level.

The auto/manual selector switch on the side of the Bailey positioner at the feedwater regulating bypass valves can be used for local pneumatic control.

The main feedwater regulating valves and bypasses are operated by utilizing the instrument air system pressure through controllers to properly position the valves.

The main feedwater regulating valves and the bypass regulating valves have OPEN and CLOSED light indications. These position indication lights are located on the RTGB adjacent to their respective controllers.

The main feedwater regulating valves rely on motor operated block valves if needed to isolate them from the main feedwater pump discharge. These valves are controlled from the RTGB and supplied power from: FW-V2-6A for "A" steam generator from MCC-5, FW-V2-6B for "B" steam generator from MCC-6, and FW-V2-6C for "C" steam generator from MCC-6.

5.1.2 Steam Generator Water Level Control

Steam generator level is programmed for operation. The normal no load level is 39% and is programmed from 39% to 52% from 0% to 20% power. From 20% to 100% power the level remains at 52%.

The three-element feedwater control system compares actual steam generator level to the program level derived from turbine first stage pressure (for power level, selected from either PT-446 or PT-447) and any difference between the signals is the level error.

The pressure compensated steam flow signal and the feedwater flow signal are compared

and produce a flow error signal. The two error signals are then used to control the position of the main feedwater regulating valves.

For normal operation, the dominate signal for controlling the valves is the level error. For large transients the flow error will be most significant.

The feedwater control system can only control the valves when in automatic. If the valves are placed in manual, then the manual pushbuttons are used in controlling valve position. These controllers are located on the RTGB. Also located on the RTGB are the controls for bypass valves. These are operated manually from potentiometers that indicate from 0% to 100%.

The no load steam generator level (39%) is to maintain the mass in the steam generator great enough to stay above the reactor trip setpoints and minimizes the consequences of a steam break accident.

The normal at power (greater than 20%) steam generator level (52%) is provided to maintain sufficient level to prevent a reactor trip from a 50% load rejection and is low enough to minimize moisture carryover.

59. 028 K5.03 001

Which ONE (1) of the following identifies the *major* source of hydrogen following a design basis LOCA with Inadequate Core Cooling AND is listed in the UFSAR criteria for ECCS performance following a LOCA?

A.✓ Zirc-water reaction.

B. Radiolysis of RCS water.

C. Radiolysis of containment sump water.

D. Degradation of non-metallic insulation material.

The correct answer is A.

A. Correct. Sources of hydrogen during a LOCA:
- Zirc-water reaction
- Metal-water reaction
- Radiolysis of RCS water
- Radiolysis of Containment sump water

B. Incorrect. Radiolysis of RCS water causes hydrogen but is not the major source described in the UFSAR criteria for ECCS performance following a LOCA.

C. Incorrect. Radiolysis of containment sump water is a contributor to hydrogen in containment but is not the major source as discussed in the UFSAR criteria for ECCS performance following a LOCA.

D. Incorrect. Degradation of Insulation Material during a design basis LOCA is a known concern for sump screen clogging issues. Insulation degradation is not a major source of hydrogen.

Question 59
Tier 2 / Group 2
K/A Importance Rating - RO 2.9 SRO 3.6

Knowledge of the operational implications of the following concepts as they apply to the HRPS: Sources of hydrogen within containment

Reference(s) - Sim/Plant design, System Description, UFSAR (ECCS Acceptance Criteria)
Proposed References to be provided to applicants during examination - None
Learning Objective - CVHVAC 004
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.5 / 45.7
Comments - K/A is met because the candidate must know the major source of hydrogen following a design basis LOCA.

15.6.2 SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

15.6.2.1 Identification of Causes and Frequency Classification and Acceptance Criteria

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the Reactor Coolant System (RCS) pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 sq. ft. This event is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 sq. ft. in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

The acceptance criteria for the loss-of-coolant accident is described in 10CFR50.46 as follows:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F.

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of zircaloy in the reactor.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

Description of Small Break LOCA Transient

Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

6.2.5.1 Design Basis

Following a design basis accident (DBA), hydrogen gas may be generated inside the containment by reactions such as radiolysis of aqueous solutions in the sump and core, zirconium metal with water, and corrosion of materials of construction.

Prior to October 2003, 10 CFR 50.44 required controls to ensure the containment hydrogen concentration remained below combustible concentrations. Such controls included the use of a hydrogen recombiner or hydrogen purge. An October 2003 revision to 10 CFR 50.44 eliminated the need for such controls for recovery from design basis accidents for containments similar to the HBR 2 containment. Therefore, HBR 2 no longer maintains access to a hydrogen recombiner and does not require hydrogen purging to recover from a design basis accident. The Post-Accident Containment Venting System is available for reduction of the containment hydrogen concentration if desired. The Post-Accident Containment Venting System is described in Section 9.4.3.

- Provide air for purging and vacuum relief.
- Provide pressure and vacuum relief for the containment vessel.
- Provide adequate removal of potentially contaminated air from the refueling water surface during refueling operations.
- Remove the normal heat loss from all equipment and piping during all phases of plant operation.
- Remove heat generated by the control rod drive mechanisms.
- Provide cooling to the reactor vessel insulation surface and for the primary concrete shield.
- Provide depressurization by removing heat from the containment following an accident.
- Provide adequate air filtration for safe access after reactor shutdown.
- Reduce the concentration of airborne radionuclides and non-radioactive particulate matter for purging and personnel access.
- Provide adequate air circulation to prevent the potential stratification of hot air, hydrogen, and other gases of low molecular weight, particularly in the dome.

2.1.2 Post Accident Containment Venting System

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as radiolysis of aqueous solutions in the sump and core, zirconium metal with water, and corrosion of materials of construction. The Post Accident Containment Venting system permits controlled venting of the containment atmosphere to maintain the hydrogen concentration at a safe level.

The function of the Post Accident Containment Venting, Makeup, and Exhaust system under post-accident conditions, after containment pressure and the concentration of airborne radioactive materials have been reduced to near normal conditions, is to provide controlled venting of the containment atmosphere to prevent the accumulation of hydrogen gas from ever reaching a potentially hazardous concentration.

2.2 Design Basis

2.2.1 CVHVAC Systems

System Design Parameters
The CVHVAC systems are based on the following design parameters.

Outdoor Air Condition
Winter: 10°F db (except cond. pol. which is 22°F db)
Summer: 95°F db, 78°F wb (except cond. pol. which is 94°F db)

Indoor Air Condition to be Maintained at the Following Average Temp.

During Normal Power Operation
Containment Vessel - General: 120°F db, max., +50°F min.(See NOTE*)

60. 034 K6.02 001

Given the following plant conditions.

- Plant is currently in Mode 6 during a scheduled refueling outage.
- Core re-load is in progress.
- The alarming radiation monitor on the refueling bridge loses power.

Which ONE (1) of the following identifies the impact to refueling operations and the proper actions of the refueling crew?

A. Fuel movement may continue as long as R-2, CV AREA, remains operable.

Replace the radiation monitor with an operable monitor with an alarm setpoint of 15 mr/hr.

B. Fuel movement may continue as long as R-2, CV AREA, remains operable.

Replace the radiation monitor with an operable monitor with an alarm setpoint of 100 mr/hr.

C.✓ Fuel movement must be suspended until the radiation monitor is replaced.

Replace the radiation monitor with an operable monitor with an alarm setpoint of 15 mr/hr.

D. Fuel movement must be suspended until the radiation monitor is replaced.

Replace the radiation monitor with an operable monitor with an alarm setpoint of 100 mr/hr.

The correct answer is C.

A. Incorrect. The Precautions and Limitations of GP-010, Refueling, states that an alarming radiation monitor must be installed on the refueling bridges with an alarm set point of 15 mrem/hr during fuel movement operations. R-2, CV AREA, is required to be operable at all times during fuel movement per Att. 10.7 of GP-010.

B. Incorrect. The Precautions and Limitations of GP-010, Refueling, states that an alarming radiation monitor must be installed on the refueling bridges with an alarm set point of 15 mrem/hr during fuel movement operations. R-2, CV AREA, is required to be operable at all times during fuel movement per Att. 10.7 of GP-010.

C. Correct.

D. Incorrect. The Precautions and Limitations of GP-010, Refueling, states that an alarming radiation monitor must be installed on the refueling bridges with an alarm set point of 15 mrem/hr during fuel movement operations. R-2, CV AREA, is required to be operable at all times during fuel movement per Att. 10.7 of GP-010.

Question 60
Tier 2 / Group 2
K/A Importance Rating - RO 2.6 SRO 3.3

Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System : Radiation monitoring systems

Reference(s) - Sim/Plant design, System Description, GP-010
Proposed References to be provided to applicants during examination - None
Learning Objective - FHS 010
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 45.5
Comments - K/A is met because candidate must know the effect a malfunction of the radiation monitor on the Refueling Bridge will have on refueling operations.

5.24 When the SFP GATE VALVE is open, it takes 1500 gallons of makeup to raise level 1 inch in the Spent Fuel Pool and Refueling Cavity.

5.25 The Fuel Transfer Cart shall be on the CV side at the end stop position before the SFP GATE VALVE is closed. [CAPR NCR 00239329/NCR 00231270]

5.26 ITS 3.9.2 requires two Source Range Neutron Flux Monitors to be operable in MODE 6. If one of the BF3 Source Range Detectors becomes inoperable, a PAM Source Range Detector may be used if the requirements of ITS LCO 3.9.2A are met.

5.27 RCP "B" and RCP "C" should **NOT** be uncoupled at the same time if the RHR System is required to be operable **AND** RCS level is at or below -50 inches. Uncoupled RCPs result in inaccurate standpipe level indication when RCS level is at or below -50 inches. (ESR 95-00649) (NCR 30599)

5.28 During fuel movement operations, ensure an alarming radiation monitor is installed on the refueling bridge(s) with an alarm set point of 15 mrem/hr. If an alarm is received notify the refueling SRO. If a sustained dose rate of 15 mrem/hr is found on the refueling bridge fuel movement operations shall be suspended until RC supervision provides approval to resume fuel movement.

5.29 Detensioning may begin with the RCS full **AND** vented however, detensioning will stop with 6 studs still tensioned (Group 25, 26) until RCS is below the vessel flange (at least four inches). (EC 74794)

Progress Energy	C Continuous Use

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 3

GP-010

REFUELING

REVISION 71

SHIFTLY CHECKS

NOTE:	These checks are completed prior to unlatching CRDMs **AND** shiftly until the Reactor Vessel head is installed except as defined below. This attachment is **NOT** required to be performed **OR** remain in effect, while the Reactor is defueled.

This revision has been verified to be the latest revision available.

Name (Print)	Initial	Signature	Date

NOTE:	Steps and substeps in this attachment may be performed in any order.

CV Checks — 07-19 19-07

1. **VERIFY** Containment Closure is being tracked by OMM-033. ____ ____

NOTE:	During periods of core alteration, one channel shall have an audible count rate indication available in the containment. ITS 3.9.2 Bases (NCR 299128)

2. **VERIFY** the following:
   - Either Source Range channel N-31 **OR** N-32 providing audible indication inside CV ____ ____
   - CV evacuation alarm tested **AND** audible in all areas ____ ____
   - R-2, CV AREA, is OPERABLE ____ ____
   - R-9, LETDOWN LINE AREA, is OPERABLE ____ ____
3. **IF** a CV Purge is in progress, **THEN VERIFY** the Refueling Surface Ducts set-up for REFUEL on the PURGE OR REFUEL DAMPERS selector switch. ____ ____
4. **IF** moving recently irradiated fuel, **THEN PERFORM** Attachment 10.8. ____ ____
5. **VERIFY** the RCDT Pumps in AUTOMATIC. ____ ____

SHIFTLY CHECKS

CV Checks (Continued)	07-19	19-07
6. **VERIFY** one of the following conditions for maintaining CV Sump level: (N/A others)		
a. At least one CV Sump Pump breaker CLOSED:		
– CV SUMP PUMP "A" on MCC 2 in CMPT 3M	____	____
– CV SUMP PUMP "B" on MCC 1 in CMPT 5H	____	____
OR		
b. CV Sump level is being controlled manually per the requirements of OP-701	____	____
7. **VERIFY** CV Sump Pump Discharge valves OPEN **AND** in AUTOMATIC:		
– WD-1728, CONTAINMENT SUMP PUMP DISCHARGE AUTO ISOLATION	____	____
– WD-1723, CONTAINMENT SUMP PUMP DISCHARGE AUTO ISOLATION	____	____
8. **IF** changing Core Geometry, **THEN VERIFY** voice communications are established between the Control Room and the CV Manipulator Crane. (TRMS 3.13)	____	____
9. A licensed Reactor Operator is in the Control Room to monitor instrumentation.	____	____

NOTE:	A Refueling SRO should be in the CV when lifting/setting the Reactor Vessel Head **OR** Reactor Vessel Internals, but is **NOT** required by 10 CFR 50.54.

10. A Refueling SRO is in the CV to supervise fuel movement. (10 CFR 50.54(m)(2)(iv))	____	____

SHIFTLY CHECKS

NOTE:	SFP Checks can be marked N/A when no Refueling evolutions are occurring in the SFP.

	SFP Checks	07-19	19-07
1.	**PERFORM** Attachment 10.9.	____	____
2.	SFP Cooling System operable.	____	____
3.	SFP Foreign Material Exclusion Area (SFP-FMEA) Wall is intact.	____	____
4.	**VERIFY** the following Radiation Monitors are OPERABLE:		
	1) Process Monitors		
	a. R-20, FUEL HANDLING BLDG LOWER LEVEL LOW RANGE (Required only for fuel movement **OR** storage on lower level of Fuel Handling Building)	____	____
	b. R-14C, NG-LO PLANT EFFLUENT	____	____
	2) Area Monitor		
	a. R-5, SPENT FUEL AREA	____	____
5.	**VERIFY** Spent Fuel Building Air Filtration System is OPERABLE with the exhaust aligned to discharge through the HEPA **AND** charcoal filters IAW OP-906.	____	____
6.	**VERIFY** a negative pressure is being maintained in the SFP Area.	____	____
7.	**IF** fuel is being transferred between the CV **AND** the SFP, **THEN VERIFY** voice communications are established between the Control Room, Spent Fuel Building and the CV.	____	____
8.	A Qualified Fuel Handler is in the SFP.	____	____

SHIFTLY CHECKS

	Initials	Name (Print)	Date
Performed By:	______	______________________	__________
	______	______________________	__________
	______	______________________	__________
	______	______________________	__________
	______	______________________	__________
	______	______________________	__________
	______	______________________	__________

Reviewed By: 07-19 ______________________ __________
STA or CRSS Date

19-07 ______________________ __________
STA or CRSS Date

Approved By: ______________________ __________
Shift Manager Date

INIT

8.3 Fuel Assembly and Core Component Movement

NOTE:	A case evaluation has been performed IAW PLP-037 for each major evolution in this section. The case determination is defined prior to each applicable evolution.

8.3.1 **CHECK** that the Reactor has been subcritical for at least 100 hours **AND** record hours subcritical.
(TRMS 3.12) ______hrs ____

8.3.2 **VERIFY** EST-001 is complete. ____ ENG

NOTE:	The following equipment tests need to be completed prior to moving fuel.

8.3.3 **VERIFY** the equipment necessary to support refueling operations has been tested IAW the applicable EST-30 series procedure. (Equipment that is **NOT** needed to support refueling operations may be marked N/A.)

– Fuel Transfer System ____ ENG

– Manipulator Crane ____ ENG

– CV RCC Change Fixture ____ ENG

– New Fuel Handling Equipment ____ ENG

– NFB New Fuel Lift ____ ENG

8.3.3 (Continued) INIT

- SFP New Fuel Monorail ____ ENG

- SFP Bridge Crane ____ ENG

- SFP New Fuel Elevator ____ ENG

- RCC Change Tool ____ ENG

- SFP GATE VALVE ____ ENG

- Polar Crane-Manipulator Crane Interference Interlock ____ ENG

- Fuel Handling Tools ____ ENG

8.3.4 **IF** fuel movement will occur in the SFP, **THEN VERIFY** the following:

1. The SFP Boron Concentration is greater than or equal to 2050 ppm. (ITS LCO 3.7.13 limit is 1500 ppm)
   _______ ppm ____

2. The SFP Cooling System is in operation IAW OP-910 **OR** the SFP is being cooled IAW another approved procedure. ____

3. An alarming Radiation Monitor is installed on the SFP Refueling Bridge with an alarm set point of 15 mrem/hr. ____ RC

8.3.4 (Continued) INIT

4. **IF** irradiated fuel movement will occur in the SFP, **THEN** **VERIFY** the following:

   a. Control Room Radiation monitor, R-1, CONTROL ROOM, is operable. (ITS LCO 3.3.7) ____

   b. The Control Room Emergency Filtration System is operable as required by ITS LCO 3.7.9 For Movement of Irradiated Fuel Assemblies. (ITS LCO 3.7.9) ____

   c. The Water Cooled Condensing Units are operable as required by ITS LCO 3.7.10 for Movement of Irradiated Fuel Assemblies (ITS LCO 3.7.10) ____

NOTE:	SR 3.7.11.1 requires the FBACS to operate for > 10 hours with heaters operating in Automatic every 31 days. If this SR has **NOT** been met in the previous 31 days, ITS requires that it be performed prior to movement of irradiated fuel in the SFP.

   d. The FBACS (HVE-15A) is operable **AND** operating. (ITS LCO 3.7.11) ____

   e. The monthly operation of the fuel building air cleanup system section of FHP-003 is complete. ____

NOTE:	Maintaining SFP water level above 35 feet, 1/4 inch satisfies the ITS required level of 21 feet above the top of the irradiated fuel assemblies seated in the storage racks.

   f. The SFP water level is greater than or equal to 35 feet, 1/4 inch. (ITS LCO 3.7.12) ____

NOTE:	Due to Spent Fuel being stored in the SFP, it may be necessary to retain some New Fuel in the New Fuel Storage Racks prior to loading the Core. The appropriate Fuel Assembly and Core Component Movement procedure will describe storage locations.

8.3.5 **IF** new fuel is to be placed in the Reactor, **THEN VERIFY** the new fuel assemblies have been removed from their shipping containers and stored in the SFP **OR** New Fuel Storage Racks IAW FHP-002. ____

8.3.6 **IF** temporary lighting is needed in the area of fuel handling operations, **THEN VERIFY** temporary lights are installed. ____

8.3.7 **VERIFY** an alarming Radiation Monitor is installed on the Manipulator Bridge with an alarm set point of 15 mrem/hr. ____ RC

8.3.8 **VERIFY** a copy of FMP-019 **AND** the Fuel Shuffle Sheets are available at the following locations:

- Control Room ____
- SFP ____
- CV ____

NOTE:	Fuel movement has been determined to involve PLP-037 Case II activities.

8.3.9 A Management Designated Monitor (MDM) shall be assigned **AND** shall give permission to perform this evolution as documented by having performed the Pre-Job briefing.

- ______________________________
  Management Designated Monitor signature

8.3.10 **VERIFY** Fuel Transfer Center Line has been tested IAW EST-030. ____ ENG

8.3.11 **MOVE** the first Fuel Assembly IAW FMP-019. ____ ENG

61. 041 K4.09 001

Which ONE (1) of the following actions is necessary to utilize the steam dumps to cooldown the plant IAW GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown?

A. Place the Steam Dump Mode Switch to RESET with Tave less than 543°F.

B. Place the Steam Dump Mode Switch to RESET with Tave less than 547°F.

C✓ Bypass the Low Tavg interlock by selecting BYPASS TAVG INTERLOCK on the Steam Dump Control switch with Tavg less than 543°F.

D. Bypass the Low Tavg interlock by selecting BYPASS TAVG INTERLOCK on the Steam Dump Control switch with Tavg less than 547°F.

Correct answer is C.

A. Incorrect - The RESET position resets the loss of load bistables that have retentive memory. The switch is taken to RESET after plant conditions have stabilized and Tave is returned to program. Taking this switch to RESET will not allow the steam dumps to operate below 543°F.

B. Incorrect - The RESET position resets the loss of load bistables that have retentive memory. The switch is taken to RESET after plant conditions have stabilized and Tave is returned to program. Taking this switch to RESET will not allow the steam dumps to operate below 543°F. 547°F is the normal RCS No Load Tave where steam dumps controls are transferred from Tave control to Steam Pressure control. Steam Pressure control is used for RCS Cooldown.

C. Correct.

D. Incorrect - The steam dumps can be operated down to 543°F without taking the Steam Dump Control switch to BYPASS TAVE INTERLOCK. The steam dumps are setup to normally control at 1005 psig which equates to 547°F when in Steam Pressure Mode during shutdown conditions.

Question 61
Tier 2 / Group 2
K/A Importance Rating - RO 3.0 SRO 3.3

Knowledge of SDS design feature(s) and/or interlock(s) which provide for the following:
Relationship of low/low T-ave. setpoint in SDS to primary cooldown

Reference(s) - Sim/Plant design, System Description, GP-007
Proposed References to be provided to applicants during examination - None
Learning Objective - SD 005
Question Source - RNP Bank
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7
Comments - K/A is met because the candidate must know how the steam dump controls are operated during a cooldown and the associated temperature limits.

INIT

8.2.21 **WHEN** RCS temperature decreases to less than 543°F, **THEN** perform the following:

1. Display ERFIS point MSF0495D, STM LINE SI FLO/LO PRESS/TEMP BLK. (ACR 93-00023) ____
2. Check MSF0495D indicates TRIP. (ACR 93-00023) ____
3. Momentarily hold the STEAM DUMP CONTROL Switch in the BYPASS T-AVG INTLK position to continue Steam Dump. ____
4. Check that the STEAM DUMP T-AVG CONTROL BLOCKED status light has ILLUMINATED. ____
5. Momentarily hold the T-AVG Block/Unblock Switch in the BLOCK position. ____
6. Check that the LO TEMP SAFETY INJECTION BLOCKED status light has ILLUMINATED. ____
7. Check that ERFIS point MSF0495D, STM LINE SI FLO/LO PRESS/TEMP BLK, indicates trip reset. (ACR 93-00023) ____

NOTE:	ERFIS indication status change corresponding to the associated equipment change satisfies the acceptance criteria for this test. (ACR 93-00023)

8. **IF** MSF0495D does not indicate TRIP RESET, **THEN** initiate a Work Request for the Computer Group. (ACR 93-00023) WR# ________ ____

The steam dump valves are gas-operated valves. Normal gas supply is through the Steam Dump System Nitrogen Accumulator, which is kept pressurized by the high-pressure nitrogen system and a backup supply from an 18-bottle nitrogen bank. The S/G PORVs are also gas-operated valves. Their normal supply is from instrument air system, but an emergency supply line exists from the steam dump section of the Nitrogen system.

3.0 COMPONENT DESCRIPTION

3.1 Steam Dump Valves

The steam dump valves are eight inch, reverse-acting, nitrogen-operated globe valves rated for 810,000 lbm/hr at 740 psia. Because 100% steam flow is 10,110,000 lbm/hr, each steam dump provides approximately 8% steam flow capacity. Their normal supply of control gas is from the Steam Dump N_2 Accumulator through three parallel pressure control valves, PCV-1093A thru C, to a single header, which supplies nitrogen to the individual dump valves

The valves are divided into three banks. Bank 1 contains PRV-1324A1, PRV-1324B1 and PRV-1324B2 and provides a total of 24% steaming capacity when the valves are full open. Bank 2 contains PRV-1324A2 and PRV-1324B3, which have a 16% steaming capacity. Bank 4 contains the S/G PORVs with a combined capacity of 17% steam flow. The original Bank 3, atmospheric steam dump valves, was removed. The banks were never renamed due to the configuration of the instrumentation and control wiring.

3.2 Power Operated Relief Valves

The S/G PORVs (Bank 4) are eight inch, air operated globe valves. Their normal supply of control gas is from Instrument Air with a backup supply from the Steam Dump Nitrogen System. Procedures for swapping the gas supply and for manual operation of the S/G PORVs can be found in AOP-017 and the Dedicated Shutdown Procedures. Each valve is rated for 580,000 lbm/hr at a S/G pressure of 790 psi.

4.0 INSTRUMENTATION

4.1 Steam Dump Controls

The steam dump controls operate the five steam dumps and, when required, the S/G PORVs. The system functions to: reduce Tavg to 547°F after a turbine trip, reduce Tavg to within 5°F of Tref following a secondary load rejection or turbine run back, and is used for RCS cooldown until Residual Heat Removal System can take over.

4.1.1 Control Switches

4.1.1.1 Steam Dump Mode Switch (SD-Figures 1 & 2)

A three position switch is provided on the RTGB to select the mode of steam dump control. The positions are:

RESET (spring return to Tavg control) - Tavg - STEAM PRESS

When selected to STEAM PRESS, two functions within the system are performed: the steam dumps (**Banks 1 and 2**) are given an arming signal provided the permissives and low Tavg interlock are satisfied and the output of the steam pressure controller is sent to the steam dump valve positioners (This will cause APP-006-F5, STEAM DUMP ARMED, to illuminate). The arming portion of the system only applies to Banks 1 and 2. **The PORVs do not "arm"**; they are **always operational**, with Instrument Air supplied to them. In the STEAM PRESS mode, the steam pressure controller that has a 10-turn potentiometer that controls pressure over the range of 0 to 1400 psig controls the steam dumps. In the Steam Pressure mode of operation, the Steam Dump control system cannot control the operation of the S/G PORVs.

Tavg is the normal at-power position of the mode switch. In this position, either of the two sub modes, Load Rejection or Turbine Trip, can arm the steam dumps via the loss of load bistables, which sense the change in turbine impulse pressure. If no turbine trip occurs, the output of the load rejection controller is sent to the steam dump valve positioners to control Tavg to the prescribed value. If a turbine trip occurs, the output of the load rejection is blocked and the turbine trip controller output is sent to the steam dump valve positioners. In the Tavg mode of operation, the steam dump control system can control the operation of the S/G PORVs as long as a turbine trip has not occurred.

The RESET position resets the loss of load bistables that have retentive memory. The switch is taken to RESET after plant conditions have stabilized and T_{avg} is returned to program. This action disarms steam dump and prevents further opening of the steam dump valves until the system is again armed by another load rejection. The logic controls the nitrogen/air supplies to the valves following a loss of load or turbine trip. The Tavg

control position places the steam dump in automatic control in order that it can function following a loss of load or turbine trip at which time the difference between Tavg and Tref controls the dumps. The steam pressure control position places the dump system under the control of the steam header pressure.

4.1.1.2 Steam Dump Control Switch (SD-Figure-2)

A three-position switch is provided to control steam dumps, with the following positions:

OFF - ON - BYPASS T_{AVG} INTLK (spring return to ON)

In the ON position, the steam dumps are ON, waiting to be armed and actuated. The BYPASS Tavg INTLK position allows the low Tavg interlock to be bypassed. When Tavg reaches 543°F(decreasing) on 2 out of 3 Tavg channels, steam dumps are disarmed. When the low temperature signal is present and the switch is placed in BYPASS Tavg INTLK position, only the bank 1-steam dump valves can now be armed. When Tavg increases above 543°F the interlock block is automatically removed and the steam dump valves will automatically arm when all arming conditions are satisfied.

The OFF position serves two functions. When in OFF the system is disarmed (Steam Dumps disabled; PORV operation not affected). Placing the switch in OFF also reinstates the low Tavg block of steam dumps and disarms the bank 1 steam dump valves if bypassed with Tavg less than 543°F.

4.1.2 Sudden Loss of Load Bistables

These bistables provide the arming signal for the steam dump valves in the Tavg mode. They sense turbine impulse pressure from PT-447 and if a decrease of the proper magnitude (10%, 10%, or 50%) is sensed during a continuous two-minute period, the appropriate bistable trips, arming the system. The circuit can also cause S/G PORVs control to shift from normal to the steam dump controller (>50% rejection with no turbine trip).

Set points are:

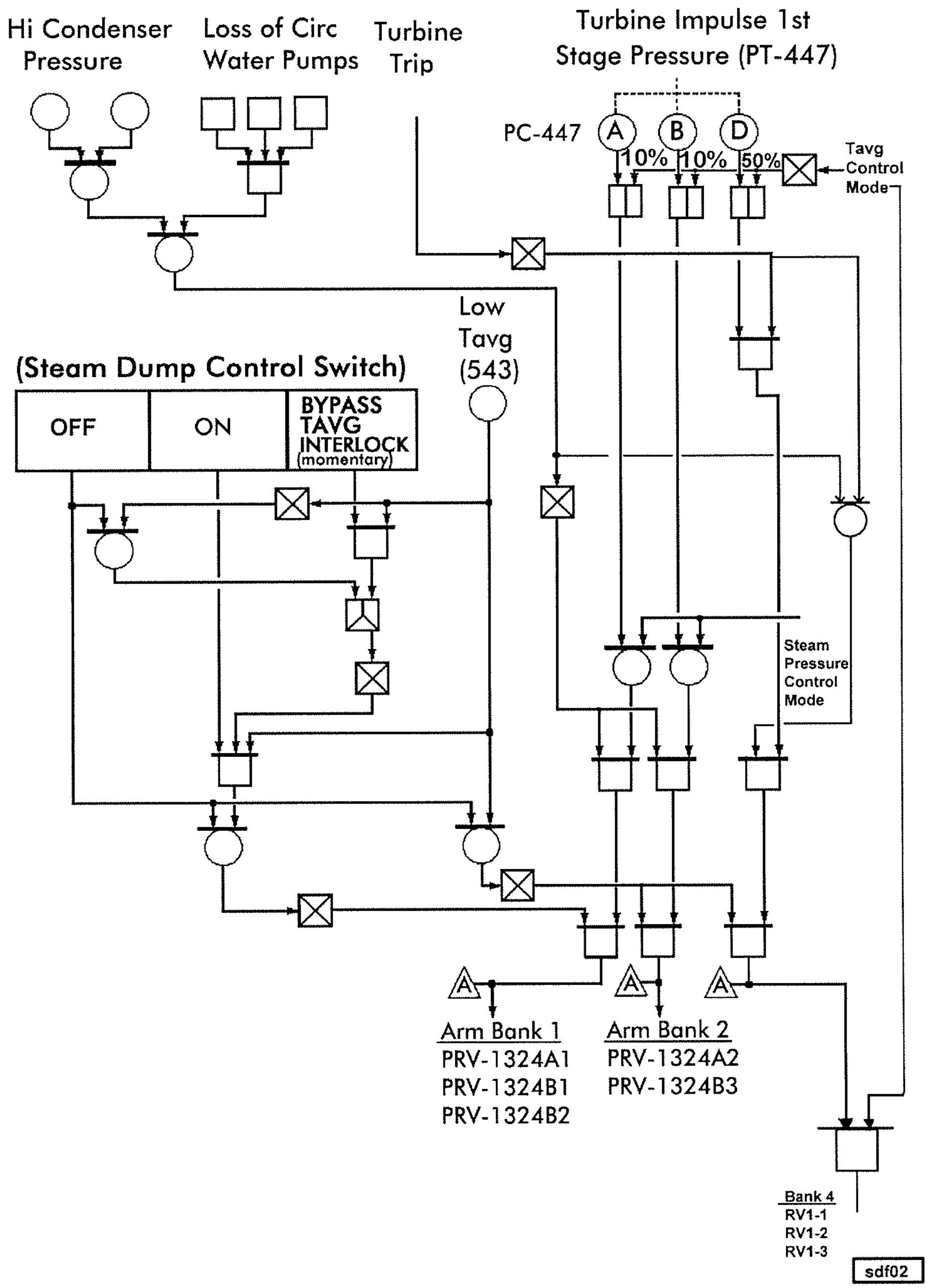
a. Hagan Derivative Unit. (Gain=1)
   (PM-447C) 105 seconds (525 sec. to return to minimum output)

b. Sudden Loss of Load Bistables

Controller No.	Arms	Bank	Setpoint

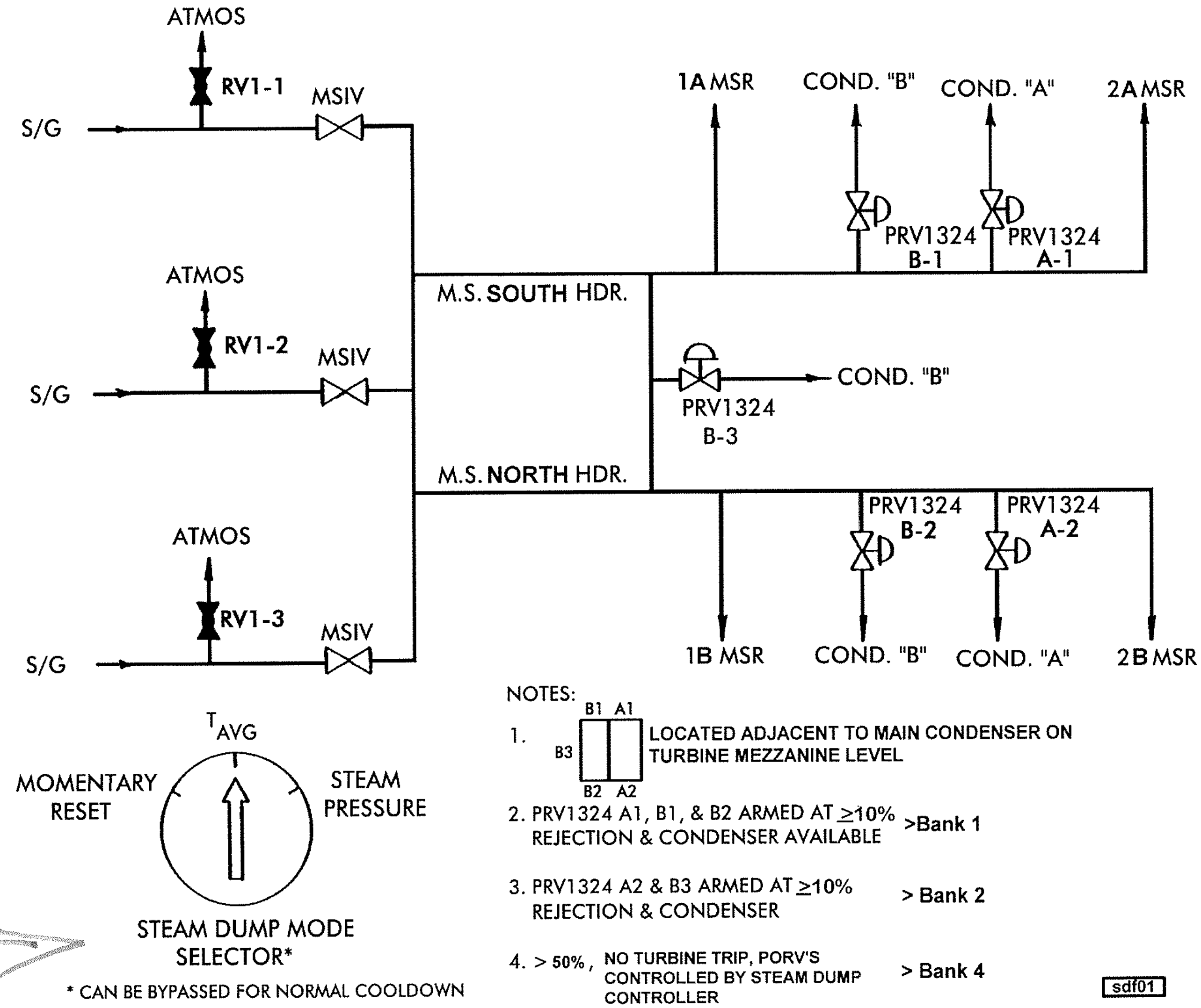
STEAM DUMP ARMING SIGNAL LOGIC

SD-FIGURE-2



INFORMATION USE ONLY

STEAM DUMP VALVES
SD-FIGURE-1



INFORMATION USE ONLY

62. 071 A1.06 001

Given the following plant conditions:

- Plant is at 100% RTP.
- Waste Gas Release of "A" Waste Gas Decay Tank is in progress.
- APP-010-B7, HVE-2A/2B AIR FLOW LOST/OVLD, is received.
- HVE-2A has tripped.

Which ONE (1) of the following describes the actions required by the operator AND the potential plant effect of a loss of the Reactor Auxiliary Building Exhaust Fan?

The operator will ____(1)____ AND the possible effect of a loss of Reactor Auxiliary Building Exhaust Fans is / has ____(2)____.

A. (1) have to manually start HVE-2B

(2) an unmonitored release from the Reactor Auxiliary Building.

B.✓ (1) verify that HVE-2B starts automatically

(2) an unmonitored release from the Reactor Auxiliary Building.

C. (1) have to manually start HVE-2B

(2) no impact since HVE-5A or HVE-5B will automatically start.

D. (1) verify that the HVE-2B starts automatically

(2) no impact since HVE-5A or HVE-5B will automatically start.

The correct answer is B.

A. Incorrect. HVE-2A and HVE-2B are interlocked so that on a motor electrical trip, the standby fan starts automatically. The standby fan will also start on air flow lost after a 20 second time delay. Plausible, since not all components have interlocks to automatically start the backup unit. The second half of the answer is correct.

B. Correct

C. Incorrect. HVE-2A and HVE-2B are interlocked so that on a motor electrical trip, the standby fan starts automatically. The standby fan will also start on air flow lost after a 20 second time delay. Plausible, since not all components have interlocks to automatically start the backup unit. HVE-5A / -5B are not operated during normal plant conditions and are not interlocked with HVE-2A/2B to start on a failure of HVE-2A/2B. However, HVE-5A/ -5B are interlocked such that they will not start unless HVE-2A or HVE-2B are running.

D. Incorrect. The first half of the question is correct. HVE-5A / -5B are not operated during normal plant conditions and are not interlocked with HVE-2A/2B to start on a failure of HVE-2A/2B. However, HVE-5A/ -5B are interlocked such that they will not start unless HVE-2A or HVE-2B are running.

Question 62
Tier 2 / Group 2
K/A Importance Rating - RO 2.5 SRO 2.8

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation system

Reference(s) - Sim/Plant design, System Description, APP-010-B7
Proposed References to be provided to applicants during examination - None
Learning Objective - WD 007
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 45.5
Comments -

ALARM

HVE-2A/B AIR FLOW LOST/OVLD

AUTOMATIC ACTIONS

1. Standby Fan STARTS

CAUSE

1. Breaker on MCC-5 (MCC-6) Open
2. Magnetic or Thermal Overload
3. Low flow due to Fan Fault
4. Belt slipping or broken

OBSERVATIONS

1. HVE-2A/B Indicating Lights

NOTE:	The low flow alarm may be reset and the fan restarted on a tripped fan by placing the fan control switch to stop and then back to start.

ACTIONS CK (✓)

1. **IF** the operating fan has tripped, **THEN VERIFY** the standby fan starts. ____
2. **IF** the operating fan has **NOT** tripped, **THEN PERFORM** the following:
   1) **STOP** the running fan. ____
   2) **START** the standby fan. ____
   3) **VERIFY** the annunciator clears. ____
3. **IF** a fan has tripped, **THEN CHECK** the affected breaker: ____
   – HVE-2A - MCC-5, CMPT 4M
   – HVE-2B - MCC-6, CMPT 3M
4. **IF** alarm is **NOT** due to breaker trip, **THEN CHECK** the following: ____
   – Damper positions
   – Fan belts
5. **IF** neither fan can be started, **THEN NOTIFY** RC Personnel of the possibility of an unmonitored release due to pressurization of the Auxiliary Building. ____

DEVICE/SETPOINTS

1. FS-A2 (HVE-2A)
2. FS-A24 (HVE-2B)

POSSIBLE PLANT EFFECTS

1. Loss of negative pressure in Auxiliary Building
2. Unmonitored release from the Auxiliary Building

REFERENCES

1. CWD B-190628, Sheet 540, Cable F

exhaust unit HVE-13. Air supplied by HVAC-1 is heated/cooled as needed and recirculated without any direct make-up or exhaust air. The men's and ladies' restrooms are ventilated by individual exhaust fans and heated by electric unit heaters, EUH-F and G. C-1 is located on the roof of the tool room adjacent to the CV access area.

2.4.8 Reactor Auxiliary Building Main Supply (HVS-1) and Exhaust (HVE-2A, HVE-2B) (See Figures 3 and 4)

The main air supply unit for the Reactor Auxiliary Building is supply unit HVS-1. HVS-1 supplies treated outdoor air during the summer and partly return/partly outdoor air during the winter.

Heating steam to coils in HVS-1 (Reactor Auxiliary Building) is supplied from the Auxiliary Steam System, and condensate is returned to the same system. Supply water to the cell type air washers is supplied from the Service Water System.

The inter-area air transfer in system HVS-1 is accomplished by maintaining a pressure differential between the supply air outlets and exhaust intakes so that the direction of air flow is always from areas of lower contamination to areas of higher contamination. Part of the air supplied by HVS-1 is collected and returned to the unit during winter. This is accomplished by a return air system that includes fan HVE-7. The total amount of air handled by HVE-7 includes air from the Electrical Equipment Area, Relay Room No. 1, CCW Surge Tank Area, and the H&V Equipment Room. The rest of the air supplied by HVS-1 is exhausted by two sets of exhaust units: HVE-5A, 5B and HVE-2A, 2B. HVE-5A and 5B exhaust to HVE-2A and 2B through a charcoal filtering unit and are not used during normal operations. HVE-2A and 2B discharge to the plant stack.

2.4.9 RHR Pump Room (HVH-8A, HVH-8B), SI (HVH-6A, HVH-6B), and AFW Pump Rooms (HVH-7A, HVH-7B) (See Figure 4)

These rooms are cooled by recirculating the air through cooling units located in the rooms. The RHR Pump Room is cooled by cooling units HVH-8A and 8B. The SI Pump Room is cooled by cooling units HVH-6A and HVH-6B. The AFW Pump Room is cooled by cooling units HVH-7A and HVH-7B. The actual cooling is accomplished by circulating the room air, and water from the Service Water System through a heat exchanger cooled by SW.

2.4.10 CCW Surge Tank Room - Electric Unit Heater EUH-1

This heater provides local heating to the Component Cooling Surge Tank Room.

2.4.11 Miscellaneous Rooms - Steam Unit Heaters SUH-1 through SUH-13

The air supply unit HVS-1 contains prefilters, steam heating coils, cell type air washer, centrifugal fan with drive and motor, and is housed within a room. The air intake of the unit is connected to dampered outdoor air louvers, and the supply air is discharged into an air distribution system which consists of ductwork, ductwork auxiliaries and air distribution terminals.

3.8.2 Exhaust Units HVE-5A and HVE-5B

Manufacturer - Motor	Westinghouse
Manufacturer - Fan	American Standard Ind. Division
Air flow rate per fan	5,750 cfm
Power requirements - per fan	5 hp
HEPA filters - number per unit	6
Carbon filters - number per unit	18

These units consist of high efficiency particulate air (HEPA) filters, activated carbon adsorbers in one sheet metal enclosure, and two 100 percent capacity axial flow fans each with drive and motor. The discharge of these units is connected to the intake of exhaust units HVE-2A and HVE-2B.

3.8.3 Exhaust Units HVE-2A and HVE-2B.

Manufacturer	Westinghouse
Air flow rate per fan	54,150 cfm
Power requirements - per fan	75 hp
Prefilters - number per unit	48
HEPA filters - number per unit	48

These exhaust air units HVE-2A and HVE-2B (standby) consist of air intake terminals, ductwork, ductwork auxiliaries, prefilters, and HEPA filters. The discharge from these units is directed to the plant stack.

3.9 RHR, SI and AFW Pump Rooms

3.9.1 HVH-8A and HVH-8B (RHR Pump Room)

Manufacturer – Motor	Westinghouse
Manufacturer - Fan	H. K. Porter Co., Inc.
Air flow rate per unit	8,200 cfm
Cooling capacity per unit - Total	71,316 Btu/hr

4.6 Diesel Generator Rooms

The supply fans HVS-5 (Diesel Room "B") and HVS-6 (Diesel Room "A") and their respective exhaust fans HVE-17 and HVE-18 in the diesel-generator rooms are started manually from a switch with indicating lights on the RTGB, or automatically when the diesel-generator starts. The outside air intake louver opens and the respective evaporative air cooler solenoid valve (EAC-1 or EAC-2) is energized when the supply fan is energized. The exhaust damper opens when the exhaust fan is energized.

An outdoor thermostat set at 55°F on temperature drop de-energizes the evaporative air cooler solenoid valve, opens the return air damper and places the exhaust fan on low speed.

A nitrogen bottle backup to the instrument air supply (for the HVE-17 and HVE-18 exhaust damper/louver actuators) is provided for fire protection considerations. The N_2 backup will be used to hold the dampers closed on loss of Instrument Air to allow CO_2 concentration to be maintained in the event of a fire requiring CO_2 actuation. An electric thermostat located in each room annunciates and sounds an alarm on the RTGB when the room temperature is above 110°F, APP-010-E5/E6.

4.7 Containment Vessel Access Area Supply and Exhaust

Unit HVAC-1 is controlled by a local thermostat located on the corridor wall. This starts the fan and energizes the compressor or heaters. Exhaust fan HVE-13 runs continuously in local control and acts to create a slightly negative pressure in the CV access area.

The men's and ladies' restroom ventilation fans run continuously while the heaters are controlled by local thermostats.

4.8 Reactor Auxiliary Building Main Supply and Exhaust

Supply air unit HVS-1 is interlocked with exhaust fans HVE-2A and HVE-2B. Starting of the exhaust fan HVE-2A or HVE-2B will start the supply air unit HVS-1. The thermostat controlled return, exhaust, and outdoor air dampers open or close depending on the outdoor air temperature. "HVS-1 Trouble", APP-010-A5, is annunciated on the RTGB on low flow, low temperature or loss of power. HVE-2A and 2B are started manually through switches with indicating lights located on the RTGB.

* Fans HVE-2A and 2B are interlocked so that on a motor electrical trip, the standby fan starts automatically and annunciator HVE-2A/B AIR FLOW LOST/OVLD at APP-010-B7 is actuated. On flow failure, an air flow switch (in the discharge of each fan) starts the standby fan after a 20 second time delay, de-energizes the controls of the fan that failed, annunciates low flow, and sounds an alarm for standby fan running.

A pitot tube sensing velocity pressure, located in the common discharge duct, modulates (through a differential pressure controller) the filter dampers to maintain constant air flow.

Booster fans HVE-5A and 5B for carbon and absolute filters are interlocked with HVE-2A and 2B fans. Manual starting of HVE-5A and 5B closes the normal flow damper automatically, opens the filter damper and starts the fan with indicating lights located on the RTGB. The damper positions are shown on the RTGB with indicating lights.

If the motor of HVE-5A electrically trips, the standby fan, HVE-5B will start automatically. An air flow switch in the filter duct also starts the standby fan on flow failure after a 20 second time delay, de-energizes the controls of the fan that failed, annunciates low flow, and sounds an alarm for standby running.

A pitot tube sensing velocity pressure located in the inlet duct to the filters, modulates the filter damper through a differential pressure controller to maintain constant air flow through the filters. The filter damper opens automatically when either exhaust fan HVE-5A or 5B is energized.

Fresh air intake louver and fan discharge dampers open and controls are energized when HVS-1 fan is energized. A modulating controller, set at 50°F, controls the air temperature leaving the steam coil by throttling the steam valve to each coil section. A thermostat in the return air readjusts the modulating controller to maintain a minimum return air temperature of 50°F. A thermostat located in the discharge of HVS-1 annunciates and sounds an alarm on the RTGB when discharge temperature is below 35°F.

Exhaust fan HVE-7 is energized when supply fan HVS-1 is energized. When the outdoor temperature is below 60°F, the EAC is not placed in service, the return air damper to HVS-1 opens, and the exhaust damper of HVE-7 closes. When the outdoor temperature is above 60°F, EAC-3 may be placed in service, the return air damper closes, and the exhaust damper opens.

Electric duct heaters EDH-2 and 3 are energized through a room thermostat when supply fan HVS-1 is energized.

4.9 RHR, SI, and AFW Pump Rooms

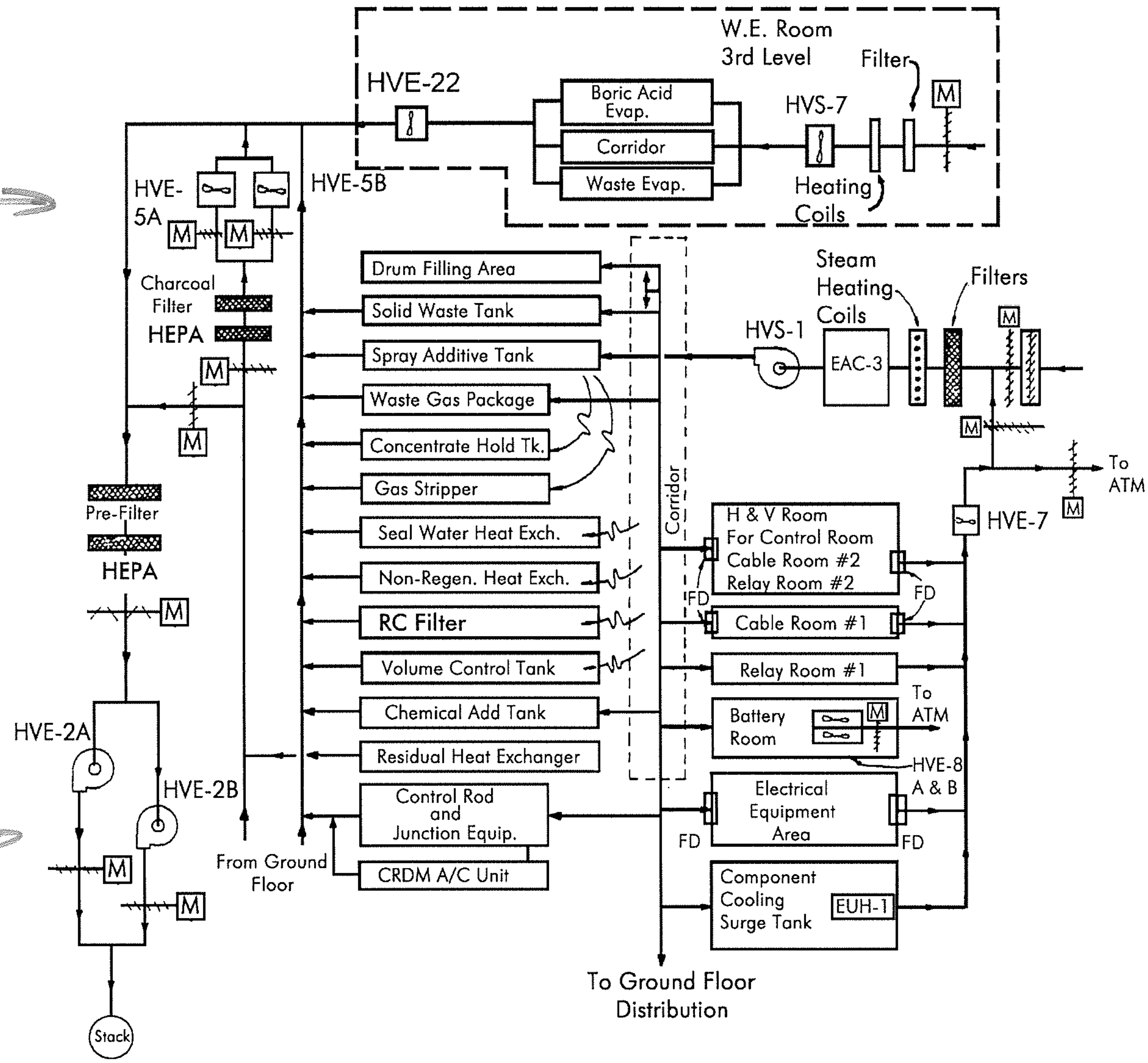
Cooler Unit Fans HVH-6A, -6B, -7A, -7B, -8A, and -8B, will start under the conditions specified in Section 6.1.9. Indicating lights on the RTGB show units off or running.

4.10 CCW Surge Tank Room - Electric Unit Heater EUH-1

EUH-1 is located in the Component Cooling Surge Tank Room and is solely controlled by

MEZZANINE LEVEL RAB VENTILATION

HVAC-FIGURE-3



INFORMATION USE ONLY

63. 072 G2.1.20 001

Given the following plant conditions:

- Plant is stable at 60% RTP.
- Charging pumps "A" & "C" are in service.
- Letdown orifice valves CVC-200A and CVC-200B are open.
- A valid alarm has been received on R-9, LETDOWN LINE AREA.

Which ONE (1) of the following identifies the correct operator response to a valid alarm on R-9, LETDOWN LINE AREA, IAW AOP-005, Radiation Monitoring System?

A.✓ Shut CVC-200B to minimize letdown flow.

B. Isolate letdown flow by closing LCV-460A / B.

C. Secure Charging pump "C" and lower Charging pump "A" to minimal flow.

D. Dispatch an operator to Loose Parts Monitoring Cabinet to monitor for abnormal indications.

The correct answer is A.

A. Correct. AOP-005 directs the operator to Verify letdown orifice valves less than or equal to one open.

B. Incorrect. AOP-005 directs the operator to Verify letdown orifice valves less than or equal to one open. The student may not remember the exact wording of the step and think that it would be more prudent to secure letdown rather than reduce the lineup to just one orifice in service as directed by AOP-005.

C. Incorrect. AOP-005 directs the operator to verify letdown orifice valves less than or equal to one open. Although maximizing purification may seem prudent, reducing letdown flow with an indication of failed fuel is more important.

D. Incorrect. Normally a Loose Parts Alarm will occur before indications of failed fuel. It is plausible to dispatch an operator to LPMS to see if loose part indications exist.

Question 63
Tier 2 / Group 2
K/A Importance Rating - RO 4.6 SRO 4.6

Ability to interpret and execute procedure steps. Area Radiation Monitoring System

Reference(s) - Sim/Plant design, System Description, APP-005, APP-036
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-005-003
Question Source - RNP Bank - Modified
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.12
Comments - K/A is met because candidate analyze the given plant conditions and associated Area Radiation Monitor alarm and determine the appropriate actions.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 9

AREA MONITOR R-9 - LETDOWN LINE AREA

(Page 1 of 2)

1. Place VLC Switch To EMERG Position

2. Place And Hold EVACUATION ALARM Switch To LOCAL Position For 15 SECONDS

3. Announce The Following Over Plant PA System:

   "ATTENTION ALL PERSONNEL. ATTENTION ALL PERSONNEL. A HIGH RADIATION ALARM HAS BEEN RECEIVED ON LETDOWN LINE AREA MONITOR R-9. ALL NON-ESSENTIAL PERSONNEL EVACUATE AUXILIARY BUILDING UNTIL FURTHER NOTICE"

4. Repeat PA Announcement

5. Place VLC Switch To NORM Position

6. Contact RC Personnel To Perform A Survey In The Following Areas To Determine Magnitude Of Radiation Source:

   - Lower level Aux Building

   - VCT area

7. Verify LTDN ORIFICE Valve(s) - LESS THAN OR EQUAL TO ONE OPEN

8. Control Charging Flow To Maintain PZR Level

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 9

AREA MONITOR R-9 - LETDOWN LINE AREA

(Page 2 of 2)

9. Request E&C To Sample The RCS To Determine The Following:
   - Gross Activity
   - Iodine
   - Gaseous Activity

10. Go To The Main Body, Step 1.b, Of This Procedure

- END -

ALARM

LPMS TROUBLE

AUTOMATIC ACTIONS

1. Auto-start of Tape Recorder (if selected)

CAUSE

1. Possible metal impact detected by one or more of five pairs of accelerometers located in the Reactor Coolant System (Three Steam Generators and the top and bottom of the Reactor Vessel)
2. Loss of power to LPMS **OR** LPMS Control Panel power OFF

OBSERVATIONS

1. R-9, Letdown Line
2. R-15, Condenser Air Ejector
3. R-19A, R-19B, and R-19C, Steam Generator Blowdown

ACTIONS	CK (✓)
1. **DISPATCH** an operator **OR** Shift Technical Advisor to the LPMS Control Panel to investigate the alarm.	____
2. **IF** impacts are present, **THEN OBTAIN** an Event Recorder printout using OP-007.	____
3. **IF** an actual loose part is suspected, **THEN NOTIFY** the On-Call Manager.	____

DEVICE/SETPOINTS

Refer to OP-007 for setpoints

POSSIBLE PLANT EFFECTS

1. Metal impact damage in affected region

REFERENCES

1. OP-007, Loose Parts Monitoring System
2. CWD B-190628, Sheet 801, Cable N

64. 075 A2.03 001

A plant shutdown is in progress with the plant at 20% RTP with the steam dumps in STEAM PRESSURE mode.

- "C" Circulating Water Pump is Out of Service for Maintenance.
- "A" and "B" Circulating Water Pumps are in operation.
- "B" Circulating Water Pump trips and V6-50B, CIRC WATER PMP "B" DISCH, remains full open and will not close.

What impact will this have on Condenser Vacuum and Steam Dump Operation AND what actions are required IAW AOP-012, Partial Loss of Condenser Vacuum or Circulating Water Pump Trip?

Condenser Vacuum will.....

A.✓ rapidly degrade. Steam dump operation may continue as long as ONE (1) Circulating Water Pump is running and Vacuum remains greater than 19.7" Hg.

Dispatch an operator to manually close V6-50B.

B. rapidly degrade. Steam dump operation may continue as long as ONE (1) Circulating Water Pump is running and Vacuum remains greater than 18" Hg.

Manually trip the turbine and go to AOP-007, Turbine Trip Below P-8.

C. slowly degrade. Steam dump operation may continue as long as ONE (1) Circulating Water Pump is running and Vacuum remains greater than 18" Hg.

Dispatch an operator to manually close V6-50B.

D. slowly degrade. Steam dump operation may continue as long as ONE (1) Circulating Water Pump is running and Vacuum remains greater than 19.7" Hg.

Manually trip the turbine and go to AOP-007, Turbine Trip Below P-8.

Correct answer is A.

A. Correct - Condenser vacuum will degrade rapidly due to a loss of CW flow. All CW flow will be routed back through V6-50B. Steam Dump operation requires greater than 19.7"Hg and at least ONE CWP to be in operation. AOP-012, Partial Loss of Condenser Vacuum or Circulating Water Pump Trip, immediate action step is to "**Verify** the tripped CWP Discharge Valve - Closed or Closing." Since the valve motor trips the BOP should dispatch an operator to locally close the valve.

B. Incorrect - 18" Hg is the setpoint for turbine trip for loss of vacuum. Tripping the turbine and going to AOP-007 would be the next appropriate act if the condenser backpressure entered a restricted region on Attachment 3, Condenser Backpressure Limit Curve. This information was not given in the stem of the question.

C. Incorrect - Condenser vacuum will degrade rapidly due to a loss of all CW flow to the water boxes. The candidate may not recognize the complete loss of CW flow. A normal trip of a CW Pump will cause condenser vacuum to degrade slowly with the reduction in CW flow. However, since V6-50B remained open all CW flow is lost. The second half of the response is correct.

D. Incorrect - Condenser vacuum will degrade rapidly due to a loss of all CW flow to the water boxes. The candidate may not recognize the complete loss of CW flow. A normal trip of a CW Pump will cause condenser vacuum to degrade slowly with the reduction in CW flow. Tripping the turbine and going to AOP-007 would be the next appropriate act if the condenser backpressure entered a restricted region on Attachment 3, Condenser Backpressure Limit Curve. This information was not given in the stem of the question

Question 64
Tier 2 / Group 2
K/A Importance Rating - RO 2.5 SRO 2.7

Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Safety features and relationship between condenser vacuum, turbine trip, and steam dump

Reference(s) - Sim/Plant design, System Description, APP-012
Proposed References to be provided to applicants during examination - None
Learning Objective - CW 008
Question Source - RNP Bank - Modified
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A is met because the candidate must predict the impact of a loss of circulating water on condenser vacuum and based on those impacts determine the appropriate course of actions.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-012

PARTIAL LOSS OF CONDENSER VACUUM OR CIRCULATING WATER PUMP TRIP

REVISION 22

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides instructions for a partial loss of Condenser vacuum or Circulating Water Pump trip.

2. ENTRY CONDITIONS

   This procedure is entered whenever a partial loss of Condenser vacuum or Circulating Water Pump trip occurs.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

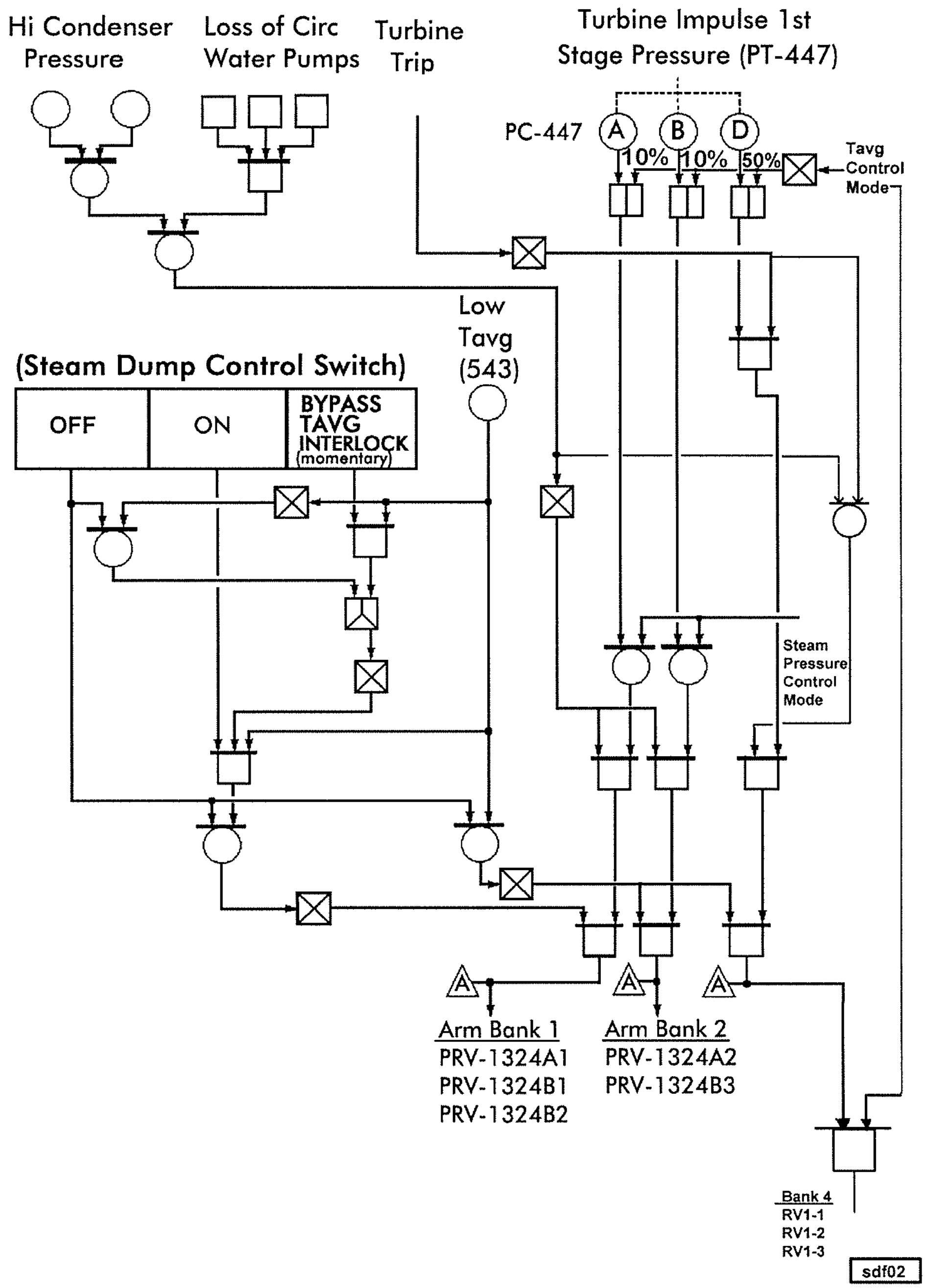
NOTE

Steps 1 and 2 are Immediate Action steps.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Circulating Water Pump - ANY TRIPPED	Go To Section A, Partial Loss of Condenser vacuum.
2.	<u>Verify</u> The Tripped Circulating Water Pump Discharge Valve - CLOSED <u>OR</u> CLOSING • V6-50A, CIRC WATER PMP "A" DISCH <u>OR</u> • V6-50B, CIRC WATER PMP "B" DISCH <u>OR</u> • V6-50C, CIRC WATER PMP "C" DISCH	
3.	Start Any Available Circulating Water Pump	
4.	Make PA Announcement For Procedure Entry	
5.	Check Liquid Waste Batch Release - IN PROGRESS	Go To Step 7.
6.	Stop Any Liquid Waste Batch Release In Progress As Follows: a. Stop the applicable running waste release pump. b. Isolate the waste release flowpath.	
7.	Check Condenser Status - VACUUM PREVIOUSLY ESTABLISHED	Implement the EALs. Return To Procedure and Step in effect.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8	Check Status Of The Tripped Circulating Water Pump Discharge Valves - COMPLETED CLOSING • V6-50A, CIRC WATER PMP "A" DISCH OR • V6-50B, CIRC WATER PMP "B" DISCH OR • V6-50C, CIRC WATER PMP "C" DISCH	WHEN the affected valve indicates closed, THEN Go To Step 9. IF the valve will NOT close from the RTGB, THEN dispatch an Operator to close the valve AND Go To Step 9.
9.	Check Plant Conditions - IN MODES 1 OR 2	Go To Step 14.
*10.	Check Condenser Back Pressure On PI-1312 AND PI-1313 - APPROACHES RESTRICTED REGION OF ATTACHMENT 3, CONDENSER BACKPRESSURE LIMIT CURVE	IF Condensor Backpressure approaches the restricted region of Attachment 3, Condenser Backpressure Limit Curve, THEN Go To Step 11. Go To Step 13.
11.	Check REACTOR TRIP BLOCK P-7 Status Light - ILLUMINATED	Perform the following: a. Trip the Reactor. b. Go To Path-1.
12.	Perform Turbine Trip Actions As Follows: a. Manually trip the Turbine b. Go To AOP-007, Turbine Trip Below P-8	
*13.	Check Condenser Backpressure On PI-1310 AND PI-1311 - INCREASING	WHEN Condenser Backpressure on PI-1310 and PI-1311 is Increasing, THEN observe the NOTE prior to Step 14 and Go To Step 14 Go To Step 15

STEAM DUMP ARMING SIGNAL LOGIC
SD-FIGURE-2



INFORMATION USE ONLY

Residual Heat Removal System can take over.

If the condenser heat sink is not available during a turbine trip, excess steam generated as a result of RCS sensible heat and core decay heat, is discharged to the atmosphere via the S/G PORVs and main steam safety valves by their normal, pressure-regulating mode of operation (not through steam dump system control).

2.3 System Flow Paths (SD-Figure-1)

Steam leaves each S/G through twenty-six inch steam lines. Just upstream of each (MSIV) is located the S/G PORV for each S/G, RV1-1, RV1-2 and RV1-3. The S/G PORVs relieve to atmosphere. The steam lines from each S/G join at a common seventy-two inch header and then split into a North and South Main Turbine Supply Header. The condenser steam dumps tap off the North and South Main Turbine Supply Headers and discharge to the Main Condenser. The North Header supplies two valves, PRV-1324A-2 and PRV-1324B-2, which are located south of the main condenser. The South Header also supplies two valves, PRV-1324A-1 and PRV-1324B-1, which are located north of the Main Condenser. The North South crossconnect header supplies PRV-1324B-3, which is located west of the "B" condenser. Valves PRV-1324B-1, PRV-1324B-2 and PRV-1324B-3 dump steam to the "B" (West) condenser and valves PRV-1324A-1 and PRV-1324A-2 dump steam to the "A" (East) condenser.

2.4 System Description

The S/G PORVs and the condenser steam dump valves have separate controls and can be operated independently of each other. The S/G PORVs only act as part of the Steam Dump System when a large load rejection occurs and the turbine does not trip. The S/G PORVs provide a means for plant cooldown by steam discharge to atmosphere if the condenser steam dumps are not available (condenser vacuum not available, MSIVs shut, etc.)

The S/G PORVs are normally in automatic sensing their respective steam line pressure. Individual steam header pressure is compared to the desired steam pressure as set on the S/G PORV controller and modulates individual S/G PORV valves open as necessary to maintain steam pressure below setpoint.

The steam dump system has two basic modes of control: "Tavg Mode" and "Steam Pressure Mode. In Steam Pressure Mode of control, the steam dump modulation is controlled by main steam header pressure. Main steam header pressure is compared to the desired steam pressure as set on the steam dump controller and modulates steam dump valves open as necessary to maintain steam pressure at setpoint.

Tavg Mode of control has basically two sub-modes: load rejection and turbine trip. The function of either of these modes is to control Tavg to some desired value set by the operator.

If the plant were to experience a load rejection there would initially exist a power mismatch between the reactor and turbine until the rods could be inserted to lower reactor power. This mismatch would cause Tavg to increase, which will cause the rod control system to insert rods to lower the temperature. If the load rejection is larger than rod control is designed to handle, the steam dumps would open to give an artificial load to limit the Tavg rise. The amount of modulation open of the steam dump valves is controlled by the deviation of actual Tavg (median Tavg is used) from program Tavg (Tref). If the deviation exceeds a preset value another signal is sent to the steam dump valves to "pop" open. With the turbine at a lower power level and impulse pressure, the program value for Tref will be lower than before the load rejection. As the rod control system inserts the rods, lowering Tavg, the deviation between actual and program temperature decreases and the steam dump valves modulate shut.

The Turbine Trip submode basically performs the same as load rejection mode. When a turbine trip occurs, a power mismatch occurs between the reactor and the secondary system, even if a reactor trip occurs. Power in the reactor does not immediately go to zero, so heat must be removed from the RCS. Since turbine impulse pressure is zero after a trip, Tref goes to its minimum value. If sufficient deviation from no load Tavg is sensed, the steam dump valves will modulate open or "pop" open if the deviation exceeds a preset value.

(SD-Figure-2)

The system has to meet three "permissives" to be able to modulate the condenser steam dump valves: a circulating water pump must be running, adequate condenser vacuum must be available (>19.7"Hg Vac; i.e. APP-008-B6, CONDENSER LO VACUUM TURB TRIP, extinguished) and the system must be turned on. In addition to the permissives being met, the steam dump system must be "armed" to allow dump valves to open. In the steam pressure mode of control, the system is armed when Steam Pressure Mode is selected and the steam dump valves can modulate as necessary to control main steam header pressure as long as the permissives are met. In the Tavg mode of control further stimulus is required to arm the steam dumps. If a load rejection is sensed, as determined by turbine impulse pressure decreasing by a prescribed amount in a preset time, the load rejection submode will "arm" and actuates to help control Tavg along the Tavg program. If a turbine trip occurs, the turbine trip submode "arms" and actuates to reduce Tavg to the no-load value. Since PT-447 arms the Steam Dumps and PT-446

65. 086 A4.01 001

Which ONE (1) of the following lists the location(s) that the Motor Driven Fire Pump (MDFP) and the Engine Driven Fire Pump (EDFP) can be manually started?

The MDFP can be started at the __(1)__ AND the EDFP can be started at the __(2)__.

A.✓ (1) Unit 2 Control Room AND Unit 2 Intake Structure.

(2) Unit 2 Intake Structure ONLY.

B. (1) Unit 2 Control Room AND Unit 2 Intake Structure.

(2) Unit 2 Control Room AND Unit 2 Intake Structure.

C. (1) Unit 2 Control Room ONLY

(2) Unit 2 Intake Structure ONLY

D. (1) Unit 2 Control Room ONLY

(2) Unit 2 Control Room AND Unit 2 Intake Structure.

The correct answer is A.

A. Correct - The MDFP can be manually started from the control room and Unit 2 Intake Structure. The EDFP can be manually started at the Intake Structure ONLY.

B. Incorrect - The MDFP can be manually started from the control room and Unit 2 Intake Structure. The EDFP can be manually started at the Intake Structure ONLY. The candidate may think that EDFP has controls in the control room similar to the MDFP. The controls for the MDFP are on the Containment Fire Control Panel in the control room.

C. Incorrect - The MDFP can be manually started from the control room and Unit 2 Intake Structure. The EDFP can be manually started at the Intake Structure ONLY. The MDFP is only started locally during the performance of tests that specifically require using the local controls.

D. Incorrect - The MDFP can be manually started from the control room and Unit 2 Intake Structure. The EDFP can be manually started at the Intake Structure ONLY. Only one fire pump can be started from the control room and the candidate may think that the EDFP is the appropriate selection.

Question 65
Tier 2 / Group 2
K/A Importance Rating - RO 3.3 SRO 3.3

Ability to manually operate and/or monitor in the control room: Fire water pumps

Reference(s) - Sim/Plant design, System Description, OP-801
Proposed References to be provided to applicants during examination - None
Learning Objective - FPW 004
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.7 / 45.5 to 45.8
Comments - K/A is met because candidate must know the location of the fire pump controls.

5.0 CONTROLS AND PROTECTION

5.1 Booster Pump

The pump runs continuously and is not connected to any instrumentation or controlling devices other than its motor controller. The local control panel is equipped with a two (2) position selector switch (ON/OFF), normally in the ON position.

5.2 Motor Driven Fire Pump

The local control switch at the intake is equipped with a three (3) position rotary snap switch (OFF, AUTO, ON). The selector switch is normally in the AUTO position. This allows for automatic starting of the fire pump by the pressure switch. Manual starting is accomplished by moving a rotary snap switch on either the local controller or on the Containment Fire Protection Panel in the Control Room to the ON position.

5.3 Engine Driven Fire Pump

The controller for the diesel engine driven fire pump is located in a separate structure west of the Intake Structure. Input power is supplied from Lighting Panel 35 Circuit 8. The controller electrical output for battery charging is 24 volts DC. The control system consists of the controller, battery charger, batteries, and associated alarms as shown in Figure 6. The controller performs the following functions:

1. In AUTO, starts the engine upon a low fire main loop pressure of 90 psig (85 psig to 95 psig). An approximate two (2) second time delay is provided so a pressure spike will not start the diesel fire pump.
2. Maintains battery charging via the two (2) battery chargers.
3. Provides for normal and emergency battery starting power.
4. Provides for local starting and stopping.
5. Accommodates MANUAL testing.
6. Monitors the following eight indications and alarms:

    A. BATTERY 1 FAILURE
    B. BATTERY 2 FAILURE
    C. LOW OIL PRESSURE
    D. HIGH WATER TEMP

8.2 NORMAL OPERATIONS

NOTE:	Emergency operations allow the starting of the Fire Water Pumps without the performance of pre-start checks.

8.2.1 Operation of the Fire Water Booster Pump

1. PLACE the local control switch on the Fire Water Booster Pump Controller in the ON position.

2. CHECK that the Fire Water Booster Pump has started by observing discharge pressure indication on PI-7049 at the pump is at 110 psig or greater.

8.2.2 Operation of the Motor Driven Fire Pump (MDFP)

NOTE:	Any start of the Motor Driven Fire Pump, either manual or automatic, will cause APP-044-C58, M.D. Fire Pump Running, to go into alarm.

AT INTAKE

1. **IF** the MDFP will be started locally, **THEN** PERFORM the following:

   a. CHECK oil level is in the MDFP bull's eye

   b. PLACE the local control switch on the MDFP in the ON position.

   c. CHECK that the Motor Driven Fire Pump has started by observing discharge pressure indication on PI-7050 is between 125 and 145 psig.

   d. VERIFY the local control switch on the MDFP is in the AUTO position.

IN CONTROL ROOM

2. **IF** the MDFP will be started from the Containment Fire Protection Panel (CFPP), **THEN** PERFORM the following:

   a. PLACE the Motor Driven Fire Pump control switch on the Containment Fire Protection Panel (CFPP) in the ON position.

   b. CHECK that the Motor Driven Fire Pump has started by the illumination of the red annunciator light on the CFPP above the Motor Driven Fire Pump control switch.

66. G2.1.28 001

Given the following plant conditions:

- The plant was at 42% RTP and rising IAW GP-005, Power Operation.
- A loss of feedwater occurs.
- All three S/G NR levels are at 0%.
- It is now 29 seconds later and the Reactor Trip Breakers have remained closed.

Which ONE (1) of the following is the expected response of the ATWS Mitigation System Actuation Circuitry (AMSAC) and the purpose of those actions?

AMSAC will __(1)__ and the purpose of AMSAC is to prevent __(2)__

A.✓ (1) TRIP the Main Turbine and START the AFW pumps

(2) voiding in the RCS during an ATWS condition and a loss of feedwater.

B. (1) TRIP the Main Turbine and START the AFW pumps

(2) exceeding the kW/ft limit of the fuel rods during an ATWS condition.

C. (1) START the AFW pumps, and close the S/G blowdown and sample isolation valves

(2) prevent voiding in the RCS during an ATWS condition and a loss of feedwater.

D. (1) START the AFW pumps, and close the S/G blowdown and sample isolation valves

(2) exceeding the kW/ft limit of the fuel rods during an ATWS condition.

The correct answer is A.

A. Correct.

B. Incorrect. The first part of answer is correct. AMSAC was not designed to provide protection for the reactor against exceeding the linear power rating of the fuel rods. This is the design basis of the Overpower DeltaT Reactor Protection circuitry.

C. Incorrect. AMSAC will start the AFW pumps, however, the AFW pump start circuitry will send the signal to close the SG blowdown valves. The second part of the answer is correct.

D. Incorrect. AMSAC will start the AFW pumps, however, the AFW pump start circuitry will send the signal to close the SG blowdown valves. AMSAC was not designed to provide protection for the reactor against exceeding the linear power rating of the fuel rods. This is the design basis of the Overpower DeltaT Reactor Protection circuitry.

Question 66
Tier 3
K/A Importance Rating - RO 4.1 SRO 4.1

Knowledge of the purpose and function of major system components and controls.

Reference(s) - Sim/Plant design, System Description,
Proposed References to be provided to applicants during examination - None
Learning Objective - AMSAC 005
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7
Comments - K/A met because the candidate must know the purpose of AMSAC and how it operates.

1.0 INTRODUCTION

The ATWS Mitigating System Actuation Circuitry (AMSAC) system is the result of a 10CFR50.62 requirement that requires a system to be installed to mitigate the effects of an Anticipated Transient Without Scram (ATWS).

2.0 GENERAL DESCRIPTION

2.1 System Purpose

The purpose of the AMSAC System is to automatically initiate a turbine trip and auxiliary feedwater start under conditions indicative of an ATWS condition and a loss of feedwater.

2.2 System Description

The AMSAC is a processor based logic system. The AMSAC System monitors the level signals from all three steam generators. There is one level transmitter per steam generator with a different power supply that provides AMSAC inputs through safety-related isolation equipment. AMSAC, according to Addendum 1 to WCAP 10858 , must be armed prior to exceeding 40% power to prevent voiding in the RCS in case of a loss of feedwater ATWS. Note that the logic diagram reflects the requirement that AMSAC be armed when greater than 35% load. The arming setpoint is conservatively set at approximately 35% load. There are two pressure signals taken from the first stage turbine which provide inputs to enable AMSAC at 191 psig (approximately 35% turbine load). Both turbine inputs are required to enable AMSAC. Upon indication that two steam generator levels are approximately 5% of narrow range span below the steam generator low-low level setpoint of 16%, AMSAC will initiate a trip signal if the plant is at or greater than approximately 35% turbine load (Note: AMSAC could initiate at less than 35% turbine load since a time delay, TD2, prevents losing the 35% signal for 360 sec.) The trip signal is fed into the turbine trip and auxiliary feedwater start circuitry through isolation relays. There is a AFW bypass keyswitch on the RTGB that will prevent AMSAC from starting the AFW pumps when the keyswitch is in the bypass position. There is a normal/bypass hand switch on the RTGB which allows the actuating signal from AMSAC to be blocked.

3.0 COMPONENT DESCRIPTION

AMSAC consists of two independent and redundant processors which are capable of initiating a trip signal. These processors perform self-diagnostics and system diagnostics automatically at preset intervals or on operator command. AMSAC will display a trouble annunciation if self diagnostic tests indicate trouble. In the event of a power failure, AMSAC will fail to the non-trip mode. AMSAC processors have a nonvolatile EPROM memory and will continue to function as power is restored. Loss

of power is not anticipated since the AMSAC has a dedicated UPS power supply and a battery internal to prevent program loss. The UPS inverter in the 4160v Switchgear Room provides the 120 VAC power to AMSAC.

3.1 AMSAC Programmable Controller

The AMSAC is built with Gould PC 0185 programmable controller components. Each processor consists of the following:

1 PC 0185 Programmable Controller
1 I/O 1103 DC input, transistor output module
4 I/O 1106 AC input, relay output modules
2 AI 1121 Analog, input modules
2 PS 100 Auxiliary power supplies

The controller enters the run mode as soon as power is applied. It can be stopped by changing the mode through the programmer or by switching off power to the unit. In normal circumstances operator attention is not required.

Operation of the unit is best described after explaining the function of the indicators and controls.

3.2 Setpoints

The following abbreviations are used for setpoints:

LT - The steam generator level below which the AMSAC triggers (1.44V). This equates to 11% NR
TD1 - The time delay between level dropping below LT and the AMSAC firing (25S).
PT - The power level below which firing of the AMSAC is inhibited (2.20V). This equates to 191 psig (approximately 35% turbine load).
TD2 - The time delay in inhibiting the AMSAC when power drops below PT (360S).

4.0 INSTRUMENTATION

4.1 Loop Instrumentation (Figure 1)

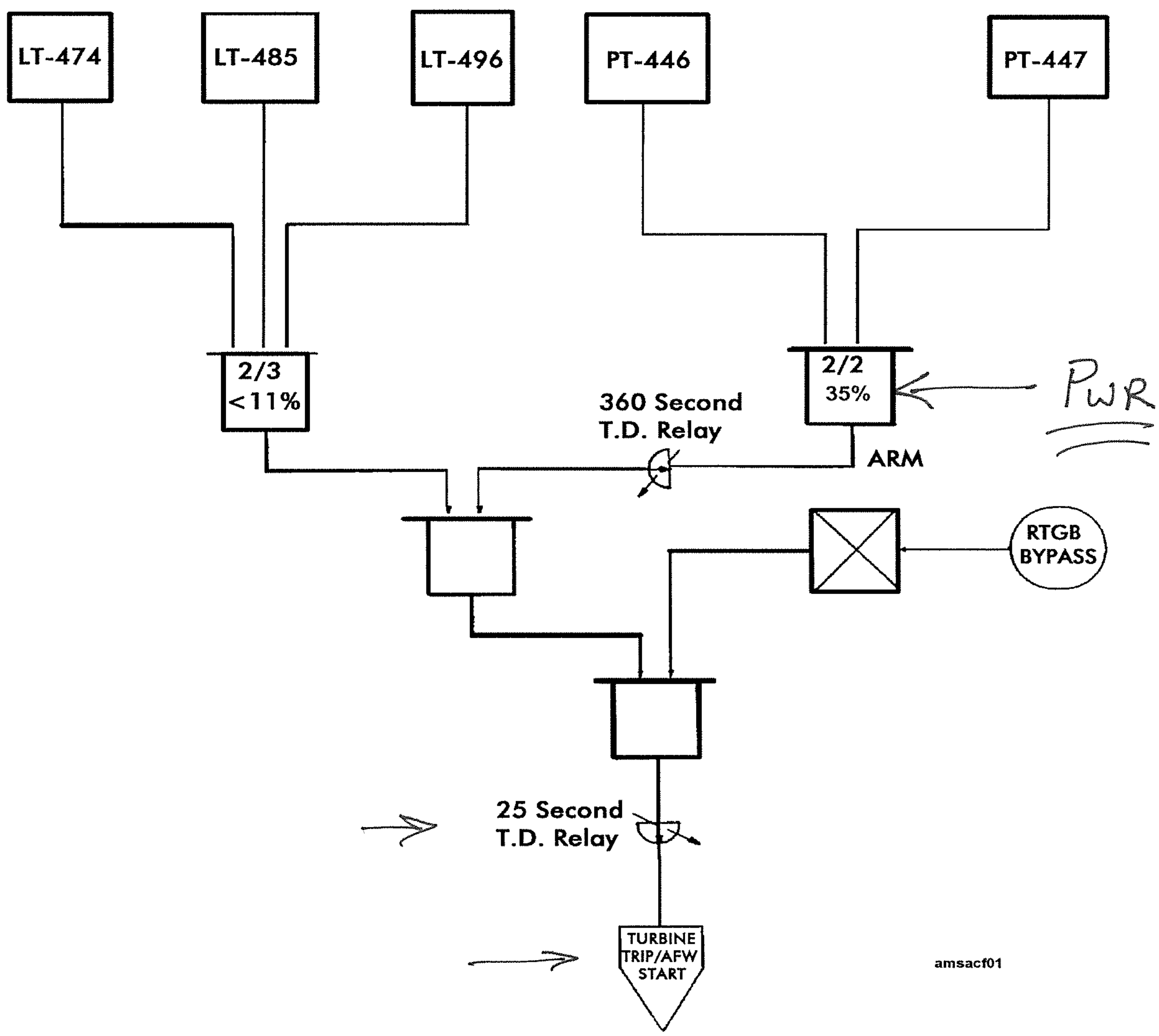
4.1.1 Loop Pressure - Turbine First Stage

There are two pressure transmitters sharing a common sensing line from the turbine first stage.

Loop	Instrument
PT-446	PM-446D

AMSAC LOGIC DRAWING

AMSAC-FIGURE-1



INFORMATION USE ONLY

67. G2.1.4 001

In order to maintain an active license and qualification, a Reactor Operator must, at a **MINIMUM,** meet the following requirements:

A. Stand a minimum of **four** 12 hour shifts per calendar quarter as RO.
Participate in LOCT and be evaluated as an RO or BOP in LOCT.
Have a current annual medical exam.

B. Stand a minimum of **four** 12 hour shifts per calendar quarter as RO or BOP.
Participate in LOCT and be evaluated as an RO in LOCT.
Have a current biennial medical exam.

C. Stand a minimum of **five** 12 hour shifts per calendar quarter as RO.
Participate in LOCT and be evaluated as an RO or BOP in LOCT.
Have a current annual medical exam.

D.✓ Stand a minimum of **five** 12 hour shifts per calendar quarter as RO or BOP.
Participate in LOCT and be evaluated as an RO in LOCT.
Have a current biennial medical exam.

The correct answer is D.

A. Incorrect. To reactivate a license an individual has to have a minimum of 40 hours of contact time with a licensed watchstander. This usually takes four 12 hour shifts to accomplish. Second sentence is correct. Third sentence is incorrect because only a biennial medical exam is required.

B. Incorrect. To reactivate a license an individual has to have a minimum of 40 hours of contact time with a licensed watchstander. This usually takes four 12 hour shifts to accomplish. May also think that watch must be stood as the RO. Second sentence is incorrect because the individual must be evaluated as an RO in LOCT. Third sentence is correct.

C. Incorrect. First sentence is incorrect because the individual may stand watch as either the RO or BOP. The second sentence is incorrect because the individual must be evaluated as an RO in LOCT. The third sentence is incorrect because the individual is only required to have a biennial medical exam.

D. Correct. Per OMM-001-5.

Question 67
Tier 3
K/A Importance Rating - RO 3.3 SRO 3.8

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Reference(s) - Sim/Plant design, System Description, OMM-001-5
Proposed References to be provided to applicants during examination - None
Learning Objective - OMM-001-5-004
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.10 / 43.2
Comments - K/A met because candidate must know the requirements to maintain an active license status.

NOTE:	For accounting purposes, a proficiency watch started in one quarter and ending in the next is credited in the quarter in which the watch is completed.

8.3 Maintaining License And Qualification Active Status

8.3.1 The minimum shift complement is specified in OMM-001-2. To stand any position an individual must be actively qualified to stand that watch station. In addition, for the SM, CRS, and CO positions, the individual must also hold an active license IAW 10CFR55.

8.3.2 To maintain active license status, individuals must stand a minimum number of hours in a licensed position, satisfactorily participate in the Licensed Operator Continuing Training Program, and have a biennial medical exam.

8.3.3 Individuals with SRO licenses may maintain an active license by standing watch a minimum of five 12 hour shifts per calendar quarter in a licensed-required position as SM, CRS, RO or BOP. In addition, the individual shall be trained and evaluated as an SRO in LOCT.

- An SRO who stands five RO watches, but less than the required number of SM or CRS watches, is not qualified for the SM or CRS supervisory position, but still maintains an active license for manipulating RTGB controls. (NUREG-1262)

8.3.4 Individuals with RO licenses may maintain an active license by standing watch a minimum of five 12 hour shifts per calendar quarter as RO or BOP and be trained and evaluated as an RO in LOCT.

8.3.5 Maintaining active qualification is not synonymous with having an active license status. An individual may have an active license status but not be qualified to stand a particular watch station. To be qualified an individual must:

- be respirator qualified as defined in the section on Maintaining License and Qualification Active Status.
- satisfactorily participate in the Licensed Operator or Auxiliary Operator Continuing Training Program
- stand the required number of watches each calendar quarter:

1. To maintain qualification as SM/CRS, active SRO licensed individuals shall stand a minimum of five 12 hour shifts in a license required position, at least one of which must be in the SM or CRS position in the Main Control Room.

   - Maintaining active SM/CRS qualification status also qualifies the individual to stand watch as the STA (must have an eligible college degree), Refueling SRO, RO, BOP or AO.

2. To maintain qualification as RO, active licensed individuals shall stand a minimum of five 12 hour shifts as the RO or BOP in the Main Control Room.

   – Maintaining active RO qualification status also qualifies individuals licensed as an RO to stand watch as the BOP and AO.

3. To maintain qualification as STA, individuals shall:

   – Stand a minimum of three 12 hour shifts as STA **OR** maintain an active SRO license

4. To maintain qualification as an AO, individuals shall stand a minimum of one 12 hour shift in any AO watch station (Inside AO, Outside AO, MWT, or FPAO). The individual is only qualified to stand watch in positions in which one 12 hour shift was stood during the previous calendar quarter.

8.3.6 Active status for licensed watchstanders will also be maintained in Cold Shutdown with staffing in excess of the TECH SPECS Staffing requirement IAW OMM-001-2, **AND** the licensed individuals are actively performing the functions of watchstander for five 12-hour shifts per calendar quarter. The positions required by OMM-001-2 are determined necessary for safe plant operations. (NUREG-1262)

8.3.7 Operators receiving new licenses are considered proficient (active) by virtue of having passed the license exam, and may stand watches in the calendar quarter which the license is issued. The requirements specified above regarding active license and qualification for specific positions will commence in the first calendar quarter after receiving the license.

8.3.8 Respiratory protection qualifications (CR 99-01605)

- All Shift Operations personnel are required to be respirator qualified.

- In order to support timely donning of SCBAs in response to toxic gas conditions, Control Room watch standers (SM, CRS, STA, RO, & BOP) shall be clean-shaven in the area where the respirator sealing surface makes contact with the face, **AND** shall not have any hair which intrudes into the sealing surface or exhalation valve area of the respirator.

- In addition to the required training and medical requirements, Control Room watchstanders and Fire Brigade members shall maintain a current (annual) SCBA quantitative and qualitative fit test .

- Shift personnel should maintain qualifications on the SCBA and at least one air purifying respirator.

- Watchstanders who require glasses shall maintain a pair of respirator glasses in the Fire Equipment Building while on watch and assigned to the Fire Brigade. (NCR 50096)

8.4 Inactive Status

8.4.1 If a licensee has **NOT** been actively performing the functions of an operator or senior operator for the periods defined in the section on Maintaining License And Qualification Active Status, then the individual's license is declared inactive and the licensee may **NOT** resume activities authorized by a license issued under 10CFR55.

8.4.2 Inactive licensees may still fulfill the functions of Fire Brigade Incident Commander and Fire Brigade Member if all fire brigade training is current, act in the capacity of WCC SRO or WCC RO and perform WCC functions, or perform valve manipulations and independent verifications with approved procedures.

8.4.3 If a non-licensed watchstander has **NOT** been actively performing the functions for which he/she is qualified for the periods defined in the section on Maintaining License And Qualification Active Status, the watchstander may **NOT** resume activities authorized by their qualifications.

8.4.4 Inactive non-licensed watchstanders may still fulfill the functions of Fire Brigade Member (if all fire brigade training is current) or perform valve manipulations and independent verifications with approved procedures.

8.5 Reactivation of a Licensed Watchstander

NOTE: A minimum of 40 hours of contact time with the licensed watchstander is required for reactivation. Time spent outside the Main Control Room can be counted toward the 40 hours, provided the reactivating individual is in the company of the assigned watchstander performing watchstander activities, or attending meetings for which the watchstander is responsible. Time spent separated from the assigned watchstander will not count toward the required 40 hours.

Time spent performing required plant tours counts toward the 40 hours.

8.5.1 **IF** an individual's license becomes inactive status, **THEN** before resumption of activities authorized by a license issued under 10CFR55, the Manager – Operations shall certify, using Attachment 10.2, that qualifications and status of the licensee are current and valid as discussed in the section on Qualification Documentation, **AND** that the licensee has completed a minimum of 40 hours of shift functions in the position in which the individual will be qualified.

8.5.2 **IF** an individual has an active license and it is desired to reactivate his qualification at another watch station, **THEN** the individual must complete a minimum of 40 hours of shift functions in that watch station. Attachment 10.2 Steps 1, 2 and 5 are used to document the watches stood and Steps 3 and 4 of the attachment should be marked N/A.

8.5.3 The following guidelines apply regarding the required number of watches stood:

1. The 40 hours shall be accumulated during a minimum of four complete shift watches including shift turnovers IAW plant procedures before and after each watch under the direction of an individual qualified to stand that watch station.

2. The 40 hours shall include a complete tour of the plant as defined in the section on Plant Tours (For Reactivation Purposes Only).

8.5.3 (Continued)

3. The 40 hours of reactivation time should take place over a maximum period of four weeks **AND** shall occur in the same calendar quarter.

4. Only actual contact hours with the active watchstander will count toward the required 40 hours. Time spent out of the Main Control Room shall be documented I Attachment 10.2 Step 2, noting the activities involved and whether the active watchstander was present.

8.5.4 Reactivation of the Refueling SRO is covered under Short Term and Special Qualifications.

8.5.5 Inactive licensees may stand watch on non-licensed watch stations provided ONE of the following conditions are met:

- The individual has reactivated their license IAW the section on Reactivation of a Licensed Watchstander.

- The individual stands one 12-hour watch, including shift turnovers and a complete set of logs, under the supervision of an active watchstander for the particular non-licensed watch station(s) **AND** completes Attachment 10.3

- Manager – Operations waives requirement to reactivate for the particular non-licensed watch station(s) **AND** documents on Attachment 10.3

8.5.6 Completed Attachments 10.2 and 10.3 shall be routed to the Operations Scheduler who will revise Attachment 10.4 to reflect the reactivation of qualification.

68. G2.1.43 001

With the plant at 100% RTP what power level will be acceptable for performance of OST-206, Comprehensive Flow Test for the Steam Driven Auxiliary Feedwater Pump and what effects will forward flowing the S/G have on Reactor Power?

In accordance with OST-206, Reactor Power must be below.....

A. 99%.

Running the SDAFW Pump results in a rise in reactor power *solely* due to the additional feedwater flow.

B.✓ 98%

Running the SDAFW Pump results in a rise in reactor power due to using steam to run the pump and cooler feedwater being supplied to the S/Gs.

C. 99%

Running the SDAFW Pump results in a rise in reactor power due to using steam to run the pump and cooler feedwater being supplied to the S/Gs.

D. 98%

Running the SDAFW Pump results in a rise in reactor power *solely* due to the additional feedwater flow.

The correct answer is B.

A. Incorrect. 99% is correct when running the SDAFW pump on recirculation. The reason given is also correct for running the SDAFW pump on recirculation.

B. Correct. See exert from OP-402 below.

C. Incorrect. 99% is correct when running the SDAFW pump on recirculation. The basis given is correct.

D. Incorrect. The power level is correct, however, the basis is not correct.

Exert from OP-402, Auxiliary Feedwater System.

5.28 When the SDAFW Pump provides forward flow from the CST to the Steam Generators, the CST water from the SDAFW Pump is significantly cooler than the Main Feedwater (60-70°F vs. 435°F), which results in an increase in steam demand and an increase in reactor power. Reactor power should be less than 98% prior to pump operation when feeding the Steam Generators in Mode 1. During this time, the evolution is designated as an R2 Reactivity Evolution IAW OPS-NGGC-1306.

Due to the increase in steam demand when running the SDAFW Pump on recirculation (steam demand is proportional to the pump flow rate), reactor power should be less than 99% prior to running the pump on recirculation in Mode 1. During this time, the evolution is designated as an R2 Reactivity Evolution IAW OPS-NGGC-1306.
Refer to NCR 263090 Assignment 14.

Question 68
Tier 3
K/A Importance Rating - RO 4.1 SRO 4.3

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Reference(s) - Sim/Plant design, System Description, OST-206, OP-402
Proposed References to be provided to applicants during examination - None
Learning Objective - AFW 008
Question Source - NEW
Question History - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.10 / 43.6 / 45.6
Comments - K/A is met because the candidate must the power limitations for operation of a SDAFW pump and forward flowing to the S/Gs and the basis for this power limitation.

5.28 When the SDAFW Pump provides forward flow from the CST to the Steam Generators, the CST water from the SDAFW Pump is significantly cooler than the Main Feedwater (60-70°F vs. 435°F), which results in an increase in steam demand and an increase in reactor power. Reactor power should be less than 98% prior to pump operation when feeding the Steam Generators in Mode 1. During this time, the evolution is designated as an R2 Reactivity Evolution IAW OPS-NGGC-1306.

Due to the increase in steam demand when running the SDAFW Pump on recirculation (steam demand is proportional to the pump flow rate), reactor power should be less than 99% prior to running the pump on recirculation in Mode 1. During this time, the evolution is designated as an R2 Reactivity Evolution IAW OPS-NGGC-1306.

Refer to NCR 263090 Assignment 14.

5.29 When feeding more than one Steam Generator with the Steam Driven AFW Pump, the potential exist to have back flow from a higher pressure feedwater line to a lower pressure feedwater line resulting in indication of greater than 500 GPM flow.

5.30 Feeding Steam Generators with the Steam Driven AFW Pump will result in FWUFM instability due to changes in flow/temperature profile and an increase in FWUFM Calorimetric calculated power. When this occurs, diverse redundant indications of reactor power should be monitored until steady state conditions are achieved. (NCR 93811)

INIT

8.2.10 **ISOLATE** Instrument Air to the SDAFW Pump Woodward Governor by performing the following:

1. **CLOSE** IA-3120, SDAFW PUMP WOODWARD AIR REGULATOR ISOLATION. ____

2. **BLEED** air from the SDAFW PRESS CTLR BYPASS REG BLOWDOWN. ____

3. **VERIFY** PI-11204, SDAFW GOVERNOR AIR PRESSURE, (on the supply line to the Woodward Governor) is indicating zero. (Gauge located near IA-3120). ____

8.2.11 **CAUTION** personnel in the area of the SDAFW pump atmosphere exhaust prior to admitting steam to the turbine. ____

CAUTION
Reactor Power, Turbine/Generator load, and Steam Generator levels should be closely monitored to ensure that no abnormal conditions occur while feeding cold water to the Steam Generators. Controlling Reactor Power between 3% and 5% will increase the steaming rate of the Steam Generators, and reduce the chances of Steam Generator levels reaching the High Level setpoint while running the SDAFW pump.
The SDAFW Pump shall **NOT** be started **IF** CST level is < 34%.

8.2.12 **VERIFY** reactor power is below 98% IAW OP-105 prior to running the SDAFW Pump. ____

69. G2.2.20 001

Given the following information.

- Plant is at 100% RTP.

- The current plant Risk Profile (EOOS profile) is an elevated GREEN due to scheduled repairs to Reactor Protection related components. These repairs are in-progress and will take 5 hours to restore, test and return to service.

The Diesel System Engineer has coordinated with the Work Week Coordinator and Plant Management to gain concurrence to open the "A" EDG Cabinet for a visual inspection to resolve a recent Vendor Notice.

Which ONE (1) of the following describes the actions that must be taken prior to authorizing the "A" EDG Cabinet inspection to occur while the Reactor Protection work continues?

A. EOOS Risk Assessment will NOT be performed since opening the cabinet does NOT render the EDG unavailable. EOOS Risk Assessment will not need to be performed. Work may be released.

B.✓ EOOS Risk Assessment will be performed assuming that the EDG will be made unavailable. **IF** the assessment results in an **ORANGE** risk condition **THEN** the emergent work SHALL NOT be released.

C. EOOS Risk Assessment will be performed assuming that the EDG will be made unavailable. **IF** the assessment results in a **YELLOW** risk condition **THEN** the emergent work SHALL NOT be released.

D. EOOS Risk Assessment will be performed assuming that the EDG will be made unavailable. **IF** the assessment results in an **ORANGE** risk condition **THEN** the work may be released with CRS approval.

The correct answer is B.

A. Incorrect. Plausible, because simply opening the cabinet for a visual inspection does not render the EDG unavailable. However, OMM-048 clearly states that Emergent Work items on safety related components be evaluated for risk assuming that the component will be made unavailable as a result of the activity even if it is not planned to do so. Opening of an EDG Cabinet is used as an example in OMM-048. This was a relatively recent change to OMM-048 to ensure that a "what if" assessment is performed when performing emergent work.

B. Correct. Per OMM-048, "Work which will place the plant in an ORANGE risk condition should the unavailability occur SHALL NOT be released."

C. Incorrect. Per OMM-048, the intent of performing a "what if" risk assessment is NOT to defer all work based on this assessment. OMM-048 assessment that could result in a change in risk status (Green to Yellow) shall be discussed with the WWC prior to release.

D. Incorrect. Per OMM-048, "Work which will place the plant in an ORANGE risk condition should the unavailability occur SHALL NOT be released." This is a plausible answer since the Shift Manager has the authority to authorize many activities, however, this is not a decision that the Shift Manager nor CRS can make independently.

Question 69
Tier 3
K/A Importance Rating - RO 2.6 SRO 3.8

Knowledge of the process for managing troubleshooting activities.

Reference(s) - Sim/Plant design, System Description, OMM-048
Proposed References to be provided to applicants during examination - None
Learning Objective - OMM-048-002
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13
Comments - K/A is met because the candidate must know the process for performing "what-if" risk assessments for emergent work activities assuming the equipment will be made unavailable. Candidate must also know the restrictions on allowing work to be performed based on the results of the risk assessment.

8.1.5 Fix-It-Now/Facilities Maintenance Risk Assessment

1. FIN Team/Facilities Maintenance work is normally minor scope work and is prioritized using ADM-NGGC-104. Items having priority E, 1, 2, and 3 will be released and assessed by the WCC-SRO. The Work Week Coordinator will review FIN Team items during the morning work coordination meeting and bring up items having plant impacts for discussion and/or closure at the afternoon schedule review meeting. FIN Team/Facilities Maintenance work will be Risk Assessed on an as needed basis by the WWC or WCC SRO and documented in the Risk Section of the 1300 Meeting Summary Sheet.

8.1.6 Emergent Work Risk Assessment

NOTE:	The intent of the Risk Assessment of emergent conditions combined with scheduled work is to evaluate and manage the overall risk and minimize the incremental increase in risk even in an acceptable yellow configuration.

1. General

Emergent conditions may result in the need for action prior to conduct of the formal assessment, or could change the conditions of a previously performed assessment. Examples include plant configuration or mode changes, additional SSCs out of service due to failures, or significant changes in external conditions (weather, offsite power availability). The following guidance applies to this situation:

The risk assessment should be performed (or re-evaluated) to address the changed plant conditions on a reasonable schedule commensurate with the risk significance of the condition. Based on the results of the assessment, ongoing or planned maintenance activities may need to be suspended or rescheduled, and SSCs may need to be returned to service.

8.1.6.1 (Continued)

For non-routine emergent work or emergent work associated with short duration LCO's (7 days or less), reference Attachment 10.6, Unplanned Schedule Change / Emergent Work Checklist. This attachment is to provide guidance prior to the implementation / schedule integration of non-routine emergent activities and emergent work associated with a short duration LCO (7 days or less). It is meant to provide guidance throughout the planning and scheduling process associated with emergent work, to help ensure a broad perspective is maintained when preparing for the activity. [CAPR 281604]

Performance or re-evaluation of the assessment should not interfere with, or delay, the operator and/or maintenance crew from taking timely actions to restore the equipment to service or take compensatory actions.

If the plant configuration is restored prior to conducting or re-evaluating the formal assessment, the formal assessment need not be conducted, or re-evaluated if already performed.

2. Up-front Assessment

a. Emergent Work items during normal business hours are added to the Emergent Work list electronically. The Work Week Coordinator will perform an up-front risk assessment and mark a "Y" in "RISK REVIEW COMP" block. If the block is blank the item has not been up-front risk assessed due to not having determined a date for work to be performed or scope of work not planned.

b. The risk assessment is performed using the Risk Assessment methodology described in this procedure.

8.1.6.2 (Continued)

c. The following guidance should be performed as soon as reasonably possible for Emergent Work items: (NCR 321420)

If work is to be released on safety related equipment, then prior to the release, EOOS will be used to quantitatively evaluate the risk assuming that the component will be made unavailable as a result of the activity even if it is not planned to do so. This information will be factored into the decision to release the work or defer it. This would not include work that is modeled and documented in the current EOOS profile as long as there are no deviations from the original assumptions.

EXAMPLE: Adjustment of equalizing voltage on an in-service battery charger – Risk will be assessed as if the charger will be made unavailable during the activity.

EXAMPLE: Opening an EDG Cabinet for inspection – Risk will be assessed assuming that the EDG will be made unavailable.

The intent is NOT to defer all work based on this assessment. It is meant to be used as a tool to assess the consequences of the maintenance if the component is made unavailable as a result. This will ensure that consequences are thoroughly evaluated and a risk-informed decision is made. General guidelines for implementation are listed below:

Work which will place the plant in an ORANGE risk condition should the unavailability occur **SHALL NOT** be released.

Work that could result in a change in risk status (Green to Yellow) shall be discussed with the WWC prior to release.

The Shift Manager and CRS must use sound judgment for releasing work when activities could result in an elevation of risk that remains within the current risk status color. (i.e. risk is yellow and unavailability results in an elevated yellow condition.

70. G2.2.22 001

Which ONE (1) of the following sets of conditions represents a violation of a Technical Specification Safety Limit and required action?

(Reference Provided)

A. RTP - 10%, PZR Pressure - 2350 psig, RCS Highest Cold Leg Temp - 650°F

Restore compliance and be in Mode 3 within 1 hour.

B.✓ RTP - 80%, PZR Pressure - 2250 psig, RCS Highest Cold Leg Temp - 640°F

Restore compliance and be in Mode 3 within 1 hour.

C. RTP - 10%, PZR Pressure - 2350 psig, RCS Highest Cold Leg Temp - 650°F

Restore compliance within 1 hour ONLY.

D. RTP - 80%, PZR Pressure - 2250 psig, RCS Highest Cold Leg Temp - 640°F

Restore compliance within 1 hour ONLY.

The correct answer is B.

A. Incorrect. Does not exceed TS safety limit. Plausible because pressure is above the PORV setpoint of 2335 psig and the temperature is above RCS Tave DNB limit of 579.4°F.

B. Correct. The parameters exceed the TS 2.1.1 Reactor Core safety limits. Required action as specified in TS 2.2.1 is to restore compliance and be in MODE 3 within 1 hour.

C. Incorrect. Does not exceed TS safety limit. Plausible because pressure is above the PORV setpoint of 2335 psig and the temperature is above RCS Tave DNB limit of 579.4°F. The second half of the answer is incorrect because it omits the requirement to be in Mode 3 within 1 hour. Plausible because some TS Action Statements are to restore within 1 hour or change modes within a specific timeframe.

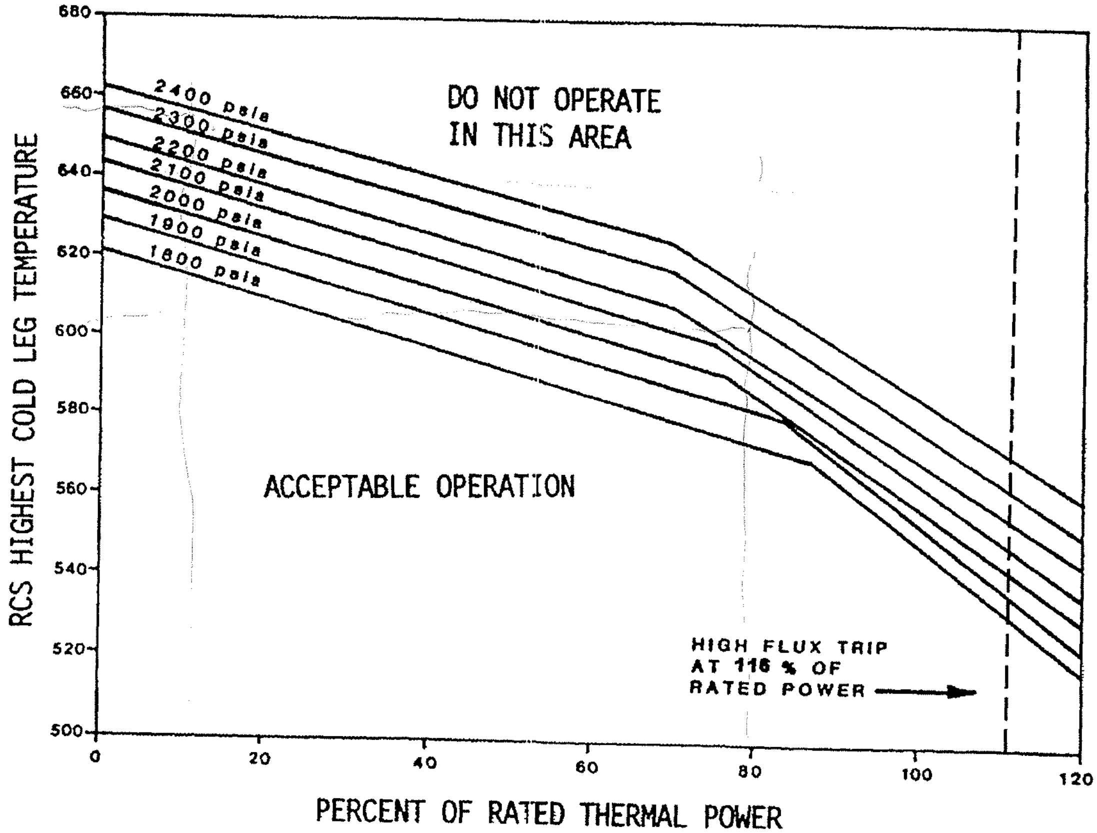
D. Incorrect. The parameters exceed the TS 2.1.1 Reactor Core safety limits. The second half of the answer is incorrect because it omits the requirement to be in Mode 3 within 1 hour. Plausible because some TS Action Statements are to restore within 1 hour or change modes within a specific timeframe.

Question 70
Tier 3
K/A Importance Rating - RO 4.0 SRO 4.7

Knowledge of limiting conditions for operations and safety limits.

Reference(s) - Sim/Plant design, System Description, ITS 2.0
Proposed References to be provided to applicants during examination - ITS Figure 2.1.1-1
Learning Objective - RCS 014
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.2 / 45.2
Comments - K/A is met because candidate must be able to determine is a safety limit is violated based on given information. Candidate must also know the ITS Action requirements for exceeding safety limits.

REFERENCE MATERIAL



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3×10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest cold leg temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

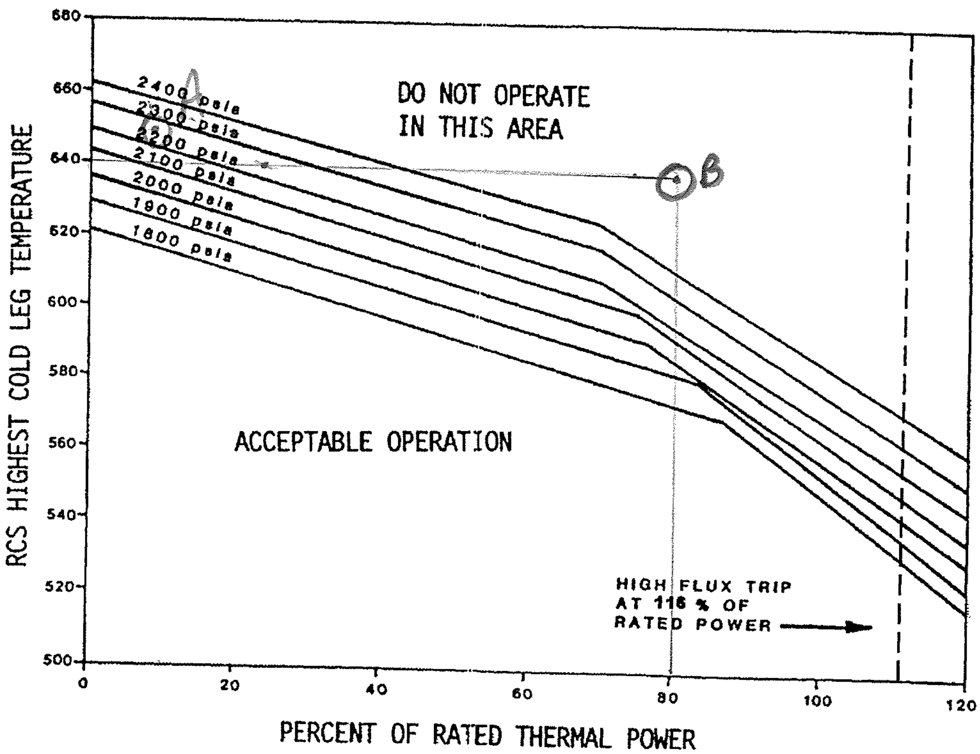
2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3×10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

71. G2.2.35 001

What Technical Specification OPERATIONAL MODE is the plant in with RCS temperature being maintained at 380°F AND how many Safety Injection Pump(s) are required to be operable?

The plant is in Mode ____(1)____ AND there must be at least _____(2)____ Safety Injection Pump(s) operable without entering a TS Action Statement.

A. (1) 3

(2) 1

B.✓ (1) 3

(2) 2

C. (1) 4

(2) 1

D. (1) 4

(2) 2

The correct answer is B.

A. Incorrect. The first part is correct. The second part is plausible since 1 SI Pump would be required when in Mode 4.

B. Correct. Mode 3 is greater than or equal to 350°F. Two SI pumps are required.

C. Incorrect. Mode 4 is plausible since the temperature at which the mode is determined is 350°F. The second part is plausible since 1 SI pump would be required in Mode 4.

D. Incorrect. Mode 4 is plausible since the temperature at which the mode is determined is 350°F. The second part is correct for Mode 3.

Question 71
Tier 3
K/A Importance Rating - RO 3.6 SRO 4.5

Ability to determine Technical Specification Mode of Operation.

Reference(s) - Sim/Plant design, System Description, ITS 3.5, ITS 1.0
Proposed References to be provided to applicants during examination - None
Learning Objective - RCS-014 / SI-SD-002-13
Question Source - Bank - Modified
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 41.1 / 43.2 / 45.13
Comments - K/A is met because the candidate must determine the plant Mode based on given RCS temperature and the candidate must also know the ITS limit for ECCS trains in this Mode.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

------------------------------------NOTES------------------------------------

1. In MODE 3, one cold leg safety injection (SI) pump flow path may be isolated by closing the isolation valves for up to 24 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. Operation in MODE 3 with one required SI pump declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. One valve identified in SR 3.5.2.1 or SR 3.5.2.7 with control power or air restored.	B.1 Verify control power is removed to all valves identified in SR 3.5.1.5. <u>AND</u>	Immediately (continued)

72. G2.3.13 001

Given the following plant conditions:

- A fuel handling accident occurs in containment and an evacuation is ordered via a PA announcement.
- R-14C, NG-LO, is reading normal with a steady reading.
- R-11 and R-12, Containment Particulate and Gas Radiation Monitors, indications are showing a slight rising trend.
- The Equipment Hatch is removed.

Which ONE (1) of the following is an action to be taken to mitigate the consequences of this event IAW AOP-013, FUEL HANDLING ACCIDENT?

A. Verify Control Room Ventilation System shifted to Emergency Recirculation Mode.

B✓ Place HVE-3 and HVE-4, CV Iodine RMVL Fan, in the PREPURGE position.

C. Place the Upper Fuel Handling Building Ventilation System in service.

D. Verify and maintain the CV Purge secured.

The correct answer is B.

A: Incorrect - The control room ventilation System is shifted to the Emergency Pressurization mode. The Emergency Recirculation mode is an abnormal lineup that the ventilation can be placed in but is not appropriate for this event.

B: Correct - HVE-3 and HVE-4 are placed in the PRE-PURGE position IAW AOP-013.

C: Incorrect - The Upper Fuel Handling Building Ventilation System is only addressed if the accident location is in the Spent Fuel Pool Area.

D. Incorrect - The CV Purge is only secured if the Equipment Hatch is installed. AOP-013 contains a NOTE that reinforces the intent of the procedure to maintain the CV Purge in service if the Equipment Hatch is <u>NOT</u> installed.

Question 72
Tier 3
K/A Importance Rating - RO 3.4 SRO 3.8

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Reference(s) - Sim/Plant design, System Description, AOP-013
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-013-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.12 / 43.4 / 45.9 / 45.10
Comments - K/A is met because the candidate must know the appropriate actions following a fuel handling accident.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-013

FUEL HANDLING ACCIDENT

REVISION 12

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides the instructions for actions to be taken for a fuel handling accident in Containment, the Spent Fuel Pit Area or the New Fuel Storage Area, OR a drop of the Spent Fuel Shipping Cask while on site.

NOTE

- For the purposes of this procedure an "Accident" is a drop of a fuel assembly OR other event that has the potential to cause damage to the assembly.
- Minor bumps of assemblies being moved AND leaning assemblies are NOT considered accidents.

2. ENTRY CONDITIONS

Upon receiving a report that a fuel handling accident, as described above, has occurred.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Evacuate Personnel From Affected Area:	
	a. Place the VLC switch in EMERG	
	b. Check accident location - INSIDE CV	b. Perform the following: 1) Place and hold EVACUATION ALARM switch in the LOCAL position for 15 seconds. 2) Go To Step 1.d.
	c. Depress and hold CV EVACUATION HORN button for 15 seconds	
	d. Announce the following over the PA system: 1) nature of the accident 2) location of the accident 3) location of personnel assembly area based on accident location: • CV - outside airlock at bottom of stairs • Spent Fuel Pit - landing halfway down the stairs • New Fuel Storage Area - outside Fuel Handling Building door	
	e. Repeat the alarm/horn for 15 seconds	
	f. Repeat the announcement	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Notify RC Personnel To Perform The Following: • Survey personnel from the affected area • Establish access control for the affected area • Take air samples and surveys of the affected area to determine when reentry may be made for damage assessment	
3.	Check Accident Location - INSIDE CV	IF the accident location is in the Spent Fuel Pit Area, THEN Go To Step 10. IF the accident location is in the New Fuel Storage Area, THEN Go To Step 15.
* 4.	Check Containment Radiation Levels As Follows:	
	a. Verify R-11/R-12 SELECTOR switch in the CV position	
	b. Check R-11 AND R-12 - EITHER MONITOR INCREASING OR IN ALARM	b. IF R-11 OR R-12 indication increases or goes into alarm, THEN Go To Step 5 Go To Step 8.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

The intent of this procedure is to maintain the CV Purge in service if the Equipment Hatch is NOT installed.

* 5. Verify Containment Ventilation As Follows:

a. Check Equipment Hatch - INSTALLED

Response Not Obtained:

a. Perform the following:

1) Verify HVE-1A AND 1B, CV PURGE FANs are running.

2) Verify V12-6 AND V12-7, CV PURGE INLETs are open.

3) Verify V12-8 AND V12-9, CV PURGE OUTLETs are open.

4) Verify CV Closure using OMM-033, CV Closure.

5) At MCC-5, open breaker AUX BUILDING SUPPLY FAN, HVS-1 (CMPT-7J).

6) Go To Step 5.c.

b. Verify CV Purge As Follows:

- HVE-1A AND 1B, CV PURGE FANs - STOPPED
- V12-6 AND V12-7, CV PURGE INLETs - CLOSED
- V12-8 AND V12-9, CV PURGE OUTLETs - CLOSED

c. Verify Vacuum AND Pressure Relief As Follows:

- V12-10 AND V12-11, CV PRESS RELIEFs - CLOSED
- V12-12 AND V12-13, CV VAC RELIEFs - CLOSED

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Verify Control Room Ventilation System Shifted To Emergency Pressurization Mode As Follows:	
	a. Verify CONT RM AIR EXHAUST Fan, HVE-16 - STOPPED	
	b. Verify CLEANING Fan HVE-19 A <u>OR</u> B - RUNNING	
	c. Verify CONTROL ROOM AIR EXHAUST FAN DISCHARGE DAMPERs - CLOSED • CR-D1A-SA • CR-D1B-SB	c. Locally verify - CLOSED • CR-D1A-SA • CR-D1B-SB
7.	Initiate CV Cleanup As Follows:	
	a. Place HVE-3, CV IODINE RMVL FAN, control switch in the PREPURGE position	
	b. Place HVE-4, CV IODINE RMVL FAN, control switch in the PREPURGE position	
8.	Notify RC personnel to monitor radiation levels in the Reactor Coolant Filter Area	
9.	Go To Step 15	
10.	Verify The Upper Fuel Handling Building Ventilation System Is In Service For Moving Irradiated Fuel Using OP-906, Heating, Ventilation, And Air Conditioning	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*11.	Determine If Unplanned Release Is In Progress Due To Breached Fuel As Follows:	
	a. Check RI-14C, NG-LO - INCREASING <u>OR</u> IN ALARM	a. <u>IF</u> RI-14C increases or alarms, <u>THEN</u> Go To Step 11.b. Go To Step 12.
	b. Perform dose projections using EPRAD-03, Dose Projections	
	c. At MCC-5, open breaker AUX BUILDING SUPPLY FAN, HVS-1 (CMPT-7J)	
12.	Notify RC Personnel To Monitor Radiation Levels In The Following Areas: • Spent Fuel Pit Cooling Pumps • Spent Fuel Pit Heat Exchanger	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
13.	Determine If Radiation Levels In The Spent Fuel Pit Demineralizer Area Should Be Monitored:	
	a. Check Spent Fuel Pit Purification System - ALIGNED TO SPENT FUEL PIT	a. Contact Engineering Personnel to evaluate the need to place the Spent Fuel Pit Purification System in service. <u>IF</u> Engineering Personnel recommend placing the system in service, <u>THEN</u> place it in service IAW OP-910, Spent Fuel Pit Cooling and Purification System <u>AND</u> Go To Step 13.b. <u>IF</u> Engineering Personnel recommend <u>NOT</u> placing the system in service, <u>THEN</u> Go To Step 14.
	b. Notify RC Personnel to monitor radiation levels in the Spent Fuel Pit Filter/Demineralizer Area	
14.	Verify Control Room Ventilation System Shifted To Emergency Pressurization Mode As Follows:	
	a. Verify CONT RM AIR EXHAUST Fan, HVE-16 - STOPPED	
	b. Verify CLEANING Fan HVE-19 A <u>OR</u> B - RUNNING	
	c. Verify CONTROL ROOM AIR EXHAUST FAN DISCHARGE DAMPERs - CLOSED • CR-D1A-SA • CR-D1B-SB	c. Locally verify - CLOSED • CR-D1A-SA • CR-D1B-SB
15.	Notify Manager - Operations <u>OR</u> Designee Of Conditions	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16.	Implement The EALs	
17.	Contact RC Personnel To Determine When Radiological Conditions Permit Entry Into The Affected Area	
*18.	Check Entry Into Affected Area - PERMITTED BY RC PERSONNEL	<u>WHEN</u> conditions permit entry into the affected area, <u>THEN</u> Go To Step 19.
19.	Notify Reactor Engineering To Make An Entry Into The Affected Area To Provide Damage Assessment And Recovery Recommendations	
20.	Check Accident Location - INSIDE CV	<u>IF</u> the accident location is in the Spent Fuel Pit, <u>THEN</u> contact Chemistry to perform the following: a. Take samples of the Spent Fuel Pit for isotopic analysis. b. Notify the SM of the results of the analysis. c. Return to procedure and step in effect. <u>IF</u> the accident location is in the New Fuel Storage Area, <u>THEN</u> return to procedure and step in effect.
21.	Contact Chemistry To Perform the Following: a. Take samples of the Refueling Cavity for isotopic analysis b. Notify the SM of the results of the analysis	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
22.	Return To Procedure And Step In Effect	

- END -

73. G2.3.4 001

Given the following:

- A Progress Energy Mechanic has been assigned to repack a valve.
- Projected dose rate in the area is 500 mR/hr.
- The Mechanic has a current yearly dose of 0.5 Rem.

What is the MAXIMUM time that the mechanic can work on the valve before reaching the Progress Energy Annual Administrative Dose Limit?

A. 1 hour

B.✓ 3 hours

C. 4 hours

D. 7 hours

The correct answer is B.

A. Incorrect. In one hour the worker would reach 1 Rem annual exposure. The Progress Energy limit is 2 Rem.

B. Correct.

C. Incorrect. In 4 hours the worker would reach 2 Rem exposure for the job. The candidate may forget that worker already had 0.5 Rem and think that the worker could stay in the Rad Area until he received the annual limit of 2 Rem.

D. Incorrect. In 7 hours the worker would reach 4 Rem annual exposure. Workers are allowed a total of 4 Rem if they have dose from non-Progress Energy sites. The limit in this case would be 2 Rem at non-Progress Energy sites plus 2 Rem at Progress Energy sites. This equates to a total of 4 Rem. The stem of the question specifies that the worker is a Progress Energy Mechanic.

Question 73
Tier 3
K/A Importance Rating - RO 3.2 SRO 3.7

Knowledge of radiation exposure limits under normal or emergency conditions.

Reference(s) - Sim/Plant design, DOS-NGGC-0004,Administrative Dose Limits
Proposed References to be provided to applicants during examination - None
Learning Objective - Rad. Worker Training.
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.12 / 43.4 / 45.10
Comments - K/A is met because the candidate know the Progress Energy Annual Administrative Dose Limits.

74. G2.4.22 001

The plant was at 100% RTP when a Small Break LOCA occurred. After verifying immediate actions of PATH-1 a loss of offsite power occurred due to an internal failure in the Startup Transformer.

- Due to several equipment malfunctions, the Critical Safety Function Status Trees indicate as follows:

- Subcriticality - Yellow
- Core Cooling - Orange
- Heat Sink - Red
- Integrity - Green
- Containment - Green
- Inventory - White

Which ONE (1) of the following identifies the proper procedure that will be entered based on priority and the purpose of the procedure? (Assume that monitoring of CSFSTs is in effect.)

A. FRP-S.2, Response to Loss of Core Shutdown, to prevent fuel damage from excessive heat generation.

B. FRP-C.2, Response to Degraded Core Cooling, to reduce RCS pressure and minimize subcooling.

C.✓ FRP-H.1, Response to Loss of Secondary Heat Sink, to respond to a loss of secondary heat sink in all Steam Generators.

D. FRP-C.1, Response to Inadequate Core Cooling, is used to reduce RCS temperature and ensure that RCS inventory is maintained.

Correct answer is C.

A. Incorrect. FRP is Yellow, and therefore not required to be performed. This is the highest priority of critical safety functions and could be selected based on that knowledge.

B. Incorrect. FRP required to be performed, a red terminus on Heat Sink is a higher priority and must be addressed first.

C. Correct.

D. Incorrect. FRP-C.1 is a red terminus on Core Cooling. The stem of the question stated that core cooling was Orange.

Question 74
Tier 3
K/A Importance Rating - RO 3.6 SRO 4.4

Knowledge of the bases for prioritizing safety functions during abnormal / emergency operations.

Reference(s) - Sim/Plant design, OMM-022, EPP-1, CSFST
Proposed References to be provided to applicants during examination - None
Learning Objective - OMM-022-003
Question Source - RNP Bank
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 43.5 / 45.12
Comments - K/A is met because the candidate must know the priorities for the Functional Restoration Procedures.

Progress Energy

I
Information Use

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3
PART 1

OPERATIONS MANAGEMENT MANUAL

OMM-022

EMERGENCY OPERATING PROCEDURES USER'S GUIDE

REVISION 33

8.2.5 (Continued)

19. Supplement Q – This supplement provides the information to determine if SW-969 closure is warranted to isolate Service Water to the Turbine Building and provides the instructions for closing the valve.

8.2.6 Use of CSFSTs

1. The monitoring of the CSFSTs is initiated when the operator:

   - is directed by an action step in Path-1, or
   - transitions to an EPP or FRP from the Path.

   Monitoring continues until the EOP Network directs that it can be terminated.

2. The step to begin monitoring of the CSFSTs is not provided in FRP-S.1 until the end of the FRP since no higher priority FRP exists. In other procedures transitioned to or from the Path, the transition is either preceded by an action step that states to reset SPDS and initiate monitoring of the CSFSTs, or this guidance is provided early in the subject EOP.

3. Monitoring of the Critical Safety Function Status Trees takes place in accordance with its own rules of usage, in parallel with the recovery actions being performed by the Operator. The monitoring may be done directly by one of the Operators in the control room or by some other member of the shift assigned to the control room (normally the STA).

4. Monitoring is accomplished by use of the Safety Parameter Display System (SPDS) feature of the ERFIS computer or by use of the manual procedure. The CRS is the designated primary SPDS user while the Shift Technical Advisor is available to assist the CRS as the secondary SPDS user.

5. Status Trees ask a series of questions about plant conditions, and in general, each question asked depends on the answer to the previous question. This dependency results in a branching pattern, which is referred to as a "tree."

8.2.6 (Continued)

6. There are six different trees, each one evaluating a separate safety aspect (Critical Safety Function) of the plant. At any given time, a Critical Safety Function status is represented by a single Path through its tree. Since each Path is unique, it is uniquely labeled at its end point, or terminus. This labeling consists of color-coding of the terminus and possible transition to an appropriate FRP, if required by that safety status. If the status is normal for a particular Critical Safety Function, no transition is specified, and the condition is clarified by the words CSF-SAT.

NOTE: WHITE is **NOT** a color coding for a terminus. This is a function of ERFIS only to indicate that ERFIS does not have sufficient input to make the determination of the proper terminus.

7. Color-coding can be either RED, ORANGE, YELLOW, or GREEN, with GREEN representing a "SAT" safety status. Each non-green color represents an action level that should be addressed according to the Rules of Priority for Status Tree Use.

8. Several special conditions also affect the CSFSTs indicated by ERFIS:

   - All CSFSTs are forced to a GREEN-condition when the plant mode is Cold Shutdown.
   - The Heat Sink Tree is forced to a GREEN-condition when the plant is less than 350°F.
   - The Subcriticality Tree is forced to a GREEN-condition when the plant mode is Power Operation or Hot Shutdown except:
     - When in the Power Operation mode, the actual Critical Safety Function Status will be displayed if the Reactor Trip and Bypass Breakers are open.
     - When in any mode, the actual Critical Safety Function Status will be displayed if a trip condition exists (as determined from Reactor Protection System inputs).

8.2.6 (Continued)

9. The six Status Trees are always evaluated in the following sequence (order of priority):

   1) Subcriticality (S) → ORANGE

   2) Core Cooling (C) → ORANGE

   3) Heat Sink (H) → RED

   4) Integrity (P)

   5) Containment (J)

   6) Inventory (I)

10. If identical color priorities are found on different trees during monitoring, the required action priority is determined by the above sequence. For example, a RED-condition on the Subcriticality Tree takes priority over a RED-condition on Core Cooling Tree.

11. The user begins monitoring with the Subcriticality Tree. Questions are answered based on plant conditions at the time, and the appropriate branch line followed to the next question. An individual Status Tree evaluation is complete when the user arrives at a color-coded terminus. With the exceptions noted below, the color and instructions of the terminus are noted and the user continues to the next tree in sequence.

    a. If any RED terminus is encountered, the operator is required to immediately stop any Path or EPP in progress, and to perform the Function Restoration Procedure (FRP) required by the terminus.

    b. If, during the performance of any RED-condition FRP, a RED-condition of higher priority arises, then the higher priority condition should be addressed first, and the lower priority RED-condition FRP suspended.

    c. If any ORANGE terminus is encountered, the operator is expected to monitor all of the remaining trees and if no RED-condition is encountered, suspend any Path or EPP in progress and perform the FRP required by the ORANGE terminus.

8.2.6.11 (Continued)

d. If during the performance of an ORANGE-condition FRP, any RED-condition or higher priority ORANGE-condition arises, then the RED or higher priority ORANGE-condition is to be addressed first, and the original ORANGE-condition FRP suspended.

e. An exception to item 8.2.6.11.d above is that if an ORANGE condition goes to RED **AND** the same FRP is applicable, then progress in the FRP should continue without starting over.

12. Once an FRP is entered due to a RED or ORANGE-condition, that FRP is performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the FRP will clear the RED or ORANGE-condition before all the Operator actions are complete. However, the FRPs should be performed to the point of the defined transition to a specific EOP or to the "procedure and step in effect."

13. When monitoring CSFSTs using the manual procedure, Status Tree monitoring should be continuous if any ORANGE or RED- condition is found to exist. If no condition more serious than YELLOW is encountered, monitoring frequency may be reduced to 10-20 minutes, unless a significant change in plant status occurs.

14. A YELLOW terminus does not require immediate Operator attention. Frequently, it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the YELLOW-condition might provide an early indication of a developing RED or ORANGE-condition.

15. Following FRP implementation, a YELLOW-condition might indicate a residual off-normal condition. The Operator is allowed to decide whether or not to implement any YELLOW-condition FRP.

16. When using the SPDS to monitor the CSFSTs, the "SPDS Reset" feature must be used prior to initiating CSFST monitoring. SPDS software "locks in" the highest priority condition occurring during the transient, regardless of whether or not the condition is still present.

8.2.6 (Continued)

17. The only requirement of the monitoring function is that the CRS in charge of recovery actions be immediately informed of RED or ORANGE-conditions, and regularly advised of YELLOW or GREEN-conditions.

18. The Path or EPP actions in progress are suspended if either a RED or ORANGE-condition is detected on a Status Tree. Path or EPP actions are not to be performed while a Critical Safety Function is being restored from a RED or ORANGE-condition, unless required by the FRP in effect. Conversely, in a few cases, the FRPs are not performed while certain EPPs are in effect. These cases will be explicitly noted in the EPP.

19. After restoration of any Critical Safety Function from a RED or ORANGE-condition, recovery actions may continue when the FRP is complete. Most often, the FRP will return the Operator to the suspended Path or EPP. At times, an FRP will require a transition to a different Path or EPP because of conditions created within the FRP.

20. Upon continuation of recovery actions, some judgement is required by the Operator to avoid inadvertent reinstatement of a RED or ORANGE- condition by undoing some critical step in a Function Restoration Procedure. The plant recovery procedures are optimal in assuming that equipment is available when required. The appearance of a RED or ORANGE-condition in most cases implies that some equipment or function required for safety is not available, and some adjustment may be required in the recovery procedures.

21. An example might be the establishing of an alternate feed Path to the Steam Generators as required by FRP-H.1. With feed flow from either the Main Feedwater or Condensate system, the Operator would **NOT** want to isolate the Main Feedwater line if required in a subsequent procedure.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3
PART 4

Emergency Operating Procedure

CSFST

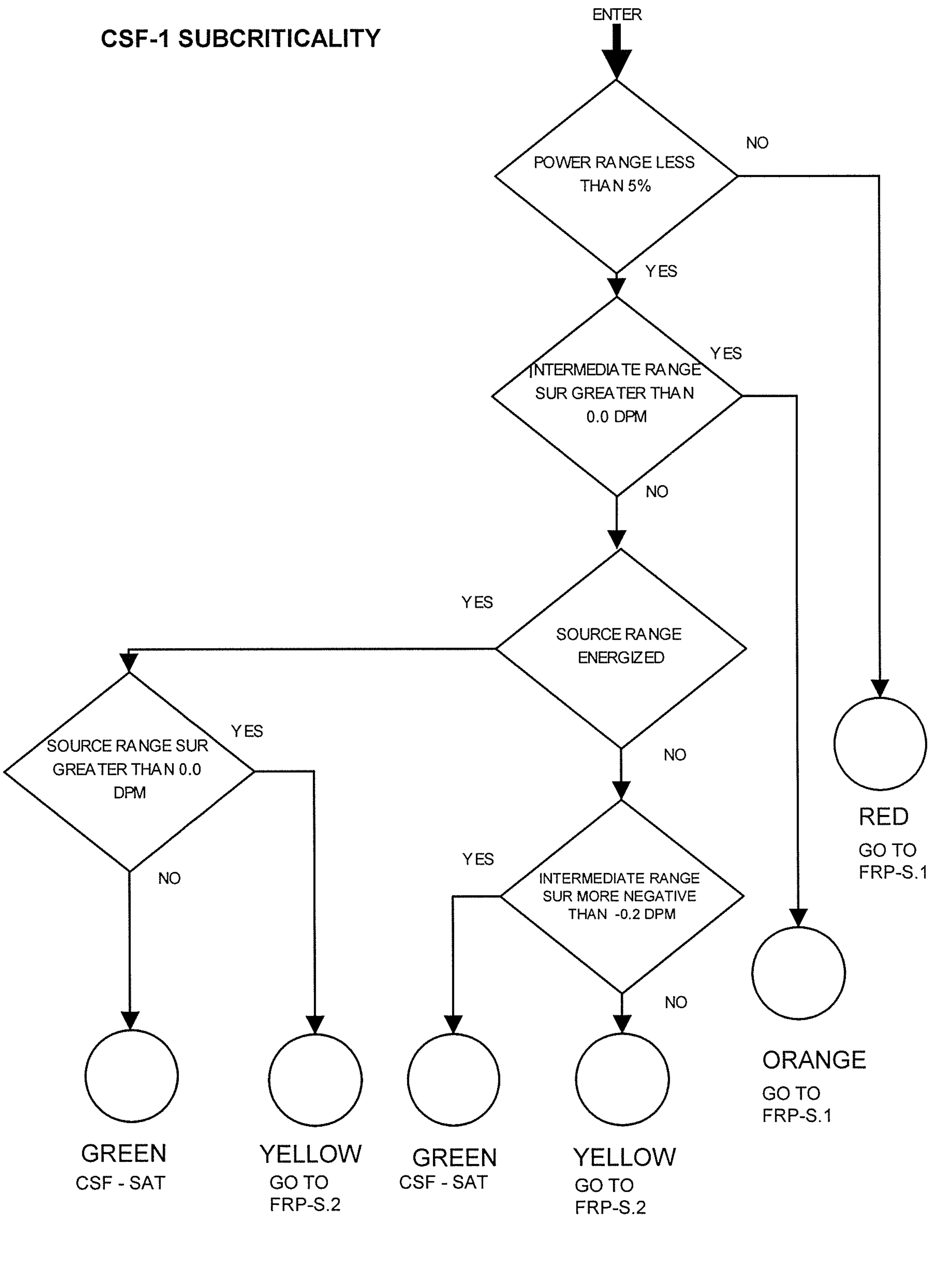
CRITICAL SAFETY FUNCTION STATUS TREES

REVISION 4

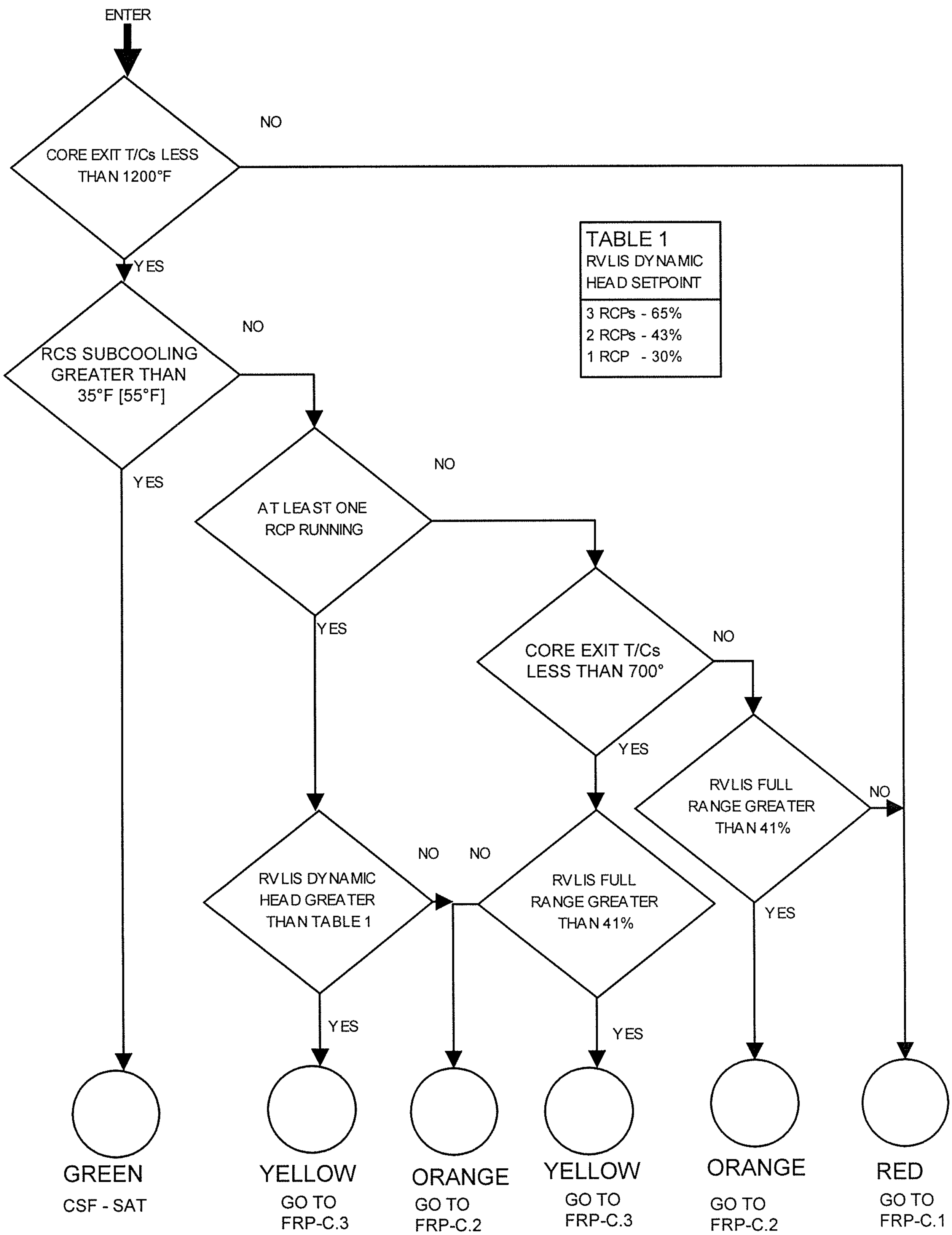
CSFST, Revision 4
PRR 273208
Summary Of Changes

CSF-5: Changed CV High Radiation value to 5 R/HR to incorporate EC 69380.

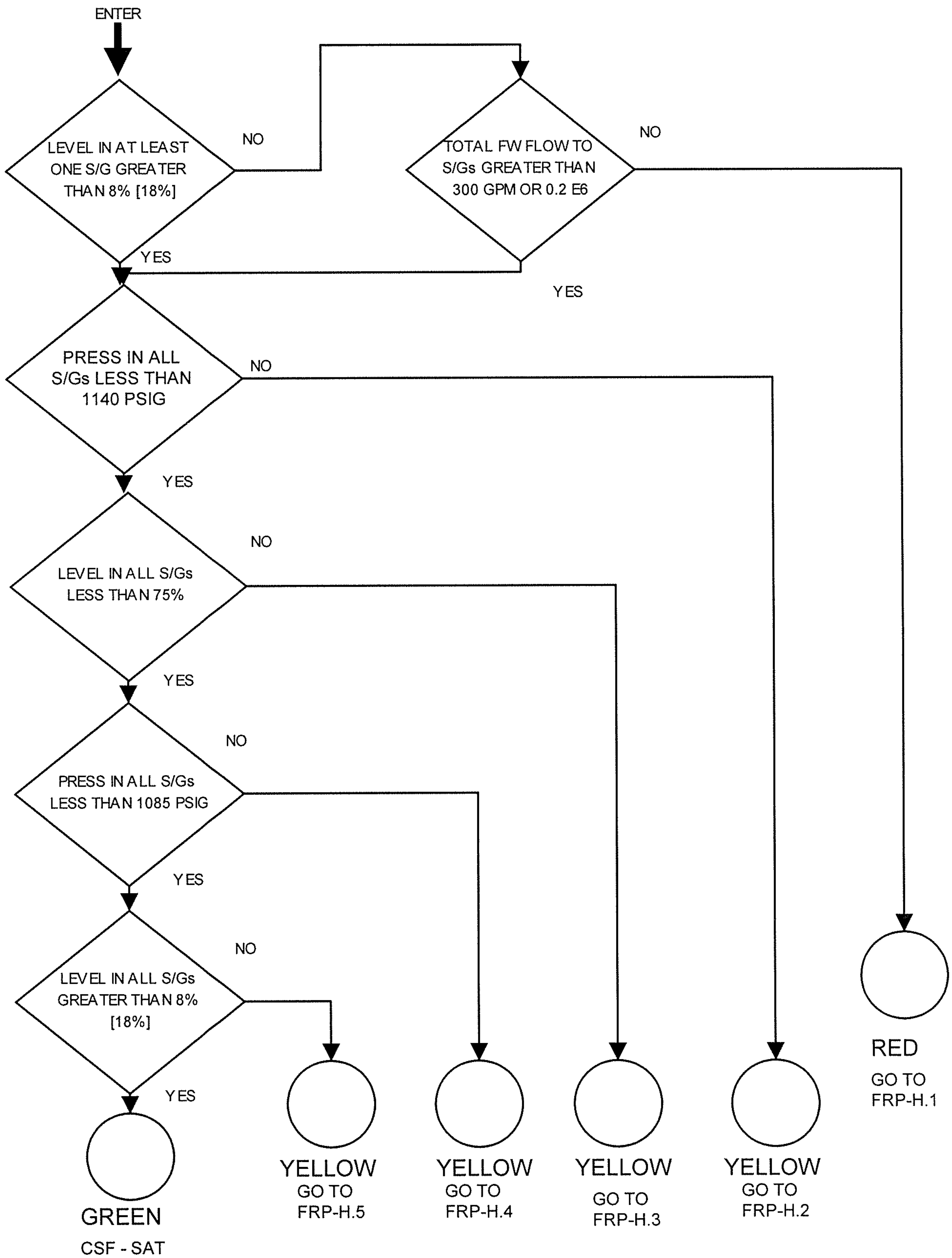
CSF-1 SUBCRITICALITY



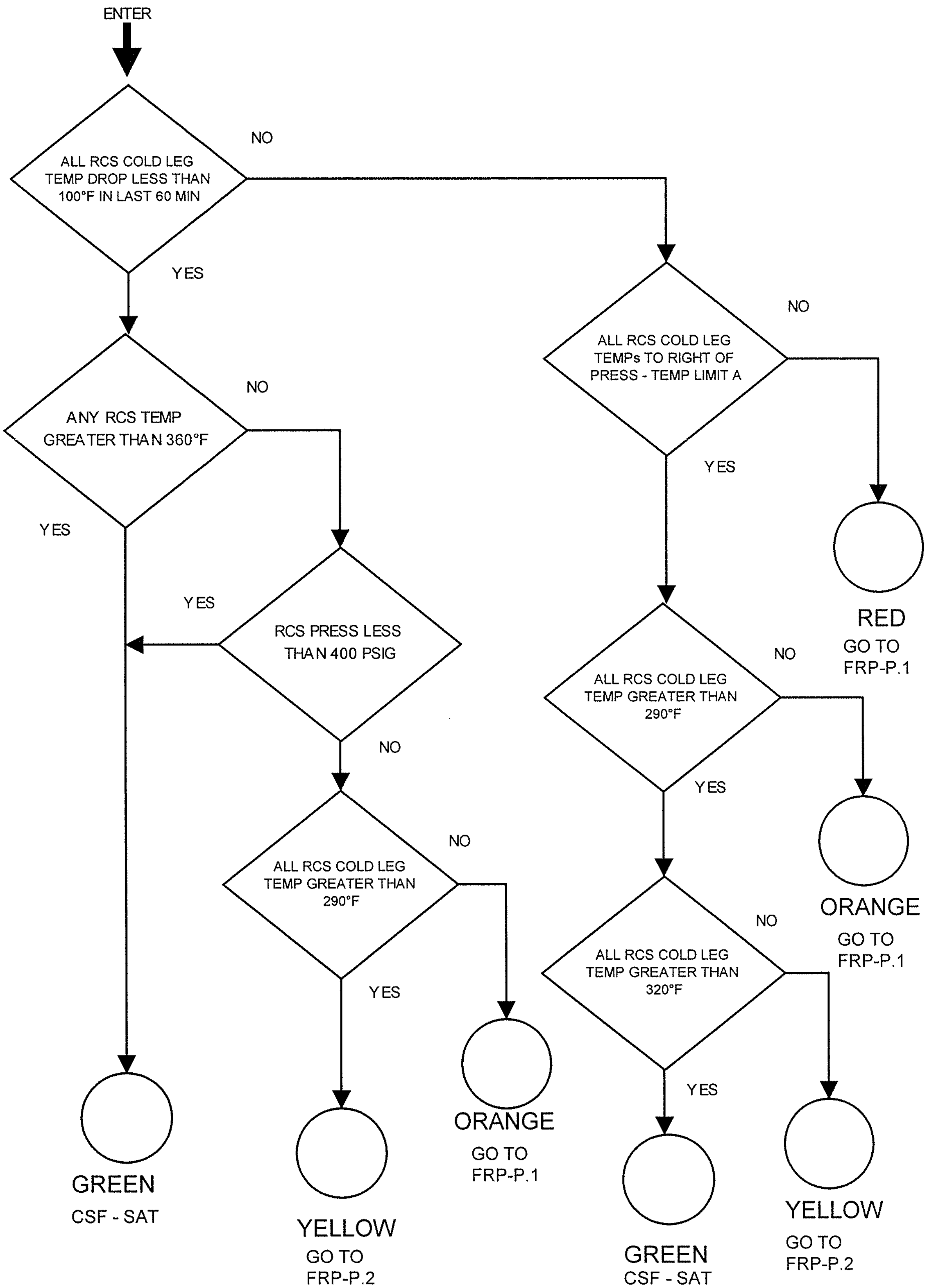
CSF-2, CORE COOLING



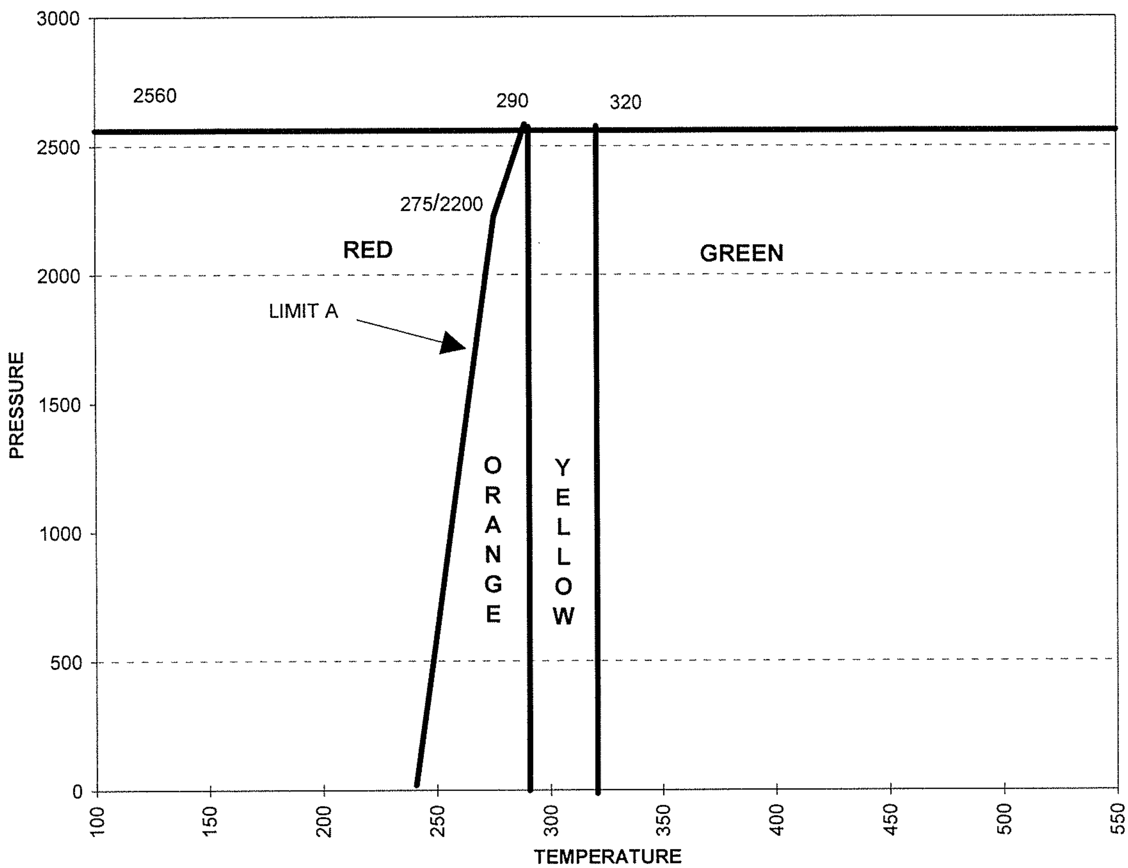
CSF-3, HEAT SINK



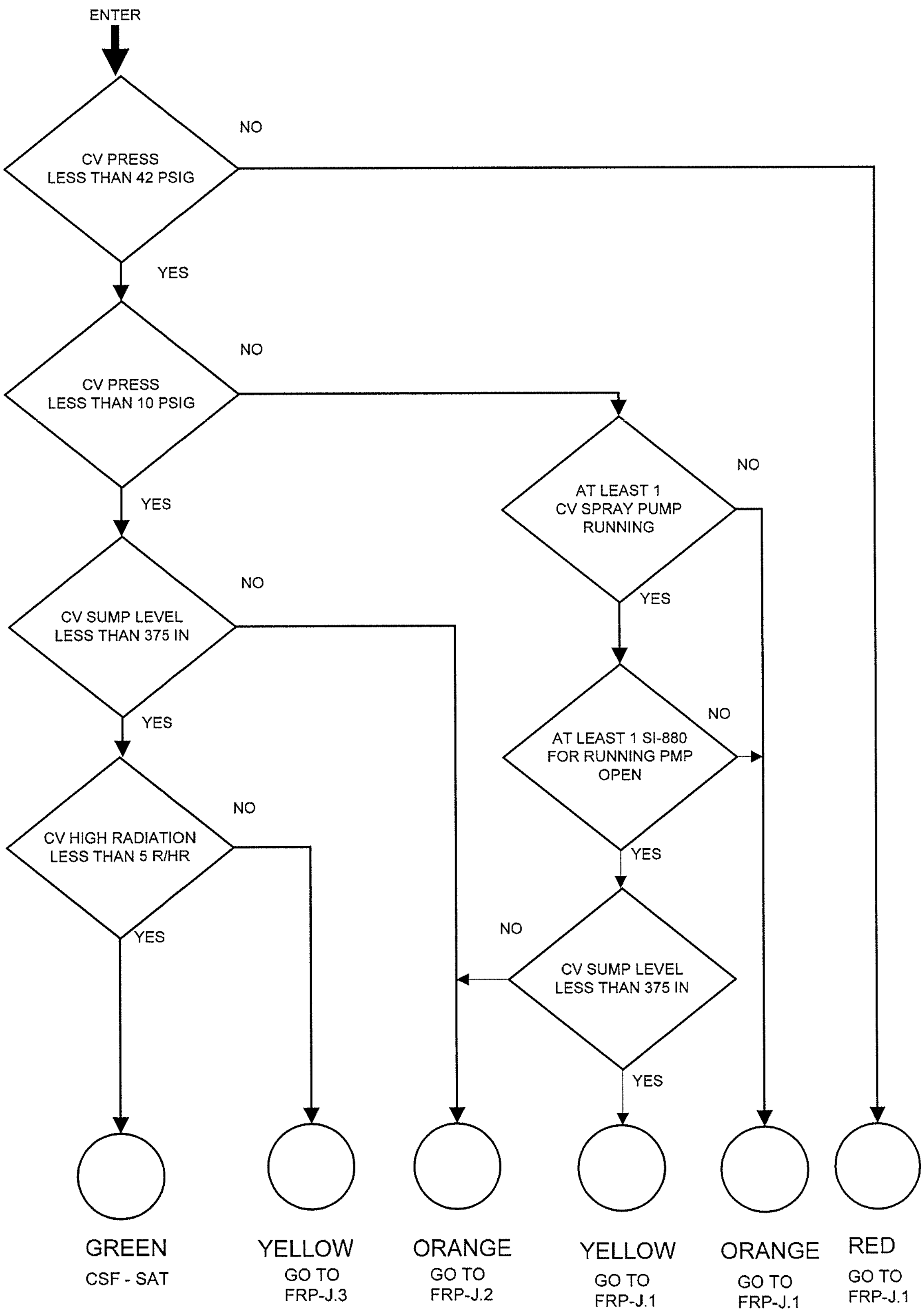
CSF-4, RCS INTEGRITY



INTEGRITY CSF-4a

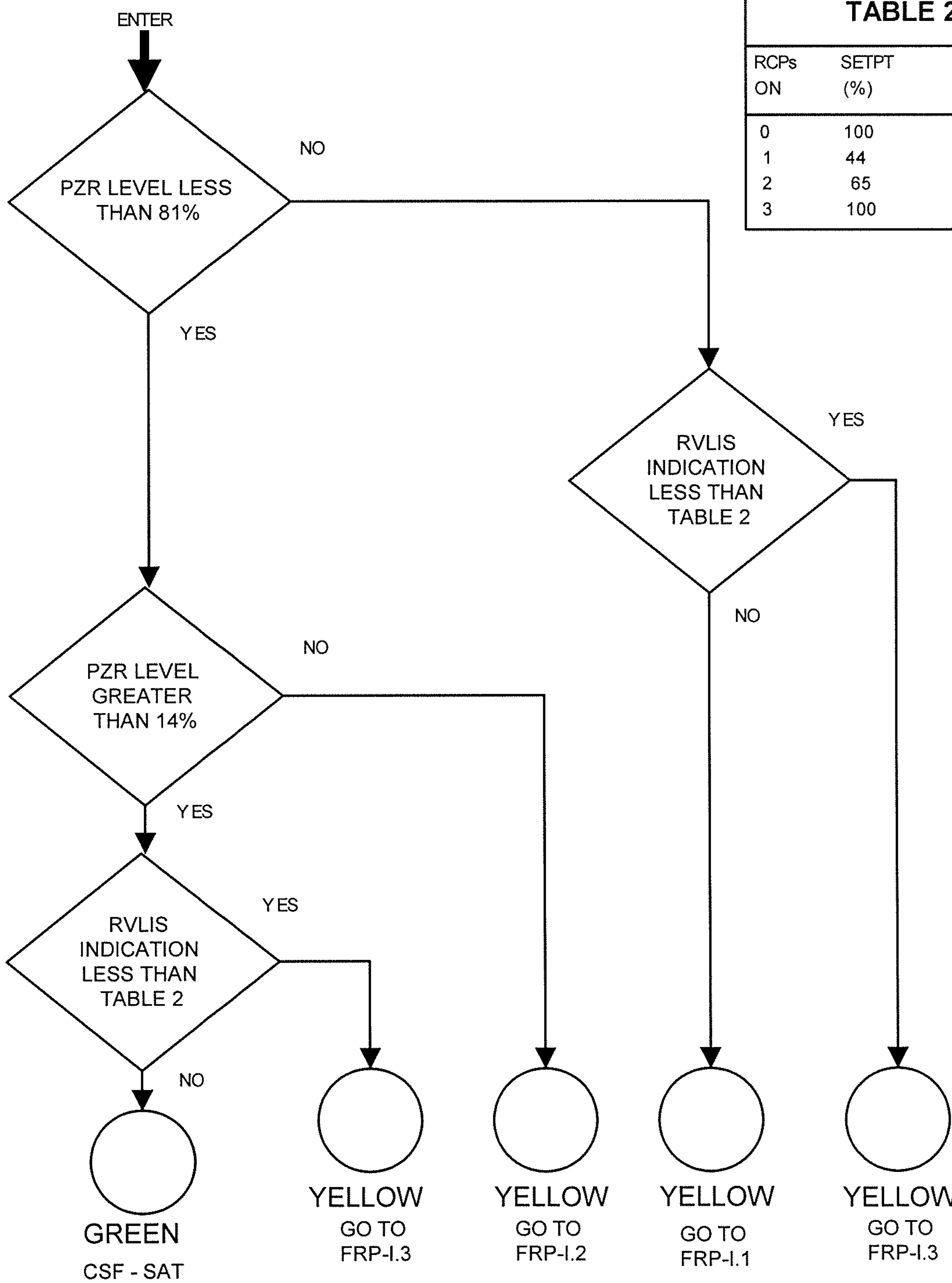


CSF-5, CONTAINMENT



CSF-6, RCS INVENTORY

TABLE 2		
RCPs ON	SETPT (%)	RVLIS INDICATION
0	100	UPPER
1	44	DYNAMIC
2	65	DYNAMIC
3	100	DYNAMIC



Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to respond to a loss of secondary heat sink in all Steam Generators.

2. ENTRY CONDITIONS

   a. PATH-1, when minimum AFW flow is not verified AND narrow range level in all S/Gs is less than 8% [18%].

   b. CSF-3, Heat Sink Critical Safety Function Status Tree on a RED condition.

- END -

DISCUSSION (From the WOG FR-C.1 Basis Document)

1. INTRODUCTION

Function Restoration Guideline, FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, has been developed to address the symptoms for inadequate core cooling. These symptoms are: 1) core exit TC temperatures greater than 1200°F, or 2) core exit TC temperatures greater than 670°Fplus adverse containment errors or 700°F, whichever is greater, and a RVLIS full range indication less than 3-1/2 feet above the bottom of the active fuel. The basis for these symptoms can be found in the Critical Safety Function Status Tree background document, F-0.2, CORE COOLING.

The guideline is entered from F-0.2 on either of two RED priorities. The major actions to be performed in this guideline include:

1) Reinitiation of high pressure safety injection
2) Rapid secondary depressurization
3) RCP restart and/or opening PRZR PORVs

These actions are to be performed sequentially. Success, as indicated by improved core cooling and increasing vessel inventory, is evaluated prior to performing the next action in the sequence. If the operator is successful in restoring adequate core cooling via high pressure safety injection, then he is returned to the guideline and step in effect. If not, he continues with this guideline. Once core cooling has been restored and adequate makeup flow has been established and verified, the operator is directed to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 16.

2. DESCRIPTION

The indication of inadequate core cooling requires prompt operator action. Inadequate core cooling is caused by a substantial loss of primary coolant resulting in a partially or fully uncovered core. Without adequate heat removal, the core decay energy will cause the fuel temperatures to increase. Severe fuel damage will occur unless core cooling is promptly restored.

Reinitiation of high pressure safety injection is the most effective method to recover the core and restore adequate core cooling. If some form of high pressure injection cannot be established or is ineffective in restoring adequate core cooling, then the operator must take actions to reduce the RCS pressure in order for the SI accumulators and low-head SI pumps to inject. Analyses have shown that a rapid secondary depressurization is the most effective means for achieving this. If secondary depressurization is not possible, or primary-to-secondary heat transfer is significantly degraded due to a loss of secondary heat sink, for example, then the operator must start the RCPs. The RCPs will provide forced two phase flow through the core and temporarily improve core cooling until some form of make-up flow to the RCS can be established.

The recovery techniques applied in this guideline were developed from transient analyses presented in the inadequate core cooling WCAPs, References 1 and 2. The expected system response to each of the recovery techniques listed above will be described below. A complete description of the computer code and modeling assumptions used in the analyses can be found in the referenced WCAPs.

2.1 Reinitiation of High Pressure Safety Injection

The introduction of subcooled safety injection into the highly voided RCS will cause steam in the cold legs to condense. Steam flow throughout the RCS will increase as a result of this condensation effect. Superheated steam forced out of the core may initially cause the core exit TC temperatures to increase. As the vessel begins to refill, heat transfer from the fuel will cause the fluid entering the core to boil vigorously. This will create a frothy two phase mixture which will eventually recover the entire core and cause the core exit TC temperatures to quickly decrease to the saturation temperature of the RCS. The RVLIS indication, which is an indirect measure of vessel level based on local pressure differences within the vessel, may fluctuate as the core recovers; however, the general trend in RVLIS indication should increase as the vessel is refilled.

DISCUSSION (From the WOG FR-C.2 Basis Document)

1. INTRODUCTION

Function Restoration Guideline FR-C.2, RESPONSE TO DEGRADED CORE COOLING, has been developed to address the symptoms for degraded core cooling. These symptoms are:

1) core exit TC temperatures greater than 670°F plus adverse containment errors or 700°F , whichever is greater, and a RVLIS full range indication greater than 3-1/2 feet above the bottom of the active fuel, or
2) core exit TC temperatures less than 670°F plus adverse containment errors or 700°F , whichever is greater, and a RVLIS full range indication less than 3-1/2 feet above the bottom of the active fuel, or
3) a RVLIS dynamic head indication corresponding to a 50% vessel void fraction, depending upon the number of RCPs running.

The basis for these symptoms can be found in the Critical Safety Function Status Tree background document F-0.2, CORE COOLING.

This guideline is entered from F-0.2, on an ORANGE priority. The major actions to be performed in this guideline include:

1) Reinitiation of high pressure safety injection
2) Controlled secondary depressurization

These actions are to be performed sequentially. Success, as indicated by improved core cooling and increasing vessel inventory is evaluated prior to performing the next action in the sequence. If, for example, the operator is successful in restoring core cooling via high pressure safety injection, then he is returned to the guideline and step in effect. If not, he continues with this guideline. Once core cooling has been restored and adequate makeup flow has been established and verified, the operator is directed to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 16.

2. DESCRIPTION

Degraded core cooling is caused by a substantial loss of primary coolant. The RCPs may or may not be running when degraded core cooling is indicated. If the RCPs are not running, the degraded core cooling symptoms indicate the core is partially uncovered. If the RCPs are running, the symptoms indicate that the potential for core uncovery exists should the RCPs fail or be manually tripped. Operator action is required to restore RCS inventory in either case.

Reinitiation of high pressure safety injection is the most effective method to restore RCS inventory and core cooling. If some form of high pressure injection cannot be established or is ineffective in restoring core cooling, then the operator must take actions to reduce the RCS pressure in order for the SI accumulators and low-head SI pumps to inject. A controlled secondary depressurization is an effective method for achieving this, while at the same time avoiding a rapid RCS cooldown that could cause problems with pressurized thermal shock.

The expected system response to both of the recovery techniques listed above, will be described below.

DISCUSSION (From the WOG FR-H.1 Basis Document)

1. INTRODUCTION

Guideline FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, provides guidance to address an extreme challenge (i.e., RED priority) to the Heat Sink Critical Safety Function that results if total feed flow is below a minimum value and level is below the narrow range in all SGs at any time. Less severe challenges to the Heat Sink Critical Safety Function that result from secondary inventory concerns on individual steam generators are discussed in the background documents for guidelines FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, and FR-H.5, RESPONSE TO STEAM GENERATOR LOW LEVEL.

An early indication that secondary heat transfer capability may be challenged is that auxiliary feedwater (AFW) flow is not available to any steam generator. Following a reactor trip and safety injection (SI), main feedwater isolation is automatically initiated. Auxiliary feedwater flow to the steam generators must be automatically or manually initiated in order to maintain adequate secondary inventory for decay heat removal. Consequently, a failure of the AFW system results in a challenge to the Heat Sink Critical Safety Function. If reactor trip and SI occur, the operation of the AFW system is verified in guideline E-0, REACTOR TRIP OR SAFETY INJECTION, prior to Status Tree monitoring. If minimum AFW flow is not being provided, the operator is directed to implement guideline FR-H.1.

The objective of guideline FR-H.1 is to maintain reactor coolant system (RCS) heat removal capability by establishing feed flow to an SG or through establishing RCS bleed and feed heat removal. Guideline FR-H.1 is entered at the first indication that secondary heat removal capability may be challenged.

This permits maximum time for operator action to restore feedwater flow to at least one steam generator before secondary inventory is depleted and secondary heat removal capability is lost. Once secondary heat removal capability is lost, RCS bleed and feed must be established to minimize core uncovery and prevent an inadequate core cooling condition.

Guideline FR-H.1 may be exited at several locations depending on the status of secondary heat sink and whether RCS bleed and feed heat removal has been initiated. In general, the operator is directed to the guideline in effect if feedwater and secondary heat sink is restored or RHR System operation is established before RCS bleed and feed heat removal is initiated. If RCS bleed and feed heat removal is initiated, the operator is directed to Step 7 of guideline ES-1.1, SI TERMINATION, following restoration of secondary heat sink and termination of RCS bleed and feed heat removal. In addition, there are two transitions to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, depending upon RCS pressure and the status of the PRZR PORVs.

2. DESCRIPTION

A loss of secondary heat sink can occur as a result of several different initiating events. Possibilities are a loss of main feedwater during power operation, a loss of offsite power, or any other scenario for which main feedwater is isolated or lost when the steam generators provide the main heat removal path. For these initiating transients a failure of the auxiliary feedwater (AFW) system to inject or a loss of AFW early in the cooldown, before RHR System operation can be established, could lead to a loss of secondary heat sink.

A loss of all feedwater transient is characterized by a depletion of secondary inventory and eventual degradation of secondary heat transfer capability. As secondary heat transfer capability degrades, a loss of secondary heat sink results and core decay heat generation will increase RCS temperature and pressure until the pressurizer power operated relief valves (PORV) or pressurizer safety valves open to relieve the increasing RCS pressure. At this point the opening and closing of the PORVs or safety valves will result in a loss of RCS inventory similar in nature to a small break loss of coolant accident. If operator action is not taken, the pressurizer PORVs or safety valves will continue to cycle open and closed at the valve setpoint pressure removing RCS inventory and a limited amount of core decay heat until eventually enough inventory will be lost to result in core uncovery.

75. G2.4.6 001

Which ONE(1) of the following identifies the mitigation strategy of EPP-12, "Post-SGTR Cooldown Using Backfill"?

EPP-12 cools down and depressurizes the plant to Cold Shutdown conditions and........

A. cools the ruptured S/G by filling to 91% WR level.

B. cools the ruptured S/G by steaming with the steam line PORV.

C. depressurizes the ruptured S/G by draining the ruptured S/G through S/G blowdown.

D✓ depressurizes the ruptured S/G by draining the ruptured S/G through the ruptured tube(s).

Correct answer is D.

A. Incorrect. S/Gs are filled to 91% WR prior to reaching 350°F to ensure the S/Gs are filled prior to the SDAFW pump losing motive force while cooling down IAW DSP-007, Cold Shutdown Using the Dedicated / Alternate Shutdown System.

B. Incorrect. This is the purpose of EPP-14, Post-SGTR cooldown using steam dump.

C. Incorrect. This is the purpose of EPP-13, Post-SGTR cooldown using blowdown.

D. Correct.

Question 75
Tier 3
K/A Importance Rating - RO 3.7 SRO 4.7

Knowledge of EOP mitigation strategies.

Reference(s) - Sim/Plant design, EPP-12, -13, -14, DSP-007
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-12-001
Question Source - NEW
Question History - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13
Comments - K/A is met because the candidate must know the mitigation strategy for Post-SGTR cooldown using backfill.

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to cooldown and depressurize the plant to Cold Shutdown conditions following a Steam Generator Tube Rupture. This recovery method depressurizes the ruptured S/G by draining it through the ruptured tube into the RCS.

2. ENTRY CONDITIONS

   a. Path-2

   b. EPP-13, Post-SGTR Cooldown Using Blowdown, when S/G blowdown is not available and the backfill method of cooldown is selected.

- END -

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to cooldown and depressurize the plant to cold shutdown conditions following a Steam Generator Tube Rupture. This recovery method depressurizes the ruptured S/Gs by draining them using S/G blowdown.

2. ENTRY CONDITIONS

   EPP-12, Post-SGTR Cooldown Using Backfill, if the backfill method of cooldown is not desired.

   EPP-14, Post-SGTR Cooldown Using Steam Dump, if the steam dump method of cooldown is not desired.

- END -

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to cooldown and depressurize the plant to cold shutdown conditions following a Steam Generator Tube Rupture.
   This recovery method depressurizes the ruptured S/Gs by dumping steam.

2. ENTRY CONDITIONS

   a. EPP-12, Post-SGTR Cooldown Using Backfill, when backfill method of cooldown is not available or desired.

   b. EPP-13, Post-SGTR Cooldown Using Blowdown, when S/G blowdown and backfill method of cooldown are not available or desired.

- END -

STEP | INSTRUCTIONS | RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 1

CSD USING THE DEDICATED SHUTDOWN BUS

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****

CAUTION

Filling the S/Gs at too great a flow rate can cause the 10°F/Hr cooldown rate to be exceeded.

****

39. Fill S/Gs To Between 85% TO 91% WR AND Maintain This Level During The Remainder Of The Cooldown

40. Verify Only ONE SI Pump Is Capable Of Injecting To The RCS

41. Maintain RCS Temperature Below 340°F By Slowly Adjusting The White Thumbwheels Of The Desired Pressure Indicating Controllers At The Secondary Control Panel

42. Verify CCW Heat Exchangers In Service As Follows:

- CC-712A, CC HX "B" INLET (CHAIN OPERATED) - OPEN
- CC-712B, CC HX "B" OUTLET (CHAIN OPERATED) - OPEN
- CC-713A, CC HX "A" INLET (CHAIN OPERATED) - OPEN
- CC-713B, CC HX "A" OUTLET (CHAIN OPERATED) - OPEN

43. Verify CCW PUMP A - RUNNING