Vogtle Initial Exam

February 2010

Draft

RO 1to 40

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HL-15R RO NRC Exam

1. 001K2.05 001/2/2/RODS-MG SETS/MEM- 2.9 / 3.2/NEW/HL-15R NRC/RO/TNT/DS

An ATWT is in progress.

- The Unit Operator has left control room to locally trip the reactor.

If unable to successfully open the reactor trip breakers, per 19211-C, "FR-S.1 Response To Nuclear Power Generation/ATWT", the UO should open___(1)___MG sets___(2)___ breakers.

A. (1) both

(2) supply

B.✓ (1) both

(2) output

C. (1) either

(2) supply

D. (1) either

(2) output

K/A

001 Control Rod Drive System

K2.05 Knowledge of bus power supplies to the following:

M/G sets. – load center

K/A MATCH ANALYSIS

The question presents a plausible scenario where an ATWT is in progress, the student must know to open both MG sets output breakers.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Both MG sets output breakers must be opened to trip the reactor as they are in parallel and one MG set can supply the rod control power cabinets. The output breakers are to be opened versus the supply breakers. Opening the output breakers immediately cuts power to the rod control power cabinets and all rods immediately drop in. Opening the supply breakers would result in rods dropping in erratically as the MG sets coast down, this could result in possible flux peaking resulting in fuel damage.

B. Correct. Both MG sets output breakers must be opened to trip the reactor as they are in parallel and one MG set can supply the rod control power cabinets. The output breakers are to be opened versus the supply breakers. Opening the output breakers immediately cuts power to the rod control power cabinets and all rods immediately drop in. Opening the supply breakers would result in rods dropping in erratically as the MG sets coast down, this could result in possible flux peaking resulting in fuel damage.

C. Incorrect. Both MG sets output breakers must be opened to trip the reactor as they are in parallel and one MG set can supply the rod control power cabinets. The output breakers are to be opened versus the supply breakers. Opening the output breakers immediately cuts power to the rod control power cabinets and all rods immediately drop in. Opening the supply breakers would result in rods dropping in erratically as the MG sets coast down, this could result in possible flux peaking resulting in fuel damage.

D. Inorrect. Both MG sets output breakers must be opened to trip the reactor as they are in parallel and one MG set can supply the rod control power cabinets. The output breakers are to be opened versus the supply breakers. Opening the output breakers immediately cuts power to the rod control power cabinets and all rods immediately drop in. Opening the supply breakers would result in rods dropping in erratically as the MG sets coast down, this could result in possible flux peaking resulting in fuel damage.

REFERENCES

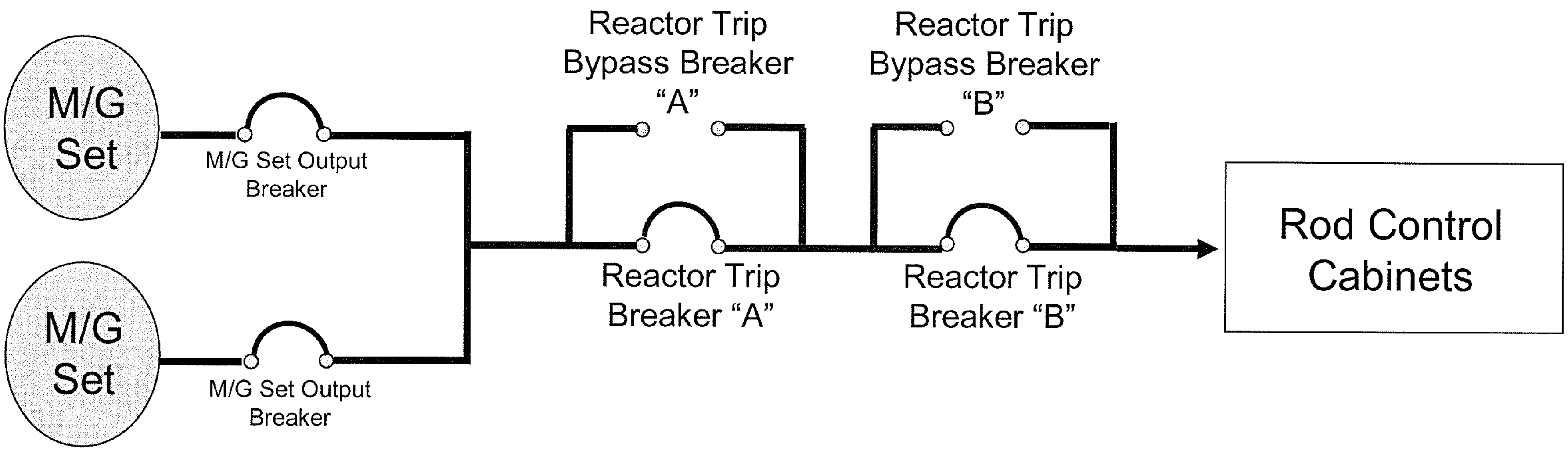
19211-C, FR-S.1 Response to Nuclear Power Generation/ATWT

V-LO-PP-28101, Solid State Protection System, slide # 67.

VEGP learning objectives:

LO-PP-27101-02, State the power supplies for the Rod Control System.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***7. Check for SI:**	
__a. SI signal - EXISTS OR ACTUATED.	__a. IF an SI signal is actuated during this procedure, THEN initiate ATTACHMENT A. __ Go to Step 8
b. Initiate ATTACHMENT A.	
8. Check the following trips have occurred:	
__a. Reactor trip.	__a. Locally trip the Reactor trip and Bypass breakers. __ IF the trip breakers will NOT open, THEN trip the Control Rod Drive MG Set output breakers at the Reactor Trip Switchgear.
__b. Turbine trip.	__b. Dispatch operator to trip turbine at the HP Turbine front standard.
***9. Check Reactor power:**	
__a. LESS THAN 5%.	__a. Go to Step 10.
__b. IR SUR - LESS THAN 0 DPM.	__b. Go to Step 10.
__c. Go to Step 24.	



Either MG set operating with output breaker closed maintains power to the trip breakers & power cabinets.
Both output breakers required to open to immediately cut power & allow Rods to drop.

IF MG supply opened, MG sets spin down allowing rods to spuriously drop as coil voltage decays.
(V.C. Summer Red Rain event)

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2. 002A2.03 001/2/2/RCS-LOSS CIRCULATION/C/A - 4.1 / 4.3/NEW/HL-15R NRC/RO/DS/TNT

Initial conditions:

The unit was operating at 100% power
The reactor was manually tripped
2NAA and 2NAB de-energize on the reactor trip
The crew implements 19001-C, "ES-0.1 Reactor Trip Recovery"

Current conditions:

2NAB has been re-energized

RCS WR Tcold - 558 F and rising
AFW flow - 150 GPM per SG
SG NR levels - 18% and slowly rising

The UO and OATC should raise...

A. AFW flow to maintain RCS WR cold leg temperatures at 557 F while attempting to start RCP 4.

B. AFW flow to maintain RCS WR cold leg temperatures at 557 F while attempting to start RCP 1.

C.✓ the steaming rate with SG ARVs to maintain RCS WR cold leg temperatures at 557 F while attempting to start RCP 4.

D. the steaming rate with Steam Dumps to maintain RCS WR cold leg temperatures at 557 F while attempting to start RCP 1.

K/A

002 Reactor Coolant System (RCS)

A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

Loss of forced circulation.

K/A MATCH ANALYSIS

The question presents a natural circulation scenario where the student must use the correct actions of porcedure 19001-C and use systems knowledge to determine the impacts on the steam dump system to properly apply the actions in 19001-C.

A. Incorrect. Raising AFW flow is plausible since SG NR levels are only at 18% and RCS temperature is increasing. Raising the AFW flow will introduce more cold feed water into the SGs and temporarily lower RCS temperatures. Step 15 of 19001-C requires starting an RCP (4 or 1 is preferred) to go back to forced flow conditions if possible.

B. Incorrect. Raising AFW flow is plausible since SG NR levels are only at 18% and RCS temperature is increasing. Raising the AFW flow will introduce more cold feed water into the SGs and temporarily lower RCS temperatures. Step 15 of 19001-C requires starting an RCP (4 or 1 is preferred) to go back to forced flow conditions if possible.

C. Correct. 2NAA and 2NAB power the RCPs and the circulating water pumps. C-9 signal will not be present, preventing the use of the steam dumps. With WR Tcold > 557 and increasing the correct action per 19001-C step 4 continuous action is to increase the rate of dumping steam. This will have to be done using the SG ARVs since the steam dumps will not arm for these conditions. Step 15 of 19001-C requires starting an RCP to go back to forced flow conditions if possible.

D. Incorrect. Raising the steaming rate is the correct action per 19001-C step 4, this will be possible with the steam dumps while 2NAB re-energized. Starting RCP 1 is an incorrect action due to still being de-energized.

REFERENCES

19001-C, "ES-0.1 Reactor Trip Response" steps 4 and 15

V-LO-PP-01101 "Electrical Distribution" presentation, slide 22

L-LO-PP-21201-12 " Steam Dumps" presentation, slide 91

VEGP learning objectives:

V-LO-PP-21201-14:

Explain the operation of the steam dump system arming circuit.

V-LO-LP-37011-02:

State how the following control systems are employed to automatically stabilize the plant after a reactor trip:

a. steam dumps
b. feedwater
c. pressurizer level and pressure
d. auxiliary feedwater

V-LO-LP-37011-04:

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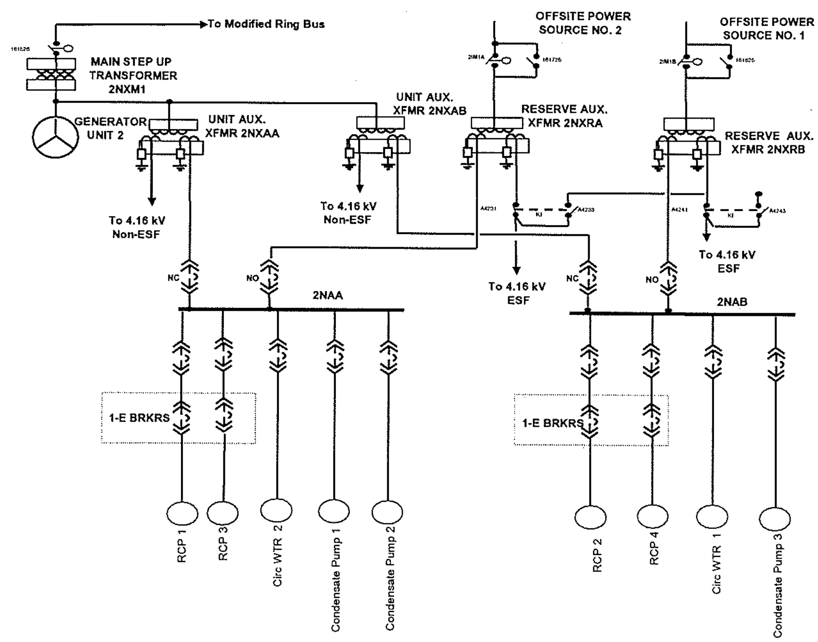
State and describe the major action categories of 19001, "Reactor Trip Recovery."

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*4. **Check RCS temperature stable at or trending to 557°F.**	*4. IF temperature is less than 557°F and lowering, THEN perform the following as necessary:
__ With RCP(s) running - RCS AVERAGE TEMPERATURE.	__a. Stop dumping steam.
-OR-	b. Perform the following as appropriate:
__ Without RCP(s) running - RCS WR COLD LEG TEMPERATURES.	__ IF at least one SG NR level greater than 10%, THEN lower total feed flow.
	-OR-
	__ IF all SG NR levels less than 10%, THEN lower total feed flow to NOT less than 570 gpm.
	__c. IF cooldown continues, THEN close MSIVs and BSIVs.
	__d. IF temperature less than 557°F and NOT trending to 557°F, THEN borate as necessary to maintain shutdown margin by initiating 13009, CVCS REACTOR MAKUP CONTROL SYSTEM.
	__e. IF temperature greater than 557°F and rising, THEN dump steam.

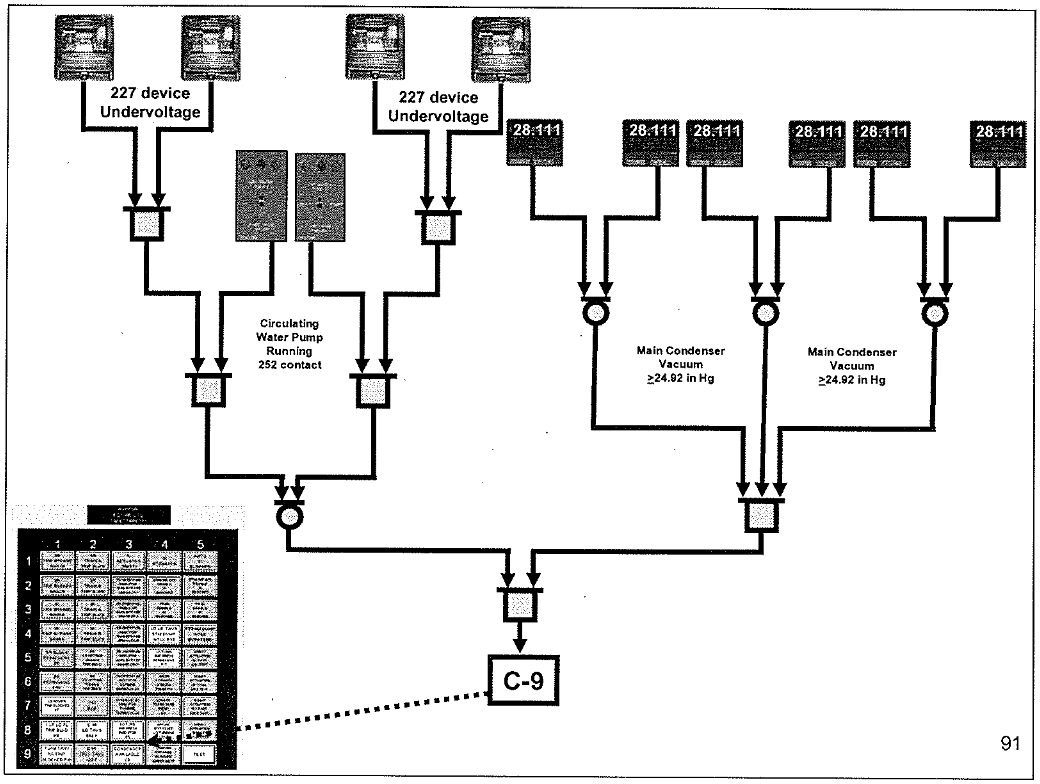
ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
e. Control Tavg: __ Manual control -OR- __ Auto control	
15. Check RCP status:	
__a. RCPs - ALL STOPPED.	__a. Go to Step 16.
__b. Start an RCP using ATTACHMENT A. (RCP 4 or RCP 1 preferred).	__b. IF an RCP can NOT be started, THEN verify natural circulation using ATTACHMENT B. __ IF natural circulation NOT established, THEN raise rate of dumping steam using Steam Dumps. __ IF Steam Dumps NOT available, THEN dump steam using SG ARVs.



V-LO-PP-01101 Ver-02 22

This drawing shows the interconnection of the UAT's and RAT's with the 13.8kV distribution system. Note to the students how the Non-1E buses have the closest breaker to the RCP's classified as 1-E. Also, the 13.8kV buses receive power from the RAT's until power (turbine power 12004-C) gets to 30%, then the buses are transferred to the UAT's.

The next few slides will deal with the RX trips associated with the 13.8kV buses. A drop in frequency would decrease RCP rotor-power output and the developed torque necessary to supply rated flow. A drop in line voltage would result in excessive reactor coolant pump motor currents which could damage the motor. Time delays on the UV and UF relays will prevent spurious signals from tripping the reactor.



OBJECTIVE LO-PP-21201-12

If no circulating water pumps are running or insufficient vacuum exists in the main condenser, dumping steam into the condenser can cause an overpressure condition which can damage the condenser. To protect against this, a permissive circuit prevents arming of the steam dumps. As can be seen in the figures below, the permissive circuit is composed of contacts which involve condenser vacuum and the circulating water pump breakers and their associated switchgear voltages (at 0 volts for 5.75 seconds). The condenser vacuum contact will be closed as long as two separate condenser pressure transmitters provide a signal indicating that at least 24.92 inches of mercury vacuum exists in the condenser. Each of the circulating water pump breaker contacts will be closed as long as the breaker for the associated pump is closed and the associated switchgear voltages are present. All of the above contacts are arranged so that the condenser must have at least 24.92 inches of mercury vacuum and at least one circulating water pump breaker closed with voltage applied. This will energize the permissive relay which closes the permissive contact in the arming circuit to permit arming of the steam dump valves. The permissive relay then energizes a **"C-9 Condenser Available"** permissive status light.

3. 003AK3.10 001/1/2/DROP ROD-RIL/PDIL/MEM - 3.2 / 4.2/NEW/HL-15R NRC/RO/TNT/DS

Given the following conditions:

- A dropped rod recovery is in progress while at power.

- 14915, "Special Condition Surveillance Logs", Data Sheet 5 for "Rod Insertion Limit Monitor Inoperable" is being performed by the OATC.

Which **ONE** of the following choices correctly lists the bank and group of the dropped rod that will render the RIL monitor inoperable during the rod recovery?

A.✓ Control Bank C, Group 1

B. Control Bank D, Group 2

C. Shutdown Bank A, Group 1

D. Shutdown Bank B, Group 2

K/A

003 Dropped Control Rod

AK3.10 Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod.

RIL and PDIL.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a dropped rod recovery is in progress. Special Condition Surveillance Date Sheet 5 for RIL Monitor inoperable is being performed. The student must recall which rods (Control Banks group 1) input into the RIL monitor via the P to A Converter.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Group 1 Control Banks input to the P to A converter which would render the RIL monitor inoperable during recovery. This requires a special condition surveillance to be performed.

B. Incorrect. Group 2 Control Banks do not input to the P to A converter. Plausible the candidate may think that Group 2 Control Banks input to the P to A converter versus the group 1 or that both groups input to the P to A converter.

C. Incorrect. Shutdown Banks Group 1 or Group 2 do not input to the P to A converter. Plausible that the student could think either group inputs to the P to A converter to

RIL.

D. Incorrect. Shutdown Banks Group 1 or Group 2 do not input to the P to A converter. Plausible that the student could think either group inputs to the P to A converter to affect RIL since all the Shutdown Banks are required to be fully withdrawn to meet RIL.

REFERENCES

V-LO-PP-27201, Digital Rod Position Indication System, slide # 8

V-LO-PP-27101, Rod Control System, slide # 93, 100, 101, 102, and 103.

18003, Rod Control Malfunction, section A for Dropped Rods in Mode 1, page # 8, step # A18 and # A19.

14915-1, Special Conditions Surveillance Logs pages # 5 and # 15.

V-LO-TX-27101, Rod Control System, page # 44.

VEGP learning objectives:

LO-PP-27101-15, State the inputs to the P to A Converter to inlude:

a. Inputs

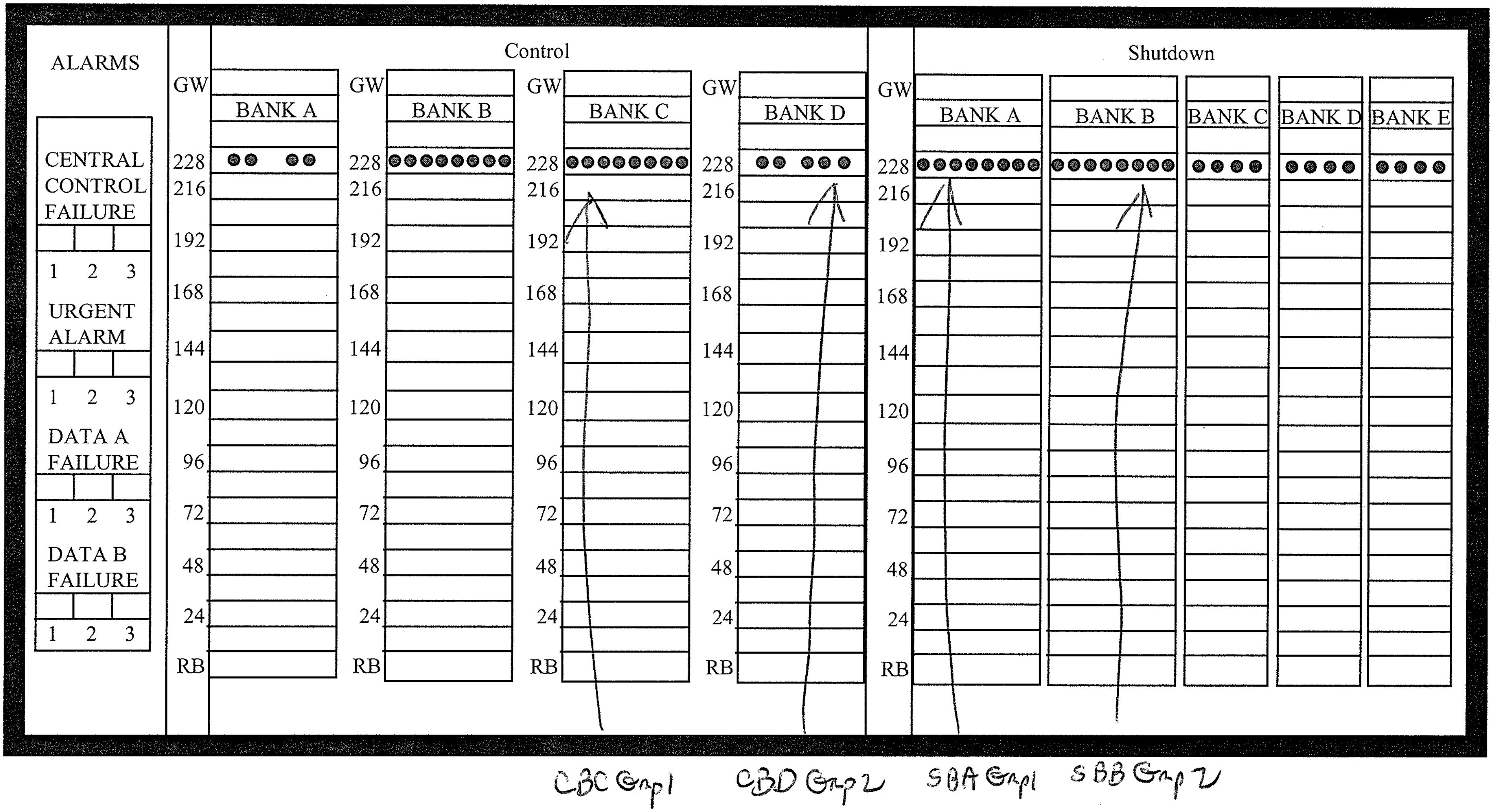
LO-LP-60303-04, Describe the effects of failing to reset the P to A Converter (Bank Demand Position Display) following a dropped rod retrieval.

LO-LP-60303-19, Given the entire AOP, describe:

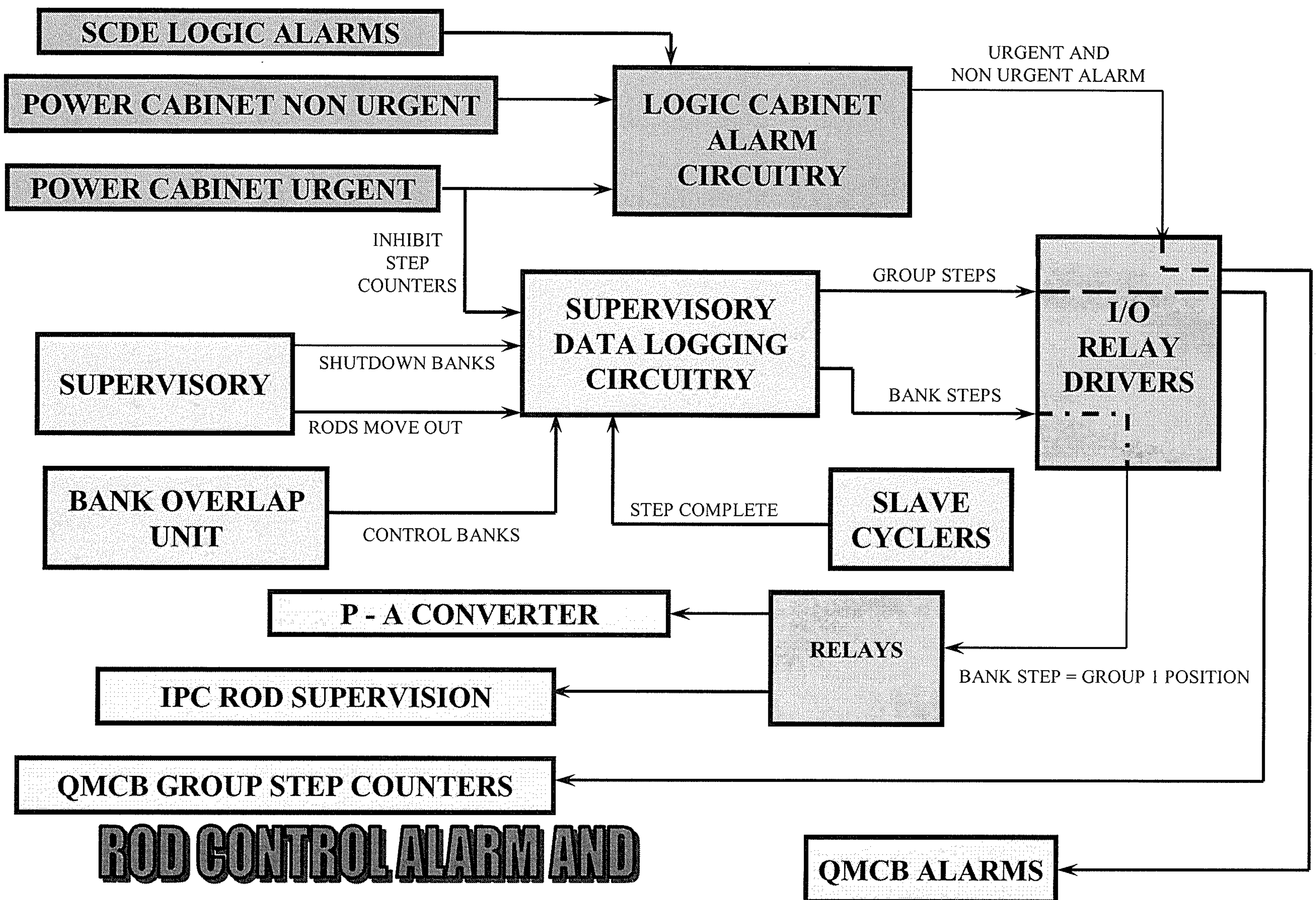
a. Purpose of each step.

b. How and why each step is performed.

Digital Rod Position Indication System



V-LO-PP-27201, slide #8



ROD CONTROL ALARM AND DATA LOGGING CIRCUITS

Only Grp 1 inputs to P/A converter (RIL)

Group 1, rod M12 has been withdrawn from 180 to 190 steps to realign it to the bank. Which of the following is true?

A. Rod Deviation Monitoring and RIL Monitoring are both operable.

B. Rod Deviation Monitoring and RIL Monitoring are both inoperable.

C. Rod Deviation Monitoring is inoperable and RIL Monitoring is operable.

D. Rod Deviation Monitoring is operable and RIL Monitoring is inoperable.

Grp 1 control Rods (control bank, not shutdown bank) Renders RIL monitor inoperable,

Group 2, rod M4 has been withdrawn from 180 to 190 steps to realign it to the bank. Which of the following is true?

A. When the P-A Converter is checked it should indicate 180 steps.

B. When the P-A Converter is checked it should indicate 190 steps.

C. When the P-A Converter is checked it should indicate 200 (190 + 10) steps.

Group 2 control banks do not affect P to A converter (RIL)

Group 1, rod M12 has been withdrawn from 180 to 190 steps to realign it to the bank. Which of the following is true?

A. When the P-A Converter and the demand position in the IPC are checked they should indicate 180 steps.

B. When the P-A Converter and the demand position in the IPC are checked they should indicate 190 steps.

C. When the P-A Converter and the demand position in the IPC are checked they should indicate 200 (190 + 10) steps.

Group 1 control bank affects P to A.

Group 2, rod M4 has been withdrawn from 180 to 190 steps to realign it to the bank. Which of the following is true?

A. When the demand position in the IPC is checked it should indicate 180 steps.

B. When the demand position in the IPC is checked it should indicate 190 steps.

C. When the demand position in the IPC is checked it should indicate 200 (190 + 10) steps.

Group 2 does not affected P to A.

A DROPPED RODS IN MODE 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐A18. Check Rod being realigned - GROUP 1 CONTROL OR SHUTDOWN BANK ROD	☐A18. Go to Step A20.
A19. Initiate 14915, SPECIAL CONDITIONS SURVEILLANCE LOGS:	data sheet 5 (RIL Monitor inop)
☐• Rod Insertion Limit Monitor (if control bank)	RIL Monitor Inop IF Group 1 Control Bank.
☐• Rod Position Deviation Monitor	
☐A20. Check Unit operation at or above 75% for at least 72 cumulative hours in a 7 day period.	☐A20. Limit dropped Rod withdrawal to 3 steps per hour.
☐A21. Record the affected bank's group step counter positions in the Unit Control Log.	

Sheet 1 of 2

Table 1 – Special Conditions Surveillance Requirements

SPECIAL CONDITION	SURVEILLANCE	APPLICABLE MODE	TECH SPEC	FREQUENCY	DATA SHEET
Reactor Critical, Tavg-Tref Dev Alarm not reset and any RCS loop Tavg <561°F	Minimum Temperature for Criticality	1 and 2	SR 3.4.2.1	Once per 30 min	1
Rod Position Deviation Monitor inoperable	Rod Group Alignment Limits	1 and 2	SR 3.1.4.1	Once per 4 hours	3
Rod Position Indication System inoperable	Rod Position Indication System - Operating	1 and 2	LCO 3.1.7 (Actions A.1, B.1)	Once per 8 hours	4a
Rod Demand Indication System inoperable	Rod Demand Indication System - Operating	1 and 2	LCO 3.1.7 (Actions C.1.1 C.1.2	Once per 8 hours	4b
Rod Insertion Limit Monitor inoperable	Control Rod Insertion Limits	1 and 2	SR 3.1.6.2	Once per 4 hours	5
Axial Flux Difference Monitor Alarm inoperable	Axial Flux Difference	1 ≥50%	SR 3.2.3.1	Once per hour	6
Quadrant Power Tilt Monitor Alarm inoperable	Quadrant Power Tilt Ratio	1 >50%	SR 3.2.4.1	Once per 12 hours	7
Quadrant Power Tilt Ratio >1.02	Quadrant Power Tilt Ratio	1 >50%	LCO 3.2.4 (Action A.2.1)	Once per 12 hours	7
One Power Range NI Channel inoperable	Quadrant Power Tilt Ratio	1 ≥75%	SR 3.2.4.2	Once per 12 hours	7
Primary or Secondary Temperature ≤70°F	S/G & RCS Press Limitations	At all times	TRS 13.7.1.1	Once per hour	8
Two Source Range NI Channels inoperable	RCS Boron	6	LCO 3.9.3 (Action B.2)	Once per 12 hours	9

DATA SHEET 5 Sheet 1 of 1

CONTROL ROD INSERTION LIMITS WITH ROD INSERTION LIMIT MONITOR INOPERABLE

1. **Record** each Control Rod Bank Step Counter Demand position and each Rod Bank Insertion limit once every 4 hours.

2. **Verify** each Rod Bank Step Counter Demand position is greater than or equal to the Bank Insertion limit.

3. If any Rod Bank Step Counter Demand position is below the Bank Insertion limit, within 15 minutes **initiate** and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required shutdown margin is restored. (Technical Requirement 13.1.1)

Date __________ Only control banks affect

Time						
CBA DEMAND						
CBA INSERTION LIMIT						
CBB DEMAND						
CBB INSERTION LIMIT						
CBC DEMAND						
CBC INSERTION LIMIT						
CBD DEMAND						
CBD INSERTION LIMIT						
VERIFIED						

Shift Supervisor Review: ______________/______________/______________

Initial Date Time

The Supervisory Logic receives the signal and generates the following when the In-Hold-Out Switch demands movement:

- signal to the Pulser oscillator to generate pulses for 48 steps/min.
- signal to the Pulser oscillator to allow oscillator to generate timing pulses.
- signal to the Bank Overlap Unit to select which control bank or banks to move. In this case, both control Banks A and B will move.
- signal to the Master Cycler allowing fast pulses.
- signal to the Master Cycler specifying the direction to move.
- signal to the Slave Cycler specifying the direction to move

The Bank Overlap Unit will receive the rod motion and direction demand and will generate the following:

- signal to the Master Cycler specifying which shutdown bank to move.
- signal to the Power Cabinet multiplexing relays specifying which group in the cabinet to move.

The Master Cycler fast steps. At step 0 the Master Cycler sends a GO pulse to slave cycler 1AC and 1BD and signal to supervisory logic that rod motion is started. The Supervisory logic then sends a signal to the Master Cycler inhibiting fast pulse and the Master Cycler continues advancing its counter at the pulser rate (48 steps/min). At counter step 3, slave cyclers 2AC and 2BD receives a go pulse. This cycle repeats until demanded motion stops.

On receipt of a Go pulse, the Slave cyclers generate current orders for one step of rod motion and send them to their respective Power Cabinet. They also generate the following:

- signal to the data logging circuits of the supervisory logic when a step is complete. The Supervisory logic sends this signal to the IPC computer for rod deviation monitoring, signals the Demand counters to step, and, if a group 1 rod, sends a signal to the P/A Converter for the RIL monitor.

- Alarm signals if a fault is detected in its monitored circuits.

The Power Cabinets will generate the currents requested by the slave cycler and the rods will move. The group 1 Control Bank A and B rods will move out a step, then the Group two rods will move out.

D. MOVEMENT OF CONTROL BANKS IN AUTO

The Bank selector switch must be selected to AUTO. For this discussion assume CBD is to be moved in from 228 steps.

The Supervisory Logic receives the signal and generates the following when the T-avg Control System demands movement:

- signal to the Pulser oscillator to generate pulses for 8 to 72 steps/min depending on the T-avg control system.
- signal to the Pulser oscillator to allow oscillator to generate timing pulses.

Vogtle Txt V-LO-TX-27101

4. 003G2.4.35 001/2/1/RCP-LOCAL SO ACTIONS/C/A - 3.8 / 4.0/NEW/HL-15R NRC/RO/TNT/DS

Given the following plant conditions:

- 19010-C, "E-1.0 Response to Loss of Reactor or Secondary Coolant" in effect.
- RCS pressure is 1670 psig and stable.

The following annunciators illuminate:

RCP SHAFT VIBRATION ALERT	**RCP SHAFT HI VIBRATION**

Which ONE of the following correctly identifies where RCP vibrations are read and the action to take based on a shaft vibration of 20.2 mils?

	Location	Action to take
A.	IPC computer points	Immediately trip the affected RCP(s)
B.	IPC computer points	Continue RCP operation and monitor vibrations
C.✓	Locally in Control Building	Immediately trip the affected RCP(s)
D.	Locally in Control Building	Continue RCP operation and monitor vibrations

K/A

003 Reactor Coolant Pump System (RCPS)

G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects:

K/A MATCH ANALYSIS

The question presents a plausible scenario where RCP vibration alarms are received during EOP performance. The student must know where the vibration is monitored and action to take in response to given vibration values.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Monitored locally in the Control Building, plausible due to other large components such as MFPTs and Main Turbine as well as several RCP parameters may be checked on the IPC. High shaft vibration indicates the immediate trip criteria has been exceeded. Tripping the RCP is the correct action.

B. Incorrect. Monitored locally in the Control Building, plausible due to other large components such as MFPTs and Main Turbine as well as several RCP parameters may be checked on the IPC. Both shaft alert and high level alarms indicate the immediate trip criteria is exceeded, allowing the pump to run after the one that is identified to have a problem is an incorrect action.

C. Correct. Control building is correct location to monitor. The high shaft vibration alarm indicates the immediate trip criteria has been exceeded. Once the problem RCP has been identified, it should be tripped.

D. Incorrect. Control building is correct location. Immediate RCP trip is required for a valid hi RCP shaft vibration alarm. The alarm is validated by locally reading the RCP vibrations. To allow the pump to continue to run is an incorrect action.

REFERENCES

17008, ARP for windows E04 and F04. Pages 38 and 39 and pages 43 and 44.

13003, Reactor Coolant Pump Operation Limitation 2.2.10

V-LO-PP-16401, Reactor Coolant Pumps slide # 26

VEGP learning objectives:

Not applicable.

	(1)	(2)	(3)	(4)	(5)	(6)
A	RCP 1 MTR OVERLOAD	RCP 1 STANDPIPE LO LEVEL	RCP 1 STANDPIPE HI LEVEL	RCP 1 NO. 2 SEAL LKOF HI FLOW	RCP 1 CONTROLLED LKG HI/LO FLOW	
B	RCP 2 MTR OVERLOAD	RCP 2 STANDPIPE LO LEVEL	RCP 2 STANDPIPE HI LEVEL	RCP 2 NO. 2 SEAL LKOF HI FLOW	RCP 2 CONTROLLED LKG HI/LO FLOW	
C	RCP 3 MTR OVERLOAD	RCP 3 STANDPIPE LO LEVEL	RCP 3 STANDPIPE HI LEVEL	RCP 3 NO. 2 SEAL LKOF HI FLOW	RCP 3 CONTROLLED LKG HI/LO FLOW	
D	RCP 4 MTR OVERLOAD	RCP 4 STANDPIPE LO LEVEL	RCP 4 STANDPIPE HI LEVEL	RCP 4 NO. 2 SEAL LKOF HI FLOW	RCP 4 CONTROLLED LKG HI/LO FLOW	RCP NO. 1 SEAL LO ΔP
E			RCP FRAME VIBRATION ALERT	RCP SHAFT VIBRATION ALERT		RCP SEAL WATER INJ FILTER HI ΔP
F			RCP FRAME HI VIBRATION	RCP SHAFT HI VIBRATION		RCP SEAL WATER INJ LO FLOW

WINDOW E04

ORIGIN	SETPOINT	
1-XE-0471C,D 1-XE-0472C,D 1-XE-0473C,D 1-XE-0474C,D	15 MILS	RCP SHAFT VIBRATION ALERT

1.0 **PROBABLE CAUSE**

1. RCS operating temperature below 500°F.
2. Pump Bearing failure.
3. Pump Impeller - shaft assembly out-of-balance.
4. Misalignment between Pump Shaft and Motor Shaft.
5. Loose connections or disconnected vibration probes.
6. Vibration Monitoring Panel power failure.
7. Local COMMON RESET not cleared.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW E04
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONSS

> NOTE
>
> The Vibration Monitoring Panel displays auctioneered high vibration levels.

Local panel

1. **Dispatch** an operator to the Vibration Monitoring Panel 1-1201-P5-VMP to:

   a. **Identify** the Reactor Coolant Pump (RCP) causing the alarm.

   b. **Check** both vibration channels and alarm setpoints for shaft and frame of each RCP (32 points in all) to verify no obvious vibration monitoring equipment problems exist.

   c. Attempt to **reset** alarm using COMMON RESET toggle switch.

Plausible if they confuse or don't know alarm meaning or hi vibe stpt for Immediate Trip Criteria.

2. **Continue operation** of affected RCP and frequently monitor vibration.

3. Refer to 13003-1, "Reactor Coolant Pump Operation" and **shut down** the affected RCP if rate of increase in vibration exceeds 1 MIL/hour.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB113, 1X6AB06-119, 1X3D-BD-M01A, 1X3D-CD-M10A, 1X6AB09-88, CX5DT101-176A, CX5DT101-176B

WINDOW F04

ORIGIN	SETPOINT	
1-XE-0471C,D 1-XE-0472C,D 1-XE-0473C,D 1-XE-0474C,D	20 MILS	RCP SHAFT HI VIBRATION

1.0 **PROBABLE CAUSE**

1. RCS operating temperature below 500°F.
2. Pump Bearing failure.
3. Pump Impeller - shaft assembly out-of-balance.
4. Misalignment between Pump Shaft and Motor Shaft.
5. Loose connections or disconnected vibration probes.
6. Vibration Monitoring Panel power failure.
7. Local COMMON RESET not cleared.

2.0 **AUTOMATIC ACTIONS**

NONE

WINDOW F04
(Continued)

NOTES

- Prompt action is required to confirm alarm validity and shut down affected RCP if required.
- The Vibration Monitoring Panel displays auctioneered high vibration levels.

3.0 INITIAL OPERATOR ACTIONS

1. Attempt to **confirm** validity of annunciator through related plant parameters.

2. **Dispatch** an operator to the Vibration Monitoring Panel 1-1201-P5-VMP to:

   a. **Identify** the Reactor Coolant Pump (RCP) causing the alarm.

   b. **Check** both vibration channels and alarm setpoints for shaft and frame of each RCP (32 points in all) to verify no obvious vibration monitoring equipment problems exist.

   c. Attempt to **reset** alarm using COMMON RESET toggle switch.

3. Refer to 13003-1, "Reactor Coolant Pump Operation" and shut **down** the affected RCP.

4.0 SUBSEQUENT OPERATOR ACTIONSS

NONE

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

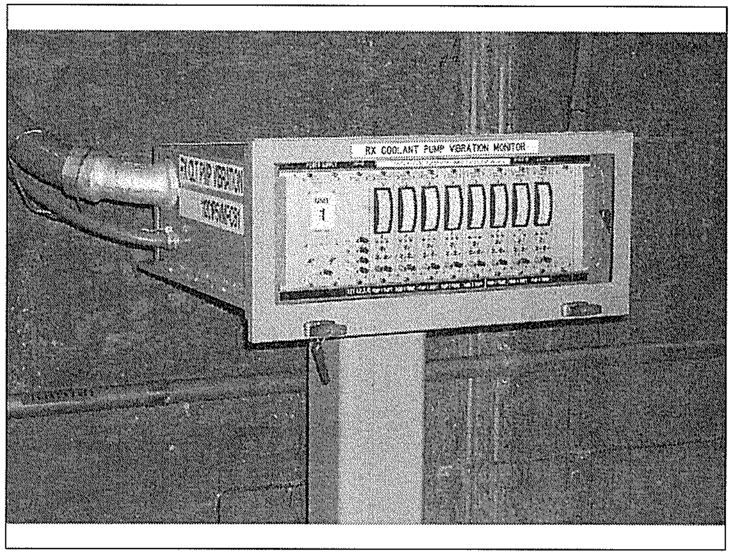
REFERENCES: 1X4DB113, 1X6AB09-119, 1X3D-BD-M01A, 1X3D-CD-M10A, 1X6AB09-88

2.2.9 During RCS filling and venting, RCS pressure must be greater than 325 psig prior to starting an RCP to verify adequate seal D/P is maintained throughout RCS fill and vent. If necessary, the RCP should be stopped prior to seal D/P dropping less than 200 psid. If the seal D/P goes below 200 psid during pump operation or coast down, the RCP should be evaluated before restarting the RCP.

2.2.10 An RCP shall be stopped if any of the following conditions exist.

- Motor bearing temperature exceeds 195°F.
- Motor stator winding temperature exceeds 311°F.
- Seal water inlet temperature exceeds 230°F
- Total loss of ACCW for a duration of 10 minutes.
- RCP shaft vibration of 20 mils or greater.
- RCP frame vibration of 5 mils or greater.
- Differential pressure across the number 1 seal of less than 200 psid.

2.2.11 If a loss of RCP seal cooling (Seal Injection and/or ACCW to Thermal barrier) occurs, resulting in RCP shutdown due to exceeding operating limits, then the unit should be cooled down to Mode 5 to facilitate recovery. Upon reaching Mode 5, ACCW to the Thermal barrier should be restored. Seal injection should then be returned to service. This sequence should prevent seal damage, RCP shaft bowing, ACCW System damage, etc. due to excessive thermal stresses.



1) RCP Shaft Vibration

- Measured by a vertical and horizontal proximity probes mounted parallel and perpendicular respectively to the pump discharge at a location near the pump coupling.
- Continuous monitoring
- Alarm in control room alert and high (Alert 15 mils / High 20 mils)
- greater than 15 mils could require pump shutdown (indicative of bearing failure)
- Must trip at 20 mils

2) RCP Frame Vibration

- 2 probes 90° apart mounted at the top of the motor frame
- Continuous monitoring
- Alarm in the control room alert and high (Alert 3 mils / High 5 mils)
- greater than 3 mils could require pump shutdown (indicative of misalignment or a out of balance condition)
- Must trip at 5 mils

Vibrations checked on a daily bases on control building rounds for trending purposes.

Vibrations should be monitored locally during RCP startup.

5. 004G2.1.23 001/2/1/CVCS-PROCEDURES/C/A - 4.3 / 4.4/NEW/HL-15R NRC/RO/DS/TNT

Given the following conditions:

The reactor is at 100% power
PRZR level is slowly lowering due to a 50 GPM RCS leak
120 GPM CVCS letdown is in service
RCP seal injection flow is 8 GPM per pump
Charging flow controller FIC-0121 is in automatic

Which one of the following lists the correct system response and procedurally directed operator actions for this condition?

A. Charging flow will automatically increase due to PRZR level dropping below the program level.

The increase in charging flow will automatically maintain PRZR level at the program level.

B.✓ Charging flow will automatically increase due to PRZR level dropping below the program level.

Letdown flow will have to be isolated and charging flow will have to be manually adjusted to approximately 62 GPM to maintain a constant PRZR level.

C. Charging flow will automatically increase due to PRZR program level changing.

The increase in charging flow will automatically maintain PRZR level at the program level.

D. Charging flow will automatically increase due to PRZR program level changing.

Letdown flow will have to be isolated and charging flow will have to be manually adjusted to approximately 62 GPM to maintain a constant PRZR level.

K/A

004 Chemical and Volume Control System

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A MATCH ANALYSIS

The question requires the student to correctly determine the CVCS system response to an RCS leak and the appropriate AOP actions for the CVCS system to stabilze PRZR level matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Charging flow will increase due to PRZR level dropping below the program level. Since 120 GPM CVCS letdown is in service, the available increase in charging flow will not keep up with the leak. Maximum charging flow will be approximately 135 GPM.

B. Correct. Charging flow will increase due to PRZR level dropping below the program level. Since 120 GPM CVCS letdown is in service, the available increase in charging flow will not keep up with the leak. The AOP will direct the operator to isolate letdown. Charging flow will then need to be adjusted to 62 GPM to offset the 50 GPM leak and the 12 GPM RCP seal leakoff flow.

C. Incorrect. Charging flow will increase due to the drop in measured PRZR level. Program level will remain the same because Tave will not change as a result of the RCS leak. Since 120 GPM CVCS letdown is in service, the available increase in charging flow will not keep up with the leak. Maximum charging flow will be approximately 135 GPM.

D. Incorrect. Charging flow will increase due to the drop in measured PRZR level. Program level will remain the same because Tave will not change as a result of the RCS leak. Since 120 GPM CVCS letdown is in service, the available increase in charging flow will not keep up with the leak. The AOP will direct the operator to isolate letdown. Charging flow will then need to be adjusted to 62 GPM to offset the 50 GPM leak and the 12 GPM RCP seal leakoff flow.

REFERENCES

AOP 18004-C, "RCS Leakage" step A3

V-LO-TX-16001, "Primary Systems" pages 73, 74, AND 75 for PRZR level control

VEGP learning objectives:

LO-PP-16302-03:

Describe how Pressurizer level control maintains level on program.

LO-LP-60304-09:

Given the entire AOP, describe:

a. Purpose of selected steps.
b. How and why the step is being performed.
c. Expected response of the plant/parameter(s) for the step.

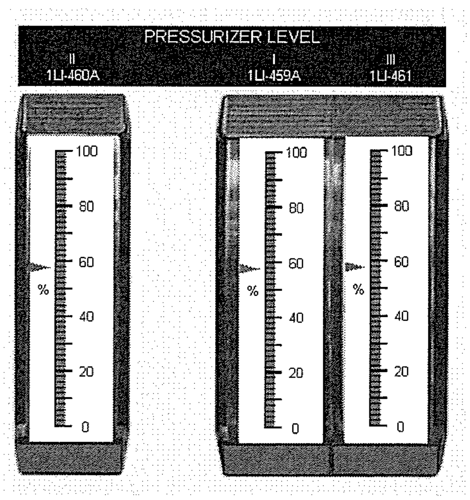
A. RCS LEAKAGE (MODE 1, 2, AND 3 WITH RCS PRESSURE >1000 PSIG)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A1. Check plant conditions:	A1. Go to the appropriate section of this procedure:
__In Mode 1 or 2.	__SECTION B. RCS LEAKAGE (MODE 3 <1000 PSIG AND 4)
-OR-	-OR-
__In Mode 3 with RCS pressure greater than 1000 psig.	__SECTION C. RCS LEAKAGE (MODE 5).
__A2. Initiate the Continuous Actions Page.	
*A3. **Maintain PRZR level:**	
__a. Adjust charging flow as necessary to maintain program level.	
__b. Check PRZR level – STABLE OR RISING.	b. Perform the following:
	1) Isolate letdown by closing:
	__a) Letdown Orifice Valves.
	__b) Letdown Isolation Valves.
	__c) Excess Letdown Valves.
	__2) Start an additional Charging Pump as necessary.

Step 3 continued on next page

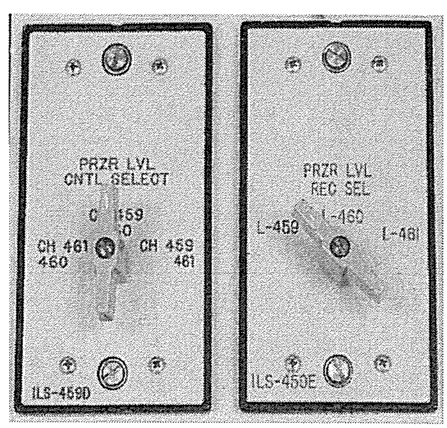
It is important to understand the principles of operation and the limitations of these levels transmitters.

ΔP type level transmitters that are calibrated for normal operating conditions may be inaccurate under abnormal conditions such as a LOCA or steam line break in the containment. Specifically, the reference leg piping and condensing pots are exposed to the containment atmosphere. At elevated containment temperatures, the reference leg pipe and the water it contains will heat up, decreasing the density of the reference leg. This causes the indicated level to be greater than the actual level. As the reference leg is heated up, the volume of the water in the reference leg increases and forces some of the liquid from the reference leg. The pressure that the reference exerts on the level transmitter is less than the pressure that the water exerted prior to being heated up. The result is an indicated change in level (increase) although the actual level may not have changed. The severity of the error will depend on the actual containment conditions. Redundant level channels of the pressurizer may uniformly present inaccurate indications and, under such conditions, must be considered unreliable. Other conditions that may affect pressurizer level indication are reference leg leaks or partial draining due to instrument calibration or other maintenance activities, and a phenomena caused by hydrogen gas coming out of solution in the reference leg. Since the reference leg temperature is cooler than the pressurizer, it has a higher affinity for absorbing hydrogen gas. The hydrogen gas could come out of solution during transients. The results from all of the above mention would be reduction in the ΔP. This reduction in ΔP would cause the pressurizer level indication being higher than actual.



16-58 <u>Pressurizer Level Control System</u>

The Pressurizer Level Control System utilizes three hot calibrated level channels (LT-459, LT-460, and LT-461) for control. Two of these channels are selected at any given time by a three-position selector switch (LS-459D). The possible combinations are: Channels LT-459 and LT-460, Channels LT-461 and LT-460, or Channels LT-459 and LT-461. Only channels LT-459 and LT-461 can be selected for primary level control and only channels LT-460 and LT-461 can be selected for secondary control. The pressurizer level control uses the primary channel input to compare its value to the calculated level set point to control pressurizer level. The secondary channel is used for protection only. A three-position recorder selector switch (LS-459E) is provided to select the actual level to be recorded along with the program level on LR-459.
The reference level signal is generated by auctioneered high Tavg (No-load Tavg 557°F, to 100% Full Power Tavg of 586.4°F) which generates a program level of 25% to 57.8%, which corresponds to the difference between No-load Tavg and full load Tavg. The program level is compared to one of the selected level channels, LT459 or LT461, to produce a level error signal.



The level error produced is used as input by the master level controller. The master controller is sensitive to both the magnitude of the difference and the time duration that the difference is present. A large level error will result in a large controller output. The integral portion of the controller will also produce a high output for small errors that are present for long time durations. To change the level in the pressurizer, either the temperature or the mass balance of the RCS must change. The master controller responds to level errors by changing CVCS charging flow.

During steady state operation with no pressurizer level change, CVCS letdown flow is equal to CVCS charging. If charging flow changes and letdown flow remains constant, then the mass balance of the RCS will change. Pressurizer level control operates on this principal. Charging flow is varied by controlling the position of the flow control valve (FCV-121).

The demand signal from the level master controller (LIC-459) is sent to the charging flow controller (FIC-121). The charging flow controller compares demand flow from the pressurizer level master controller with actual flow. If there is a difference between the two, the controller will position FCV-121 accordingly to correct the error. Both controller MANUAL/AUTO stations are located on the "C" panel in the control room.

The Pressurizer Level Control System will automatically isolate CVCS letdown when pressurizer level decreases to 17%. Both letdown isolation valves, LV-459 and LV-460, close as well as the letdown orifice isolation valves. This prevents draining the pressurizer if a leak occurs in CVCS system. Damage would occur if the heaters were energized and not fully immersed in water. Therefore, the level control system also de-energizes the pressurizer heaters when the water level decreases to 17%. This prevents damage to the wall of the pressurizer vessel due to overheating and to the heaters themselves. The heaters would be exposed if the pressurizer level decreased below 14%. (See Pressurizer Level Control Logic Drawing) The pressurizer Level control system is designed to accommodate the following without a reactor trip:

a. Ramp unloading rate of 5% per minute with auto rod control.

b. Instantaneous load reduction of 10% with auto rod control.

c. Step load reduction of 50% with both auto rod control and steam dump control.

Level control selector switch LS-459D

To further explain its operation the following example is given:

Level transmitter 459/460 is selected on LS-459D

LT-459 is selected as the primary channel for the master level control. If the level that is sensed by LT-459 drops to = 17% it will cause the following to occur:

a. CVCS Charging Flow increases by opening FCV-121

b. All pressurizer heaters will automatically trip.

c. CVCS Letdown Isolation valve LV-459 will automatically close.

d. All three CVCS Letdown Orifice Isolation valves will automatically close.

LT-460 is selected for the secondary channel. If level sensed by LT-460 drops to = 17% it will cause the following to occur:

a. All Pressurizer heaters will automatically trip.

b. CVCS Letdown Isolation valve LV-460 will automatically close.

c. All three CVCS Letdown Orifice Isolation valves will automatically close.

If the primary level control channel sense pressurizer level = 5% above program pressurizer level, a signal is generated that energizes the pressurizer backup heaters. Alarm ALB11-C01 "Przr Hi Level Dev and heaters on" annunciates. The purpose for this design is to heat the in surge of water to saturation in anticipation of a possible sudden out surge to maintain pressurizer pressure. Typical pressurizer temperatures are as follows:

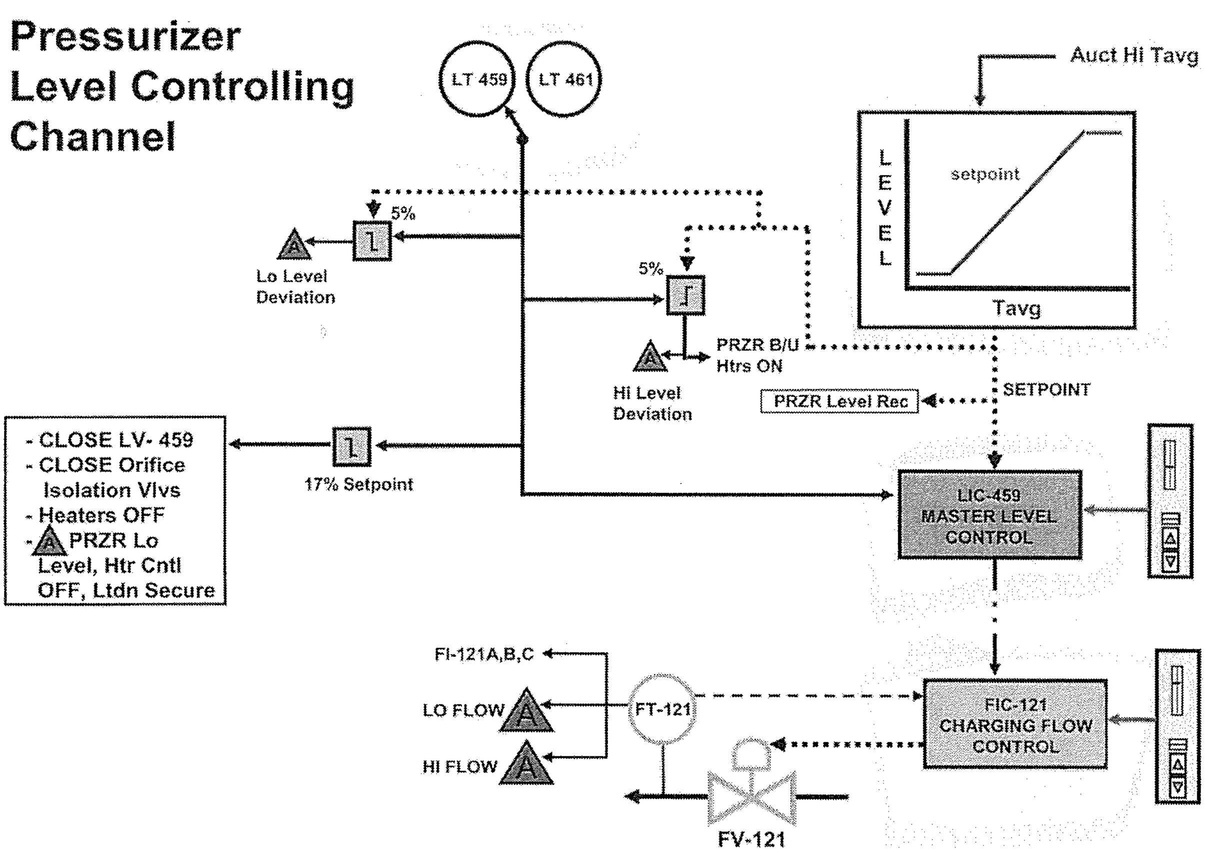
- Pressurizer Surge Line Temperature ~ 645°F

- Pressurizer Liquid Temperature ~ 650°F
- Pressurizer Steam Space Temperature ~ 650-655°F

This example applies to all possible selections on LS-459D.

16-59 Pressurizer Level Protection System

The Pressurizer Level Protection System also utilizes the same level transmitters as the Control System. The level indications provide the information to the Reactor Protection System (RPS). The Reactor Protection System will automatically trip the reactor if the pressurizer level reaches a high level set point of 92% when the reactor is above 10% power. This function however looks at all three level channels and is not based on the switch position of LS-459D. Reactor trip will occur if two out of the three level transmitters are indicating =92%. This Reactor trip function protects the RCS from the over pressurization



that might occur if the pressurizer were to go water solid (Loss of bubble). When the plant is shut down, the pressurizer is cooled down and is allowed to go water solid. The high level trip is automatically disabled by the RPS trip permissive P-7, which happens when the reactor power decreases to <10%.

6. 004K6.31 001/2/1/CVCS-SEAL INJ LIMITS/C/A - 3.1 / 3.5/M- LOIT BANK/HL-15R NRC/RO/TNT/DS

Which one of the following choices lists the correct action to take for the given RCP seal injection flow?

	RCP Seal Injection Flow	Action to take
A.	7.8 gpm	Press UP arrow to throttle open HV-0182.
B.	13.4 gpm	Press UP arrow to throttle closed HV-0182.
C.✓	13.4 gpm	Press DOWN arrow to throttle open HV-0182.
D.	7.8 gpm	Press DOWN arrow to throttle closed HV-0182.

HL-15R RO NRC Exam

K/A

004 **Chemical and Volume Control System**

K6.31 **Knowledge of the effect of a loss or malfunction on the following CVCS components.**

Seal injection system and limits on flow range.

K/A MATCH ANALYSIS

The question presents a plausible scenario where seal injection flows are out of limits. Proper OATC actions and HV-182 response to these actions are required.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. UP arrow raises flow by closing valve. Correct button to depress, however HV-182 closes, not opens. Wrong response to the action.

B. Incorrect. UP arrow raises flow by closing valve. Wrong button to depress, HV-182 response to the action is correct.

C. Correct. DOWN arrow lowers flow by opening valve. Depress the down arrow to open HV-182 and divert less flow to the seals.

D. Incorrect. DOWN arrow lowers flow by opening valve. Wrong button to depress, HV-182 opens, not closes. Wrong response to the action.

REFERENCES

004A4.11, LOIT Exam Bank previously used on HL-13 RO Retake Exam (not in last 2). Question used as base for modification.

Vogtle Text Chapter # 9 for CVCS pages # 21 and # 26

SOP-13003-1/2, Reactor Coolant Pump Operation" limitation 2.2.3.

VEGP learning objectives:

LO-PP-16401-03 Describe the control room indications for a failure of a RCP seal.

1. 004A4.11 001

Which **ONE** of the following actions would be **CORRECT** to take using HC-0182 given the provided seal injection flow indication?

A. seal injection flows - 13.8 gpm per RCP, depress the red UP arrow to throttle open HV-0182 to lower seal injection flow.

B.✓ seal injection flows - 7.4 gpm per RCP, depress the red UP arrow to throttle closed HV-0182 to raise seal injection flow.

C. seal injection flows - 7.4 gpm per RCP, depress the green DOWN arrow to throttle open HV-0182 to raise seal injection flow.

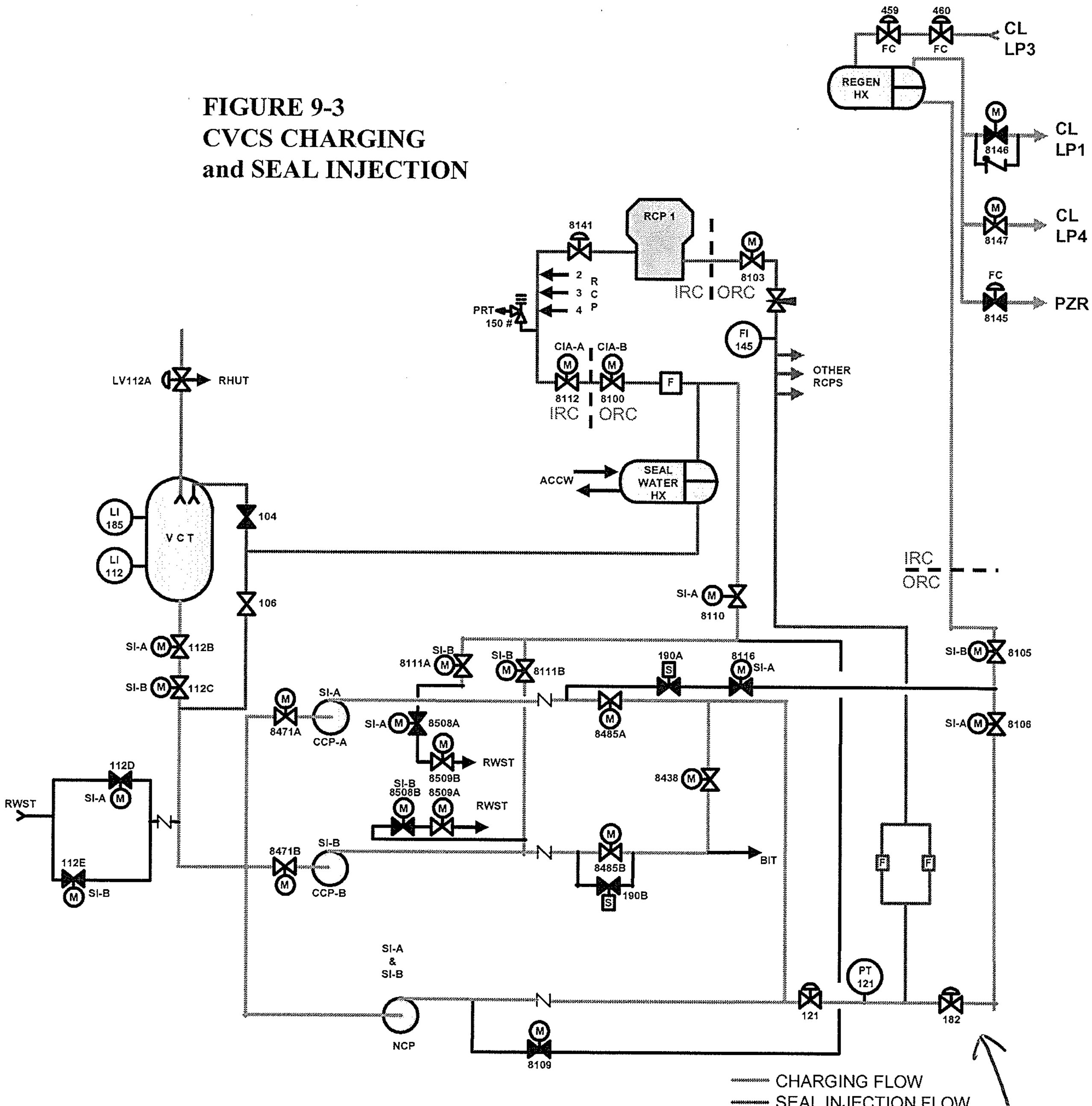
D. seal injection flows - 13.8 gpm per RCP, depress the green DOWN arrow to throttle closed HV-0182 to lower seal injection flow.

Question used as base for modification. Used on 2006 RO NRC Retake (not in last 2).

Question fits KA 004K6.31

through the seal leak off isolation valves (HV-8141A, B, C, and D), through a motor-operated isolation valve (HV-8112), and then exits the containment building. The seal return flow immediately passes through a second motor-operated isolation valve (HV-8100) upon exiting the containment. Both of these motor-operated isolation valves serve to isolate the containment upon receiving a Containment Isolation Actuation(CIA) signal. Seal water return flow next passes through the seal water return filter which removes any insoluble material picked up as the seal water passed through the reactor coolant pump seals. It is then reduced in temperature from approximately 175°F to 130°F as it passes through the tube side of the seal water heat exchanger before returning through an isolation valve to the suction header of the charging pumps.

FIGURE 9-3
CVCS CHARGING
and SEAL INJECTION



close – diverts more to seals,
open – diverts less to

Charging Flow Control Valve FV-121

The discharge of the NCP and CCPs combine into a single charging flow path. Charging flow is controlled by the position of FV-121. In automatic, FV-121 controls the total flow directed toward the normal charging header. The position of FV-121 is determined by the output error signal from pressurizer level controller FIC-121. This error signal is determined by the difference between pressurizer program level(determined by auctioneered high Tavg) and actual pressurizer level. If an output error signal indicates that pressurizer level is below program level, FV-121 will open to provide more charging flow to eliminate that error signal. Conversely, if pressurizer level is above program level, then FV-121 valve position will throttle more closed to lower charging flow. Charging flow controller FIC-121 can be operated locally in the Auxiliary Building. Charging flow indication is provided in the control room on panel A and C.

Seal Flow Control Valve, HV-182

This hand controller air-operated valve in the charging header maintains sufficient backpressure in the charging header to ensure adequate flow of seal water to the reactor coolant pumps. The flow indicators, (FI-142, -143, -144, and -145) for each RCP seal injection are used to adjust the setting of this valve so that approximately 8-13 gpm seal injection flow is maintained to each RCP. The valve is manually controlled from the main control board. The valve fails open on loss of power or air. If more seal injection flow is required, the operator depresses the UP pushbutton on HC-182. This causes HV-182 to be in a more shut position, thus forcing more flow toward the RCP seal injection line. This has the immediate effect of lowering charging flow directed toward the normal charging header. With letdown flow the same and now less charging flow going through the normal charging header, letdown temperature out of the regenerative heat exchanger will increase. Consequently, anytime seal injection flow is adjusted, the effect on letdown parameters must be evaluated. Vice versa, anytime charging flow is changed, the effect on seal injection and letdown parameters must be evaluated also and appropriate actions to restore system parameters to their normal operating band.

Normal Charging header Isolation Valves, HV-8106, HV-8105

Charging flow that does not flow toward the seal package(based on the position of HV-182) flows past HV-182 toward the RCS penetration past series charging isolation valves HV-8106(Train A) and HV-8105(Train B). These valves are operated from the main control room and from the shutdown panels. Each valve is powered from a 1E MCC. On a Safety Injection actuation signal, these valves receive a CLOSE signal from their respective train related SI signal. This isolates the normal charging header and allows the safety-related CCPs to direct their flow through the BIT into all RCS cold legs.

Normal/Alternate Charging to RCS Isolation Valve (HV-8146 and HV-8147)

Control switches for these motor-operated isolation valves are located on the QMCB and the Remote Shutdown panels. These switches are two position (Close/Open). An additional switch Control Room/Local transfer switch is located on the Remote Shutdown panel. This switch must be in the Control Room position to enable the QMCB switches. To equalize thermal stresses, SOP 13006-1/2 states that normal charging valve HV-8146 should be in service during even numbered fuel cycles and alternate charging valve HV-8147 should be used during odd-numbered fuel cycles. This should be performed at cold shutdown conditions to avoid thermal transients.

CVCS Pressurizer Auxiliary Spray Valve (HV-8145)

This air-operated isolation valve is operated from the QMCB and the Remote Shutdown panel. These switches are two position (Close/Open). There is also a

2.1.5 When starting the first RCP with a bubble in the Pressurizer, the additional RCP heat input may cause an insurge of cooler RCS water into the pressurizer. Surge line temperature may be controlled by monitoring surge line temperature and adjusting RHR cooling and charging flow to verify a net outsurge from the pressurizer.

2.1.6 With Westinghouse and Operations management approval, RCPs may be started without ACCW flow to perform 30 second and 1 minute air sweeps per 13001, "Reactor Coolant System Filling and Venting" or to verify proper rotation following electrical maintenance (less than 1 minute). General Manager approval will be required for starting RCPs without ACCW for any other operation. Operation without ACCW in service for more than 10 minutes is prohibited.

2.1.7 Seal Injection flow should be maintained to coupled RCPs when RCS level is greater than the 190 foot elevation, however, if necessary, seal injection may be secured to RCPs above the 190 foot elevation provided RCS level is maintained constant.

2.1.8 RCPs should NOT be uncoupled and placed on their back seat until the RCS is depressurized and vented.

2.2 LIMITATIONS

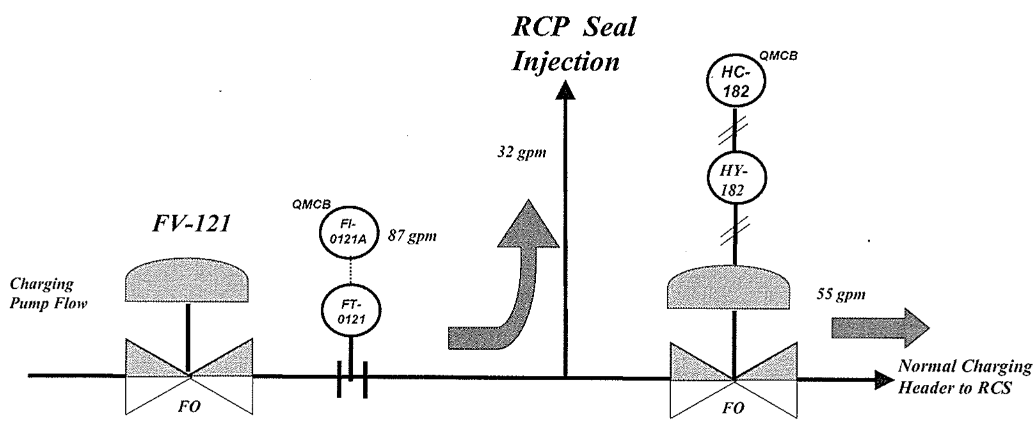
2.2.1 If seal injection is NOT in service AND the reactor coolant temperature is greater than 150°F, Auxiliary Component Cooling Water shall be supplied to the thermal barrier.

2.2.2 When the reactor coolant pressure is less than 100 psig, the No. 1 Seal Leakoff Valves should be closed.

2.2.3 The RCP seal injection flow should be maintained greater than 8 gpm and less than 13 gpm any time seal injection is required.

2.2.4 With the reactor coolant temperature greater than 400°F, the seal injection temperature should be maintained less than 135°F.

HV-182



V-LO-PP-09200-02.1 151

Charging FV-121 and HV-182 are air operated valves that FAIL OPEN on loss of air. Seal injection flow is controlled by manually positioning HV-182 to provide backpressure against the charging header to force flow towards the RCP seal injection header.

Consequently, when both valves fail open, then charging flow goes to its maximum value. If this malfunction was due to a loss of instrument air, procedure 18028-C provides guidance to establish Safety Grade Charging per SOP 13006-1/2.

Charging flow is controlled using the Safety Grade charging flow controller(HC-190A or HC-190B). Seal injection is controlled locally by locally opening 1208-U6-151 or 152.

7. 005K2.03 002/2/1/RHR-MOV POWER/MEM - 2.7 / 2.8/M- LOIT BANK/HL-15R NRC/RO/TNT/DS

Which **ONE** of the following **CORRECTLY** describes the power supplies to the RHR loop suction isolation valves ?

A. All 4 loops suctions are powered from 1E 480V MCCs.

B. All 4 loops suctions are powered from 1E 25KVA Inverters.

C. Two loop suctions on one train are powered from 1E 480V MCCs
Two loop suctions on one train are powered from 1E 25KVA Inverters.

D✓ One loop suction on each train is powered from 1E 480V MCCs.
One loop suction on each train is powered from 1E 25KVA Inverters.

HL-15R RO NRC Exam

K/A

005 **Residual Heat Removal System (RHRS)**

K2.03 **Knowledge of bus power supplies to the following:**

RCS pressure boundary motor-operated valves.

K/A MATCH ANALYSIS

The question presents a plausible scenario where the candidate must pick the correct power supplies to both trains RHR loop suction isolation valves.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible to think all 4 valves are powered from 480V 1E MCCs since most safety related valves are powered from 480V 1E MCCs, however 2 of the loops suctions are powered from 480V 1E Inverters CD1I5 and DD1I6.

B. Incorrect. Plausible to think all 4 valves are powered from 480V 1E Inverters. There are 4 green annunciators associated with these valves (2 for starters and 2 inverters)

C. Incorrect. Plausible to think one train powered from inverters and the other powered from MCCs since there are 2 valves powered from each.

D. Correct.

REFERENCES

V-LO-PP-12101, Residual Heat Removal System, slide # 42.

13011-1, Residual Heat Removal, pages # 15 and # 22. Step s 4.1.4 and 4.2.4.

LOIT BANK QUESTION 005K2.03-001

VEGP learning objectives:

LO-PP-12101-12 Briefly describe the RHR system alignment during normal power operations and during RCS cooldown.

1. 005K2.03 001

Regarding the two following valves:

* HV-8701B, RHR PMP-A UPSTREAM SUCTION FROM HOT LEG LOOP 1

* HV-8702B, RHR PMP-B UPSTREAM SUCTION FROM HOT LEG LOOP 4

Which ONE of the following CORRECTLY describes the power supplies to the RHR upstream loop suction isolation valves ?

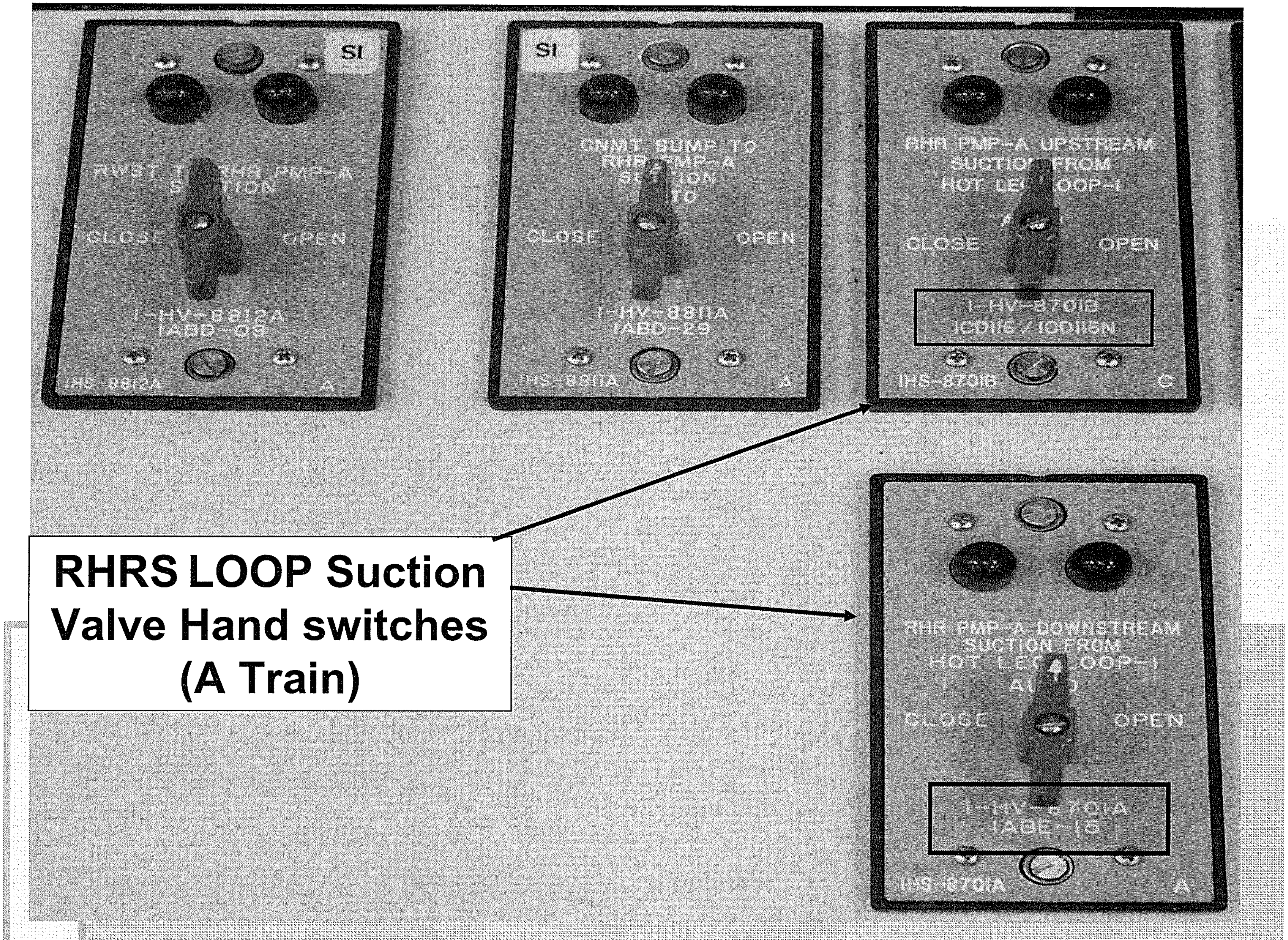
A.✓ 125V DC Inverter CD1I5 and 480V 1E MCC BBE.

B. 480V 1E SWGR AB05 and 125V DC Inverter DD1I6.

C. 480V MCC NBE and 125V DC Inverter DD1I4.

D. 125V DC Inverter CD1I3 and 480V 1E SWGR BB07.

Question used as base for modification. from LOIT Bank



V-LO-PP-12101, Residual Heat Removal, slide # 47.

INITIALS

Critical

p. **Open** the RWST TO RHR PMP-A SUCTION 1-HV-8812A. ________

____
CV

q. **Place** RHR Pump A 1-HS-0620 in AUTO. ________

r. If required, **open** RHR PMP-A TO COLD LEG 1&2 ISO VLV 1-HV-8809A. ________

4.1.3.2 **Restore** power to RHR Train A to CCP Suction as follows:

a. **Remove** tags and **close** K2 link for breakers 1ABB-05. ________

b. **Close** 1ABB-05 to Valve 1-HV-8804A. ________

4.1.3.3 **Align** RHR TRN-A for standby per Checklist 3. ________

4.1.4 IF RHR is being placed in standby for MODE 3 entry, perform the following: (IV REQUIRED)

a. **Shut down** Inverter 1CD1I5 per 13405-1, "125V DC 1E Electrical Distribution System." ________

b. **Open** and **lock** the power supplies to the RHR Loop 1 Inlet Isolations:

(1) **Turn off** 1ABE-15 for 1-HV-8701A. ________

(2) **Open** the K2 links for breaker 1ABE-15. ________

ALB34 E06 STARTER 1CD1I5N TROUBLE

(3) At 1CD1I5N **open** and **lock** the disconnect for 1-HV-8701B. ________

(4) **Remove** and **store** the annunciator card associated with ALB34 E06 per 10018-C, "Annunciator Control." ________

INITIALS

q. **Place** RHR Pump B Handswitch 1-HS-0621 in AUTO. ________

r. If required, **open** RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B. ________

4.2.3.2 **Restore** power to RHR Train B to SIP Suction as follows:

a. **Remove** tags and **close** K2 link for breakers 1BBB-05. ________

b. **Close** 1BBB-05 to Valve 1-HV-8804B. ________

4.2.3.3 **Align** RHR TRN-B for standby per Checklist 4. ________

4.2.4 IF RHR is being placed in standby for MODE 3 entry, perform the following:

a. **Shut down** Inverter 1DD1I6 per 13405-1, "125V DC 1E Electrical Distribution System." (IV REQUIRED) ________

b. **Open** and **lock** the power supplies to the RHR Loop 4 Inlet Isolations:

(1) **Open** 1BBE-13 for 1-HV-8702B. (IV REQUIRED) ________

(2) **Open** K2 links for breaker 1BBE-13. (IV REQUIRED) ________

ALB34 E07 STARTER 1DD1I6N TROUBLE

(3) At 1DD1I6N, **open** and **lock** the disconnect for 1-HV-8702A. (IV REQUIRED) ________

(4) **Remove** and **store** the annunciator card associated with ALB34 E07 per 10018-C, “Annunciator Control.” ________

8. 005K6.03 001/2/1/RHR-HX/C/A - 2.5 / 2.6/M - LOIT BANK/HL-15R NRC/RO/TNT/DS

Given the following:

- The plant is in Mode 6.

- RHR Train "B" is in service.

- RHR Hx Bypass Valve FV-619 is in auto set to maintain minimum Tech Spec flow.

- RHR Hx Outlet Valve HC-607 demand is set at 30%.

- The instrument air supply to HC-607 severs and is completely detached.

- No other air operated valves are impacted by the failure.

Which **ONE** of the following describes the following system parameter changes from the initial steady state condition to the final steady state condition?

	RCS Cooldown Rate	RHR Hx Bypass flow
A.	Lower	Lower
B.	Lower	Higher
C.✓	Higher	Lower
D.	Higher	Higher

K/A

005 Residual Heat Removal System (RHRS)

K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS:

RHR heat exchanger.

K/A MATCH ANALYSIS

The question presents a plausible scenario where an air line breaks to the RHR Hx outlet valve (HV-607). This would result in maximum cooling water flow through the RHR Hx (higher RCS Cooldown rate) and lower RHR Hx Bypass flow.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RCS cooldown rate will actually increase due to HV-607 failing open on loss of air. It will cause flow through the Hx to increase. HV-607 failing open will cause student determines that the RHR valves fail shut on loss of air.

B. Incorrect. RCS cooldown rate will actually increase due to HV-607 failing open on loss of air. It will cause flow through the Hx to increase. HV-607 failing open will cause FV-619 to throttle shut to maintain a constant total RHR flow. This choice is plausible if the student incorrectly determines that HV-607 fails shut on a loss of air.

C. Correct. RCS cooldown rate increases due to HV-607 failing open on loss of air. It will cause flow through the Hx to increase. HV-607 failing open will cause FV-619 to throttle shut to attempt maintain a constant total RHR flow.

D. Incorrect. RCS cooldown rate increases due to HV-607 failing open on loss of air. It will cause flow through the Hx to increase. HV-607 failing open will cause FV-619 to throttle shut to maintain a constant total RHR flow. This choice is plausible if the student determines that bypass flow goes up with HX flow due to loss of air.

REFERENCES

LO-PP-12101-08-002 from LOIT Exam Bank.

1X4-DB-122, Residual Heat Removal System (excerpt included).

18028-C, Loss of Instrument Air, Attachment B, Loss of Instrument Air in Modes 4, 5, 6

VEGP learning objectives:

LO-LP-12102-03, Describe the RHR system response and the appropriate corrective actions for the following.

a. Loss of air or electrical power to FCV 606 or 618.

LO-LP-60321-02, State the fail position of the following valves on a loss of instrument air.

i. RHR heat exchanger outlet valve.

1. LO-PP-12101-08 002

Which **ONE** of the following is **CORRECT** concerning the effect of a loss of instrument air with RHR Train A in service while in Mode 5?

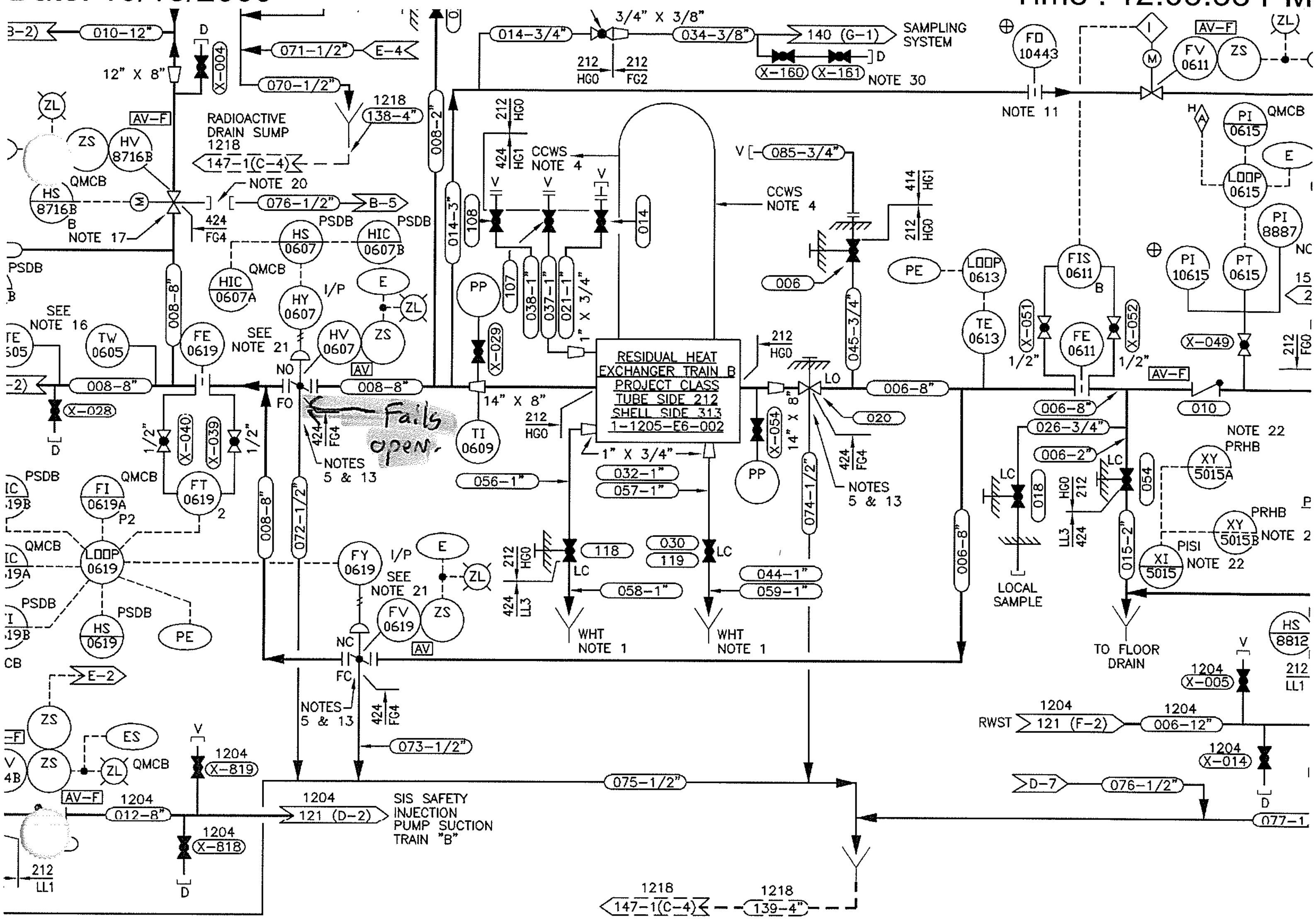
A. The RHR Heat Exchanger Outlet valve, HV-606, would fail SHUT, and the RHR Flow Control valve, FV-618, would fail SHUT.

B. The RHR Heat Exchanger Outlet valve, HV-606, would fail OPEN, and the RHR Flow Control valve, FV-618, would fail OPEN.

C. The RHR Heat Exchanger Outlet valve, HV-606, would fail SHUT, and the RHR Flow Control valve, FV-618, would fail OPEN.

D.✓ The RHR Heat Exchanger Outlet valve, HV-606, would fail OPEN, and the RHR Flow Control valve, FV-618, would fail SHUT.

Question used as base for modification.



FORMATION SEE DWGS 1X4DB143, 144-1,
·DB183.
ALVES FURNISHED BY WESTINGHOUSE EXCEPT:
LL MANUAL VALVES 2" AND SMALLER, AND
D ⊕.
R COOLING WATER CONNECTIONS, SEE DWGS
57.
SHELL SIDE CONNECTIONS SEE DWG 1X4DB137.
LE SPOOL.
CLASS CHANGE AT FIRST WELD OF
NSIDE THE CONTAINMENT.
NGE ON INLET OF EACH 14"ø PIPE FOR
DVIDE ONE MATING BLIND FLANGE TO BE DRILLED
CONNECTION. MATING FLANGE MUST BE REMOVED
N.
ED TO SEISMIC CATEGORY 1 AND ASME III CLASS MC.
STEM ARE SUSPECTED TO CAUSE STRESS
REFER TO THE APPLICABLE DESIGN DRAWING AND
IFICATIONS FOR MATERIAL AND HEAT
NTS.

10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON BPC DWG CXDG001.
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. FOR TYPICAL DETAIL SEE STD. DWG AX4DD000.
14. DELETED.
15. HIGH POINT VENT VALVE (BECHTEL-FURNISHED) FOR SEAL PIPING - SEE 1X6AF02-25.
16. TE IS A SURFACE TYPE CLAMP ON RTD, TW IS INSTALLED BUT NOT USED.
17. VENT HOLE IS DRILLED ON RESPECTIVE RHR PUMP SIDE OF DISC.
18. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.
19. ALARM ON OPEN HOT LEG VALVE AND HIGH RCS PRESSURE.
20. DELETED.
21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.
23. 4" VIEW PORT TO BE USED TO VERIFY MOTOR OPERATED VALVE STROKE TEST WITHOUT REMOVING THE VESSEL HEAD.
24. VENT HOLE IS DRILLED ON CONTAINMENT SIDE OF DISC.
25. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-249-01.
26. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-251-01.
27. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-036-01.
28. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-049-02.
29. A SECTIOI
REMOVED
30. THESE AR
ALLOW RE
31. SUMP SC
(TRAIN "A

7 6 5

ATTACHMENT B

LOSS OF INSTRUMENT AIR IN MODES 4, 5, OR 6

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B1. Check Instrument Air supply header pressure on PI-9361 - LESS THAN 100 PSIG.	__B1. Go to Step B11.

CAUTION

Loss of instrument air will cause CHARGING LINE CONTROL FV-0121 and SEAL FLOW CONTROL HV-0182 to fail open.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B2. Check RCS inventory – SOLID.	B2. Perform the following: __a. IF needed to maintain RCS level, THEN establish safety grade charging by initiating 13006, CHEMICAL AND VOLUME CONTROL SYSTEM. __b. Go to Step B5.
__B3. Trip all charging pumps.	
__*B4. **Monitor No. 1 seal leakoff temperature and flow until charging pump is restarted.**	

CAUTION

Loss of instrument air pressure will cause the RHR HX outlet valves to fail full open and the HX bypass valves to fail fully closed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B5. Check plant Mode - MODE 4 OR MODE 5.	__B5. Suspend all fuel movement.

9. 006K6.02 001/2/1/ECCS-ACCUM/C/A - 3.4 / 3.9/M - WATTS BAR 2008/HL-15R NRC/RO/TNT/DS

Given the following:

- The ECCS Accumulator isolation valves are CLOSED and in AUTO with power on the valves.

- PRZR pressure has been raised from 930 psig to 2020 psig.

- Safety Injection (SI) is actuated.

Which ONE of the following identifies the effect on the ECCS Accumulator isolation valves:

1. during PRZR pressure increase to 2020 psig, and

2. the SI signal

	PRZR Pressure Increase	SI signal
A.	valves remain closed.	Open signal is generated to the valves.
B.	valves remain closed.	Open signal is NOT generated to the valves.
C.✓	valves automatically open.	Open signal is generated to the valves.
D.	valves automatically open.	Open signal is NOT generated to the valves.

K/A

006 Emergency Core Cooling System (ECCS)

K6.02 Knowledge of the effect of a loss or malfunction on the following will have on the ECCS:

Core flood tanks (accumulators).

K/A MATCH ANALYSIS

The question presents a plausible scenario where ECCS Accumulator isolation valves are inadvertently left closed as RCS / PRZR pressure is raised to 2020 psig. Then, a manual SI signal is generated. The student must determine that the isolation valves would have opened on P-11 (2000 psig) prior to receipt of the SI signal. Also, the student must determine a manual SI signal would have generated an open signal to the valves.

ANSWER / DISTRACTOR ANALYSIS

choice is incorrect. The second part of the choice is correct as an open signal is generated on receipt of SI signal. Plausible the students may not realize P-11 sends an open signal and know that an SI signal does.

B. Incorrect. Valves will automatically open on P-11 (2000 psig) so this part of the choice is incorrect. The second part of the choice is incorrect as an open signal is generated on receipt of SI signal. Plausible the students may not realize P-11 or SI sends an open signal. These valves are usually de-energized and open prior to 1000 psig in the RCS.

C. Correct. Pressure > P-11 (2000 psig) and an SI signal will cause the valves to open by system design.

D. Incorrect. Valves will automatically open on P-11 (2000 psig) so this part of the choice is correct. The second part of the choice is incorrect as an open signal is generated on receipt of SI signal. Plausible the students may not realize P-11 or SI sends an open signal.These valves are usually de-energized and open prior to 1000 psig in the RCS.

REFERENCES

Watts Bar 2008 NRC RO Exam question # 31 used as base for modification (included).

V-LO-PP-13101, Emergency Core Cooling System (ECCS) slides # 112 and # 113. (included).

V-LO-PP-28103, Reactor Trip and ESFAS Signals, slides # 47 and # 51 (included).

VEGP learning objectives:

LO-PP-13101-12, Describe the control logic for the accumulator isolation valves in response to:

a. SI signal

b. Permissive P-11.

WRITTEN QUESTION DATA SHEET

Question Number: 31

K/A: 006 K6.02
Emergency Core Cooling
Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Core flood tanks (accumulators).

Tier:	2	RO Imp:	3.4	RO Exam:	Yes	Cognitive Level:	HIGH
Group:	1	SRO Imp:	3.9	SRO Exam:	Yes	Source:	NEW

Applicable 10CFR55 Section: 41.7/45.7
Learning Objective: 3-OT-SYS063A Objective 24: Given a set of plant conditions, determine the correct response of the Emergency Core Cooling System.
References: 1-47W611-63-7, Rev 2.

Question:

Given the following plant conditions:

- Plant startup is in progress.
- During performance of GO-1, Unit Startup from Cold Shutdown to Hot Standby, the CLA isolation valves were left CLOSED with power on the valves as pressurizer pressure was raised from 900 psig to 1900 psig.
- A manual safety injection (SI) is initiated.

Which ONE of the following identifies the position of the CLA isolation valves before the SI is initiated and how the MANUAL Safety injection will affect the valves?

	Before SI	Effect of the SI signal
A.	Valves will have automatically opened.	An open signal will be generated to the valves.
B.	Valves will have automatically opened.	An open signal will NOT be generated to the valves.
C.	Valves will have remained closed.	An open signal will be generated to the valves.
D.	Valves will have remained closed.	An open signal will NOT be generated to the valves.

DISTRACTOR ANALYSIS

a. Incorrect. The valves would not have automatically opened prior to the SI because the P-11 permissive has not been made. However, an open signal would be generated by the SI. Plausible because the valves do automatically open if pressure is greater than P-11, and an SI would generate an open signal.
b. Incorrect. The valves would not have automatically opened prior to the SI because the P-11 permissive has not been made and an open signal would be generated by the SI. Plausible because the valves do automatically open if pressure is greater than P-11 and the candidate could conclude that since the valves are normally opened manually and power removed that the SI does not generate an open signal due to the CLAs being a passive sub-system in the ECCS.
c. CORRECT. The valves would be closed until the pressure rose above P-11 (1970 psig). When P-11 permissive was met the valves would then automatically open. With pressure at 1900 psig the valves would still be closed, but would open when the SI was actuated.
d. Incorrect. The valves would have remained closed because the P-11 permissive has not been made and an open signal would be generated by the SI. Plausible because the valves would remain closed with the pressure less than P-11 and the candidate could conclude that since the valves are normally opened manually and power removed that the SI does not generate an open signal due to the CLAs being a passive sub-system in the ECCS.

Question used as base for modification, stem changed to change answer.

ACCUMULATOR OUTLET MOV INTERLOCKS

If valves have been powered up and are SHUT, they will automatically OPEN when:

- **P-11, or**
- **SI**

However, they are normally de-energized open.


ACCUM
ACCUM-I N2
SUPPLY/VENT VLV
CLOSE
OPEN
I-HV-8875A
IHS-8875A
A
ACCUM-I N2
SUPPLY/VENT VLV
CLOSE
OPEN
I-HV-8875E
IHS-8875P
B
ACCUM-I WTR
FILL VLV
CLOSE
OPEN
I-HV-8878A
IHS-8878A
SI
ACCUM-I
CLOSE
OPEN
I-HV-8808A
IABE-19
IHS-8808A
A
ACCUM-2 N2
SUPPLY/VENT VLV
OPEN
I-HV-8875B
IHS-8875B
A
ACCUM-2 N2
SUPPLY/VENT VLV
OPEN
I-HV-8875F
IHS-8875S
B
ACCUM-2 WTR
FILL VLV
CLOSE
OPEN
I-HV-8878B
IHS-8878B
SI
ACCUM-2
ISO VLV
AU
CLOSE
OPEN
IBBC-19
IHS-8808B
B

What will produce a P-11 signal?

2/3 pressurizer pressure inst. ≤ 2000 psig.

What is the function of P-11?

-Allows manual block of pressurizer low pressure SI.

-Allows manual block of low steam line pressure SI and SLI.

V-LO-PP-28103-6.2 47

What automatically occurs when Pressurizer pressure increases above 2000 psig?

- SI Accumulator Outlet Valves receive an Open signal.
- Enables alarms “ACCUM TANK 1-4 ISO VLV 8808A-D NOT FULLY OPEN”
- Enables “RWST TO SI PUMP ISOLATION VALVE 8806 NOT FULL OPEN” alarm

V-LO-PP-28103-6.2 51

Point out to the students that the Accumulator outlet valves are normally de-energized in the open position when at power.

Caution: >P-11

When pressurizer pressure goes above the P-11 set point the SI signal from low pressurizer pressure and the SI and SLI from low steam line pressure automatically unblock.

10. 007EG2.4.31 002/1/1/RX TRIP-ALARMS/ARPS/C/A - 4.2 / 4.1/NEW/HL-15R NRC/RO/TNT/ DS

Given the following plant conditions:

- The plant tripped from 20% power during a loss of grid event.
- All RCS FLOW TRIP 90% bistable lights are LIT.
- The following First Out annunciator is "gallop flashing".

LOW FLOW / RCP / P7 PERMISSIVE REACTOR TRIP

- All RCP 1E and non-1E handswitch red lights are LIT.
- 19001-C, "ES-0.1 Reactor Trip Response" is in effect.

Which **ONE** of the following is **CORRECT** regarding the:

1) event which caused the reactor trip and

2) indications the operators will use to monitor RCS temperature?

	Cause of Reactor Trip	**Temperature monitoring**
A.	RCP underfrequency	RCS AVERAGE TEMPERATURE
B.	RCP underfrequency	RCS WR COLD LEG TEMPERATURES
C.	RCS two loop low flow	RCS AVERAGE TEMPERATURE
D✓	RCS two loop low flow	RCS WR COLD LEG TEMPERATURES

K/A

007 Reactor Trip - Stabilization - Recovery

EG2.4.31 Knowledge of annunciator alarms, indications, or response procedures

K/A MATCH ANALYSIS

The question presents a plausible scenario where a grid disturbance has resulted in a reactor trip from power. The student must determine from given indications the cause of the reactor trip (2 loop low flow) and the proper method to monitor RCS temperature per 19001-C (RCS WR Tc's). Annunciator alarms, control board indications, and Rx. Trip Response procedure 19001-C are used in the question stem.

At 20% power as given in the stem, the 13.8 kV electrical buses are powered from the

RATs (offsite power) so a loss of offsite power would lead to an RCS Lo Flow Trip when > P-7. At 20% power, the reactor is > P-7 (10% set point).

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RCP underfrequency would cause the low flow bistables to be LIT. However, RCP underfrequency would cause all the RCP breakers to trip open so RCP UF is incorrect. RCS Tave is plausible as it is the normal parameter used to monitor RCS temperature, however, with no RCPs, 19001-C specifies RCS WR Tc's.

B. Incorrect. RCP underfrequency would cause the low flow bistables to be LIT. However, RCP underfrequency would cause all the RCP breakers to trip open so RCP UF is incorrect. RCS WR Tc's is the correct parameter to use per 19001-C.

C. Incorrect. Two loop low flow is correct. RCS Tave is plausible as it is the normal parameter used to monitor RCS temperature, however, with no RCPs, 19001-C specifies RCS WR Tc's.

D. Correct. Two loop low flow would cause the indications and alarms. RCS WR Tc's are the parameter to monitor per 19001-C.

REFERENCES

V-LO-PP-28103, Reactor Trip and ESFAS signals slides # 116, 117, and 120 (included).

V-LO-TX-28101, Reactor Protection System section B for Reactor Trip and ESFAS signals, page # 27 (included).

V-LO-PP-16101, RCS Temperature Instrumentation slide # 28 (included).

V-LO-PP-16401, Reactor Coolant Pumps slide # 32 and blowup of slide # 32 (included).

V-LO-TX-16001, Primary Systems page # 33 for RCP UV included.

19001-C, ES-0.1 Reactor Trip Response step # 4 (included).

17009-1, Annunciator Response Procedure for ALB 09 on Panel 1C1 on MCB, window E03 (included, note - this is the first out panel)

12004-C, Power Operation (Mode 1) steps 4.1.39 from power ascent section (included) and 4.2.10 from power descent section (included).

VEGP learning objectives:

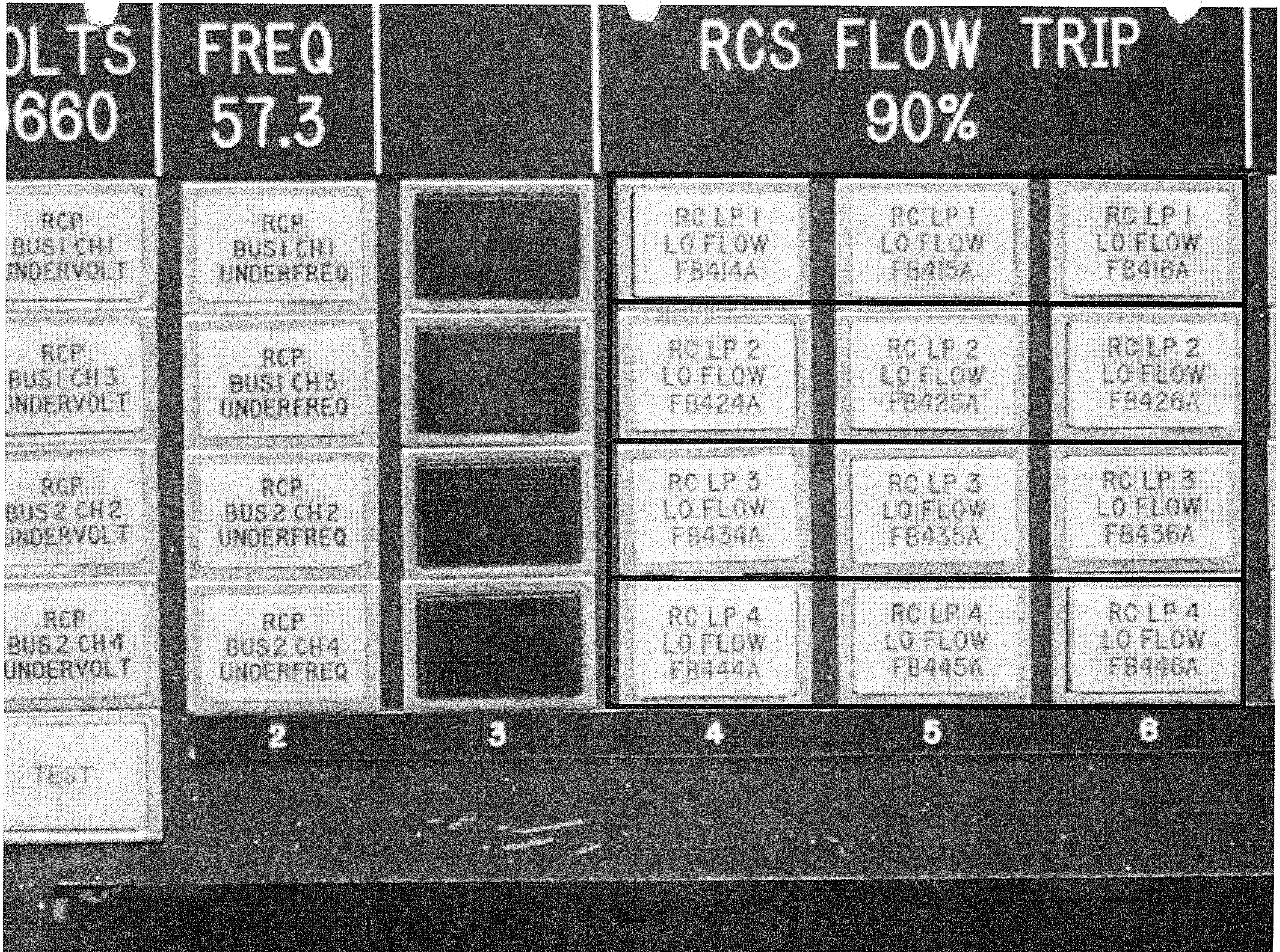
LO-PP-28103-03, List all reactor trip signals, set points, coincidences, permissives, and blocks.

LO-PP-16101-03, State the conditions when wide range temperature indications must be used instead of narrow range instrumentation.

LO-PP-16401-09, Describe the following for the RCP supply breakers.

c. Protection features.

LO-PP-61203-01, Describe the basic steps involved with transfer of the 13.8 kV and 4160 kV buses from the Reserve Auxiliary Transformers (RATs) to the Unit Auxiliary Transformers (UATs)



V-LO-PP-28103 slide # 116 Stem mentions RCS FLOW TRIP 90% bs lights

TWO LOOP LOW FLOW TRIP

2/3 CHANNELS $\leq$ 90% ON 2/4 LOOPS.

AUTO BLOCKED BELOW P-7.

BASES: DNB PROTECTION.

RCP UF TRIP

1/2 13.8 BUSSES ≤ 57.3 HZ

AUTO BLOCKED BELOW P-7

WILL ALSO TRIP ALL RCPs

BASES: DNB PROTECTION.

V-LO-PP-28103-6.2 120

The bases behind tripping the RCPs during an under frequency condition is that the inertia of the flywheel on each RCP provides enough forced flow for decay heat removal. This holds true if the RCPs are at their normal speed before the trip occurs. The assumptions are that if the RCP frequency is lowering, then eventually the RCPs are going to trip anyway. So the reactor and RCPs are tripped early to ensure proper decay heat removal is provided.

Plausible on a grid disturbance to get an RCP UF Trip. However stem gave breakers closed (Red lights LIT). So, trip could NOT have been from UF.

Bases: Prevents DNB conditions resulting from loss of one or more reactor coolant pumps.

13) **Two loop loss of flow** **2** out of **3** channels **= 90% flow ****
2 out of **4** loops
enabled above P-7; ≥ 10% reactor power

Plant at 20% in Q stem

Bases: Prevents DNB conditions resulting from loss of two or more reactor coolant pumps.

14) **RCP Under Voltage Reactor Trip** **1** out of **2** buses **= 9660 V**
enabled above P-7; ≥ 10% reactor power

Bases: Provides protection against DNB as a result of loss of forced coolant flow. Ensures a reactor trip signal occurs before the low flow trip set point is reached.

15) **RCP Under Freq. Reactor Trip** **1** out of **2** buses = **57.3 Hz**
enabled above P-7; ≥ 10% reactor power

Makes RCP UV plausible but NOT correct as brkrs trip

Also trips all RCPs feeder breakers. The bases behind tripping the RCPs during an under frequency condition is: The inertia of the flywheel on each RCP provides enough for force flow for decay heat removal. This hold true if the RCPs are at their normal speed before the trip occurs. The assumptions are that if the RCP frequency is lowering that eventually the RCPs are going to trip anyway. So the reactor and RCPs are tripped early to ensure proper decay heat removal is provided.

Bases: Provides protection against DNB as a result of loss of forced coolant flow. Ensures a reactor trip signal occurs before the low flow trip set point is reached.

16) **Safety Injection Reactor Trip** Any Safety Injection Signal

Bases: Ensures subcriticality during accident conditions. The ECCS only rated for decay heat removal only.

17) **Turbine Trip/Reactor Trip** **4** out of **4** Main = **96.7 % open**
Turbine Stop Valves
enabled above P-9; ≥ 40% reactor power
-or-
2 out of **3** ETS header = **580 psig**
pressure channels low
enabled above P-9; ≥ 40% reactor power

Bases: Assures reactor trip upon reduction of turbine power in excess of that which can be handled by steam dumps and rod control system, including single failures of steam dump or pressurizer spray valve controls.

18) **S/G Low-Low level reactor trip** **2** out of **4** channels = **38%**
on **1** out of **4** S/Gs

Bases: Protects the reactor from loss of heat sink by tripping with sufficient water level to allow for starting delays of Auxiliary Feed Water System.

RCS Wide Range Temperature

Includes hot leg and cold leg temperature indications

1 each per loop, 2 total per loop (no spare RTD) 0-700 °F Range

Measured with RTD's located in dry thermowells

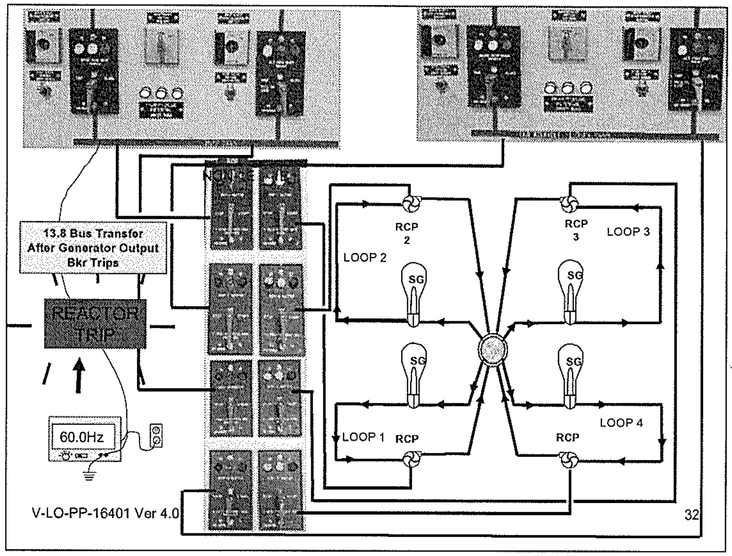
Provide protection and minimizes system leakage

Response fairly slow to change in RCS temperature under natural circulation conditions

Used when off scale on narrow range instruments (e.g., plant cool down) or during loss of forced RCS loop flow, or where NR inst. otherwise unavailable

Use WR vs NR when RCS loses Forced Flow,

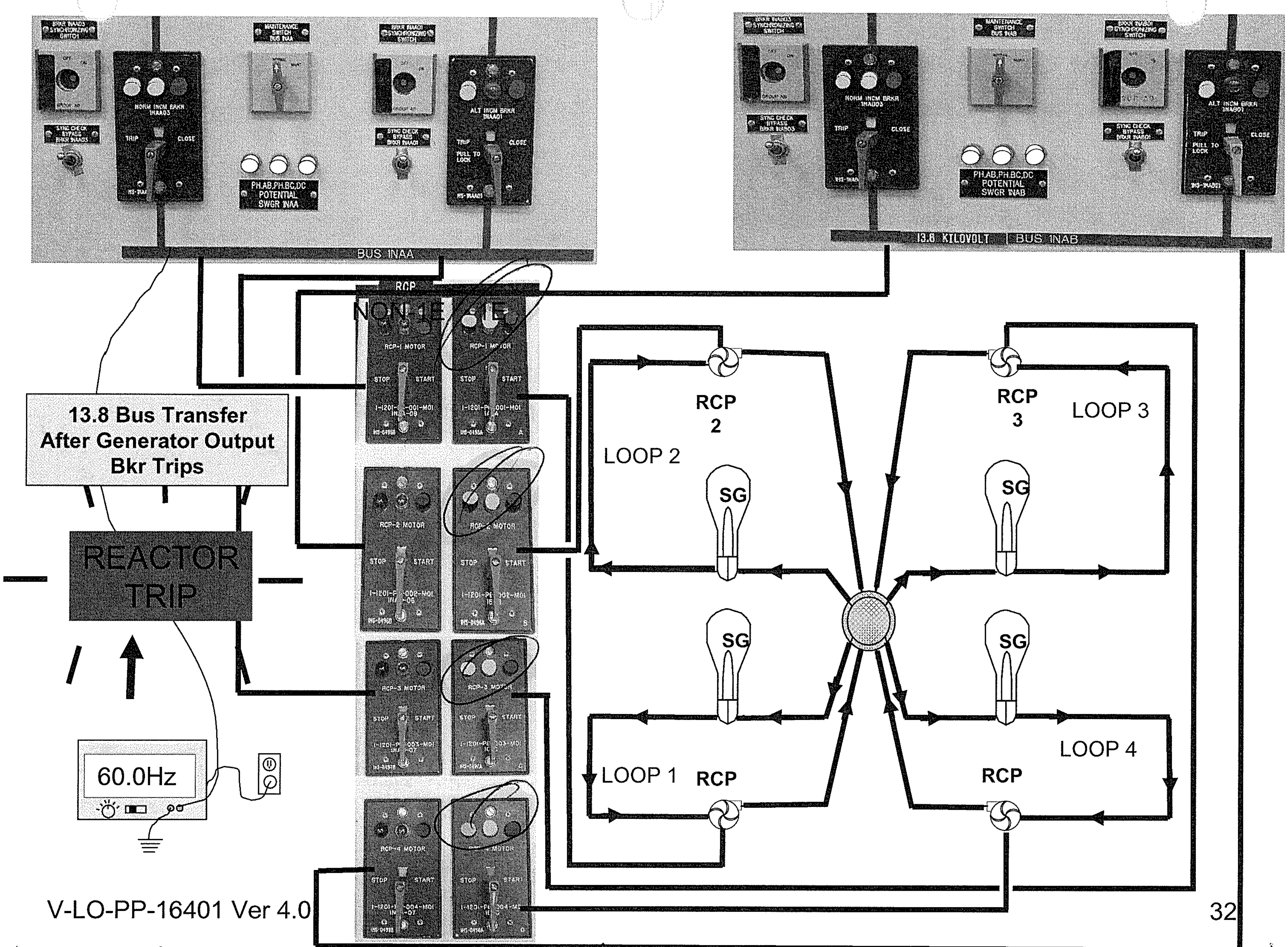
W.R. Temperature indication will continue to be representative of RCS temperature (partial) in the loop at the RTD thermowells



This drawing shows the frequency reading coming off the 13.8kV Bus for simplicity. In reality, the taps are similar to the UV taps in the previous slide. Where a combination of UF on [(1 OR 2) AND (3 OR 4)] >P7 will give the RX trip and the opening of the 1E breakers. The drop in frequency would decrease RCP rotor power output and the developed torque necessary to supply rated flow.

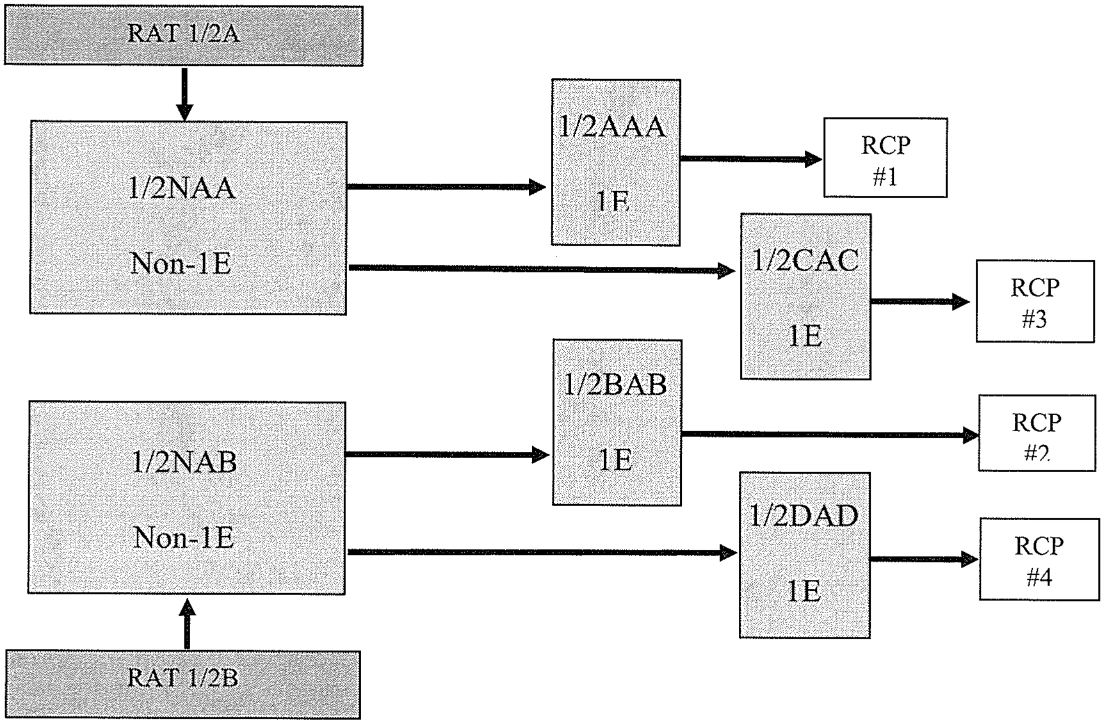
A very important note to the students: Watch how only the 1-E breakers open after the Under frequency trip. You could also talk about the flywheel on the RCP's to keep them pumping momentarily. In the drawing I left then running a little while after the breakers open.

RCP breaker trip open on UF.



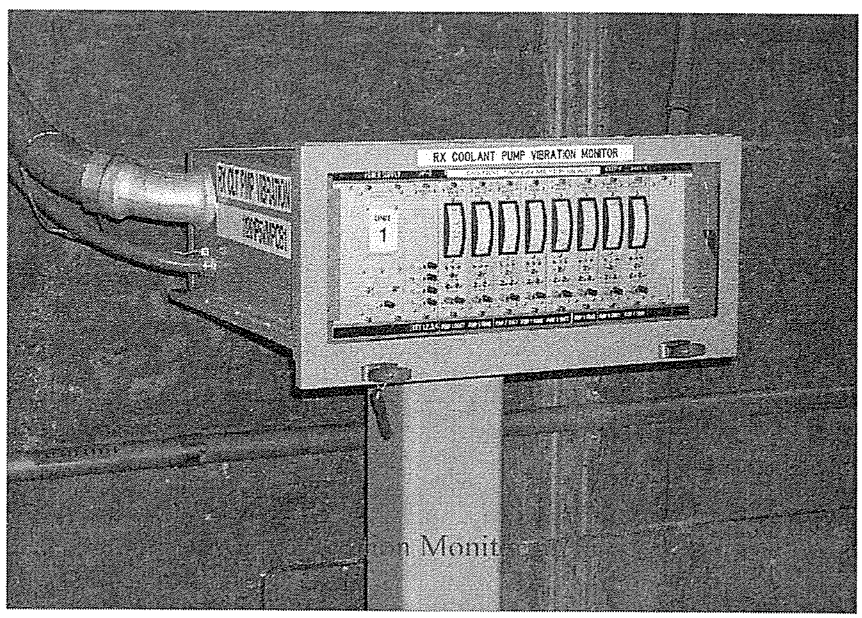
Blowup of photo on slide #32. Right hand colum of lights (1E handswitches are green and amber).

or phase differential current. The reactor coolant pump class 1E motor breakers (HS-0495A, 0496A, 0497A, 0498A) receive their control power from 125 VDC ESF busses train A, B, C, and D, respectively. Their breakers will automatically trip on under frequency or instantaneous or time delay over current.



16.23 VIBRATION MONITORING

Each RCP is equipped with both “Frame” and “Shaft” vibration monitoring. The RCP frame vibration monitors consist of two probes that are mounted 908 apart on the top of each RCP motor frame. The RCP shaft vibration monitors are measured by a vertical and horizontal proximity probe mounted parallel and perpendicular respectively to the pump discharge at a location near the pump coupling. Both frame and shaft vibrations are continuously monitored. Alarms are generated in the control room if the frame vibration exceeds 3 mils and 5 mils. At 5 mils the operators are required to trip the associated RCP. High frame vibrations are indicative of a misalignment or an out of balance condition. Shaft vibration alarms are also generated if they exceed 15 mils and 20 mils. At 20 mils the operators are required to trip the associated RCP. High shaft vibrations are indicative of a possible bearing failure. Local monitoring at the RCP vibration panel is required to determine which RCP has a vibration problem. The Control room is only provided with common alarms. During normal plant



ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*4. **Check RCS temperature stable at or trending to 557°F.**	*4. IF temperature is less than 557°F and lowering, THEN perform the following as necessary:
__With RCP(s) running - RCS AVERAGE TEMPERATURE.	
-OR-	__a. Stop dumping steam.
__Without RCP(s) running - RCS WR COLD LEG TEMPERATURES.	b. Perform the following as appropriate:
	__ IF at least one SG NR level greater than 10%, THEN lower total feed flow.
	-OR-
	__ IF all SG NR levels less than 10%, THEN lower total feed flow to NOT less than 570 gpm.
	__c. IF cooldown continues, THEN close MSIVs and BSIVs.
	__d. IF temperature less than 557°F and NOT trending to 557°F, THEN borate as necessary to maintain shutdown margin by initiating 13009, CVCS REACTOR MAKUP CONTROL SYSTEM.
	__e. IF temperature greater than 557°F and rising, THEN dump steam.

No RCP's are running so RCS Tc should be used.

Tave is plausible since this is what we normally use.

WINDOW E03

ORIGIN	SETPOINT	LOW FLOW/RCP/P7 PERMISSIVE REACTOR TRIP
2 of 4 RCP Loops Low Flow when above P7	90% of normal flow	

1.0 **PROBABLE CAUSE**

Loss of all offsite power would cause this annunciator,

1. Loss of 13.8kV bus 1NAA or 1NAB

2. Two or more Reactor Coolant Pump Breakers tripped.

2.0 **AUTOMATIC ACTIONS**

NOTE

This trip function is blocked below the P-7 permissive.

1. Reactor Trip.

2. Turbine Trip.

3. Feedwater Isolation if Tavg is less than 564°F.

4. Steam Dump Armed.

3.0 **INITIAL OPERATOR ACTIONS**

Go to 19000-C, "E-O Reactor Trip or Safety Injection".

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: FSAR Section 7.2, 1X6AA02-229, PLS

INITIALS

l. IF HDT High Level Dump Valves are in MANUAL, **restore** HDT Level Control to NORMAL. ________

m. If necessary, **adjust** SGBD Condensate Return Temperatures for the present power level per Step 4.1.10b. ________

n. IF in service, **shut down** Feed Water preheating using the 5th Stage Feed Water heaters per 13615, "Condensate and Feed Water System." ________

o. **Start up** and **test** the second Main Feed Pump per 13615, "Condensate And Feed Water System" up through and including performance of 14993, "Steam Generator Feed Pump Turbine Lube Oil System Test." and 14992, "MFPT Trip Mechanism Test." ________

4.1.39 BETWEEN 30 and 50% Reactor Power, perform the following:

Raising power, xfer to UATS when 30-50%, <30% bus on RATS. LOSP ~~would~~ de-energ 13.8KV buses causing loss power to RCPs and a loss ~~of~~ RCS Flow,

a. At SS direction, **transfer** the 13.8kV busses per 13420, "13.8kV AC Electrical Distribution System."

b. At SS direction, **transfer** the 4160V AC busses per 13425 A/B/C, "4160V AC Non 1E Electrical Distribution System."

c. **Verify** I&C has **reset** the PR high level trip bistables (NC306) for at least 3 of the 4 power range channels, per Step 4.1.38.i, prior to exceeding 42% reactor power. ________

d. **Check** PREFERRED LINE lamp is lit, on the EHC STATIC TRANSFER SWITCH 1615-D3-001 in CB Rooms A-78(U1) and A-80 (U2). IF PREFERRED LINE lamp is NOT lit, **notify** System Engineer. ________

4.1.40 IF NOT required, **isolate** the Auxiliary Steam header per 13761-C, "Auxiliary Steam System." ________

INITIALS

4.2.10 BETWEEN 50% and 30% Turbine Power, perform the following:

a. **Remove** one Main Feed Pump from service per 13615, "Condensate And Feed Water System." ________

b. **Transfer** the 13.8kV busses from the Unit Auxiliary Transformers to the Reserve Auxiliary Transformers per 13420, "13.8kV AC Electrical Distribution System." ________

Same comment as step 4.1.39

c. **Transfer** the 4160V AC Non 1E busses to the Reserve Auxiliary Transformers per 13425 A/B/C, "4160V AC Non 1E Electrical Distribution System." ________

d. **Monitor** Circulating Water Tower Basin level during the power decrease. ________

e. **Adjust** Tower blowdown and/or Tower makeup as required to **control** basin level per 13724, "Circulating Water System." ________

4.2.11 WHEN between 50% power and 20% power AND WHEN a calorimetric is performed per 14030, "Nuclear Instrument Calorimetric Calibration", IF any PR NIs are adjusted in the downward direction, **notify** I&C to **adjust** the PR high level trip bistables (NC306) for channels N41, N42, N43 and N44 to =90%.

________________ / ________ / ________
Person Contacted Date Time ________

4.2.12 **Maintain** operation of the Condensate Demineralizer System per 13616, "Condensate Filter Demineralizer System."

4.2.13 At approximately 46% Reactor Power, **verify** 1 LP LO FL TRIP BLKD P-8 illuminates. ________

4.2.14 At approximately 38% Reactor Power, **verify** the TURB TRIP/RX-TRIP BLOCKED P-9 illuminates. ________

4.2.15 At approximately 37% Turbine power, **verify** C20 clears (automatically blocks AMSAC - ATWAS Mitigation System Actuation Circuitry) by **observing** permissive light AMSAC BYPASSED LO TURBINE LOAD C20 illuminates. ________

11. 007K1.03 001/2/1/PRT-RCS/C/A- 3.0 / 3.2/B-SEQUOYAH 2007/HL-15R NRC/RO/TNT/DS

Given the following plant conditions:

- A reactor trip has occurred.

- RCS pressure is 1830 psig and lowering.

- Containment pressure is 2.3 psig and rising.

Which **ONE** of the following describes the flow path of the RCPs # 1 seal leakoffs?

Seal leakoff flows are currently directed to the...

A. VCT

B.✓ PRT

C. RCDT

D. CTMT sump

HL-15R RO NRC Exam

K/A

007 **Pressurizer Relief Tank/Quench Tank System (PRTS)**

K1.03 **Knowledge of the physical connections and/or cause effect relationships between the PRTS and the following systems:**

RCS.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a reactor trip and SI have occurred on low PRZR pressure. The candidate must choose the correct RCP # 1 seal leakoff flow path.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RCP # 1 seal leakoff normally flows to the VCT but on SI due to low PRZR pressure CIA will isolate this flow path and the leakoff will flow to the PRT via a relief valve.

B. Correct. RCP # 1 seal leakoff normally flows to the VCT but on SI due to low PRZR pressure CIA will isolate this flow path and the leakoff will flow to the PRT via a relief valve.

C. Incorrect. RCP # 1 seal leakoff is plausible to align to the RCDT but this is a manual operation on the QMCB. No indication a manual alignment has been performed is provided in the stem.

D. Incorrect. RCP # 3 seal leakoff normally flows to the Containment sumps and sometimes on seal failures # 2 and # 1 seals could flow to the sumps so this path is plausible. However, no indications of seal failure are provided in the stem.

REFERENCES

Sequoyah 2007 NRC Exam question # 11

V-LO-PP-16401, Reactor Coolant Pumps

V-LO-PP-09200, CVCS Charging System

VEGP learning objectives:

V-LO-PP-16401-04, State the effects of closing the # 1 seal leakoff valve.

11. Given the following plant conditions:

    - A reactor trip has occurred.
    - RCS pressure is 1810 psig and lowering.
    - Containment Pressure is 1.5 psig and rising.

    Which ONE (1) of the following describes the status of RCP #1 seal leakoff?

    Directed to...

    A. VCT

    B. PRT

    C. RCDT

    D. Reactor Building Floor and Equipment Sump

Bank

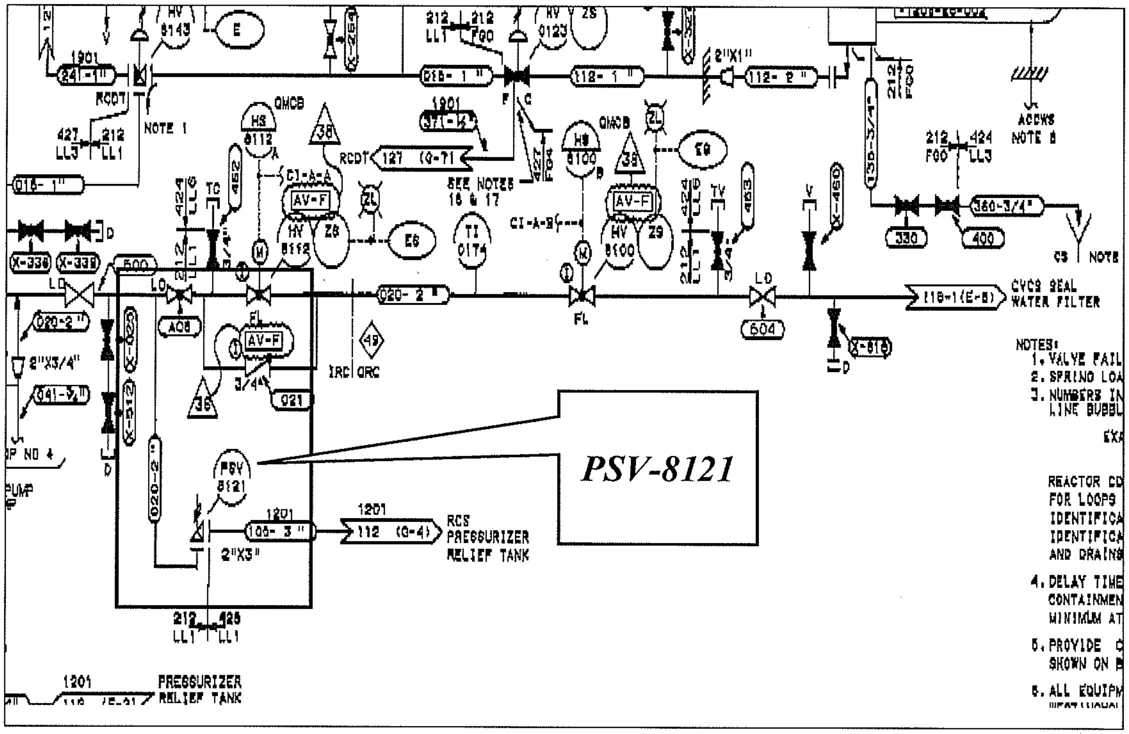
Sequoyah 2007 RO NRC Exam.

Only question I could find that met KA.

Note: Valves close automatically on train-related Containment Isolation Signal (CIA). A CIA signal is generated every time by a Safety Injection Signal. Therefore, a SI signal will cause a CIA signal, which will result in RCP #1 seal leakoff flow path being isolated. Overpressurization is prevented by the relief valve PSV-8121.

↓

Relief to PRT.



PSV-8121 located inside containment prevents over-pressurizing the seal return line if CIA valves located on the line isolate and seal injection flow remains in service. Setpoint is 150 psig. Relieves to PRT

Relieves to PRT

12. 007K4.01 001/2/1/PRT-COOLING/C/A - 2.6/2.9/NEW/HL-15R NRC/RO/DS / TNT

Given the following:

PRT temperature high due to leakage from a PRZR PORV
The unit is at 100% power
All systems are in their normal alignment

The PRT ____(1)_____ directly cooled by ___(2)____.

A. (1) is automatically; (2) ACCW via the RCDT heat exchanger

B. (1) is automatically; (2) NSCW via the RCDT heat exchanger

C.✓ (1) must be manually aligned to be; (2) ACCW with the RCDT heat exchanger

D. (1) must be manually aligned to be; (2) NSCW with the RCDT heat exchanger

HL-15R RO NRC Exam

K/A

007 Pressurizer Relief Tank/Quench Tank System (PRTS)

K4.01 Knowledge of the PRTS design feature(s) and/or interlock(s) which provide for the following:

Quench tank cooling.

K/A MATCH ANALYSIS

The question requires the student to identify if PRT cooling is automatically or manually aligned and what the cooling meduim is, which meets the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Normal at power PRT alignment is with the recirculation and fill valves shut. With these two valves closed cooling to the PRT must be manually aligned. ACCW is used to cool the PRT contents via the RCDT heat exchanger.

B. Incorrect. Normal at power PRT alignment is with the recirculation and fill valves shut. With these two valves closed cooling to the PRT must be manually aligned. ACCW is used to cool the PRT contents via the RCDT heat exchanger. ACCW rejects it heat load to the NSCW system making this cooling choice plausible.

C. Correct. Normal at power PRT alignment is with the recirculation and fill valves shut. With these two valves closed cooling to the PRT must be manually aligned. ACCW is used to cool the PRT contents via the RCDT heat exchanger.

D. Incorrect. Normal at power PRT alignment is with the recirculation and fill valves shut. With these two valves closed cooling to the PRT must be manually aligned. ACCW is used to cool the PRT contents via the RCDT heat exchanger. ACCW rejects it heat load to the NSCW system making this cooling choice plausible.

REFERENCES

13004-1, "Pressurizer Relief Tank Operation" pages 3, 27, and 28.

P&ID drawing 1X4DB-112 for the PRT showing the normal at power valve alignments to the PRT.

VEGP learning objectives:

LO-PP-16301-09:

Describe the methods for cooling the PRT.

3.0 PREREQUISITES AND INITIAL CONDITIONS

3.1 The Gaseous Waste Processing System is available to provide processing of gases from the PRT.

3.2 The Auxiliary Gas System-Nitrogen or nitrogen from the Waste Gas Decay Shutdown Tank is available to provide a nitrogen blanket for the PRT.

3.3 Reactor Make-Up Water is available to provide a cooling spray for the PRT.

3.4 ACCW is available if cooling the PRT with the RCDT Heat Exchanger.

INITIALS

4.4.3 PRT Cooldown Using Spray And Drain (One Hour Cooldown)

> NOTE
>
> Two methods for cooling the PRT exist. Cooling the PRT by spray and drain is designed to cool the PRT in 1 hour. This method uses makeup water and drains to the Waste Processing System. Cooling the PRT by recirculation through the RCDT Hx is designed to cool the PRT in 8 hours. This method minimizes makeup water use and waste processing of liquid. The time required to cool the PRT and water usage should be considered before deciding which method to use.

4.4.3.1 **Establish** communications between the Liquid Waste Processing System Panel (WPSL) and the Control Room. ________

4.4.3.2 **Verify** the PRT pressure less than or equal to 50 psig as indicated by PRESSURIZER RELIEF TANK 1-PI-0469 to prevent RCDT System over pressurization. ________

4.4.3.3 **Verify** open WPSL RCDT PUMPS DISCH TO RECYC EVAP 1-1901-U6-327. ________

4.4.3.4 **Realign** RCDT Pump Suction to the PRT and **initiate** spray as follows:

a. **Stop** the running REACTOR COOLANT DRAIN TANK PUMP

#1 1HS-1003A (WPSL) ________

#2 1HS-1003B (WPSL) ________

> **CAUTION**
>
> The RCDT level should be monitored to prevent tank flooding.

b. **Place** REACTOR COOLANT DRAIN TANK LEVEL 1-LC-1003 in MANUAL and **open** the valve (WPSL). ________

c. **Close** REACTOR COOLANT DRAIN TANK RECIRCULATION VALVE 1-HV-7144 (WPSL). ________

INITIALS

d. **Close** REACTOR COOLANT DRAIN TANK PUMP SUCTION VALVE 1-HV-7127 (WPSL). ________

e. **Open** PRT FILL ISO VLV 1-HV-8030. ________

f. **Open** PRT RECIRC ISO VLV 1-HV-8031. ________

4.4.3.5 **Initiate** PRT drain and **maintain** level and pressure as follows:

a. **Verify** ACCW is available to the RCDT Heat Exchanger. ________

b. **Start** REACTOR COOLANT DRAIN TANK PUMP

#1 1HS-1003A (WPSL) ________

#2 1HS-1003B (WPSL) ________

c. During cooldown, **maintain** PRT level greater than or equal to 58% as indicated by PRESSURIZER RELIEF TANK 1-LI-0470. ________

CAUTION

During the PRT cooldown, cooling of the liquid will cause a corresponding pressure decrease.

d. During cooldown, **maintain** PRT N2 pressure at 3 to 5 psig as indicated by PRESSURIZER RELIEF TANK 1-PI-0469. ________

4.4.3.6 At a PRT temperature of 110°F as indicated by PRESSURIZER RELIEF TANK 1-TI-0468, **secure** spray and **realign** RCDT System to normal as follows:

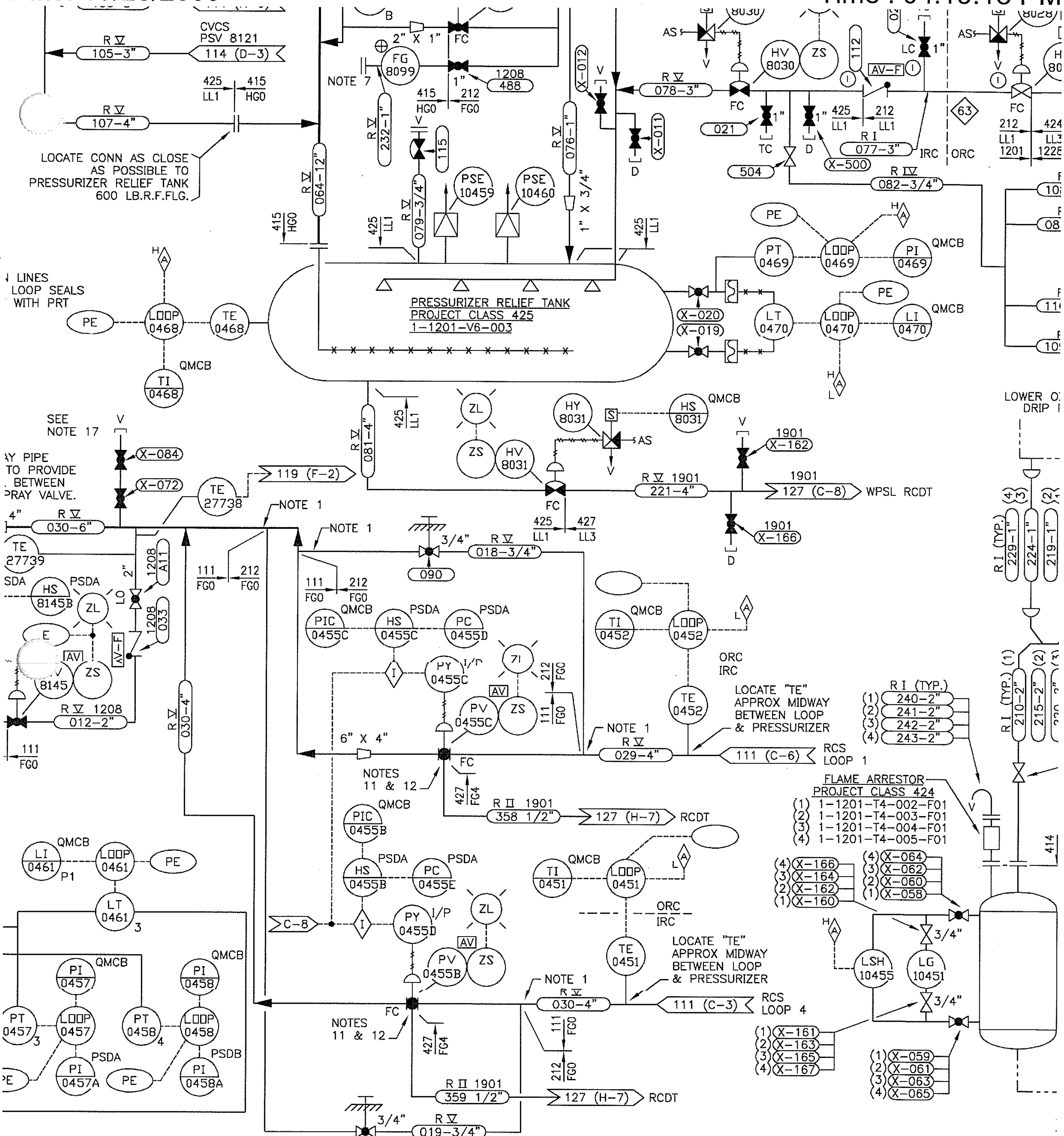
a. **Close** the PRT FILL ISO VLV 1-HV-8030. ________

b. **Stop** the running REACTOR COOLANT DRAIN TANK PUMP

#1 1HS-1003A (WPSL). ________

#2 1HS-1003B (WPSL). ________

c. **Close** PRT RECIRC ISO VLV 1-HV-8031. ________



NOTES:

1. PROVIDE 0.375" ID RESTRICTION AS SHOWN ON BPC DWG NO. CX4DG001.

2. FOR TYPICAL MAKE-UP PIPING ARRANGEMENT TO RCP STAND-PIPE SEE DWG 1X4DB114 (D-6).

3. ALL EQUIPMENT AND VALVES TO BE FURNISHED BY WESTING-HOUSE EXCEPT THE FOLLOWING:
   A. ALL PIPING AND MANUAL VALVES 2" AND UNDER (SURGE LINE, PRESSURIZER RELIEF, AND LOOP SEAL PIPING FURNISHED BY WESTINGHOUSE).
   B. SAMPLE VESSEL AND ATTACHED PIPING FITTINGS

QMCB
TI 0453

NOTES – CONT.:

9. LINE BETWEEN PRESSURIZER AND VALVE, INCLUDING THE VALVE IS PROJECT CLASS 212.

10. DELETED.

11. PROVIDE 12" REMOVAL SPOOL.

12. FOR TYPICAL DETAL SEE STD. DWG. AX4DD000.

13. PROVIDE 0.191 ID RESTRICTION AS SHOWN ON WESTINGHOUSE DWG. NO. 1548E34 SH. 1-4 (V.P. NO. X6AB17-134 THRU 137).

REACTOR COOLAN
LUBE OIL DRAIN
PROJECT CLAS
(1) 1-1201-T
(2) 1-1201-T
(3) 1-1201-T
(4) 1-1201-T

REACTOR COOLA

13. 008AA1.04 001/1/1/PRZR VAPOR-FW PUMPS/C/A - 2.8 / 2.5/NEW/HL-15R NRC/RO/TNT/ DS

Given the following sequence of events:

- The reactor is tripped due to a PRZR Safety valve failing open.

- Several minutes later Tave lowers to 557°F.

- PRZR pressure drops to 1830 psig and is now slowly rising.

The UO notes the following:

- Both MFPTs are tripped and all FWI valves are closed.

Which **ONE** of the following choices CORRECTLY lists the **first** initiating signals for the FWI valves closure and the MFPTs trip?

	FWI Valves Closure	MFPTs Trip
A.	Low PRZR pressure SI	Low PRZR pressure SI
B.✓	Reactor trip coincident with low Tave	Low PRZR pressure SI
C.	Low PRZR pressure SI	Reactor trip coincident with low Tave
D.	Reactor trip coincident with low Tave	Reactor trip coincident with low Tave

K/A

008 Pressurizer Vapor Space Accident

AA1.04 Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident:

Feedwater pumps

K/A MATCH ANALYSIS

The question presents a plausible scenario where a PRZR Safety Valve has lifted. The student has to diagnose that an SI would have occurred from given conditions and also determine the reasons for the trip of the MFPTs and the closure of the FWI valves.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. The MFPTs would have tripped on low PRZR pressure SI, the FWI valves would have closed on P-4 with Lo Tave.

B. Correct. P-4 with Lo Tave will cause the FWI valves to close, SI will cause the

C. Incorrect. While PRZR lo pressure SI would cause a FWI closure and is plausible, the first signal would have been from P-4 with Lo Tave. P-4 with Lo Tave does not cause a MFPT trip but causes a FWI. It is plausible the students may think P-4 with Lo Tave could have caused the MFPTs to trip.

D. Incorrect. The FWI valves closure on P-4 with Lo Tave is correct. P-4 with Lo Tave does not cause a MFPT trip but causes a FWI. It is plausible the students may think P-4 with Lo Tave could have caused the MFPTs to trip.

REFERENCES

V-LO-PP-18101, Condensate and Feedwater System slide # 146 (included)

V-LO-PP-28193, Reactor Trip and ESFAS Signals slides # 129, 132, 135, and 136 (included)

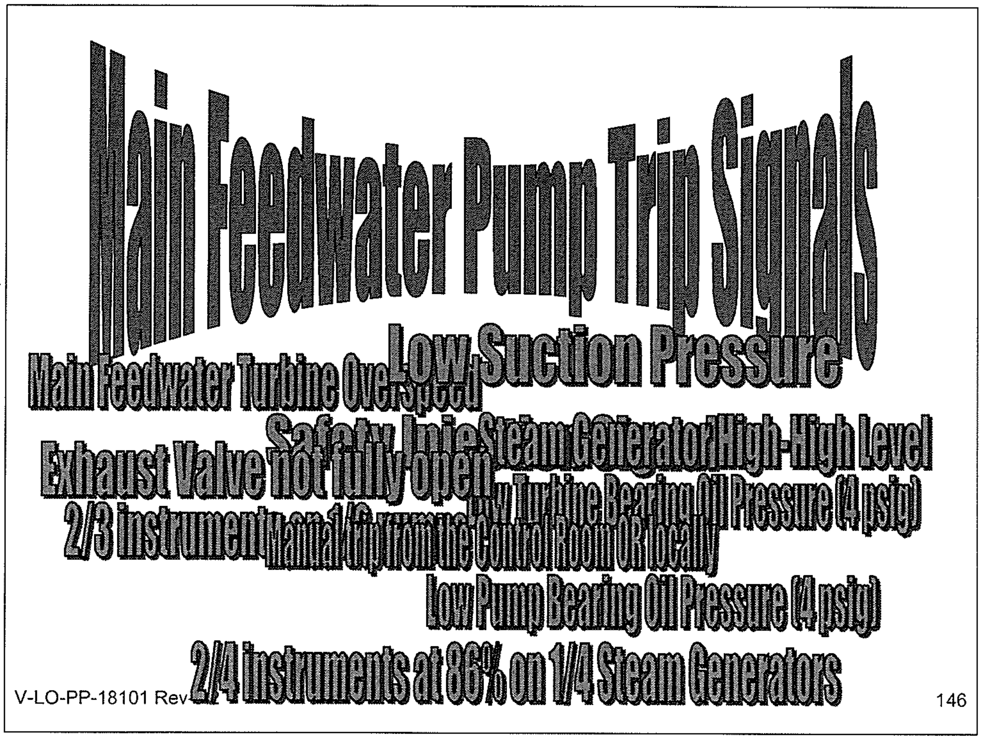
V-LO-TX-28101, Reactor Protection System, section B for Reactor Trip and ESFAS Signals, pages # 28, 29, and 30. (included)

VEGP learning objectives:

LO-PP-18101-12, Describe the operation of the Main Feedwater Pump Turbine to include:

h. How the Main Feedwater Pump Turbine will respond to a "Safety Injection signal".

LO-PP-28103-05, List all ESF actuation signals with applicable set points, coincidences, permissives, blocks, and discuss the systems response to each ESF actuation signal.



Conditions that will automatically Trip a Main Feedwater Pump

- **Safety Injection (either train)**
- **Steam Generator High-High Level (2 out of 4 instruments $\geq$82% on 1 out of 4 Steam Generators)**
- **Turbine overspeed (110%)**
- **Low suction Pressure (2 out of 3 instruments on 1 out of 2 Pumps @ 255 psig with a 20 second time delay)**
- **Turbine exhaust valve not full open**

  Main Feedwater Pump Turbine tripped is permissive to enable Turbine Exhaust Valve operation using local hand switch.
- **Low pump bearing oil pressure (4 psig)**
- **Low turbine bearing oil pressure (4 psig)**
- **Turbine thrust bearing wear**
- **Manual trip from the Main Control Room**
- **Manual trip locally**
- **Low Main Condenser Vacuum (13.5 inches Hg Vacuum)**

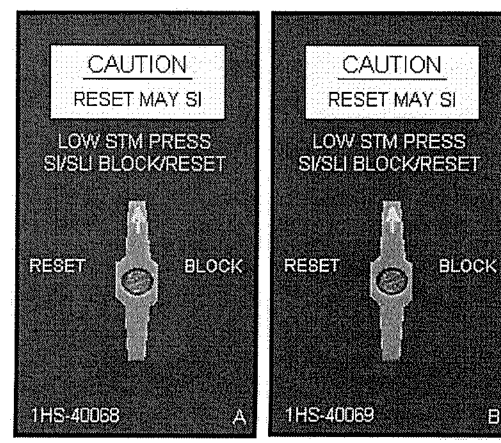
Can be overridden to allow the Main Feedwater Pump reset to establish Steam Generator Blowdown Flow.

What actuation signals input to SI?

-High-1 containment pressure
2/3 channels $\geq$ 3.8 psig.

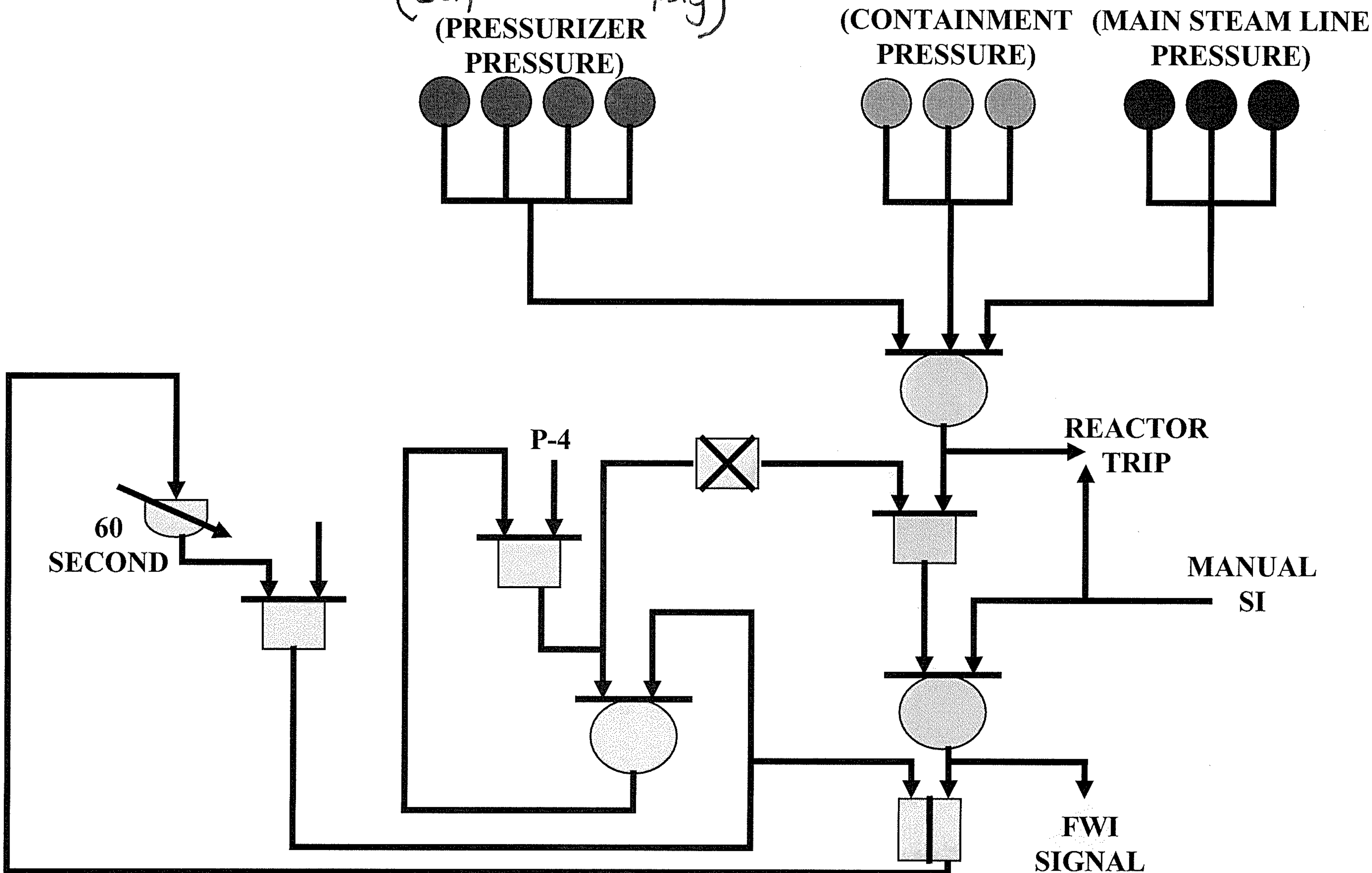
-Low pressurizer pressure
2/4 channels $\leq$ 1870 psig.
(may be manually blocked
below P-11)

-Low steam line pressure
2/3 channels $\leq$ 585 psig.
(may be manually blocked
below P-11)



In question stem, PRZR lowers to 1810 psig which is < 1870 psig.

(Setpoint is 1870 psig)



(This will close all FWI valves)

What actuation signals input to Feed Water Isolation?

P14/SI
S/G Hi-Hi
LVL FWI

- -SI
- -P-14
- -P-4 with low T_{avg} (2/4 channels < 564°F)

LO TAVG AND
REACTOR TRIP
FW VLVS CLOSE

What is the function of the FWI signal generated by SI or P-4 with low T_{avg}?

(Reactor Trip indication)

Prevents excessive RCS cooldown by isolating feedwater to the Steam Generators

FWI auto close: Both SI or Reactor trip with lo Tave closes valves,

Main and Bypass Feed Water Isolation Valves and the Main and Bypass Feed Water Regulating Valves.

Main Feed Water Regulating Valves require both trains to isolate.

19) **General Warning Reactor Trip** — **2** out of **2** SSPS General Warning Alarms

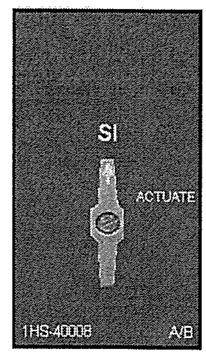
Bases: Prevents reactor operation while both trains of SSPS indicate abnormal conditions.

28.14 ESFAS Actuation Signals

Safety Injection (SI)

Purpose: To protect the core from a loss of coolant water to prevent overheating and damage to the fuel and/or fuel cladding and to provide boric acid to the core for emergency boration.

SI Actuation Signals	Coincidence	Set point
Low Pressurizer Pressure SI	**2** out of **4** channels	= **1870 psig**
Can be manually blocked below P-11		
Containment Pressure High 1 SI	**2** out of **3** channels	= **3.8 psig**
Low Steam Pressure SI/SLI	**2** out of **3** channels **1** out of **4** steam lines	= **585 psig***
Can be manually blocked below P-11		
Manual SI	**1** out of **2** hand switches	

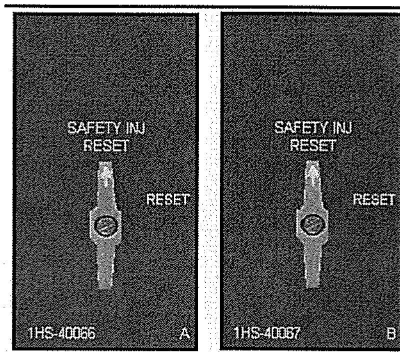


*** Set point is rate sensitive**

Systems affected by Safety Injection Signal

1) Reactor Trip
2) Turbine Trip
3) Main Feed Pump Trip
4) Feed Water Isolation
5) Motor Driven Auxiliary Feed Water Pump Start
6) MDAFW pumps discharge valve open signal
7) Steam Generator Blowdown valves close signal
8) Steam Generator Sample Valves close signal
9) **C**ontainment **I**solation Phase **A** (CIA)
10) **C**ontainment **V**entilation **I**solation (CVI)
11) **C**ontrol **R**oom **I**solation (CRI)
12) Essential Chillers start
13) Diesel Generator Emergency start
14) CVCS normal charging and safety grade charging isolates
15) **E**mergency **C**ore **C**ooling **S**ystem (ECCS) start:
    - CCPs
    - SIPs
    - RHR pumps
    - ECCS valve alignment
16) NSCW pump starts

17) NSCW cooling tower blow down isolates.
18) Containment Coolers supply and return valves receive open signals
19) Containment Coolers start in slow stopped
20) Reactor cavity and aux containment coolers supply and return valves receive close signals.
21) Non-1E buses 1NB01 and 1NB10 load shed (Stub Buses)



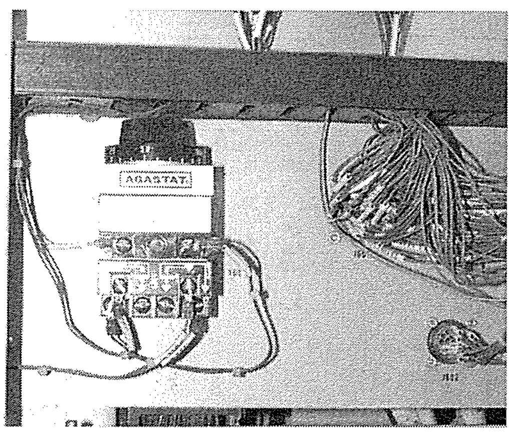
Safety Injection Reset and Block

Safety Injection signals can be reset even if the actuation signal is still present if the following is satisfied:

1) 60 seconds has past since the actuation (Timer TD-1 on both trains of SSPS)

2) Both trains P-4 must be present to seal in the reset signal. (prevents subsequent re-actuation if initiating signal is still present or from another automatic actuation signal)

3) Reset SI by using both "A" and "B" train SI reset hand switches (1 for each train of SSPS) Located on the Main Control Board "C panel"



When the Safety Injection is reset no equipment changes status. The reset only allows the operator to secure equipment auto started by the actuation.

Feed Water Isolation (FWI)

Purpose:

Isolates feed water to the Steam Generators to prevent rapid cool down of the reactor coolant system also prevents overfilling the steam generators and introduction of water in the steam lines and main turbine.

FWI Actuation Signals

1) P-4 (Reactor Trip) Lo Tavg 564°F on **2** out of **4** Loops

2) **S**afety **I**njection (also trips Main Feed Pumps and the Main Turbine)

3) P-14 Hi-Hi Steam Generator Level = 82 % on **2** out of **4** channels in **1** out of **4** S/Gs (also trips Main Feed Pumps and the Main Turbine)

Equipment affected by the Feed Water Isolation signal

1) Main Feed Water Isolation Valves receive close signals

2) Bypass Feed Water Isolation Valves receive close signals

3) Main Feed Water Regulating Valves receive close signals *

4) Bypass Feed Water Regulating Valves receive close signals *

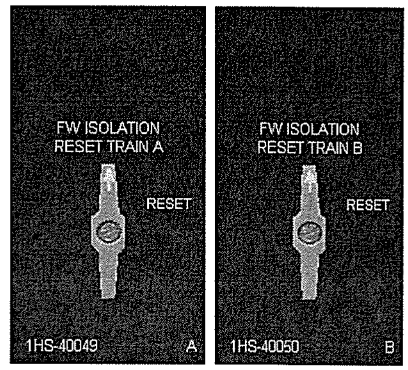
FWI closes valves.

SI trips MFPTs

*** Main Feed Water Regulating Valves are unique in the fact that they require a FWI actuation signal from both SSPS trains to auto close. This is due to the two trains of instrument air solenoid valves being arranged in parallel. The parallel arrangement minimizes the chance of inadvertent feed water isolation on a single solenoid failure.**

Resetting Feed Water Isolation

Depending on what caused the feed water isolation will determine the method in which it can be reset.



Feed Water Isolation due to Lo Tavg / reactor trip can be reset simply by the use of the "Feed Water Isolation Reset" switches. (one hand switch for each train of SSPS).

Feed Water Isolation due to SI or P-14 (a.k.a. Full FWI) is different.

1) The actuation signal must be reset or clear

2) The P-4 seal in must be taken away. This is performed by what is know as cycling the reactor trip breakers (Closing the reactor trip breakers and allowing them to re-open) Caution: If the FWI is due to a Safety Injection the actuation signal must be cleared before cycling the trip breakers. If not Safety Injection actuation will re-occur due to the auto SI block being removed.

3) Reset FWI using the "FWI Reset" Hand Switches

Main Steam Line Isolation (MSLI)

Purposes: 1) Prevents excessive cool down of the RCS for main steam line breaks down stream of the isolation valves.

2) Limits the blow down to the affected faulted steam generator this limits the cool down of the RCS and limits the amount of containment pressurization.

MSLI Actuation Signals

	Coincidence	Set point
1) **Low Steam Pressure SI/SLI**	**2** out of **3** channels **1** out of **4** steam lines	**= 585 psig***
*** Set point is rate sensitive**		
Can be manually blocked below P-11		
2) **Hi Steam Pressure negative Rate**	**2** out of **3** channels	= **100 psig**

14. 008K2.02 001/2/1/CCW-PMP POWER/MEM - 3.0/3.2/NEW/HL-15R NRC/RO/DS / TNT

Which one of the following choices lists all the procedurally allowable power supplies for Unit 1 CCW pumps 1, 3, and 5 per 13427A-1, "4160V AC Bus 1AA02 1E Electrical Distribution System"?

A.✓ RAT-1A, RAT-1B, SAT, EDG-1A

B. RAT-1A, UAT back feed, SAT, EDG-2A

C. RAT-1A, RAT-2A, SAT, EDG-1A

D. RAT-1A, UAT back feed, SAT, EDG-2A

K/A

008 Component Cooling Water System (CCWS)

K2.02 Knowledge of the power supplies to the following:

CCW pump, including emergency backup.

K/A MATCH ANALYSIS

The question requires the student to identify all possible power sourcecs for CCW pumps 1, 3, and 5 on Unit 1, including emergency backup power supplies matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. All 3 of these pumps are powered from bus 1AA02. The allowable power feeds to this bus are from its normal source - RAT 1A, emergency source - EDG-1A, and alternate sources RAT-1B and the SAT per procedure 13427A-1.

B. Incorrect. There is no procedurally directed method to connect Unit 2 DGs to any unit 1 1E busses. UAT back feed is used only for Non-1E 4160V AC busses, making this choice plausible but incorrect.

C. Incorrect. There is no procedurally directed method to connect Unit 2 RATs to any unit 1 1E busses, however there are several phyically possble connections making this choice plausible.

D. Incorrect. There is no procedurally directed method to connect Unit 2 DGs to any unit 1 1E busses. The UAT back feed is used only for Non-1E 4160V AC busses, making this choice plausible but incorrect.

REFERENCES

13427A-1, "4160V AC Bus 1AA02 1E Electrical Distribution System"
pages 2, 5, 7, and 40,

13425A-1, "4160V AC Non 1E Bus 1NA01 Electrical Distribution System" page 2

V-LO-TX-10101 "CCW System" page 10

VEGP learning objectives:

LO-PP-01101-01:

List all offsite electrical power sources.

LO-PP-01101-02:

Describe switchyard configuration (including SAT alignment) when:

a. main generator is on-line
b. main generator is shutdown

LO-PP-01101-07:

Describe the requirements for energizing a de-energized 13.8kV or 4.16 kV bus.

LO-PP-01101-08:

Describe the requirements for transferring from the alternate incoming to the normal incoming supply for 13.8 kV or 4.16kV busses

TABLE OF CONTENTS

2.2.3 During MODEs 1, 2, 3, and 4, in order to ensure that one RAT (NOT the SAT) has adequate capacity and capability to start and run both trains of ECCS loads during the connection, the following conditions shall be met:

a. Grid voltage shall be maintained at or above the minimum expected 100% grid voltage (as determined by Power Control Center [PCC]) while the busses are interconnected to one RAT.

Should grid voltage degrade below minimum, the transfer must be completed as expeditiously as possible or the alignment returned to separate sources.

b. No additional non 1E 4160V AC loads, other than those normally fed from the class 1E 4160V AC busses, shall be manually connected to the one RAT feeding both class 1E 4160V AC busses.

c. During MODE 1, the automatic bus transfer schemes for the non 1E 4160V AC busses shall be disabled during the connection of both 1E 4160V AC trains to one RAT. (The 13.8kV fast and residual voltage bus transfer schemes for the remaining RAT in service need not be disabled.)

2.2.4 During MODEs 5 and 6, one qualified circuit between offsite transmission network and onsite Class 1E Distribution System shall be operable per Technical Specification LCO 3.8.2 and a minimum of one Class 1E 4160V AC bus shall be energized per Technical Specification LCO 3.8.10.

2.2.5 The Standby Auxiliary Transformer (SAT) shall only be used as <u>one</u> offsite power source for only <u>one</u> unit.

2.2.6 During MODEs 5 and 6 both Class 1E 4160V AC buses may be manually connected to the RAT <u>OR</u> SAT. During that configuration the following precautions apply:

a. When connected to the RAT, the loads on the secondary side shall not exceed:

(1) 4160V AC 1E loads less than 1350 amps total.

(2) 4160V AC Non-1E loads less than 7500kVA total.

(3) 4160V AC Non-1E loads less than 1000 amps total.

INITIALS

3.0 PREREQUISITES OR INITIAL CONDITIONS

3.1 **Verify** 4160V AC Bus 1AA02 is aligned per 11427-1, "4160V AC 1E Electrical Distribution System Alignment." ________

3.2 **Verify** 125V DC electrical power is available to supply breaker control power. ________

4.0 INSTRUCTIONS

4.1 STARTUP

> NOTE
>
> Unless otherwise noted, all switch manipulations are performed at the QEAB in the Control Room.

4.1.1 Energizing 4160V AC Bus 1AA02 From Normal Incoming Source [RAT or SAT]

4.1.1.1 **Verify** a Normal Incoming Source is available:

a. IF 1AA02 will be energized from RAT 1NXRA, **verify** applicable sections of 13415-1, "Reserve Auxiliary Transformers," have been performed PRIOR to performing this section. ________

OR

b. IF 1AA02 will be energized from SAT, **verify** applicable sections of 13418-C, "Standby Auxiliary Transformer," have been performed PRIOR to performing this section. ________

INITIALS

4.4.2 Transferring 4160V Bus 1AA02 From The Normal Incoming Source (RAT or SAT) To The Alternate Incoming Source

CAUTIONS

- Performance of Section 4.4.2 is allowed during MODEs 1, 2, 3, and 4 by Technical Specification LCO 3.8.9 for transfer of offsite power sources only. Otherwise, it is prohibited in MODEs 1, 2, 3, and 4 by Technical Specification LCO 3.8.1 and LCO 3.8.9.
- The interconnection of both Class 1E 4160V AC busses is intended to be temporary and the time in this configuration will be minimized to the extent necessary in order to achieve a safe transfer of offsite power sources. The 4160V AC 1E breakers are expected to remain closed for approximately 2 hours during this temporary alignment.

4.4.2.1 **Verify** an Alternate Incoming Source is available.

a. IF 1AA02 will be energized from RAT 1NXRB, **verify** applicable sections of 13415-1, "Reserve Auxiliary Transformers," have been performed prior to performing this section. ________

OR

b. IF 1AA02 will be energized from SAT, **verify** applicable sections of 13418-C, "Standby Auxiliary Transformer," have been performed prior to performing this section. ________

4.4.2.2 **Verify** Limitations in Section 2.2 are met before performing this section. ________

4.4.2.3 **Verify** 14230-1, "AC Source Verification," has been successfully performed within 12 hours prior to interconnection. ________

Critical

4.4.2.4 IF Non-Class 1E 4160V AC Buses associated with RAT 1NXRB are energized from UATs by backfeed, **place** handswitch 1HS-1NA0401 Alternate Incoming Breaker in PULL-TO-LOCK and **Caution Tag.** ________

____
CV

1.0 PURPOSE

This procedure provides the instructions for the operation of the 4160V AC Non 1E Electrical Distribution System. Instructions are included in the following sections:

4.1.1 Energizing 4160V Bus 1NA01 from Alternate Incoming Source (RAT or SAT)

4.2.1 Transferring 4160V Bus 1NA01 from Alternate Incoming Source (RAT Or SAT) to Normal Incoming Source (UAT)

4.2.2 Transferring 4160V Bus 1NA01 from Normal Incoming Source (UAT) to Alternate Incoming Source (RAT/SAT)

4.3.1 De-Energizing 4160V Bus 1NA01

4.4.1 Energizing 4160V Bus 1NA01 from Normal Incoming Source (UAT)

UAT Backfeed distractor

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

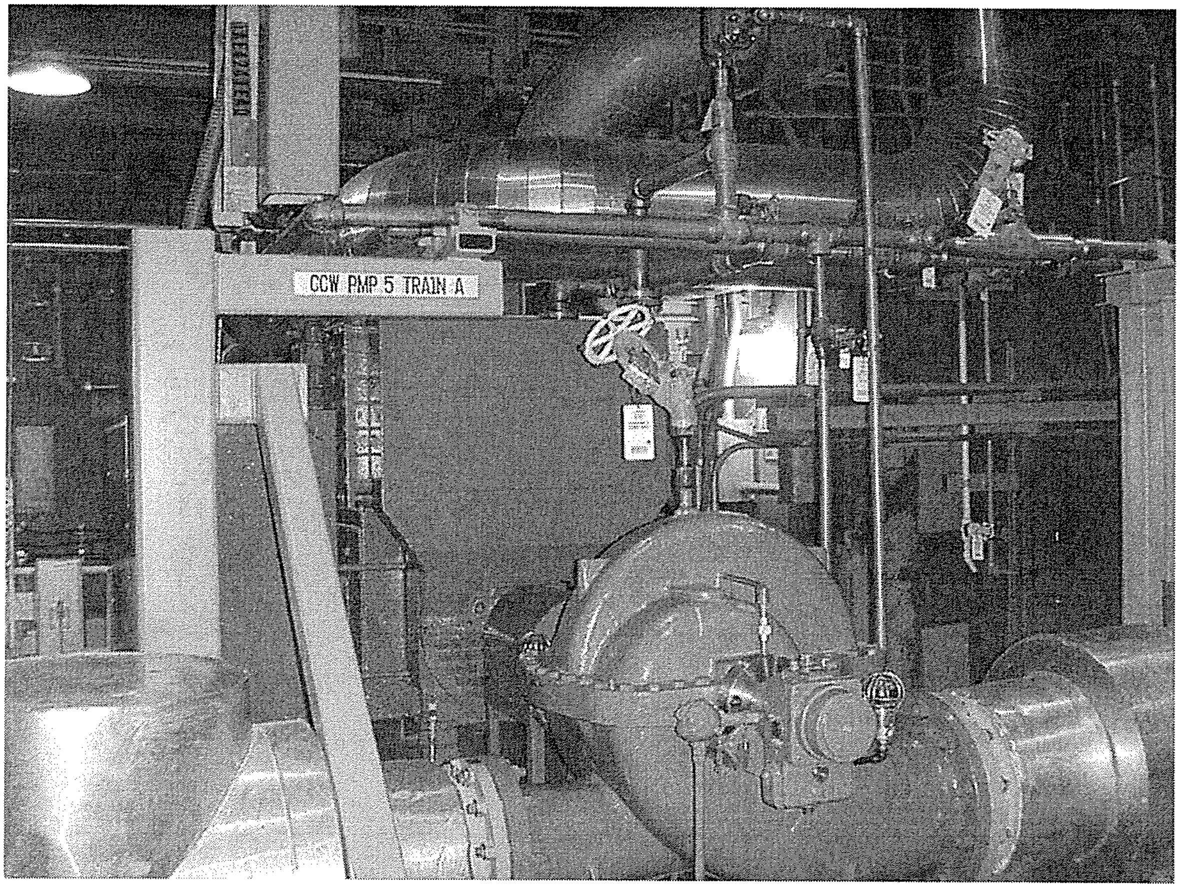
If a bus is not aligned to its normal supply for current plant conditions, Caution Tags should be generated and installed on the applicable handswitches to alert the operator to the off normal alignment.

2.2 LIMITATIONS

2.2.1 During MODEs 5 and 6 with both Class 1E 4160V AC buses manually connected to the Standby Auxiliary Transformer (SAT), all the 4160V AC Non-Class 1E buses fed by the SAT are shed and the automatic bus transfer schemes shall be disabled to prevent any auto swaps from the UAT to the SAT, and the automatic Safety Injection Signal from SSPS for one train of ECCS is blocked.

2.2.2 During MODEs 5 and 6 with both Class 1E 4160V AC buses manually connected to the same Reserve Auxiliary Transformer [RAT] and a UAT backfeed established to the Non-class 1E buses, the automatic bus transfer schemes shall be disabled for all the Non-Class 1E busses [13.8 and 4.16 KV] fed by that RAT to prevent any auto swaps from the UATs to the RATs.

V-LO-TX-10101



10.3.1 Component Cooling Water Pump & Motor

There are six component cooling water pumps arranged three in parallel per train (see Figure 10-1). Each train has three pumps for backup protection and redundancy. The pumps are 50% capacity each. Therefore, during normal operation, two pumps are running and one pump is in standby. This ensures sufficient flow for adequate heat removal to the system components during all modes of operation. The standby pump will start automatically on low pressure, when the pump discharge header reaches 65 psig (reference 17003-1 ALB A06), and another pump is running or when a running pump is tripped by protective functions.

The pumps are Ingersoll-Rand single-stage, double-suction centrifugal pumps driven by 300 HP Westinghouse motors at 1761 rpm. They are rated for 5000 gpm each at a discharge head of 160 ft. Mechanical seals are used to prevent leakage from the pump casing. Cooling for the motor is provided by circulating NSCW through the motor cooler and transferring the heat to the Ultimate Heat Sink. A 120V, 2A space heater is used to keep the motor windings dry when shut down.

Power to the component cooling water pumps is supplied by the 4.16 kV ESF busses. Component cooling pumps 1, 3, and 5 receive power from 4.16 kV ESF bus AA02 with component cooling water pumps 2, 4, and 6

15. 009EK2.03 001/1/1/SMALL LOCA-S/G/C/A - 3.0 / 3.3/NEW/HL-15R NRC/RO/TNT/DS

Given the following:

- A 200 gpm RCS leak is in progress.
- RCS pressure is 1465 psig and stable.
- Containment pressure is 2.1 psig and rising very slowly.
- The crew transitions to 19012-C, "E-1.2 Post LOCA Cooldown & Depressurization".

Which ONE of the following is CORRECT regarding **minimum** S/G NR water level required for these plant conditions and why?

A. 10%, ensures S/G tubes are covered to promote reflux boiling.

B. 32%, ensures S/G tubes are covered to promote reflux boiling.

C✓ 10%, ensures S/G inventory to ensure a secondary heat sink.

D. 32%, ensures S/G inventory to ensure a secondary heat sink.

K/A

009 **Small Break LOCA**

EK2.03 **Knowledge of the interrelations between the small break LOCA and the following:**

S/Gs

K/A MATCH ANALYSIS

The question presents a plausible scenario where a small break LOCA is in progress with some given plant parameters. The candidate must determine the minimum S/G NR level required for plant conditions (non-adverse Containment) and the basis for these levels. With a small break LOCA in progress with the given leak rate, the candidate must be aware that S/G levels are necessary for secondary heat sink and reflux cooling would not be heat removal mechanism for a small break LOCA of this size.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. 10% S/G NR level is required due to non-adverse Containment. Reflux cooling would NOT be a heat removal mechanism for a small break LOCA of this size.

B. Incorrect. 32% S/G NR level is NOT required due to non-adverse Containment. Ensures adequate feedwater flow or secondary heat sink is basis.

C. Correct. 10% S/G NR level required part is correct. Reflux cooling would NOT be a heat removal mechanism for a small break LOCA of this size.

D. Incorrect. 32% S/G NR level is NOT required and to ensure adequate feedwater flow or secondary heat sink is the basis.

REFERENCES

V-LO-HO-37111-001, Loss of Reactor or Secondary Coolant pages # 4 and # 8 (included).

VEGP learning objectives:

LO-LP-37111-02, State the effect of various size breaks on the primary system with respect to temperatures and pressures.

DESCRIPTION

Small RCS LOCA with one train of ECCS WORST case

One train of safety injection is assumed, and loss of offsite power is assumed to occur at the reactor trip time. The only means of venting steam on the secondary side is through the steam generator safety valves. Minimum auxiliary feedwater is assumed available one minute after the reactor trip time.

For these break sizes the normal makeup system cannot maintain level and pressure. The RCS will depressurize and an automatic reactor trip and safety injection signal will be generated. Provided that a secondary side heat sink exists, the RCS will reach an equilibrium pressure which corresponds to the pressure at which the liquid phase break flow equals the high pressure pumped safety injection flow, which is above the SG pressures.

RCS stable and above SG pressure.

Core heat is removed through the steam generators by continuous single or two phase natural circulation.

Once equilibrium pressure is established there is no further net loss of liquid volume in the RCS. The natural circulation heat removal mode continues until the time that the break can remove all the decay heat. Prior to this time, auxiliary feedwater is required to maintain the heat sink.

Abnormal indications should be present in the containment for this category of LOCA although the response will be slower and milder than for larger break size LOCAs. Containment pressure will probably not reach the containment High 1 pressure.

Adverse NOT expected.

Figure 1 shows the RCS pressure transient for this case. The RCS pressure stabilizes slightly above the steam generator safety valve set pressure. Figures 2 and 3 show the safety injection and break flows which are both stabilized at a flowrate of approximately 340 gpm. Figure 4 shows that the pressurizer empties at approximately 10 minutes and does not refill. The system remains in a stable condition with the core covered and decay heat being adequately removed.

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 ECCS flowrate, in order to depressurize to a cold shutdown condition it is necessary to cool the primary fluid further while stepping down the ECCS flowrate. Long term cooldown/depressurization of the plant is performed using 19012-C, ES 1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION.

V-LO-HO-37111

Medium LOCA Breaks (1in^2 to 1 ft^2)

During the early stages of the depressurization, the break flow is not capable of removing all the decay heat. At this pressure, 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 ECCS flow and break size.

Prior to the start 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 ceases, and core heat is removed through condensation of steam in the steam generator. This method of heat removal is called reflux.

Makes Reflux plausible

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. 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 primary fluid 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.

As soon as the break flow becomes all steam flow for breaks in this range of size, steam generated in the core can exit out the break, and further system depressurization occurs. Safety injection flow increases to greater than the break flow, and there is no longer a net loss of mass from the RCS. No further core uncovery will occur under these conditions. The steam generator may still be relied upon for heat removal by the condensation mode. However, only a small amount of heat removal by the steam generator is necessary and, with minimum auxiliary feedwater available, the steam generator secondary side will now begin to slowly depressurize below the steam generator safety valve set pressure. The primary system will also slowly depressurize along with the secondary side.

The RCS pressure plot, Figure 6, shows a rapid depressurization to approximately 1200 psig at 5 minutes. Immediately after the draining of the crossover leg, the break uncovers and the break flow becomes all steam. This can be seen by a rapid decrease in break flow at that time on the plot in Figure 7, indicating a change from two phase to all steam flow. Because the location of the break in this sample transient is the cold leg, the core level also decreases in conjunction with the crossover leg draining (Figure 8). As steam is relieved out the break, the core pressure decreases relative to the downcomer pressure, and the hydrostatic head in the downcomer recovers the core. Note that this core level

V-LO-HO-37111

16. 010K1.08 001/2/1/PRZR PRESS-PZR LCS/C/A - 3.2 / 3.5/NEW/HL-15R NRC/RO/DS / TNT

Complete the following statement for the PRZR Pressure Control System:

When PRZR level _____ the PRZR _____ heaters will de-energize.

A.	drops below 17%	backup only
B.	drops 5% below program level	backup and proportional
C.✓	drops below 17%	backup and proportional
D.	drops 5% below program level	backup only

HL-15R RO NRC Exam

K/A

010 **Pressurizer Pressure Control System (PZR PCS)**

K1.08 **Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems:**

PZR LCS.

K/A MATCH ANALYSIS

The question requires to student to determine the effects of high or low PRZR level on the PRZR pressure control system, matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Backup heaters would de-energize, however, the proportional heaters would also de-energize.

B. Incorrect. At 5% above program the PRZR backup heaters will automatically energize. At 5% below program PRZR heaters do not change state.

C. Correct. At 17% PRZR level all PRZR heaters are interlocked off to protect the heaters and / or the heater well from damage due the water level dropping below the heaters.

D. Incorrect. At 5% above program the PRZR backup heaters will automatically energize. At 5% below program PRZR heaters do not change state.

REFERENCES

17011-1, "Annunciator Response Procedures for ALB 11 on Panel 1C1 on MCB" Windows:

B01, "PRZR LO LEVEL HTR CNTL OFF LTDN SECURED"

C01, "PRZR CONTROL HI LEVEL DEV AND HEATERS ON

D01, "PRZR LO LEVEL DEVIATION"

VEGP learning objectives:

LO-PP-16302-04:

Describe the Hi and Low Pressurizer level protection features including the set points, coincidence, and reason for each.

WINDOW B01

ORIGIN	SETPOINT	
1-LT-0459 1-LT-0460 1-LT-0461	17%	PRZR LO LEVEL HTR CNTL OFF LTDN SECURED

1.0 **PROBABLE CAUSE**

1. Pressurizer level Control System Malfunction.
2. Charging - Letdown System Malfunction.
3. RCS cooldown.
4. Reactor Coolant System leak.

2.0 **AUTOMATIC ACTIONS**

1. All Pressurizer Heaters turn off.
2. Letdown isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** pressurizer level instrumentation.
2. IF instrument malfunction is indicated **Go To** 18001-C, "Primary Systems Instrumentation Malfunctions".
3. IF a Reactor Coolant System leak is indicated, **Go To** 18004-C, "Reactor Coolant System Leakage".

WINDOW C01

ORIGIN	SETPOINT	
1-LT-0459 1-LT-0461	5% above level program	PRZR CONTROL HI LEVEL DEV AND HEATERS ON

1.0 PROBABLE CAUSE

1. Pressurizer Level Control System malfunction.
2. Charging-Letdown System malfunction.
3. Rapid reduction in secondary steam demand.

2.0 AUTOMATIC ACTIONS

Pressurizer Backup Heaters energize.

3.0 INITIAL OPERATOR ACTIONS

Check pressurizer level using 1-LR-0459 recorder and if a Pressurizer Level Control System malfunction is indicated, **initiate** 18001-C, "Primary Systems Instrumentation Malfunction".

4.0 SUBSEQUENT OPERATOR ACTIONS

<u>IF</u> Pressurizer Level Control System is not correcting level, **take manual control** and **adjust** as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, 1X6AU01-183, 168, 1X6AX01-106, PLS

WINDOW D01

ORIGIN	SETPOINT	
1-LT-0459 1-LT-0461	5% below level program	PRZR LO LEVEL DEVIATION

1.0 PROBABLE CAUSE

1. Pressurizer Level Control System malfunction.

2. Charging-Letdown System malfunction.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

Check pressurizer level using 1-LR-0459 recorder and if a Pressurizer Level Control System malfunction is indicated, **initiate** 18001-C, "Primary Systems Instrumentation Malfunction".

4.0 SUBSEQUENT OPERATOR ACTIONS

IF Pressurizer Level Control System is not correcting level, **take manual control** and **adjust** as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, 1X6AU01-183, 168, PLS

17. 011EG2.2.38 001/1/1/LARGE LOCA-LICENSE/MEM - 3.6 / 4.5/NEW/HL-15R NRC/RO/TNT/DS

To maintain Containment parameters within the accident analysis assumptions for a DBA LOCA, the Containment Pressure and Containment Air Temperature LCO limits for Mode 1 are...

A.✓ +1.8 psig and 120°F

B. +1.8 psig and 130°F

C. -3.0 psig and 120°F

D. -3.0 psig and 130°F

K/A

011 Large Break LOCA

G2.2.38 Knowledge of conditions and limitations in the facility license.

K/A MATCH ANALYSIS

The question presents a question where the student must know the LCO limits of Tech Specs 3.6.4 and 3.6.5 for Containment Pressure and Containment Air Temperature to prevent exceeding DBA LOCA analyzed design limits.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. +1.8 psig is the upper Containment Pressure limit and 120°F is the Containment Air Temperature limit.

B. Incorrect. +1.8 psig is the correct upper Containement Pressure limit but 130°F is NOT the correct containment Air Temperature limit. 130°F was used since it is the temperature where SFP high temperature alarms and would be a plausible figure the students may recall. 130°F is also a common set point for various plant systems such as bypass valves operations for demin diverts, etc.

C. Incorrect. -3.0 psig is NOT the correct negative pressure limit. -3.0 is plausible as it is the figure used in the Tech Spec bases for the accident analysis but is NOT the LCO limit. Plausible the candidate could confuse this number with the LCO limit or be confused by the -0.3 psig negative LCO limit. 120°F is the correct Containment Air Temperature limit.

D. Incorrect. -3.0 psig is NOT the correct negative pressure limit. -3.0 is plausible as it is the figure used in the Tech Spec bases for the accident analysis but is NOT the LCO limit. Plausible the candidate could confuse this number with the LCO limit or be confused by the -0.3 psig negative LCO limit. 130°F was used since it is the temperature where SFP high temperature alarms and would be a plausible figure the students may recall. 130°F is also a common set point for various plant systems such as bypass valves operations for demin diverts, etc.

REFERENCES

Tech Spec 3.6.4 Containment Pressure and the bases.

Tech Spec 3.6.5 Containment Air Temperature.

17005-1, window A06 for SFP Hi Temp.

VEGP learning objectives:

LO-LP-39210-01, For any item in section 3.6 of Tech Specs, be able to:

a. State the LCO

LO-LP-39210-02, Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode.

a. Whether any LCO limits of section 3.6 of Tech Specs has been exceeded.

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be $\geq$ -0.3 psig and $\leq$ +1.8 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 17.7 psia (3.0 psig). This resulted in a maximum peak pressure from a LOCA of 36.5 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, 36.5 psig, does not exceed the containment design pressure, 52 psig.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The containment was also designed for an external pressure load equivalent to -3 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 14.093 psia. This resulted in a minimum pressure inside containment of 11.77 psia, which is less than the design load. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2). Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature

(continued)

WINDOW A06

SPENT FUEL PIT HI TEMP

ORIGIN	**SETPOINT**
1-TISH-626	130°F

Common alarm stpt for items.

1.0 PROBABLE CAUSE

1. Spent Fuel Pool Pump trip.
2. Loss of Component Cooling Water (CCW) flow to Spent Fuel Pool Heat Exchanger.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

Go To 18030-C, "Loss Of Spent Fuel Pool Level Or Cooling."

4.0 SUBSEQUENT OPERATOR ACTIONS

NONE

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB130, PLS

18. 012K1.05 001/2/1/RPS-ESFAS/C/A - 3.8 / 3.9/NEW/HL-15R NRC/RO/DS / TNT

Five minutes following a reactor trip and safety injection, the OATC places the Train A SI reset handswitch in the "reset" position.

P-4 train A will be generated when the Train A reactor trip breaker is open __(1)__ .

SI reset will block __(2)__ .

A. (1) **and** the Train B bypass breaker is open

(2) only the Train A PRZR Low Pressure SI and Low Steamline Pressure SI signals

B.✓ (1) **and** the Train A bypass breaker is open

(2) all Train A automatic SI signals

C. (1) **or** the Train A bypass breaker is open

(2) only the Train A PRZR Low Pressure SI and Low Steamline Pressure SI signals

D. (1) **or** the Train B bypass breaker is open

(2) all Train A automatic SI signals

K/A

012 Reactor Protection System

K1.05 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following:

ESFAS.

K/A MATCH ANALYSIS

The question requires the student to correctly identify the effects on the ESFAS system and the position of the reactor trip and bypass breakers when resetting the SI signal, matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Both the Train A reactor trip and bypass breakers must to be open in would be blocked. The Low PRZR Pressure and Low Steamline Pressure SI signals may be manually blocked when PRZR pressure is < P-11 (2000 PSIG). This operation is performed in the UOPs and EOPs.

B. Correct. P-4 Train A signal is generated when both the train A reactor trip and byapss breakers are open. Once SI is reset with the P-4 signal present, all automatic SI signals are blocked.

C. Incorrect. Both the train A reactor trip and bypass breakers must to be open in order to generate a Train A P-4 signal. If P-4 were present all automatic SI signals would be blocked. The Low PRZR Pressure and Low Steamline Pressure SI signals may be manually blocked when PRZR pressure is < P-11 (2000 PSIG). This operation is performed in the UOPs and EOPs.

D. Incorrect. Once SI is reset with the P-4 signal present, all automatic SI signals are blocked. Both the Train A reactor trip and bypass breakers must to be open in order to generate a Train A P-4 signal.

REFERENCES

V-LO-TX-28101 " Reactor Protection System" page 18

VEGP learning objectives:

LO-PP-28103-07:

Discuss SI reset to include:

a. Time delay
b. SI reset with P-4
c. SI reset without P-4
d. Auto and Manual actuation capabilities following reset

SECTION B

REACTOR TRIP AND ESFAS SIGNALS

28.11 PERMISSIVE INTERLOCKS

Permissive interlocks provide input to the protection systems to allow or prevent protective functions from occurring under certain plant conditions.

P-4 Indicates reactor tripped

Set point or conditions that give P-4

RTA and its bypass (BYA) both open give P-4 Train A
RTB and its bypass (BYB) both open give P-4 Train B

Function:

1) **Trips** the Main Turbine to limit the RCS cool down

- P-4 Train A generates a "Mechanical Turbine Trip"
- P-4 Train B generates an "Electrical Turbine Trip"

2) Steam Dumps

- P-4 Train A generates a Steam Dump Arming signal
- P-4 Train B transfers Steam Dump controllers from "Load reject" mode to the "Plant trip" mode

3) **F**eed **W**ater **I**solation (FWI)

- P-4 in conjunction with Lo Tavg of 564°F

4) Seals in FWI if caused by a Safety Injection or Hi-Hi Steam Generator water level (P-14).

5) SI reset logic

- After Safety Injection has been reset, P-4 blocks any future automatic safety injection signals.

P-6 Source Range Block Permissive

Set point:

2.0 x 10-5% POWER on any **1 / 2** IR NIS detector.

Function:

1) Allows the operator to manually block SR high flux trip. (both TRN A and TRN B switches, QMCB-C)

2) Loss of P-6 (either train no SSPS) will automatically unblock the Source Range Trip Permissive status light on BPLP (QMCB-C) Illuminates when P-6 is present.

P-6 Resets:

When 2/2 IR NIS detector drop < **7.0 X 10-6% POWER**

HL-15R RO NRC Exam

19. 012K5.01 001/2/1/RPS-DNB/MEM - 3.3 / 3.8/M - CATAWBA 08/HL-15R NRC/RO/DS/TNT

Which one of the following correctly matches the reactor trip signals to their limiting accident / protection?

	Reactor Trip Signal	Limiting Accident / Protection
A.	Overpower DT	DNBR
	Overtemperature DT	Excessive fuel heat generation rate (kW/ft)
	PRZR High Pressure	RCS integrity
	PRZR Low Pressure	DNBR
B.	Overpower DT	Excessive fuel heat generation rate (kW/ft)
	Overtemperature DT	DNBR
	PRZR High Pressure	RCCA drive housing rupture
	PRZR Low Pressure	Excessive RCS cooldown
C.✓	Overpower DT	Excessive fuel heat generation rate (kW/ft)
	Overtemperature DT	DNBR
	PRZR High Pressure	RCS integrity
	PRZR Low Pressure	DNBR
D.	Overpower DT	DNBR
	Overtemperature DT	Excessive fuel heat generation rate (kW/ft)
	PRZR High Pressure	RCCA drive housing rupture
	PRZR Low Pressure	Excessive RCS cooldown

K/A

012 Reactor Protection System

K5.01 Knowledge of the operational implications of the following concepts as they apply to the RPS:

DNB.

K/A MATCH ANALYSIS

The question requires the student to match 4 different reactor trip functions with their correct bases. Two of the trip bases are for DNBR protection matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. The bases for the OTDT and OPDT have been reversed, both of these choices are plausibe since the DT trips do provide this protection.

B. Incorrect. The PRZR high pressure trip bases is incorrectly tied to the PRNI high +rate flux trip. The PRZR low pressure trip is tied to the bases for the FWI ESFAS be an expected precusor to an RCCA drive housing rupture and low PRZR would be expected as a result of an excessive RCS coolodown.

C. Correct. OPDT trip bases is excessive fuel heat generation rate (kW/ft). OTDT & PRZR low pressure reactor trips bases are DNBR. PRZR high pressure provide RCS integrity protection.

D. Incorrect. The bases for the OTDT and OPDT have been reversed, both of these choices are plausibe since the DT trips do provide this protection. The PRZR high pressure trip bases is incorrectly tied to the PRNI high +rate flux trip. The PRZR low pressure trip is tied to the bases for the FWI ESFAS function on low Tave. Both of these choices are plausible since high pressure would be an expected precusor to an RCCA drive housing rupture and low PRZR would be expected as a result of an excessive RCS coolodown.

REFERENCES

VEGP LCO 3.3.1 bases pages 13, 16, 20, and 23 for:

- PRNI - High Positive Rate Trip
- OTDT Trip
- OPDT Trip
- PRZR High Pressure Trip
- PRZR Low Pressure Trip

VEGP LCO 3.3.2 bases page 24 for FWI - Low RCS Tavg conincident with reactor trip

Catawba 2008 NRC exam question 38

VEGP learning objectives:

LO-PP-28103-04:

Discuss the bases for each reactor trip signal.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Power Range Neutron Flux — Low (continued)

setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux — High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux — Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux — High Positive Rate

The Power Range Neutron Flux — High Positive Rate trip uses the same channels as discussed for Function 2 above.

The Power Range Neutron Flux — High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux — High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux — High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux — High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux — High Positive Rate trip Function does not have to be

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

subcritical, boron dilution (see LCO 3.3.8) and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux — Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the Source Range Neutron Flux trip is blocked.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the Rod Control System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. Source range detectors also function to monitor for high flux at shutdown. This function is addressed in Specification 3.3.8. Requirements for the source range detectors in MODE 6 are addressed in LCO 3.9.3.

6. Overtemperature ΔT

The Overtemperature ΔT trip Function (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A) is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A) ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux — High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature — including dynamic compensation for RTD response time delays.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. This results in a two-out-of-four trip logic. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Pressurizer Pressure (continued)

the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function.

a. Pressurizer Pressure — Low

The Pressurizer Pressure — Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure — Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure — Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure — High

The Pressurizer Pressure — High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires four channels of the Pressurizer Pressure — High to be OPERABLE.

The Pressurizer Pressure — High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Turbine Trip and Feedwater — Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Under specific conditions, a single inoperable actuation relay does not require that the affected automatic actuation logic function be declared inoperable. Specific guidance is provided in this section under the heading "Actuation Relays."

b. Feedwater Isolation — Low RCS T_{avg} Coincident with Reactor Trip

Since T_{avg} is used as an indication of bulk RCS temperature, this Function meets redundancy requirements with one OPERABLE channel in each loop. Thus, this function is specified as a total of four channels and not on a per loop basis. The channels are used in a two-out-of-four logic. The Low RCS T_{avg} signal is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. The P-4 interlock is discussed in Function 8.a.

c. Turbine Trip and Feedwater Isolation — Steam Generator Water Level — High High (P-14)

LOOP 1	LOOP 2	LOOP 3	LOOP 4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554

NOTE: Steam Generator Water Level channels are required OPERABLE by the Post Accident Monitoring Technical Specification.

The setpoints for this Function on Table 3.3.2-1 are in % of narrow range instrument span.

(continued)

C

QuestionBank #	KA_system	KA_number
544	SYS012	K5.01

KA_desc
Knowledge of the operational implications of the following concepts as the apply to the RPS: (CFR: 41.5 / 45.7)□DNB ....

Which one of the following selections correctly matches the reactor trip signals to their limiting accident/protection?

	Reactor Trip Signal	Limiting Accident/Protection
A.	OPDT	DNB
	OTDT	Excessive fuel centerline temperature
	Pzr High Level	NC system integrity
	Pzr Low Pressure	DNB
B.	OPDT	Excessive fuel centerline temperature
	OTDT	DNB
	Pzr High Level	DNB
	Pzr Low Pressure	NC system integrity
C.	OPDT	Excessive fuel centerline temperature
	OTDT	DNB
	Pzr High Level	NC system integrity
	Pzr Low Pressure	DNB
D.	OPDT	NC System integrity
	OTDT	Excessive fuel centerline temperature
	Pzr High Level	DNB
	Pzr Low Pressure	DNB

CATAWBA 2008

20. 013K5.02 001/2/1/ESFAS-LOGIC/C/A - 2.9 / 3.3/NEW/HL-15R NRC/RO/DS / TNT

A loss of 120V AC vital bus 1BY1B has occurred with the unit at 100% power. Which one of the following correctly decribes the impact on SSPS?

A. Only SSPS Train A channel II Input relays are **de-energized.**

The Train B Logic cabinet is **de-energized.**

The Train B Slave relays are **inoperable.**

B. Only SSPS Train A channel II input relays are **de-energized.**

The Train B Logic cabinet is **de-energized.**

The Train B Slave relays are **operable.**

C. SSPS Train A and Train B channel II Input relays are **de-energized.**

SSPS Train A and Train B Logic cabinets are **energized.**

The Train B Slave relays are **operable.**

D✓ SSPS Train A and Train B channel II Input relays are **de-energized.**

SSPS Train A and Train B Logic cabinets are **energized.**

The Train B Slave relays are **inoperable.**

K/A

013 **Engineered Safety Features Actuation System (ESFAS)**

K5.02 **Knowledge of the operational implications of the following concepts as they apply to the ESFAS:**

Safety system logic and reliability.

K/A MATCH ANALYSIS

The question presents a scenario where a loss of a vital AC bus occurs. The student must correctly determine the impact on both trains of ESFAS actuation circuitry including the logic circuits.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. A loss of 1BY1B will de-energize all channel II input relays on **both** trains of SSPS. Both trains of logic cabinets will remain energized via redundant power supplies. The Train B only slave relays will all become inoperable. This choice is plausible due to the train B impacts listed.

B. Incorrect. A loss of 1BY1B will de-energize all channel II input relays on **both** trains of SSPS. Both trains of logic cabinets will remain energized via redundant power supplies. The Train B only slave relays will all become inoperable. This choice is plausible due to the train B impacts listed and that the train B master relays still have power available.

C. Incorrect. A loss of 1BY1B will de-energize all channel II input relays on **both** trains of SSPS. Both trains of logic cabinets will remain energized via redundant power supplies. The Train B only slave relays will all become inoperable. This choice is plausible due train B master relays still have power available.

D. Correct. A loss of 1BY1B will de-energize all channel II input relays on **both** trains of SSPS. Both trains of logic cabinets will remain energized via redundant power supplies. The Train B only slave relays will all become inoperable.

REFERENCES

AOP 18032-1, "Loss of 120V AC Instrument Power", page 44

LO-LP-60324, "Loss of 120V AC Instrument Power", page 11

V-LO-TX-28101, "SSPS" text, pages 42, 44, and 45

V-LO-PP-28101, "SSPS" presentation, pages 23, 29, and 30

VEGP learning objectives:

LO-PP-28101-01:

State the sources of power to the SSPS cabinets.

LO-PP-28101-02:

Determine how the loss of a power supply will affect SSPS.

ATTACHMENT C

Sheet 4 of 5

TABLE 2 – PANEL 1BY1B LOAD LIST

BREAKER	LOAD
03	NIS CHANNEL II CONTROL POWER
04	SOLID STATE PROTECTION SYSTEM OUTPUT CABINET 2, TRAIN B
05	PROCESS RACK PROTECTION SET II
06	SPARE
07	SOLID STATE PROTECTION SYSTEM CABINET II, TRAIN A
08	NIS CHANNEL II INSTRUMENT POWER
09	DIESEL GENERATOR 1B WHM PULSE AMPLIFIER
10	SOLID STATE PROTECTION SYSTEM CHANNEL II, TRAIN B
11	BOP PROTECTION PANEL CHANNEL II 12CQPP2
12	NFMS AMPLIFIER / NFMS OPTICAL ISOLATOR
13	HVAC PANEL 1BCQHVC2
14	SPARE
15	SAFEGUARD TEST CABINET, TRAIN B
16	SPARE
17	HVAC INSTRUMENT PANEL 1-1500-V7-002-CBB
18	SWITCHGEAR 1BD1 TRANSDUCER POWER
19	PLASMA DISPLAY B
20	SPARE
21	SPARE
22	SPARE
23	MISC EQUIPMENT PANEL QPCP XMFR OF ZLB-16
24	DIST PANELS 1BD11, 1BD12 CONTROL POWER

3) CRI

C. Loss of Vital Instrument Panel 1BY1B

1. Symptoms - ***<u>Same as on loss of 1AY1A just Train B</u>***

   a. All Channel II trip status lights energized.

      1) Probably the most significant and easiest method of determining the loss of 1BY1B.

   b. Loss of N32, N36, N42 simultaneously.

   c. 1BY1B Trouble Alarm

   d. Inverter 1BD1I1 Trouble Alarm

2. Concerns and major effects of loss of 1BY1B ***<u>Same as 1AY1A except Train B equipment</u>***

   Objective 1

   a. SG 2 and 3 ARV will not operate from the control room or remote shutdown panel(NOTE).

   b. May lose letdown and have max charging if LT-460 is selected for control.

   c. Steam Generator levels and steam lows fail causing significant transient on all the steam generators.

   Normally all selected to channel 1 instruments

   d. If unable to stabilize plant conditions may require plant shutdown or trip.

   Objective 3

   e. Plant will trip if below P-10 due to N36 trip signal when de-energized.

   f. General Warning on Train B SSPS.

   Objective 4

   Without power from 1BY1B there is no power to Train B SSPS slave relays.

   Any actuation signal requiring operation of slave relays will be blocked.

   If SI signal present, only Train A equipment will Auto align.

   Train B equipment must be manually re-aligned in this case.

   g. Plant will trip if another channel instrument bistable is tripped and/or channel is in test.

COMMITMENTS:

FF 86.008
FF 89.026
FF 89.021

OTHER:

10CFR50, APPENDIX A, CRITERIA 20 THROUGH 25
DCPs 94.059 / 060

DCP 2001-032.doc

Action Item # 2003202685

DCP105387131 and 2053871401 - Revise P-9 to 40%

SECTION C

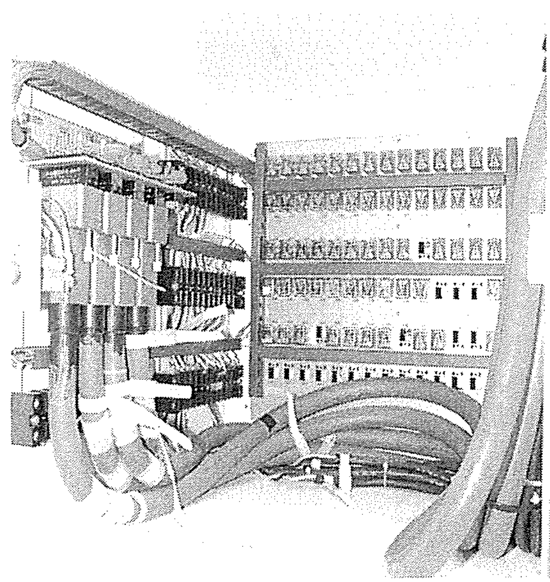
SOLID STATE PROTECTION SYSTEM

28.15 Layout and description of SSPS

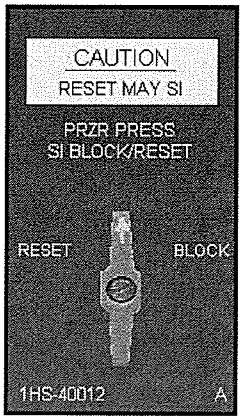
Two separate trains of SSPS exist on each unit for reactor protection. SSPS is divided in the following manner: input relay bay, logic bay, output relay bay, and the test panels. This section will discuss the purpose, automatic and testing operation portions of SSPS.

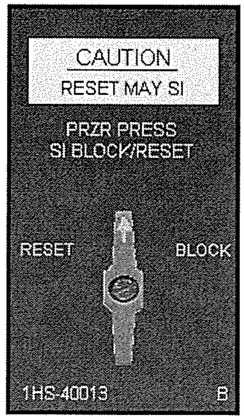
28.16 Input Relay Bays

SSPS input relay bays acts as an isolation device between the various plant inputs and SSPS. It is divided into 4 compartments (one for each protection channel) to provide separation between each input channels. There are three different types of inputs to the input relay bays: (1) NSSS and BOP protection, (2) **N**uclear **I**nstrumentation **S**ystem (NIS), and (3) Field Contacts. The input relays associated with NIS are supplied with 120 VAC from their respective NI channels. The field contacts are instruments that input directly into SSPS, such as Main Turbine Stop Valve position and RCP under frequency relays. The field contacts are powered directly from the input relay bay itself. Each SSPS input relay bay is supplied from its respective channel 120 VAC power source.



Input Relay Bay	Source
Channel I	1AY1A
Channel II	1BY1B
Channel III	1CY1A
Channel IV	1DY1B



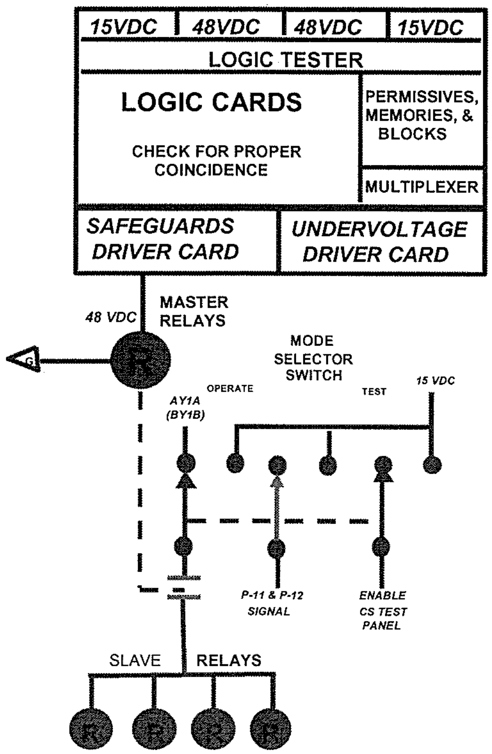


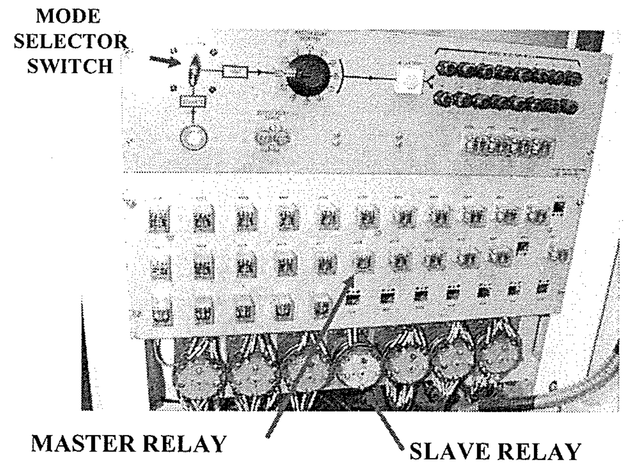
The switches shown are typical for all block and reset switches found on the main control board. Each hand switch will reset and block its associated train of SSPS only. This noted by the train designator in the bottom right hand corner.

The multiplexer portion of the logic bay monitors the input and output of the logic cards and transfers the status of each parameter to a de-multiplexer for the main control board or computer. Identical information from both train "A" and "B" SSPS is transmitted over a common "OR" cable. The multiplexer illuminates the appropriate **T**rip **S**tatus **L**ight **B**ox (TSLB) when its parameter bi-stable trips. Also it indicates the status of all permissives and control interlocks on the **B**ypass **P**ermissive **L**ight **P**anel (BPLP).

28.18 OUTPUT RELAYS BAYS

There are two types of output relays; (1) Master relays, and (2) Slave relays. The master relays can be energized from either the Safeguard Driver Card from the Logic Bay or directly from the actuation switch located on the main control board. The master relays must be energized to actuate which require 48 VDC that comes from the logic bay. Each master relay can control up to four slave relays that are dedicated for a given actuation.

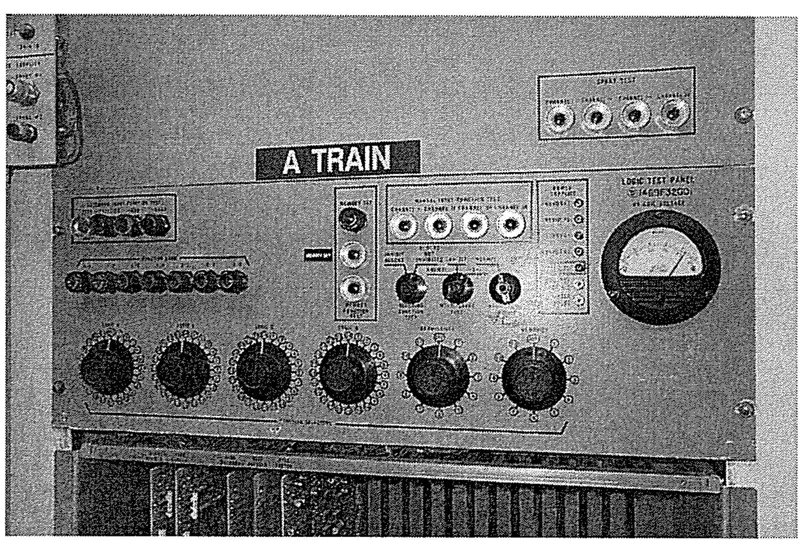




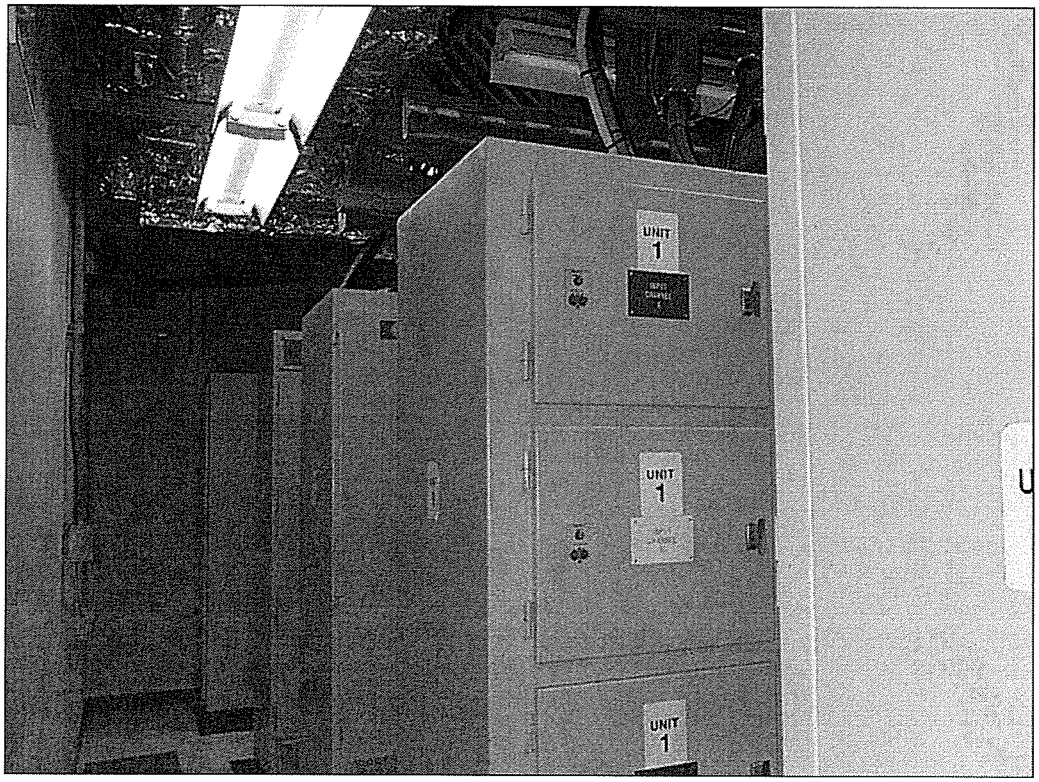
The slave relays are used to send actual start/stop signals to specific plant equipment when actuated. Slave relays require 120 VAC to actuate. Many are arranged with dual operating coils. The “latching coil” is used to actuate, which occurs when its master relay energizes. This type relay will remain in the latch position even when de-energized. The second coil called the “unlatch coil” energizes when the operator reset the actuation from the control board. If both coils are actuated at the same time the relay will remain in the Latch (actuate) position.

28.19 TEST CIRCUITS

SSPS is just like all other safety related systems, it must be tested to prove its operability. The SSPS test circuits allows testing from an individual slave relay to testing a reactor trip circuit while the unit remains online. This text will cover the different switches and circuits that allow such testing to be performed.



V-LO-PP-28101



Outside view of the input bays for “A” train SSPS.

Input relay bays are divided into 4 isolated compartments (one for each channel) to provide separation between the input channels.

Plant inputs are applied to the input relay coils and the SSPS input comes from the input relay contacts to provide electrical isolation between the two (no electrical connections between coil and contacts).

Three types of inputs:

1) NSSS and BOP protection
2) NIS
3) Field Contacts

Field contacts examples are:
1) Main Turbine Stop Valve position
2) RWST level
3) RCP under frequency Relay

All field contacts are powered by its associated input bay.

Input Relay Bay power supplies:

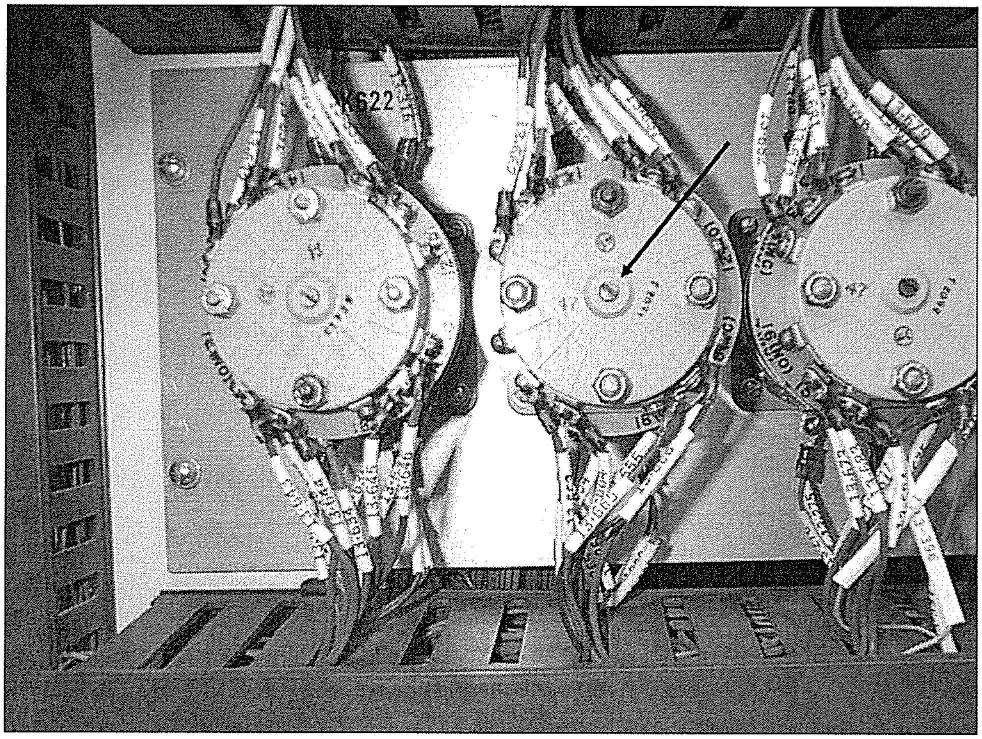
Channel I 1/2AY1A

Channel II 1/2BY1B

Channel III 1/2CY1A

Channel IV 1/2DY1B

V - LO - PP - 28101



Close up picture of slave Relay K622, K623, and K624.

Demonstrate how to determine if a relay is reset or not.

Train A Slave Relays are powered by 1/2AY1A

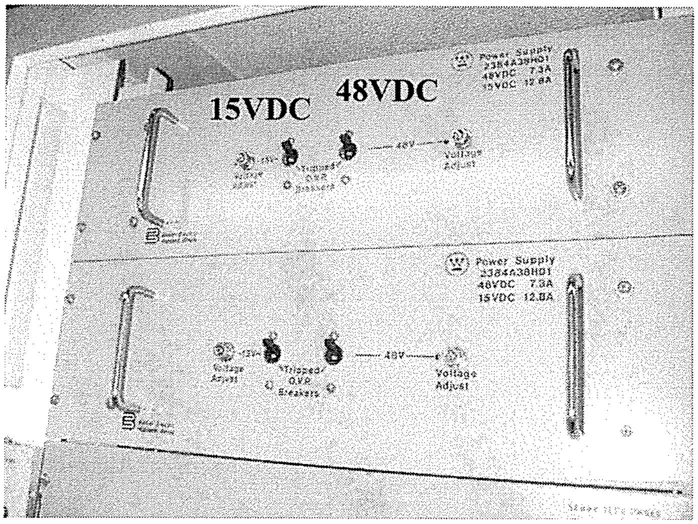
Train B Slave Relays are powered by 1/2BY1B

With a loss of slave relay power all actuations from that train that are generated from SSPS will not occur.

V-LO-PP-28101

LOGIC BAY POWER SUPPLIES

"A" TRAIN - AY1A/CY1A
"B" TRAIN - BY1B/DY1B



30

Two 15/48 VDC power supplies.

Only one needed to perform its logic operation.

Train A SSPS

15 VDC/48 VDC #1 Power supply is powered from 1/2AY1A 120 VAC

15 VDC/48 VDC #2 Power supply is powered from 1/2CY1A 120 VAC

Train B SSPS

15 VDC/48 VDC #1 Power supply is powered from 1/2BY1B 120 VAC

15 VDC/48 VDC #2 Power supply is powered from 1/2DY1B 120 VAC

21. 015AK2.07 001/1/1/RCP MALF-RCP SEALS/C/A - 2.9 / 2.9/M-LOIT/HL-15R NRC/RO/TNT/DS

Given the following:

- The plant is at 17% power.

The following conditions exist on Reactor Coolant Pump # 2.

- # 1 seal D/P is 190 psig.

- # 1 seak leakoff flow is 5.2 gpm.

Which **ONE** of the following describes the required sequence / response to these conditions?

A. Shutdown the RCP, enter AOP-18005-C, "Partial Loss of Flow".

B. Commence a unit shutdown per UOP-12004-C, shutdown the RCP within 8 hours.

C.✓ Trip the Reactor, enter 19000-C, "Reactor Trip or Safety Injection", shutdown the RCP.

D. Maintain current power, shutdown the RCP with engineering / management concurrence.

<u>K/A</u>

015 Reactor Coolant Pump (RCP) Malfunctions

AK2.07 Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following:

RCP Seals

<u>K/A MATCH ANALYSIS</u>

The question presents a plausible scenario where an RCP seal immediate trip criteria has been reached (seal D/P). Seal leakoff of 5.2 gpm is below the immediate trip criteria set point and the student must recognize the D/P is at an immediate trip set point. The student must determine the correct actions and procedure to use for the listed conditions.

<u>ANSWER / DISTRACTOR ANALYSIS</u>

A. Incorrect. Immediate RCP trip criteria present due to low seal d/p, >15% power requires trip reactor, enter E-0, then trip RCP. These actions are plausible to enter the Partial Loss of Flow AOP if the candidate does not recall power > 15% requires

B. Incorrect. Immediate RCP trip criteria present due to low seal d/p, >15% power requires trip reactor, enter E-0, the trip RCP. However, this choice is plausible if the candidate does not recognize seal dP as too low and thinks the RCP does not require an immediate trip. Commence a unit shutdown and stop RCP within 8 hours is a choice on the seal abnormality flow chart.

C. Correct. Immediate RCP trip criteria present due to low seal d/p, >15% power requires trip reactor, enter E-0, then trip the RCP.

D. Incorrect. Immediate RCP trip criteria present due to low seal d/p, >15% power requires trip reactor, enter E-0, then trip the RCP. Management and engineering concurrence very plausible if student does not recognize 200 psid as trip criteria and realizes 5.2 gpm below immediate trip threshold.

REFERENCES

015/017AK1.02 used as base for modification from LOIT exam bank (included).

13003-C, Reactor Coolant Pump Operation, Precautions and Limitations, section 4.2 for "Pump Operation With A Seal Abnormality", and Figure 1 - RCP Seal Abnormalities Decision Tree.

VEGP learning objectives:

LO-PP-16401-03, Describe the Control Room indications for failure of an RCP seal.

1. 015/017AK1.02 001

Given the following:

- The plant is at 17% power. (Requires Rx trip, not AOP entry)
- RCP # 3 seal d/p is 190 psig (Requires Imm Trip but relatively new for us)
- RCP # 3 seal leakoff flow is 5.2 gpm (below Imm Trip criteria)
- ~~TSLB-4 bistables lights for P-8 are NOT illuminated~~. (Took out, just fluff)

Which **ONE** of the following describes the required sequence/response to these conditions?

A.✓ Trip the Reactor, trip RCP # 3, enter 19000-C, "Reactor Trip or Safety Injection"

B. Trip RCP # 3, commence a unit shutdown per UOP-12004-C, "Power Operation"

C. Commence a unit shutdown per UOP-12004-C, stop RCP # 3 within 8 hours.

D. Trip RCP # 3, enter AOP-18005-C, "Partial Loss of Flow".

Base question used.

Choice "A", the answer changed due to 13003 changes made not exactly correct.

Choice "B" changed to make only 2 choices have "trip" the RCP. Made more plausible with shutdown with management/engineering concurrence.

Re-sequenced order of choices. Answer now "C"

2.2.5 The following primary to secondary temperature limitations apply for RCP start:

- In order to prevent a low temperature RCS overpressure event, Technical Specification LCO 3.4.6, Note 2 requires that during MODE 4 operation below the COPS arming temperature as specified in the PTLR, the secondary side water temperature of each Steam Generator Temperature be less than 50°F above each of the RCS cold leg temperatures prior to the start of an RCP. Additionally, while in MODE 4 with no other RCPs running, this differential temperature limit is reduced to 25°F at an RCS temperature of 350°F and varies linearly to 50°F at an RCS temperature of 200°F as shown in figure 3. This verifies RHR system design pressures are not exceeded when the RHR suction reliefs are used for cold overpressure protection.

- To verify the above limits are not exceeded, an administrative limit, FSAR 5.2.2.10.2.c, is established such that an RCP shall NOT be started if its associated Steam Generator secondary water temperature is greater than 10°F above its RCS cold leg loop temperature.

- SGBD temperatures are preferred to SG skin temperatures when establishing conditions for starting a Reactor Coolant Pump. However, in Mode 5 SGBD is not required to be in service and SG skin temperatures can be used instead.

2.2.6 An RCP should NOT be started with the reactor critical. (Ref 18005-C)

2.2.7 The following conditions for the No. 1 Seal must be established prior to RCP start:

- 200-psid minimum differential pressure across No. 1 Seal.

- A minimum VCT pressure of 18 psig.

- Minimum No. 1 Seal Leakoff as obtained from Figure 2.

Plausible the student could think this only applies to an RCP start and Not Realize an immediate trip criteria is violated

2.2.8 The following starting duty cycle for the RCP should be observed:

- Only one RCP shall be started at any one time.

- Two successive starts are permitted, provided the motor is permitted to coast to a stop between starts.

- A third start may be made when the winding and core have cooled by running for a period of 20 minutes, or by standing idle for a period of 45 minutes.

2.2.9 During RCS filling and venting, RCS pressure must be greater than 325 psig prior to starting an RCP to verify adequate seal D/P is maintained throughout RCS fill and vent. If necessary, the RCP should be stopped prior to seal D/P dropping less than 200 psid. If the seal D/P goes below 200 psid during pump operation or coast down, the RCP should be evaluated before restarting the RCP. Plausible the student may think dP only applies to Fill + vent

2.2.10 An RCP shall be stopped if any of the following conditions exist.

- Motor bearing temperature exceeds 195°F.
- Motor stator winding temperature exceeds 311°F.
- Seal water inlet temperature exceeds 230°F
- Total loss of ACCW for a duration of 10 minutes.
- RCP shaft vibration of 20 mils or greater.
- RCP frame vibration of 5 mils or greater.
- Differential pressure across the number 1 seal of less than 200 psid.

dp > 200 is an immediate trip criteria. Relatively new to us.

2.2.11 If a loss of RCP seal cooling (Seal Injection and/or ACCW to Thermal barrier) occurs, resulting in RCP shutdown due to exceeding operating limits, then the unit should be cooled down to Mode 5 to facilitate recovery. Upon reaching Mode 5, ACCW to the Thermal barrier should be restored. Seal injection should then be returned to service. This sequence should prevent seal damage, RCP shaft bowing, ACCW System damage, etc. due to excessive thermal stresses.

4.2 SYSTEM OPERATION

4.2.1 Pump Operation With A Seal Abnormality

4.2.1.1 IF the Plant Computer is available, **trend** the computer data points listed in Table 2. ________

4.2.1.2 IF the Plant Computer is NOT available, **perform** the following:

a. **Monitor** the QMCB indication listed in Table 2 at least hourly for the next 8 hours. ________

b. IF NO further seal degradation exists after 8 hours, **consult** with the Shift Supervisor (SS) for less frequent monitoring. ________

4.2.1.3 **Monitor** the No. 1 seal for further degradation using Figure 1 and RCP Trip Criteria as follows:

a. **Evaluate** the monitored indications using Figure 1, "RCP Seal Abnormalities Tree". ________

b. IF evaluation of the monitored indications using Figure 1 requires immediate pump shutdown, **Go to** Step 4.2.1.4. ________

c. IF any of the following RCP Trip Criteria is exceeded, **Go To** Step 4.2.1.4 for immediate RCP shutdown. ________

RCP TRIP CRITERIA	
Motor bearing temperature	>195°F
Motor stator-winding temperature	>311°F
Seal water inlet temperature	>230°F
RCP shaft vibration	=20 mils
RCP Frame vibration	=5 mils
#1 seal Differential Pressure	<200 psid
#1 seal leakoff flow (sum of #1 seal leakoff as indicated on the MCB and #2 seal leakoff read locally in containment)	< minimum on Figure 2 with pump bearing / seal inlet temperature increasing
Total loss of ACCW for a duration of 10 minutes	

d. WHEN directed by Figure 1, **stop** the affected RCP within 8 hours as follows:

(1) **Establish** 9 gpm or greater seal injection flow to the affected pump. _______

(2) **Stop** the affected RCP by continuing with Step 4.2.1.4. _______

4.2.1.4 WHEN directed by the SS, **perform** an RCP shutdown as follows:

a. **Start** the RCP Oil Lift Pump for affected RCP, if available. _______

b. IF Reactor Power is greater than 15% Rated Thermal Power:

(1) **Trip** the Reactor and **initiate** 19000-C, "E-0 Reactor Trip Or Safety Injection". _______

(2) WHEN the immediate operator actions of 19000-C are complete, **Go to** Step 4.2.1.4.d. _______

c. IF Reactor Power is less than 15% Rated Thermal Power, **initiate** 18005-C, "Partial Loss Of Flow".

Plausible if ~~candidate does~~ Not Recognize 15% power level as trip Reactor threshold. Makes "A" plausible

d. **Stop** the RCP by placing the RCP Non-1E Control Switch in STOP and then placing the RCP 1E Control Switch in STOP:

	RCP	Non-1E Control Switch	1E Control Switch	
•	Loop 1	1-HS-0495B	1-HS-0495A	_______
•	Loop 2	1-HS-0496B	1-HS-0496A	_______
•	Loop 3	1-HS-0497B	1-HS-0497A	_______
•	Loop 4	1-HS-0498B	1-HS-0498A	_______

CAUTION

If RCP #1 or #4 is stopped, the associated Spray Valve is placed in manual and closed to prevent spray short cycling.

e. IF RCP #1 OR #4 is stopped, **verify** its associated spray valve is placed in MANUAL AND closed.

- RCP 1: 1-PIC-0455C ________
- RCP 4: 1-PIC-0455B ________

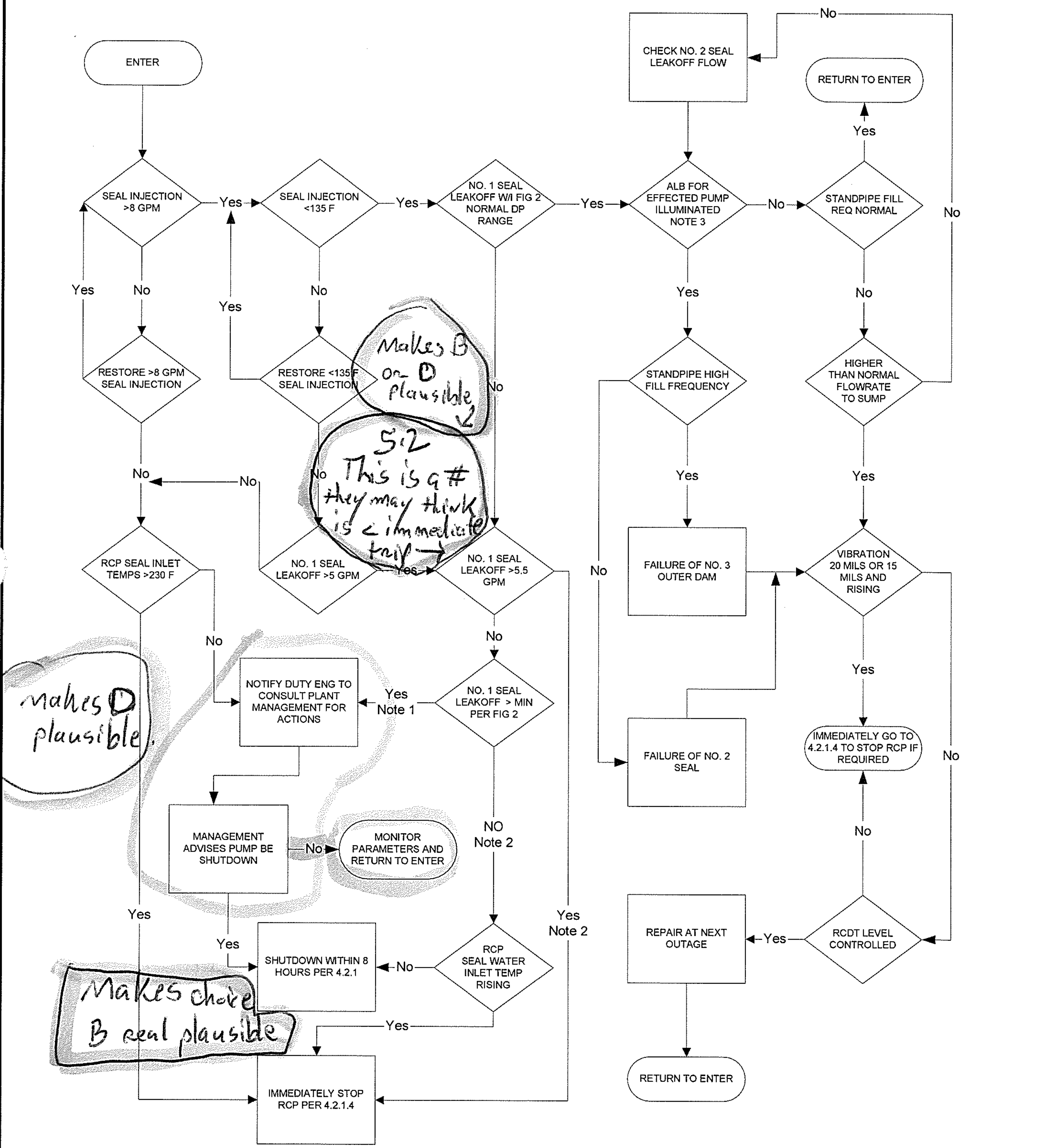
f. WHEN the RCP comes to a complete stop (as indicated by reverse flow), **close** the RCP Seal Leakoff Isolation valve for the affected pump.

- RCP 1: 1-HV-8141A ________
- RCP 2: 1-HV-8141B ________
- RCP 3: 1-HV-8141C ________
- RCP 4: 1-HV-8141D ________

g. **Secure** the associated RCP Oil Lift Pump. ________

h. IF RCP shutdown was due to loss of RCP seal cooling, **review** Limitation 2.2.11 for recovery action. ________

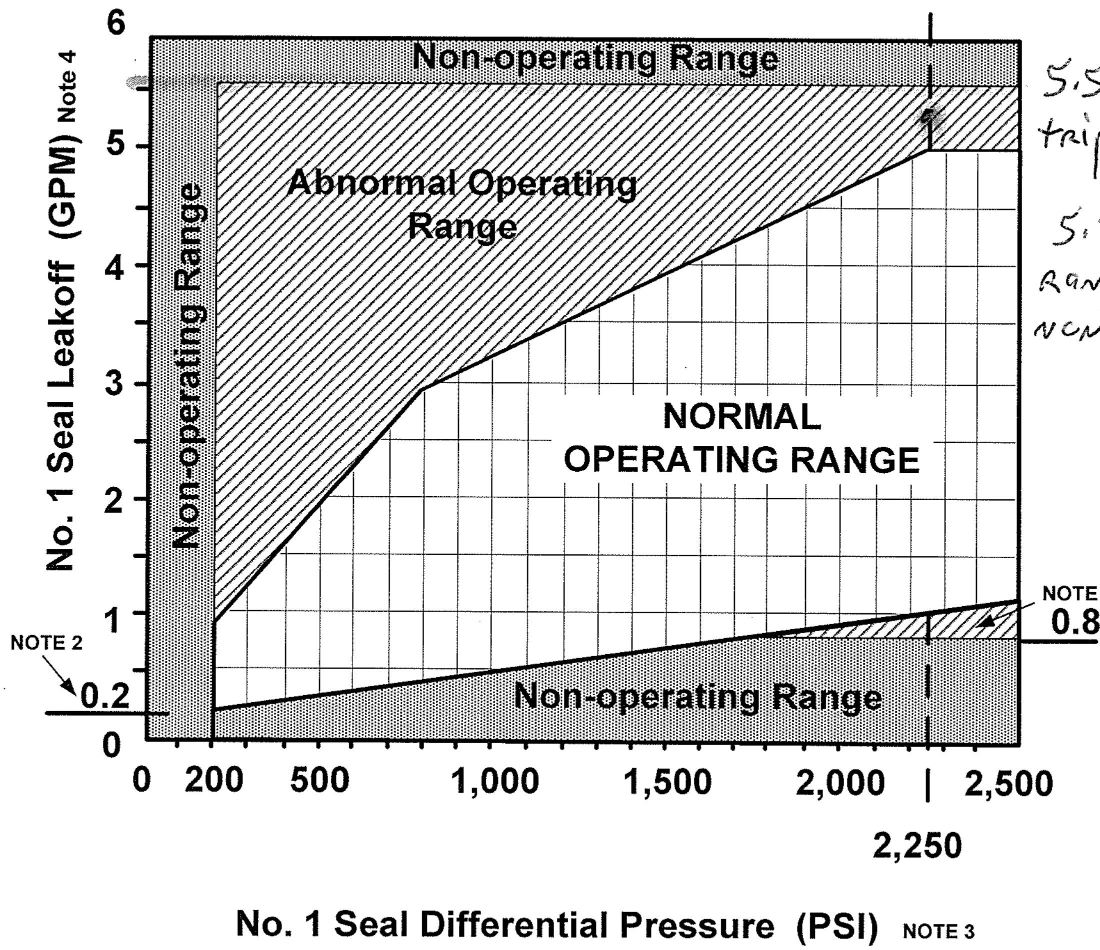
FIGURE 1 - RCP SEAL ABNORMALITIES DECISION TREE



Note 1: Abnormal Operating Range of Figure 2
Note 2: Non-operating Range of Figure 2
Note 3: ALB08 A-04, B-04, C-04 or D-04

FIGURE 2

NO. 1 SEAL NORMAL OPERATING RANGE



5.5 is immediate trip thresheld.

5.2 is abnormal Range NOT non-operating Range

1. If the No. 1 seal leak rates are outside the normal (1.0-5.0 gpm) but within the operating limits ((0.8-5.5 gpm), continue pump operation. VERIFY that seal injection flow exceeds No. 1 seal leak rate for the affected RCP. Closely monitor pump and seal parameters and contact Engineering for further instructions.

2. Minimum startup requirements are 0.2 gpm at 200 PSID differential across the No. 1 seal. For startups at differential pressures greater than 200 PSID, the minimum No. 1 seal leak rate requirements are defined in the NO. 1 SEAL NORMAL OPERATING RANGE (e.g., at 1000 psi differential pressure, do not start the RCP with less than 0.5 gpm).

3. No.1 Seal Differential Press = RCS WR Press – VCT Press.

4. Per Westinghouse Technical Bulletin ESBU-TB-93-01-R1, total #1 seal leakoff is the sum of #1 seal leakoff and #2 seal leakoff. #1 seal leakoff is read directly at the MCB and #2 seal leakoff can be obtained from instrumentation in Containment.

22. 016K1.08 001/2/2/INSTR-PZR PCS/C/A - 3.4 / 3.4/NEW/HL-15R NRC/RO/TNT/DS

Given the following conditions:

- The unit is at 100% power.

- PRZR pressure control is selected to the 457 / 456 position.

The OATC determines that the controlling channel for Pressurizer Pressure control has failed.

If NO action is taken by the crew, which **ONE** of the following describes a **CORRECT** plant response for the failure given?

A. PT-457 fails high, PRZR pressure cycles between 2185 psig and 2200 psig.

B. PT-456 fails low, PRZR pressure cycles between 2345 psig and 2325 psig.

C. PT-456 fails low, reactor trips on the high PRZR Pressure setpoint.

D.✓ PT-457 fails high, reactor trips on the low PRZR Pressure setpoint.

K/A

016 Non-Nuclear Instrumentation System (NNIS)

K1.08 Knowledge of the physical connections and/or cause effect relationships between the NNIS and the following systems:

PZR PCS

K/A MATCH ANALYSIS

The question presents a plausible scenario where a Pressurizer Pressure channel has failed, the student must determine the correct plant response.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. This is the correct plant response for PT-456 failing high causing PORV 456 to open. As RCS pressure lowers, the block valves would cycle near 2185 psig. Plausible the student could invert the channels and plant response.

B. Incorrect. This is the correct plant response for PT-455 failing low causing the PORV to cycle around the high pressure setpoint. However, PORV 456 would cycle from 2335 to 2315, this is the setpoint for the wrong PORV. Plausible the student could invert the channels and plant response.

C. Incorrect. PT-456 failing low would have no effect on the plant response. However, if the student inverts the controlling channels, it is plausible to think controlling channel could rise and cause a reactor trip.

D. Correct. PT-457 failed high would result in spray valves and a PORV failing open. RCS pressure would lower to the 2185 setpoint at which point the block valves would attempt to stabilize the plant, however, sprays failed open would lower RCS pressure to the low pressure trip setpoint.

REFERENCES

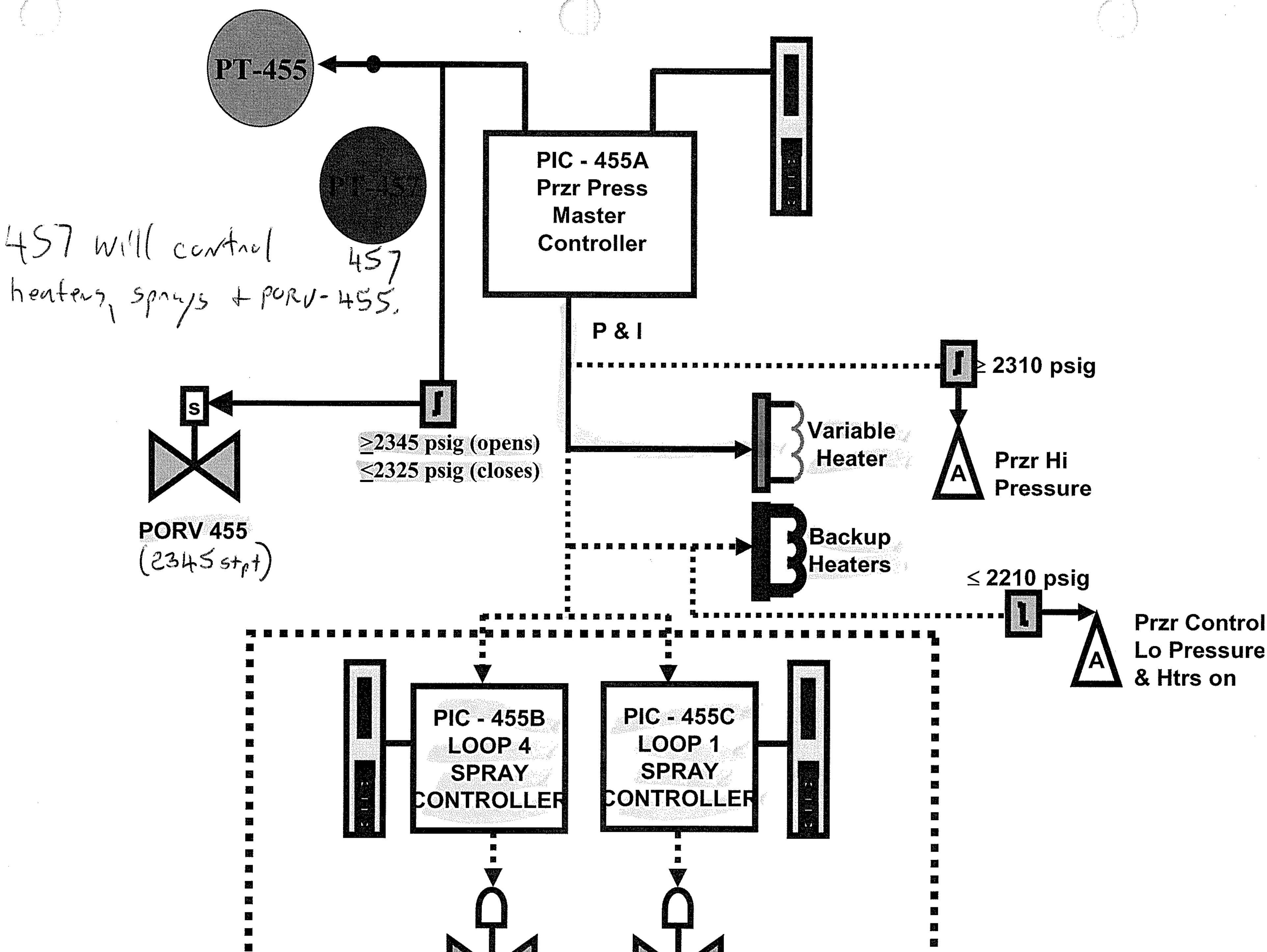
V-LO-PP-16303, Pressurizer Pressure Control System, slides 40, 42, 51, 71, 72, 153, 154, 156, and 157.

VEGP learning objectives:

LO-PP-16303-02, Describe how the Pressurizer Pressure Control System responds to the following failures.

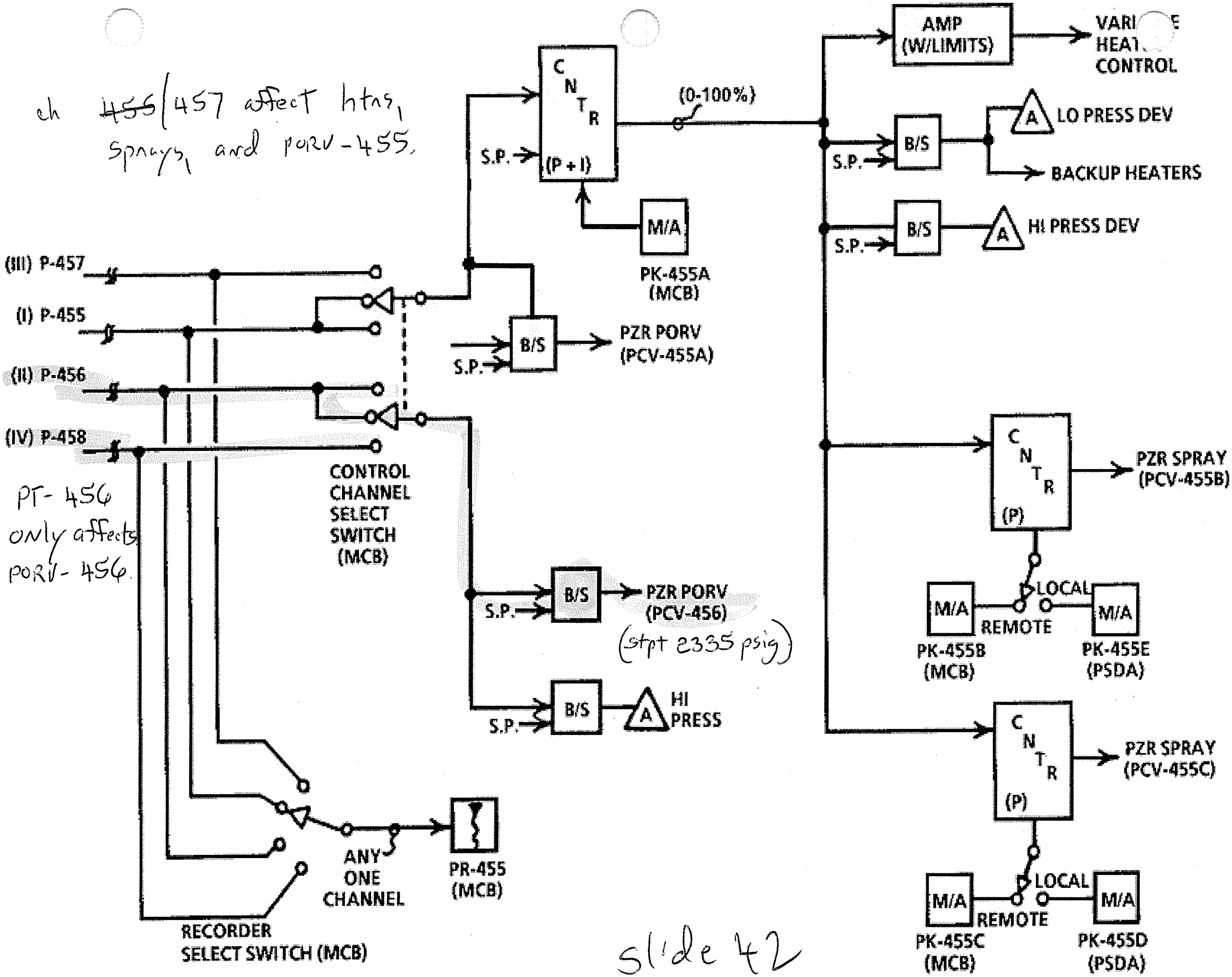
a. Controlling (primary and secondary) channel fails low.

b. Controlling (primary and secondary) channel fails high.



Pressurizer Controlling
Pressure Control

ch ~~455~~/457 affect htrs, sprays, and PORV-455.



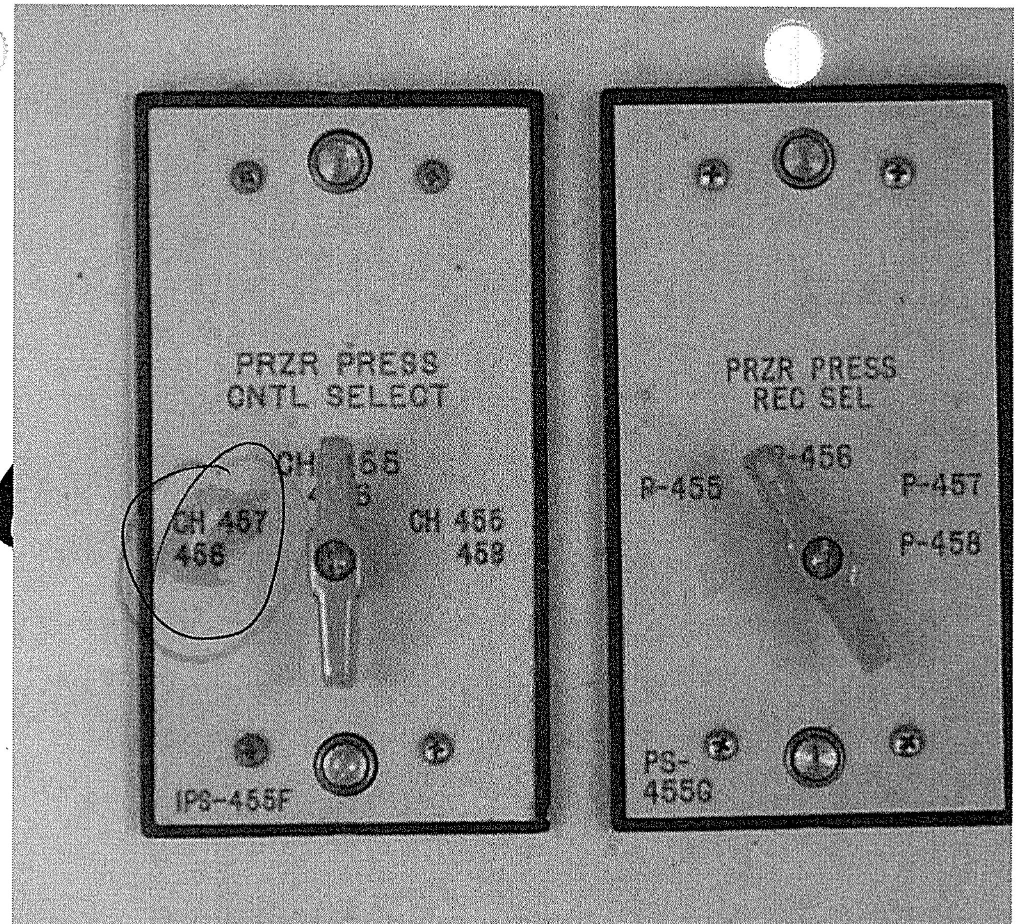
PT-456 only affects PORV-456.

(stpt 2335 psig)

slide 42

Example:

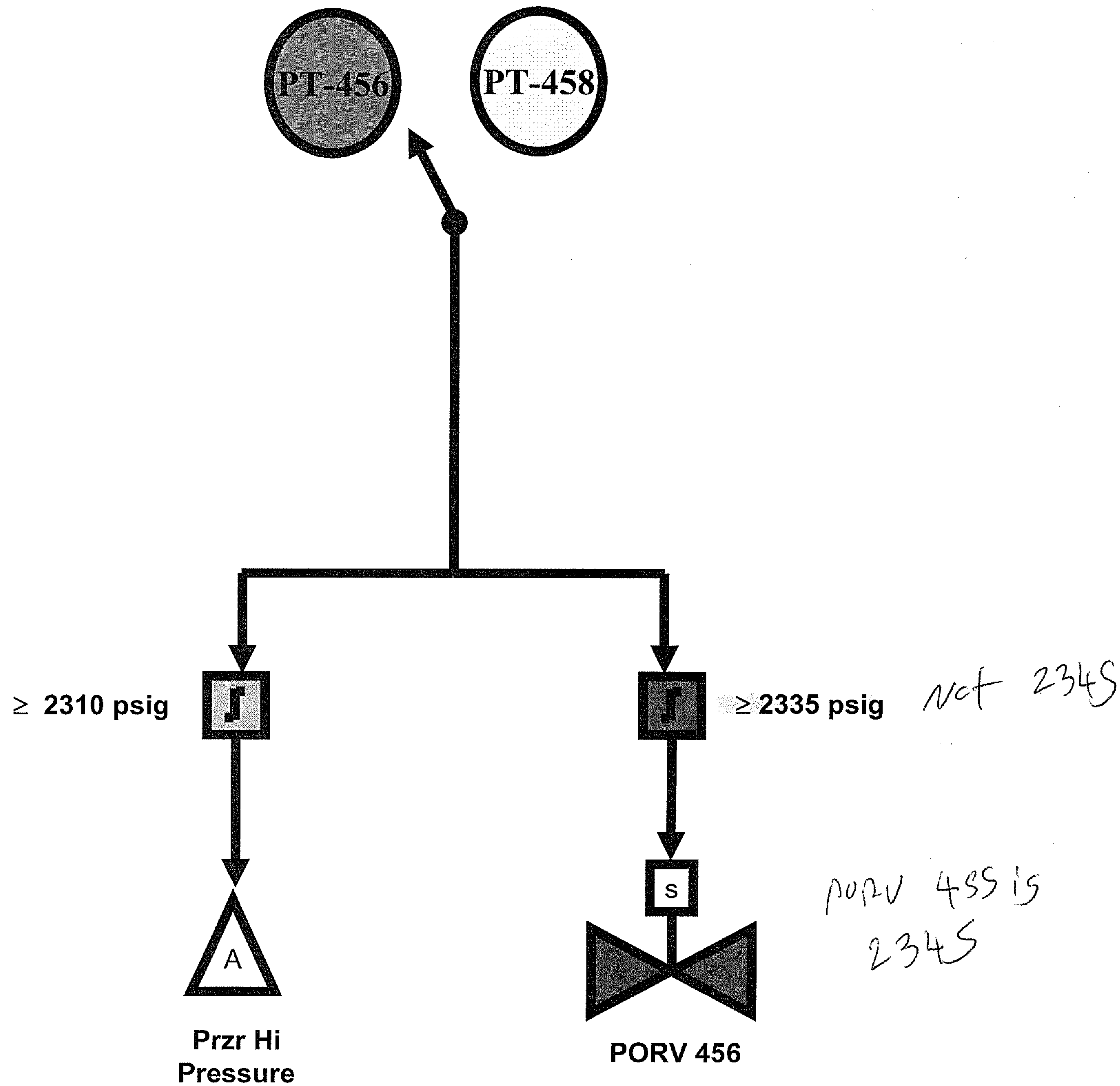
455/456



PT-455 is the primary channel. PT-456 is the secondary channel.

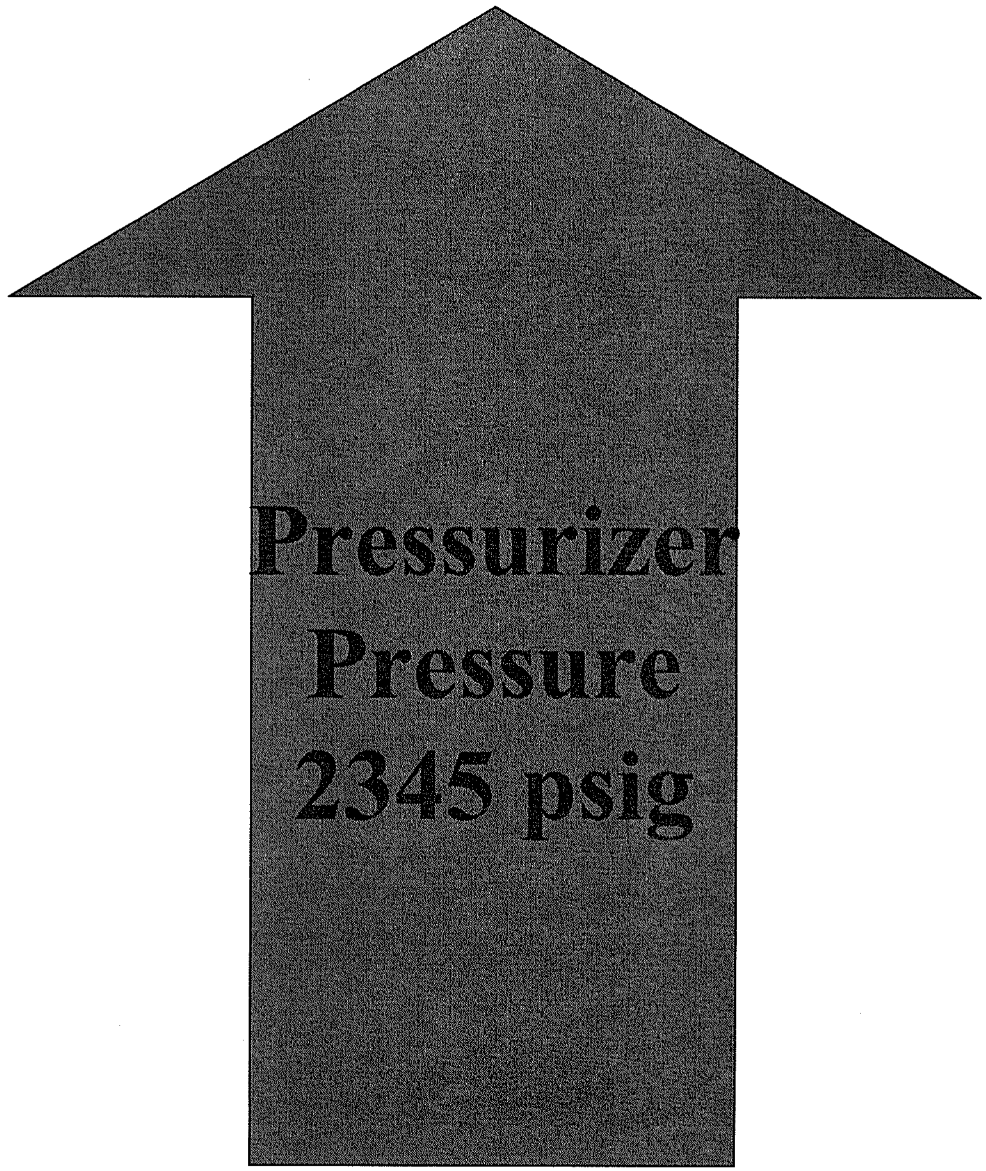
457/456

457 controls



Pressurizer Non Controlling
Pressure Control

PORV-455



457

455/456

457

- ***PT-455 fails high,***
  - ***PORV-455 opens (When Pressurizer pressure lowers to 2185 psig the PORV interlock function will close the PORV),*** choice "A" Plausible to think cycles at 2185 (spray valves cause Rx trip)
  - ***Both Pressurizer proportional spray valves will fully open,*** Pressure lowers to Rx trip setpoint,
  - ***Proportional heaters will turn off,***
  - ***Pressurizer High Pressure Alarm,***
  - ***Pressurizer High Pressure Alert,***

457

455/456

- ***PT-456 fails high:*** Also makes A plausible,
  - ***PORV-456 opens (When Pressurizer pressure lowers to 2185 psig the PORV interlock function will close the PORV),*** This would cause cycle ~ 2185
  - ***Pressurizer High Pressure annunicator,***
  - ***Pressurizer High Pressure Alert annunicator,***

457

455/456

457

- ***PT-455 fails low,*** Makes B plausible but the wrong PORV setpoint is in choice (2345 → 2325 is for the other PORV)
  - ***Proportional heaters will go to maximum output,***
  - ***All Pressurizer backup heaters will energize,***
  - ***Pressurizer Control Low Pressure and Heaters on annunciator,***
  - ***Pressure Low Pressure Channel Alert annunciator,***
  - ***Pressurizer Low Pressure SI Alert annunciator,***

457

455/456

choice C

Plausible to invert channels & think heaters on, pressure ↑ to Rx trip setpoint,

- ***PT-456 fails low:***
  - ***Pressure Low Pressure Channel Alert annunicator,***
  - ***Pressurizer Low Pressure SI Alert annunicator,***

23. 022A1.01 001/2/1/CNMT COOL-CNMT TEMP/MEM - 3.6 / 3.7/NEW/HL-15R NRC/RO/TNT/DS

CNMT HI TEMP alarm has just annunciated.

The UO notes that CNMT air temperature is rising with CNMT coolers 1, 2, 5, and 6 in service on high speed.

The unit is shutdown with RCS temperature 375 F.

What is the correct action for the UO to take to stop the CNMT air temperature rise?

A.✓ Start CNMT coolers 3 and 4 simultaneously on high speed.

B. Start CNMT coolers 3 and 7 simultaneously on high speed.

C. Start CNMT coolers 3 and 4 sequentially on high speed.

D. Start CNMT coolers 3 and 7 sequentially on high speed.

K/A

022 Containment Cooling System (CCS)

A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:

Containment temperature.

K/A MATCH ANALYSIS

The question presents a scenario where the student must determine the correct operation of the containment cooling system to lower containment temperature below its technical specification limit of 120 F as indicated by the high temperature alarm.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Per ARP 17001-1 window E06, an additional pair of containment coolers should be started. SOP 13120-1 furhter states that each pair of coolers should be started simultaneously. This is done to prevent reverse flow through a cooler since each pair shares common discharge plenums and the back draft dampers are locked open.

B. Incorrect. This would be the correct action to take per the ARP 17001-1 however this pair of coolers is not allowed by the SOP 13120-1.

C. Incorrect. This is the correct pair of coolers to start. However they should be started associated pairs of containment coolers.

D. Incorrect. This would be the correct action to take per the ARP 17001-1 however this pair of coolers is not allowed by the SOP 13120-1. Additionally, the coolers should be started simultaneously rather than sequentially. Sequential starting is required between associated pairs of containment coolers.

REFERENCES

ARP 17001-1, Window E06

SOP 13120-1, "Containment Building Cooling System" pages 8 and 9

LO-PP-29101, "Containment HVAC Systems" presentation, pages 17 and 18

VEGP learning objectives:

LO-PP-29101-08:

Describe routine actions taken to adjust Containment pressure and temperature.

WINDOW E06

ORIGIN	SETPOINT	
1-TSH-2563 1-TSH-2612 1-TSH-2613	120°F	CNMT HI TEMP

1.0 **PROBABLE CAUSE**

Insufficient number of Containment Building Cooling Units operating.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Start** an additional pair of Containment Cooling Units or a Containment Auxiliary Cooling Unit per 13120-1, "Containment Building Cooling Systems".

2. **Verify** Nuclear Service Cooling Water flow to coolers, and if necessary, **dispatch** an operator to inspect the Containment Heat Removal System.

3. Refer to Technical Specification LCO 3.6.5 and 3.6.6.

4. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB212, CX5DT101-66, CX5DT101-71

INITIALS

4.2 CONTAINMENT HEAT REMOVAL SYSTEM STARTUP

NOTES

- Normal operation is four fans running in high speed. Running all eight fans in high speed or operation of fans in low speed may be performed at SS direction.
- Containment Coolers should be operated in one of the combinations specified.
- After start of the first two Coolers in a specified combination, the operator should wait 20 seconds before starting the second pair in that combination. This will limit voltage drop on the respective switchgear.

4.2.1 To start fans in High speed **perform** Step 4.2.3. ________

4.2.2 To start fans in Low speed **perform** Step 4.2.4. ________

4.2.3 **Select** one of the following four combinations and **start** the Containment Coolers one pair at a time, in high speed, by **simultaneously** placing both handswitches to the start position:

a. Combination 1 High speed operation

Train A

(1) Fan 1, 1-HS-12582D (B24) QHVC ________

Fan 2, 1-HS-2582D (B25) QHVC ________

(2) Fan 5, 1-HS-12584D (D24) QHVC ________

Fan 6, 1-HS-2584D (D25) QHVC ________

INITIALS

b. Combination 2 High speed operation

Train B

(1) Fan 3, 1-HS-12583D (B26) QHVC ________

Fan 4, 1-HS-2583D (B27) QHVC ________

(2) Fan 7, 1-HS-12585D (D26) QHVC ________

Fan 8, 1-HS-2585D (D27) QHVC ________

c. Combination 3 High speed operation

Train A

(1) Fan 1, 1-HS-12582D (B24) QHVC ________

Fan 2, 1-HS-2582D (B25) QHVC ________

Train B

(2) Fan 7, 1-HS-12585D (D26) QHVC ________

Fan 8, 1-HS-2585D (D27) QHVC ________

d. Combination 4 High speed operation

Train A

(1) Fan 5, 1-HS-12584D (D24) QHVC ________

Fan 6, 1-HS-2584D (D25) QHVC ________

Train B

(2) Fan 3, 1-HS-12583D (B26) QHVC ________

Fan 4, 1-HS-2583D (B27) QHVC ________

Containment Cooling Units

- The fans are normally operated in pairs from the QHVC during normal operation with backup capability from their associated shutdown panels. Pairs are selected on the basis that their discharges are tied together to a common duct and to ensure balance flows to all cooled areas.

V-LO-PP-29101

Containment Cooling Units

- Motorized discharge dampers are de-energized and locked open to preclude loss of fan capability during an emergency.

V-LO-PP-29101

24. 022A2.03 001/2/1/CNMT COOL-HI SPEED/C/A - 2.6 / 3.0/NEW/HL-15R NRC/RO/DS/TNT

Initial conditions:

A steamline break inside containment has occurred
EOP 19010-C, "Loss of Reactor or Secondary Coolant" is being implemented

The following sequence of events occurs:

The SI signal is reset
CNMT pressure is 8.6 psig and lowering
A loss of both RATs occurs
Both EDGs start and re-energize their respective busses

The correct action to take is to...

A. verify the sequencers start all CNMT coolers on low speed.

Then shift the CNMT coolers to high speed after the sequencing is completed.

B. verify the sequencers start all CNMT coolers on low speed.

Then restart the SI and RHR pumps as needed to maintain RCS inventory.

C✓ verify the sequencers start all CNMT coolers on high speed.

Then shift the CNMT coolers to low speed to prevent a CNMT cooler fan overcurrent condition.

D. verify the sequencers start all CNMT coolers on high speed.

Then shift the CNMT coolers to low speed to prevent an EDG overcurrent condition.

K/A

022 Containment Cooling System (CCS)

A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

Fan motor thermal overload/high-speed operation.

The question presents a scenario where a UV condition following a SI actuation and reset occurs. The student must determine the correct action to take to prevent an over current condition on CNMT cooler motors for the conditions given.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. The SI signal was reset prior to the loss of the RATs. This will result in the sequencers running the UV sequence instead of the SI sequence. Since CNMT pressure is > 3.8 psig (adverse) the CNMT coolers should be shifted to low speed to prevent exceeding the motor current limits on the coolers due to the high density fluid conditions in CNMT. This choice is plausible since the CNMT coolers would start in low speed if the SI signal were not reset.

B. Incorrect. The SI signal was reset prior to the loss of the RATs. This will result in the sequencers running the UV sequence instead of the SI sequence. Since CNMT pressure is > 3.8 psig (adverse) the CNMT coolers should be shifted to low speed to prevent exceeding the motor current limits on the coolers due to the high density fluid conditions in CNMT. This choice is plausible since the CNMT coolers would start in low speed if the SI signal were not reset. Additionally the SI and RHR pumps would have to be manually started to help maintain RCS inventory for these conditions.

C. Correct. The SI signal was reset prior to the loss of the RATs. This will result in the sequencers running the UV sequence instead of the SI sequence. Since CNMT pressure is > 3.8 psig (adverse) the CNMT coolers should be shifted to low speed to prevent exceeding the motor current limits on the coolers due to the high density fluid conditions in CNMT.

D. Incorrect. The SI signal was reset prior to the loss of the RATs. This will result in the sequencers running the UV sequence instead of the SI sequence. Since CNMT pressure is > 3.8 psig (adverse) the CNMT coolers should be shifted to low speed to prevent exceeding the motor current limits on the coolers due to the high density fluid conditions in CNMT. This choice is plausible since loading of the EDGs is a concern and the reason for running the SI or UV sequences.

REFERENCES

EOP 19010-C, "E-1 Loss of Reactor or Secondary Coolant" page 11

LO-PP-29101, "CNMT HVAC Systems" presentation, pages 20, 22, and 23

VEGP learning objectives:

LO-PP-29101-04:

State how the Containment atmosphere density changes following a LOCA and why.

HL-15R RO NRC Exam

State why two speeds are provided for the Containment Coolers and when each speed is used.

LO-PP-29101-14:

State all auto start signals for the Containment Cooling including set points and coincidence where applicable.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

If offsite power is lost after SI reset, action is required to restart the following ESF equipment if plant conditions require their operation:

- RHR Pumps
- SI Pumps
- Post-LOCA Cavity Purge Units
- Containment Coolers in low speed (Started in high speed on a UV signal).
- ESF Chilled Water Pumps (If CRI is reset).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. Check if RHR Pumps should be stopped:	
__a. RHR Pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST.	__a. Go to Step 15.
b. RCS pressure:	
__1) Greater than 300 psig.	__1) Go to Step 16.
__2) Stable or rising.	__2) Go to Step 15.
__c. Reset SI.	
__d. Stop RHR Pumps.	
__*14. **IF RCS pressure lowers in an uncontrolled manner to less than 300 psig, THEN restart RHR Pumps.**	

Containment Cooling Units <u>Normal</u> Operations

- The Containment Cooling Units help maintain containment temperature with four of the eight units running in **fast speed (97,000 cfm).**
- The eight fans are located at the 240 feet elevation, four on each side of the containment behind the SG enclosures.

V-LO-PP-29101

Containment Cooling Units <u>Emergency</u> Operations Safety Injection

- All eight fans start in **slow speed**
  - (43,500 cfm)
  - Start signal from SI Sequencer
  - **Slow speed is used rather than fast speed due to the adverse containment conditions. The higher density pressurized air could cause over current conditions (and possible damage) if the fast speed windings are energized.**
- Four fans (single Train) in slow speed provide adequate heat removal

V-LO-PP-2961

page 22

Containment Cooling Units <u>Emergency</u> Operations Loss of Offsite Power

- Fans will be shed from the affected buss(es) and restarted in fast speed (97,000 CFM)
- All fans get at start signal at 30.5 secs
- Fans 1&2 (Train A) and 3&4 (Train B)

  Time Delayed 20 secs

    Prevents overloading DG & Bus voltage swings

V-LO-PP-29101

25. 022AK1.01 001/1/1/LOSS CHARGE-T SHOCK/C/A - 2.8 / 2.5/NEW/HL-15R NRC/RO/TNT/DS

Given the following conditions:

- A loss of all AC power occurs in Mode 1.

- The plant is currently in Mode 3.

- HV-8103A, B, C, D Seal Injection Isolation Valves are CLOSED.

Which **ONE** of the following is **CORRECT** regarding:

1) The Mode in which seal injection will be re-established and

2) the reason for closing the seal injection isolation valves?

A. Maintain Mode 3

To prevent steam binding the charging pumps via back leakage in the seal lines.

B. Maintain Mode 3

To prevent seal damage and RCP shaft bowing due to excessive thermal stresses.

C. Cooldown to Mode 5

To prevent steam binding the charging pumps via back leakage in the seal lines.

D.✓ Cooldown to Mode 5

To prevent seal damage and RCP shaft bowing due to excessive thermal stresses.

<u>K/A</u>

022 Loss of Reactor Coolant Makeup

AK1.01 Knowledge of the operational implications of the following concepts as they apply to the Loss of Reactor Coolant Makeup:

Consequences of thermal shock to RCP seals.

<u>K/A MATCH ANALYSIS</u>

which is our most likely cause of Loss of Charging and Seal Injection. The question asks the student to recall the strategy for re-establishing seal cooling and the possible consequences if this is not done properly.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Per WOG background documents for Vogtle, a plant cooldown will be performed to reduce RCP seal temperatures. SOP-13003-1/2 Limitation 2.2.11 also states the plant should be cooled down to Mode 5 if a loss of RCP seal cooling has occurred. This is to prevent seal damage and RCP shaft bowing due to excessive thermal stresses. The charging pump binding via seal lines is plausible since it sounds like the steam binding of the ACCW system which is mentioned in our procedures and background documents, plausible the student could confuse this.

B. Incorrect. Per WOG background documents for Vogtle, a plant cooldown will be performed to reduce RCP seal temperatures. SOP-13003-1/2 Limitation 2.2.11 also states the plant should be cooled down to Mode 5 if a loss of RCP seal cooling has occurred. This is to prevent seal damage and RCP shaft bowing due to excessive thermal stresses.

C. Incorrect. Per WOG background documents for Vogtle, a plant cooldown will be performed to reduce RCP seal temperatures. SOP-13003-1/2 Limitation 2.2.11 also states the plant should be cooled down to Mode 5 if a loss of RCP seal cooling has occurred. This is to prevent seal damage and RCP shaft bowing due to excessive thermal stresses. The charging pump binding via seal lines is plausible since it sounds like the steam binding of the ACCW system which is mentioned in our procedures and background documents, plausible the student could confuse this.

D. Correct. Plant should be cooled down to Mode 5 to prevent possible seal damage and bowing of RCP shafts.

REFERENCES

SOP-13003-1, Reactor Coolant Pump Operation, Limitation 2.2.11.

LO-HO-37031-001, Loss of All AC Power page 1-3.

VEGP learning objectives:

LO-LP-37031-02, State why the RCP is a primary concern during a Loss of All AC Power condition

LO-PP-16401-005, Given a loss of RCP seal injection, describe the indictions that would be monitored and impact to continued operation of the RCP.

2.2.9 During RCS filling and venting, RCS pressure must be greater than 325 psig prior to starting an RCP to verify adequate seal D/P is maintained throughout RCS fill and vent. If necessary, the RCP should be stopped prior to seal D/P dropping less than 200 psid. If the seal D/P goes below 200 psid during pump operation or coast down, the RCP should be evaluated before restarting the RCP.

2.2.10 An RCP shall be stopped if any of the following conditions exist.

- Motor bearing temperature exceeds 195°F.
- Motor stator winding temperature exceeds 311°F.
- Seal water inlet temperature exceeds 230°F
- Total loss of ACCW for a duration of 10 minutes.
- RCP shaft vibration of 20 mils or greater.
- RCP frame vibration of 5 mils or greater.
- Differential pressure across the number 1 seal of less than 200 psid.

2.2.11 If a loss of RCP seal cooling (Seal Injection and/or ACCW to Thermal barrier) occurs, resulting in RCP shutdown due to exceeding operating limits, then the unit should be cooled down to Mode 5 to facilitate recovery. Upon reaching Mode 5, ACCW to the Thermal barrier should be restored. Seal injection should then be returned to service. This sequence should prevent seal damage, RCP shaft bowing, ACCW System damage, etc. due to excessive thermal stresses.

To evaluate the most severe consequences of a loss of all A.C. power to the RCP seal system, a conservative maximum RCP leakage rate has been estimated to be 300 gpm. This rate was estimated by assuming that total RCS pressure of 2235 psig exists across the RCP thermal barrier labyrinth seals with the controlled leakage seals totally ineffective in controlling leakage flow.

The high RCS temperatures and pressures characteristic of a plant no-load condition can lead to eventual RCP seal degradation and increased RCS inventory loss. This seal degradation can be mitigated by reducing the RCS pressure and temperature consistent with other plant constraints. Reducing RCS pressure reduces leakage flow through the RCP seals, thereby reducing RCS inventory loss for a given seal condition.

Reducing RCS temperature reduces the thermal degradation of materials and thermal expansion effects that tend to degrade the seal system sealing capability and sealing life. Consequently, any actions to reduce RCS pressure and temperature during a loss of all A.C. power event will reduce RCS inventory loss and will increase the time to core uncovering.

<u>RCP Seal System Cooling NON-Restoration</u>

The effect of restoring ACCW to the thermal barrier heat exchanger following an extended loss of seal cooling event cannot be fully understood without performing detailed analyses. Therefore, the only conclusions that can be made is that restoring seal cooling following an extended loss of all ac power event could jeopardize the integrity of the ACCW system. The plant that the generic Westinghouse Emergency Response Guidelines are based on provides thermal barrier cooling by the CCW system. This plant does not have a separate ACCW system. Since CCW is a safety-related system, and while restoring seal cooling is good practice, it is not necessary to ensure the health and safety of the public. **Therefore, the Westinghouse Owners Group Operations Subcommittee has taken the position that the integrity of the CCW system should not be jeopardized to restore seal cooling. In that light, we will not restore ACCW cooling the the thermal barrier heat exchangers here at Vogtle.**

Without thermal barrier cooling, two options are available to resolve the seal cooling issue; reestablish seal injection regardless of the potential damage that will occur to the RCP seal package and shaft, or find an alternate method of cooling the seal package. **The Operations Subcommittee has decided that an alternate method of cooling the seal package should be employed to minimize damage to the RCP. This alternate method will consist of reducing primary system temperature, which will reduce the temperature of the water flowing through the pump seals.** Reducing the seal temperature via a controlled RCS cooldown has several advantages. First, the seal package should cool evenly, minimizing thermal gradients placed on the seal package and minimizing the potential for RCP shaft warping. Second, as seal temperatures are reduced, the seal leakage is likely to decrease due to the associated decrease in system pressure and differential pressure across the seals. Finally, using this alternate seal cooling approach will not jeopardize any plant safety systems.

Note that relying on an RCS cooldown to reduce seal temperatures will result in continued seal leakage until seal injection can be reestablished. This leakage will not result in core uncovery and should be within the capacity of the normal charging system. However, the leakage will be diverted to the PRT and will eventually cause the rupture disc to fail. This will result in the spilling of reactor coolant fluid on to the containment floor. However, this concern would also exist when restoring seal cooling using ACCW to the thermal barrier heat exchanger since the cooldown rate of the seals is limited to 1°F per minute regardless of the method used to cool the seals (i.e., thermal barrier cooling would not reduce leakage any faster than an RCS cooldown). Also, the thermal barrier heat exchanger may not even have the capacity to cool the seals to the necessary temperature unless done concurrently with an RCS cooldown. Finally, the consequences of rupturing the PRT are much less severe than the consequences of failing the entire ACCW system. Therefore, cooling the RCP seals via an RCS cooldown instead of using the thermal barrier heat exchanger should not have a significant impact on plant safety, and may actually improve plant safety by maintaining the integrity of the ACCW system.

Based on the above arguments, during the recovery from an extended loss of all ac power event, no attempt will be made to restore seal cooling via the thermal barrier heat exchanger. Instead, seal cooling will be restored via a controlled RCS cooldown. The limits on restoring seal injection contained in the RCP vendors manual will still be observed.

26. 025AA1.19 001/1/1/LOSS RHR-BLOCK VALVE/C/A - 2.6 / 2.4/NEW/HL-15R NRC/RO/TNT/DS

Given the following sequence:

- The plant is in Mode 6 at midloop.
- RHR pump "A" trips due to a loss of RCS inventory.
- The RCS has been refilled and RHR pump "B" is ready to be started.

Complete the following two sentences:

1) To start the pump, the RHR Hx Bypass Valve controller (FIC-0619) should be...

2) To ensure compliance with Tech Spec flow requirements, per procedure the potentiometer setting for the RHR Hx Bypass Valve controller (FIC-0619) should be set at...

Given: Formula for Potentiometer setting in gpm is (desired flow / 5000)^2 X 10.

	RHR Hx Bypass valve	Potentiometer setting
A.	in automatic.	3.6
B.	in automatic.	4.1
C.	in manual and closed.	3.6
D✓	in manual and closed.	4.1

K/A

025 **Loss of Residual Heat Removal System (RHRS)**

AA1.19 **Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System:**

Block orifice bypass valve controller and indicators.

K/A MATCH ANALYSIS

The question presents a plausible scenario where an RHR pump is to be started while at midloop. The student must know the required position of the RHR Hx Bypass Valve (FIC-0619) during pump start, and the potentiometer setting to ensure 3000 gpm flow to comply with Technical Specifications.

NOTE: A reference with the RHR Hx Bypass valve calculation formula is to be provided to the students. (Page # 33 of SOP-13011, Residual Heat Removal) is

where this formula is given.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Pump should be started with the RHR Hx Bypass Valve (FIC-0619) closed. Plausible the student could choose automatic in order to control flow at the Tech Spec requirement. However, AOP-18019, Loss of RHR and SOP-13011, RHR both specify pump start with the Hx outlet and Bypass valves closed.

The potentiometer setting is calculated by $(\text{desired flow} / 5000)^2 \times 10$. The # 3.6 is plausible due to $(3000 / 5000)^2 \times 10 = (0.6)^2 \times 10 = (0.36) \times 10 = 3.6$.

However, to ensure 3000 gpm flow under all conditions, Tech Spec Rounds and procedures call for 3200 gpm flow. the setting for this is 4.1.

The potentiometer setting is calculated by $(\text{desired flow} / 5000)^2 \times 10$. The # 4.1 is correct due to $(3200 / 5000)^2 \times 10 = (0.64)^2 \times 10 = (0.41) \times 10 = 4.1$.

The student is required to recall whether flow should be set at 3200 vs 3000 gpm.

B. Incorrect. Pump should be started with the RHR Hx Bypass Valve (FIC-0619) closed. Plausible the student could choose automatic in order to control flow at the Tech Spec requirement. However, AOP-18019, Loss of RHR and SOP-13011, RHR both specify pump start with the Hx outlet and Bypass valves closed.

The potentiometer setting is calculated by $(\text{desired flow} / 5000)^2 \times 10$. The # 3.6 is plausible due to $(3000 / 5000)^2 \times 10 = (0.6)^2 \times 10 = (0.36) \times 10 = 3.6$.

However, to ensure 3000 gpm flow under all conditions, Tech Spec Rounds and procedures call for 3200 gpm flow. the setting for this is 4.1.

The potentiometer setting is calculated by $(\text{desired flow} / 5000)^2 \times 10$. The # 4.1 is correct due to $(3200 / 5000)^2 \times 10 = (0.64)^2 \times 10 = (0.41) \times 10 = 4.1$.

The student is required to recall whether flow should be set at 3200 vs 3000 gpm.

C. Incorrect. Pump should be started with the RHR Hx Bypass Valve (FIC-0619) closed. Plausible the student could choose automatic in order to control flow at the Tech Spec requirement. However, AOP-18019, Loss of RHR and SOP-13011, RHR both specify pump start with the Hx outlet and Bypass valves closed.

The potentiometer setting is calculated by $(\text{desired flow} / 5000)^2 \times 10$. The # 3.6 is plausible due to $(3000 / 5000)^2 \times 10 = (0.6)^2 \times 10 = (0.36) \times 10 = 3.6$.

However, to ensure 3000 gpm flow under all conditions, Tech Spec Rounds and procedures call for 3200 gpm flow. the setting for this is 4.1.

The potentiometer setting is calculated by (desired flow / 5000)^2 X 10. The # 4.1 is correct due to (3200 / 5000)^2 X 10 = (0.64)^2 X 10 = (0.41) X 10 = 4.1.

The student is required to recall whether flow should be set at 3200 vs 3000 gpm.

D. Correct. Bypass valve should be in manual and closed. Pot setting for Tech Specs is set at 4.1.

REFERENCES

SOP-13011, Residual Heat Removal System section 4.4 for Placing Trn-B RHR in Service For RCS Cooldown From Standby Readiness.

AOP-18019, Loss of RHR, section B for a loss of RHR capability or imminent loss of RHR due to RCS leakage while in Mode 5 or 6 with RCS level below the PRZR indication range or with SG nozzle dams installed. Steps B16 through B19.

OSP-14000, Operations Shift and Daily Surveillance Logs, Data Sheet 3 - Modes 5 & 6 sheet 2 of 6 (page 23).

VEGP learning objectives:

LO-PP-12101-09, Describe how the RHR heat exchanger bypass valve automatically controls flow to the RCS.

LO-PP-12101-16, Briefly describe the operator actions required for placing RHR in service for shutdown cooling.

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Use either of the following: __ ATTACHMENT C Section A - VALVE LINEUP FOR RHR PUMP COLD LEG INJECTION. -OR- __ ATTACHMENT C Section B - VALVE LINEUP FOR RHR PUMP HOT LEG INJECTION.	
__c. Check RV Head - REMOVED.	__c. Go to Step B16.
__d. Use the Refueling Water Purification Pump per ATTACHMENT B.	
__B16. Identify and isolate any RCS leakage.	

> **NOTES**
>
> - The time to boiling in the RCS should be taken into consideration when determining how much time should be spent venting the RHR system prior to taking additional actions for alternate cooling sources.
> - If adequate time to completely vent the RHR system is not available, air can be swept out of the RHR lines by filling the RCS to 188 feet 3 inches and running an RHR Pump at a flowrate greater than 3000 gpm (3200 gpm indicated.)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B17. Vent any RHR Pump that experienced cavitation:	
__a. Maintain RCS level while venting RHR system.	

Step 17 continued on next page

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

b. Vent RHR Pump(s) at high point vent until water is discharged:

UNIT 1:

__ 1-HV-10465 RHR SUCT VENT LINE TRN A (AB-B08)

__ 1-HV-10466 RHR SUCT VENT LINE TRN B (FHB-B13)

UNIT 2:

__ 2-HV-10465 RHR SUCT VENT LINE TRN A (AB-B131)

__ 2-HV-10466 RHR SUCT VENT LINE TRN B (FHB-B03)

c. Vent RHR Pump(s) using casing vents until water is discharged:

UNIT 1:

__ Train A - 1-1205-U4-235 (AB-D48)

__ Train B - 1-1205-U4-236 (AB-D49)

UNIT 2:

__ Train A - 2-1205-U4-235 (AB-D22)

__ Train B - 2-1205-U4-236 (AB-D21)

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B18. Establish conditions to start RHR Pump:	
__a. RCS level - GREATER THAN 188 FEET.	__a. Consult TSC if applicable and return to Step B6.
__b. RHR Pump - AVAILABLE.	__b. Consult TSC if applicable and return to Step B6.
c. Check RHR system valves - IN PROPER ALIGNMENT: TRAIN A __• HV-8701A - OPEN __• HV-8701B - OPEN __• HV-0606 - CLOSED __• FV-0618 - CLOSED __• HV-0128 - CLOSED __• HV-10465 - CLOSED __• FV-0610 - OPEN __• HV-8809A - OPEN __• HV-8716A - CLOSED TRAIN B __• HV-8702A - OPEN __• HV-8702B - OPEN __• HV-0607 - CLOSED __• FV-0619 - CLOSED __• HV-0128 - CLOSED __• HV-10466 - CLOSED __• FV-0611 - OPEN __• HV-8809B - OPEN __• HV-8716B - CLOSED	__c. Align valves as required.
__d. Check CCW cooling to RHR system - IN SERVICE.	__d. Restore CCW cooling by initiating 18020-C, LOSS OF COMPONENT COOLING WATER.

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Starting an RHR Pump may result in an RCS level reduction due to shrink or void collapse.

B19. Restore RHR flow:

__a. Start one RHR pump.

__b. Control charging flow to maintain RCS level above 188 feet.

__c. Slowly raise RHR bypass flow to 3000 gpm.

__d. Check RHR Pump - NOT CAVITATING.

d. Perform the following:

__1) Reduce flow to stop cavitation.

__2) IF flow must be reduced to less than 1500 gpm to stop cavitation, THEN stop RHR Pump and return to Step B6.

__e. Check RHR flow - RESTORED.

__e. Consult TSC if applicable and return to Step B6.

__f. Establish desired RCS cooldown rate.

INITIALS

4.4 PLACING TRN-B RHR IN SERVICE FOR RCS COOLDOWN FROM STANDBY READINESS

4.4.1 **Notify** HP that this RHR system change could affect area radiation levels so that surveys can be taken and personnel made aware of the changed condition. ________

4.4.2 **Restore** power to RHR PMP-B SUCTION FROM HOT LEG LOOP 4 Inlet Isolations and air to RHR System Flow Control Valves as follows: (IV REQUIRED)

a. IF shutdown, **place** Inverter 1DD1I6 in service per 13405-1, "125V DC 1E Electrical Distribution System." ________

b. **Install** the annunciator card associated with ALB34 E07 and **check** ALB34 E07 illuminates. ________

c. At 1DD1I6N **unlock** and **close** the disconnect for 1-HV-8702A. ________

d. **Check** ALB34-E07 extinguishes. ________

e. **Close** the K2 link for breaker 1BBE-13. ________

f. **Unlock** and **close** RHR PMP B SUCTION FROM HOT LEG LOOP 4, 1-HV-8702B Supply Breakers 1BBE-13. ________

g. **Close** INST AIR LINE 136 DRAIN 1-2420-U4-152 (RC-89). ________

h. **Restore** air to RHR System Flow Control Valves by opening INSTR AIR ISOLATION TO LINE 136 1-2420-U4-151 (RC-85 overhead). ________

INITIALS

NOTES

- When in Mode 1, 2, or 3, 1-HV-8809A/B should not be shut simultaneously.
- One train of RHR at a time should be aligned for shutdown cooling.
- An operator should be stationed at RHR B high point vent 1-HV-10466 to monitor for water flow to the floor drain when Step 4.4.3 h. is performed.

4.4.3 **Align** the RHR for shutdown cooling as follows:

a. **Close** the RHR TRN-B TO HOT LEG CROSSOVER ISO 1-HV-8716B. ______

b. **Close** RHR TRN-B HEAT EXCH OUTLET 1-HV-0607 and **check** closure at Group 1 MLB 02 2.2 or by computer point UD8703. ______

c. **Close** RHR TRN-B HEAT EXCH BYPASS 1-FV-0619 and **check** closure by computer point UD8698. ______

d. **Verify** open RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B. (IV REQUIRED) ______

e. **Place** RHR PMP-B 1-HS-0621 in PULL-TO-LOCK. ______

f. **Close** the RWST TO RHR PMP-B SUCTION 1-HV-8812B. ______

g. **Open** the RHR PMP-B SUCTION FROM HOT LEG LOOP 4 1-HV-8702A. ______

h. **Open** RHR SUCTION VENT LINE TRN-B 1-HV-10466. ______

i. <u>WHEN</u> operator at 1-HV-10466 reports water flowing to the floor drain, **close** 1-HV-10466. ______

j. **Open** the RHR PMP-B SUCTION FROM HOT LEG LOOP 4 Valve 1-HV-8702B. ______

k. **Place** RHR PMP-B 1-HS-0621 in AUTO position. ______

INITIALS

4.4.4 **Remove** power from the RHR to SI Pump Isolation Valve as follows:

a. **Check** 1-HV-8804B CLOSED. ________

b. **Open** breaker 1BBB-05 to valve 1-HV-8804B. ________

c. **Open** the K2 links for breaker 1BBB-05 and **tag** per NMP-AD-003, "Equipment Clearance And Tagging."

4.4.5 **Verify** train related CCW System is in service per 13715-1, "Component Cooling Water System." ________

4.4.6 **Start up** one train of RHR as follows:

a. **Verify** open the RHR PMP-B MINIFLOW ISO 1-FV-0611. ________

> **CAUTION**
>
> In order to prevent excessive RHR heat up and possible pump damage, RHR HEAT EXCH OUTLET for Train B 1-HV-0607 and RHR HEAT EXCH BYPASS for Train B 1-FV-0619 must be closed. Actual valve position should be monitored at Group 1 MLB 02 2.2 or by computer point prior to pump start.

b. **Start** RHR PMP-B. ________

c. **Establish** RHR Letdown per Section 4.5. ________

4.4.7 **Warm up** the TRN-B RHR as follows:

> NOTE
>
> Due to leak-by of the RHR Hx Outlet and Bypass Valves, RHR warming will begin as soon as the pump is started. However, due to miniflow cooling back to the suction of the pump, the temperature rise at the Hx inlet is only expected to reach approximately 200°F with the RCS at approximately 350°F. A rapid temperature rise should be expected when the miniflow valve goes closed.

a. **Monitor** RHR TRN-B Heat Exchanger Inlet Temperature using Plant Computer T0631, until the temperature stabilizes. ________

INITIALS

CAUTION

If the RCS is under vacuum, a minimum flow rate of about 1200 gpm for 3 minutes is needed to refill the voided section of RHR discharge piping. 1500 gpm should NOT be exceeded during the refill period. Flow rates are to be adjusted very SLOWLY any time flow is being increased due to possible water hammer concerns.

b. **Throttle** open the RHR TRN-B HEAT EXCH BYPASS 1-FV-0619 until RHR PMP-B MINIFLOW ISO VLV 1-FV-0611 closes. ________

c. **Complete** RHR warm-up by monitoring RHR Hx Train B Inlet Temperature using Plant Computer T0631, until the temperature stabilizes. ________

4.4.8 WHEN RHR warm-up is completed, **initiate** full flow to the RCS as follows:

NOTES

- >3200 gpm indicated flow ensures >3000 gpm actual flow for all temperatures.
- 3000 gpm RHR flow is required for Mode 6.

CAUTION

If the RCS is under vacuum, a minimum flow rate of about 1200 gpm for 3 minutes is needed to refill the voided section of RHR discharge piping. 1500 gpm should NOT be exceeded during the refill period. Flow rates are to be adjusted very slowly any time flow is being increased due to possible water hammer concerns.

a. **Throttle** open the RHR HEAT EXCH BYPASS for Train B 1-FV-0619 to the desired flow rate (nominally 3000 gpm). ________

b. **Verify** the RHR PMP-B MINIFLOW ISO VLV 1-FV-0611 closes. ________

INITIALS

CAUTION

The RHR Heat Exchanger Train B Bypass Flow Controller Potentiometer should be set for a minimum flow of 3000 gpm (Pot setting: 3.6 for 3000 gpm, 4.1 for 3200 gpm) prior to placing controller in AUTO. The potentiometer setting for the desired flow rate (gpm) is approximately equal to $(\text{Desired Flow}/5000)^2 \times 10$.

c. **Place** the RHR TRN-B HEAT EXCH BYPASS Flow Controller 1-FIC-0619A in AUTO, if desired. ________

NOTE

During Solid Plant conditions only 1-PIC-0131 should be used for letdown flow control and 1-HV-0128 should remain in the FULL OPEN position.

d. **Adjust** the LOW PRESSURE LETDOWN Controller 1-PIC-0131 and/or LETDOWN FROM RHR Control Valve 1-HV-0128 as required to maintain desired letdown flow. ________

e. **Slowly throttle** open RHR TRN-B HEAT EXCH OUTLET 1-HV-0607 to establish desired RCS cooling. ________

4.4.9 IF RCS cooling using both RHR trains is desired, **place** the second train in service:

IF RHR A is in STANDBY READINESS, use Section 4.3. ________

IF RHR A is NOT in STANDBY READINESS, use Section 5.3. ________

4.4.10 **Establish** RCS Cool down per 12006-C, "Unit Cool down To Cold Shutdown." ________

DATA SHEET 3 - MODE 5 & 6 **MODE** ______________ **Sheet 2 of 6**

DATE ______________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER		INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
AT LEAST 1 RHR TRAIN SHALL BE IN OPERATION AND THE REQUIRED RHR TRAINS OR THE REQUIRED SGS OPERABLE VERIFY RHR CIRCULATION AND OR SG LEVELS	3.4.7.1 3.4.7.2 (MODE 5, LOOPS FILLED) 3.4.8.1 (MODE 5 LOOPS NOT FILLED)	RHR TRAINS (A, B)		ENTER TRAINS OPERABLE			***	3.4.7
		RHR FLOW (GPM)		1FIC-0618A			* ≥3200	
		RHR FLOW (GPM)		1FIC-0619A			* ≥3200	
		STEAM GENERATOR LEVEL (%)	1	1LI-0501				3.4.8
			2	1LI-0502				
			3	1LI-0503				
			4	1LI-0504				
	SR 3.9.5.1 (MODE 6 ≥23' ABOVE FLANGE) SR 3.9.6.1 (MODE 6 <23' ABOVE FLANGE)	*** MODE 5, LOOPS FILLED - AT LEAST 1 RHR TRAIN IN OPERATION AND ONE ADDITIONAL RHR TRAIN SHALL BE OPERABLE OR THE SECONDARY SIDE water level OF AT LEAST TWO STEAM GENERATORS ≥63% WIDE RANGE. Steam Generators may not be used as an option to an RHR train unless the RCS is filled greater than 15% Pressurizer level and RCS pressure has been maintained >100 psig since the most recent fill & vent. MODE 5, LOOPS NOT FILLED - AT LEAST 2 RHR TRAINS OPERABLE WITH 1 TRAIN IN OPERATION. MODE 6, >23 FT ABOVE FLANGE - AT LEAST 1 RHR TRAIN OPERATING WITH >3000 GPM# FLOW. MODE 6, <23 FT ABOVE FLANGE - AT LEAST 2 RHR TRAINS OPERABLE WITH 1 TRAIN IN OPERATION WITH >3000 GPM# FLOW. * ≥3200 GPM ENSURES ≥3000 GPM ACTUAL FLOW AT ALL TEMPERATURES RHR FLOW OF ≥3000 GPM IS ONLY REQUIRED IN MODE 6 AND IS N/A IN MODE 5						3.9.5 3.9.6

COMPLETED BY: **DAY:** ______________ **TIME:** ________ **NIGHT:** ______________ **TIME:** ________

SS REVIEW: **DAY:** ______________ **TIME:** ________ **NIGHT:** ______________ **TIME:** ________

27. 026A3.02 002/2/1/C.SPRAY-HS COOLING/MEM - 3.9 / 4.2/NEW/HL-15R NRC/RO/DS/TNT

Which one of the choices correctly lists **ALL** the locations where the control room crew can monitor and control the CNMT Coolers following an RCS LOCA?

MLBs - Monitor Light Boxes on the vertical section of the main control board

QMCB (NSCW) - Sloping portion of the NSCW section of the main control board

QHVC - Main Control Room HVAC panel

		Indications	Controls
A.	Fan speed	MLBs, QHVC	QHVC
	Cooling water valves	QMCB (NSCW)	QMCB (NSCW)
B.	Fan speed	MLBs, QMCB (NSCW)	QMCB (NSCW)
	Cooling water valves	MLBs	QMCB (NSCW)
C.	Fan speed	QHVC	QHVC
	Cooling water valves	QMCB (NSCW)	QMCB (NSCW)
D✓	Fan speed	MLBs, QHVC	QHVC
	Cooling water valves	MLBs, QMCB (NSCW)	QMCB (NSCW)

K/A

026 Containment Spray System (CSS)

A3.02 Ability to monitor automatic operation of the CSS, including;

Verification that cooling water is supplied to the containment spray heat exchanger.

K/A MATCH ANALYSIS

The question requires the student to properly identify what control room cooling water indications are available to monitor the performance of the CNMT coolers.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. The fan speed indications and controls listed are correct. The NSCW control and indication locations listed are correct making this choice very plausible. However, the list for NSCW is incomplete, making this choice incorrect.

B. Incorrect. This choice is plausible but incorrect due to the fan speed control being listed as from the NSCW section. The fans controlled from this section of the main control boards are the NSCW cooling tower fans.

C. Incorrect. This choice is plausble soince the locations listed are all correct. However, the list is incomplete since the MLBs has not been included in the list.

D. Correct. Fan speeds can be monitored from the MLBs on the main control board. Fan sppeds can be monitored and controlled on the HVAC panel. The HVAC panel has the high and low speed handswitches as well as the breaker position inidcating lights. The NSCW CNMT coolers cooling water isolation valves are indicated on the MLBs. The NSCW CNMT coolers cooling water isolation valves are controlled from the sloping section of the NSCW system on the main control board with motor operator hand switches and position indicating lights.

REFERENCES

VEGP P&ID 1X4DB135-1, NSCW System

VEGP P&ID 1X4DB212, CNMT Heat Removal

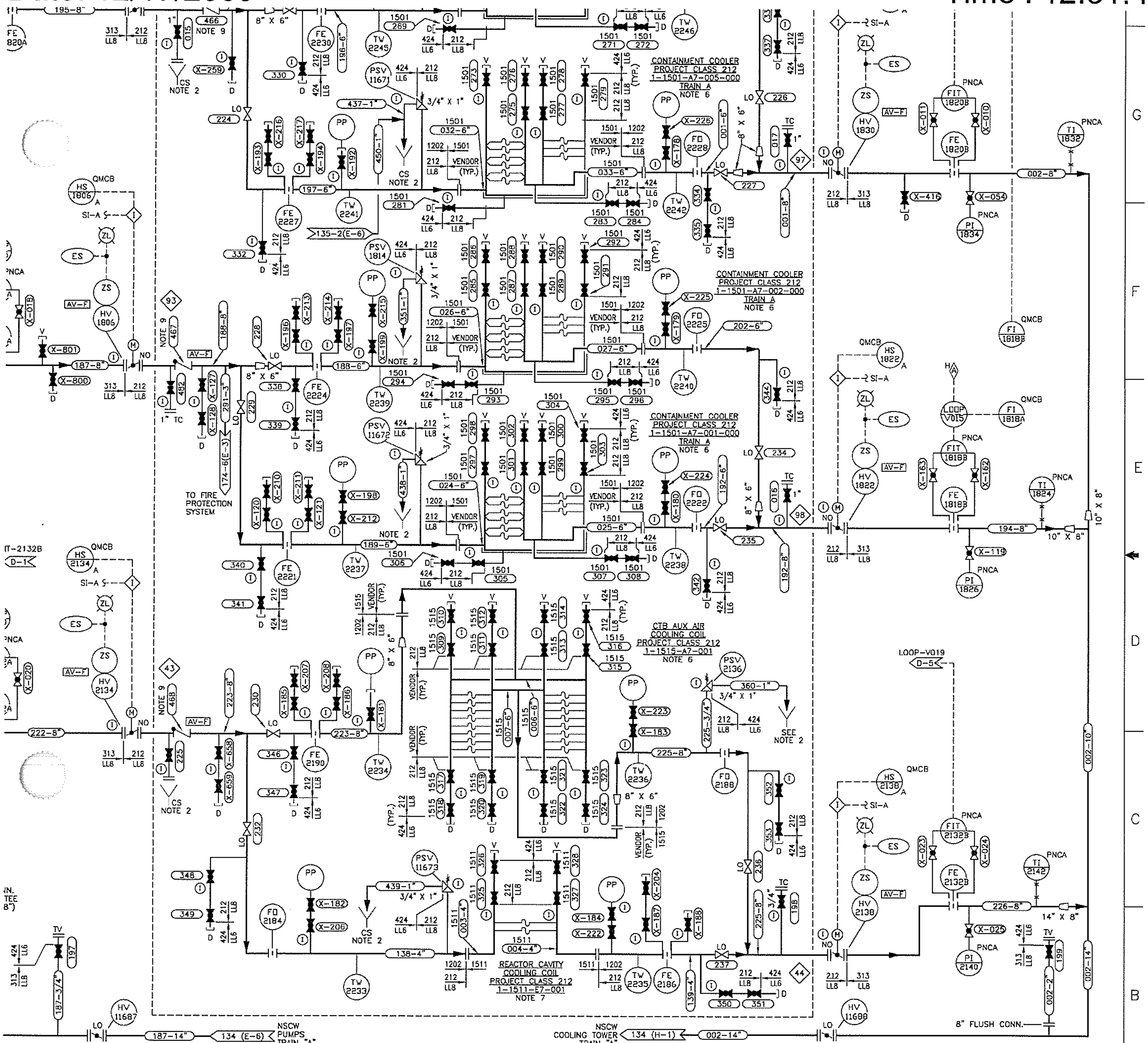
V-LO-TX-06101, NSCW System Text page 13

V-LO-TX-29101, CNMT HVAC systems text, page 15

VEGP learning objectives:

LO-PP-29101-10:

State the cooling water systems that support Containment HVAC systems.



HEADER INFORMATION SEE DWG'S 1X4DB146-1 & 1X4DB147-2.

HEADER INFORMATION SEE DWG 1X4DB143, 1X4DB144-1 &
2.

MPONENT COOLING WATER SYSTEM AND SHELL SIDE CONN'S
X4DB138-1.

NENT COOLING WATER SYSTEMS AND SHELL SIDE CONN'S SEE
136.

GEN. BLDG SUMP INFORMATION SEE DWG. 1X4DB146-2.

6. FOR CONTAINMENT COOLERS AND CTB AUX AIR COOLER AIR SIDE DRAINS SEE DWG. 1X4DB212.
7. FOR REACTOR CAVITY COOLER AIR SIDE DRAINS SEE DWG 1X4DB214-1.
8. CHECK VALVE TO BE LOCATED CLOSE TO TEE.
9. CHECK VALVE TO BE CLOSE TO PENETRATION.
10. LINE SPACER TO BE INSTALLED.
11. FOR NORMAL OPERATION, VALVE TO BE LOCKED OPEN AT SETTING OF 42.5 DEGREES FROM FULL CLOSED POSITION.

CAD 1X4DB135-1
ACAD2002 NJ - 01

NO.		DATE	DR	CHK	APPV	DTL
29.0	REDRAWN PER ABN-52215, VER. 1.0	9-7-05	NJ	LCF	LKB	X

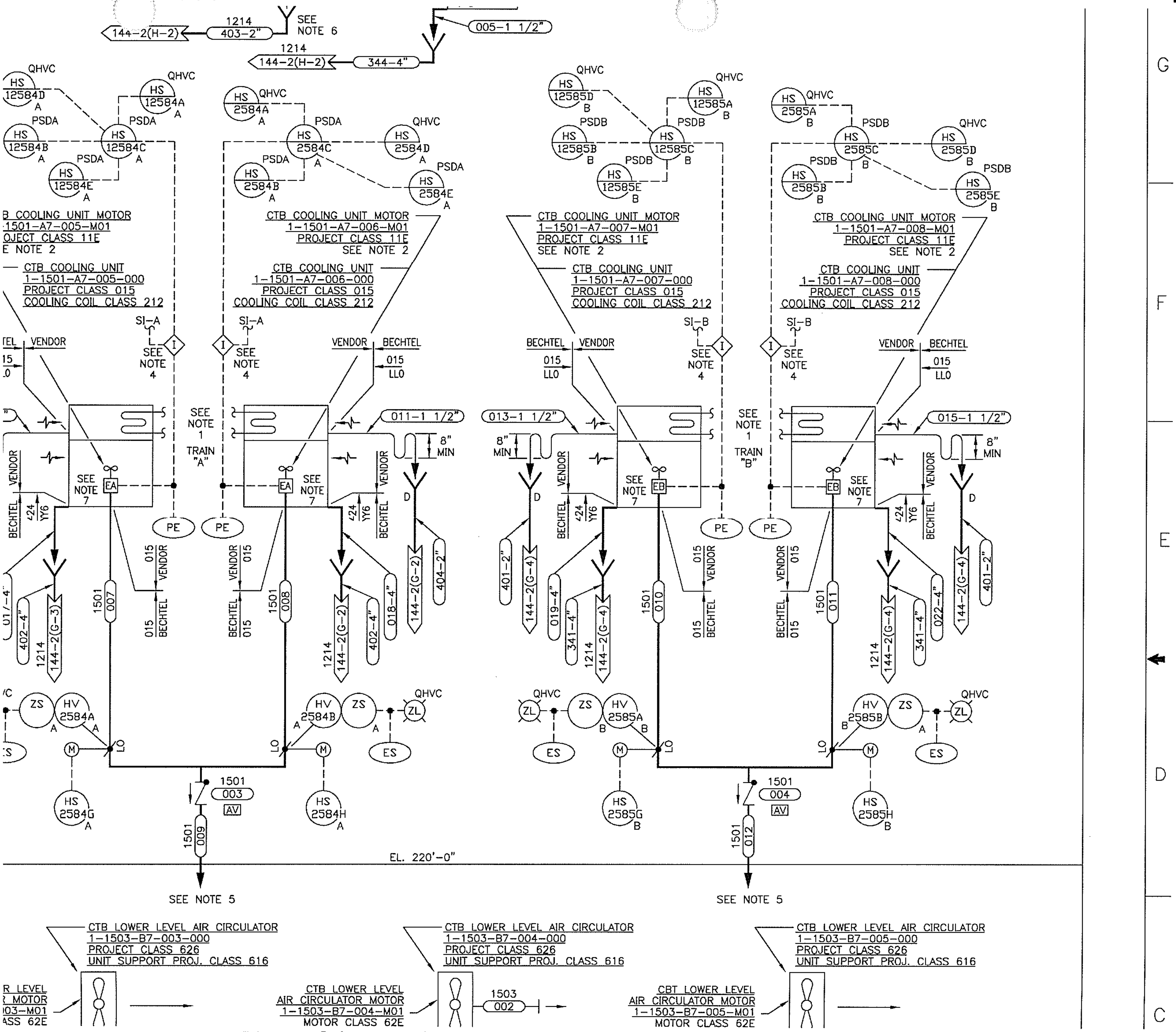
SOUTHERN COMPANY
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
NUCLEAR SERVICE
COOLING WATER SYSTEM
SYSTEM NO. 1202

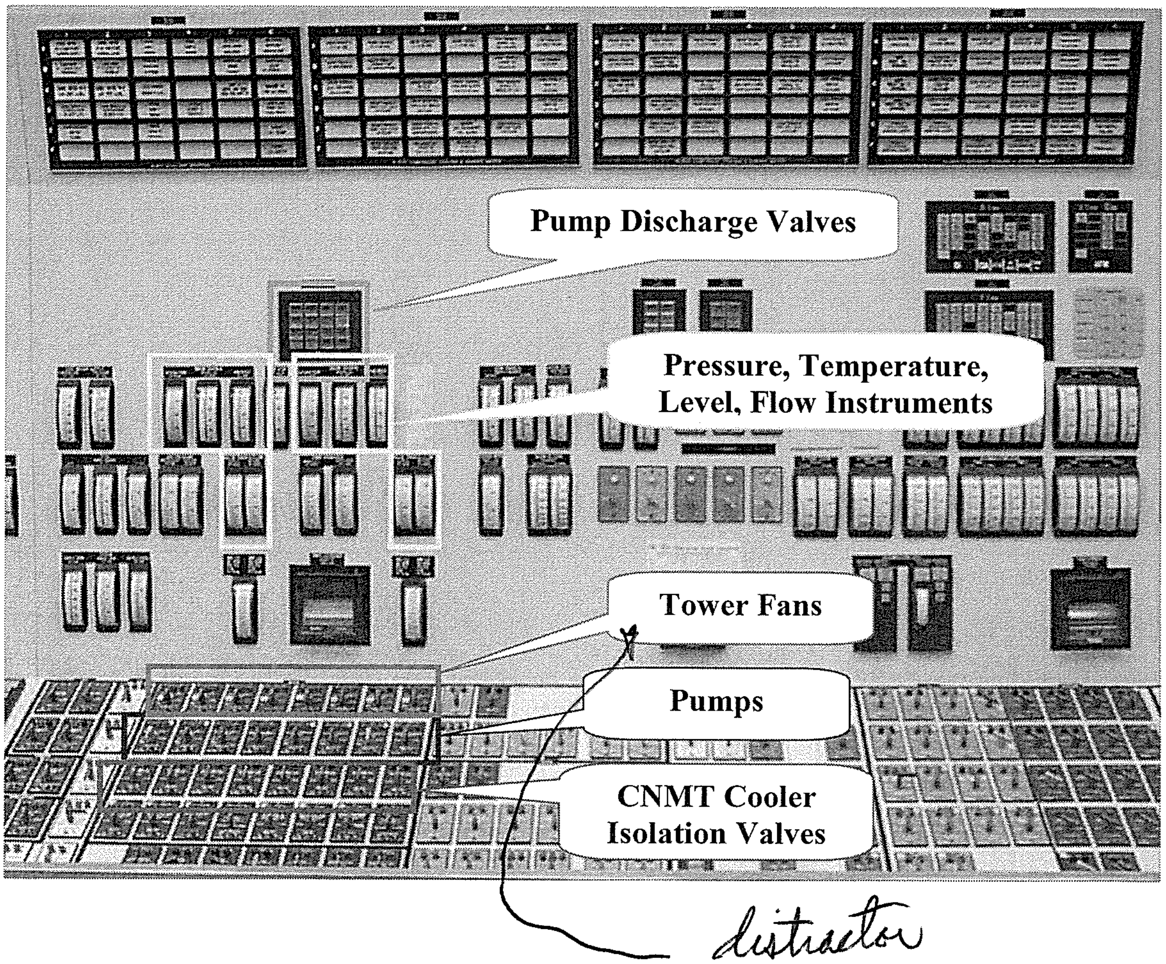
SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB135-1	29.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL


SEE NOTE 6
005-1 1/2"
344-4"
403-2"
144-2(H-2)
1214
CTB COOLING UNIT MOTOR
1-1501-A7-005-M01
PROJECT CLASS 11E
SEE NOTE 2
CTB COOLING UNIT
1-1501-A7-005-000
PROJECT CLASS 015
COOLING COIL CLASS 212
CTB COOLING UNIT MOTOR
1-1501-A7-006-M01
PROJECT CLASS 11E
SEE NOTE 2
CTB COOLING UNIT
1-1501-A7-006-000
PROJECT CLASS 015
COOLING COIL CLASS 212
CTB COOLING UNIT MOTOR
1-1501-A7-007-M01
PROJECT CLASS 11E
SEE NOTE 2
CTB COOLING UNIT
1-1501-A7-007-000
PROJECT CLASS 015
COOLING COIL CLASS 212
CTB COOLING UNIT MOTOR
1-1501-A7-008-M01
PROJECT CLASS 11E
SEE NOTE 2
CTB COOLING UNIT
1-1501-A7-008-000
PROJECT CLASS 015
COOLING COIL CLASS 212
SEE NOTE 1
TRAIN "A"
TRAIN "B"
SEE NOTE 4
SEE NOTE 7
011-1 1/2"
013-1 1/2"
015-1 1/2"
8" MIN
EL. 220'-0"
SEE NOTE 5
CTB LOWER LEVEL AIR CIRCULATOR
1-1503-B7-003-000
PROJECT CLASS 626
UNIT SUPPORT PROJ. CLASS 616
CTB LOWER LEVEL AIR CIRCULATOR
1-1503-B7-004-000
PROJECT CLASS 626
UNIT SUPPORT PROJ. CLASS 616
CTB LOWER LEVEL AIR CIRCULATOR MOTOR
1-1503-B7-004-M01
MOTOR CLASS 62E
CTB LOWER LEVEL AIR CIRCULATOR
1-1503-B7-005-000
PROJECT CLASS 626
UNIT SUPPORT PROJ. CLASS 616
CBT LOWER LEVEL AIR CIRCULATOR MOTOR
1-1503-B7-005-M01
MOTOR CLASS 62E
G
F
E
D
C

which carry 5 gpm of NSCW from the discharge of the respective pump and return the water directly to the NSCW basin.



6.3.2 NSCW Cooling Towers

The nuclear service water mechanical draft cooling towers are the ultimate heat sinks for the plant. Each NSCW tower (train) is designed to accommodate the heat load from the unit under both normal and emergency conditions. Normally, both towers will be operating. Although designed for one in operation and one in standby, both in operation ensures the reliability and readiness of all components served by NSCW, if required.

The mechanical draft cooling tower uses four fans to induce the movement of air up through the cooling tower. The nuclear service cooling water enters near the top of the cooling tower. A distribution pipe disperses the return water over the entire area of the tower. As the water is released from the distribution pipe, it falls through the fill (material used to break the water into fine droplets). Curved asbestos cement board hung vertically is used for the fill material. Air is drawn through the windows surrounding the

The following systems use NSCW supplied water in cooling coils to remove heat in containment:

a. Reactor Cavity Cooling Fans
b. Containment Fan Coolers
c. Containment Auxiliary Coolers

4. Normal Chilled Water

During refueling outages normal chilled water is supplied to the train B supplied Containment Auxiliary Cooler and Reactor Cavity Cooling Fan

5. 480 VAC 1E and Non 1E power

See the individual component descriptions for power supplies.

29.3 INSTRUMENTATION AND CONTROL

Instrumentation and control, indications, alarms and interlocks will be discussed in this section system by system. All hand switch controls stop-auto-start, spring return to auto for fans or close auto-open, spring return to auto for dampers unless otherwise noted.

Containment Coolers System 1501

Controls and Instrumentation

The Containment Coolers have hand switch controls on the QHVC panel and their respective Remote shutdown panels. There are separate hand switch controls for the low and high speed fans in both locations. On the QMCB, there are supply and return flow indicators for each pair of coolers. There are also hand switch controls for the NSCW supply and return MOVs on the QMCB. There are status lights on the monitor light boxes (MLBs) for the Containment Cooler low speed operation, and for the NSCW MOVs.
There are cooler low flow alarms for each pair of coolers on ALB02 and ALB03.

Control Functions and Interlocks

The fans may be manually started from the control room in either high or low speed. The high and low speed controls are interlocked so only one speed may be energized. The fans must be run in pairs in specific combinations as outlined in the procedure to allow for even cooling and prevent backflow through idle fans.

On a loss of offsite power, the fans in auto will be started in high speed. The Sequencer gives all fans a start signal at 30.5 secs, but delay timer delays the start of two fans by 20 seconds to prevent voltage swings on the 4160VAC 1E busses.

On an SI signal, SSPS and the sequencer will trip off any fans running in High speed and the SI sequence will restart the fans in low speed. All fans start at 30.5 secs.

28. 027AA1.05 001/1/1/PZR PRESS-HTR POWER/C/A - 3.3 / 3.2/M-ANO 2005/HL-15R NRC/RO/TNT/DS

Given the following:

- The plant is at full power when a loss of offsite power causes a plant trip.

- Both EDGs start and tie onto their respective ESF buses. All equipment sequences on as expected.

Which **ONE** of the following is **CORRECT** for PRZR heater banks available for RCS pressure control?

A. All Backup Heater Banks only.

B. Proportional Heater Bank only.

C.✓ Backup Heater Banks A and B only.

D. All Proportional and Backup Heater Banks.

K/A

027 Pressurizer Pressure Control System (PZR PCS) Malfunctions

AA1.05 Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions:

Transfer of heaters to backup power supply.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a Loss of Offsite Power occurs resulting in a plant trip with the DG's re-energizing the 1E emergency buses. The student must choose which PRZR heaters are still available for use (powered from the backup power via DG's). The other heaters powered from the non-1E buses will lose power on the reactor trip when fast bus transfer occurs due to the RATs being de-energized.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. All backup heaters are not available, only backup heaters A and B are powered from the stub buses which would re-energize on an LOSP.

B. Incorrect. The proportional heaters are not available, only backup heaters A and B are power from the stub buses which would re-energize on an LOSP.

C. Correct. Backup heaters A and B are powered from the stub buses which would re-energize on an LOSP.

D. Incorrect. All backup heaters are not available, only backup heaters A and B are powered from the stub buses which would re-energize on an LOSP. The proportional heaters are not available either as they are powered from non-1E buses.

REFERENCES

ANO 2005 RO NRC exam question # 57 (included).

HL-15 RO Audit exam (April 2009) question # 16 (included).

V-LO-PP-16303, Pressurizer Pressure Control, slide # 25 (included).

One line 1X3D-AA-E01A (NB01), Myriad drawings in refereneces.
One line 1X3D-AA-E10A (NB10), Myriad drawings in refereneces.
One line 1X3D-AA-E08A (NB08), Myriad drawings in refereneces.
One line 1X3D-AA-E09A (NB09), Myriad drawings in refereneces.
One line 1X3D-AA-F13A (PRZR heater panels), Myriad drawings.

VEGP learning objectives:

LO-LP-39208-01, For any item in section 3.4 of Tech Specs, be able to:

a. State the LCO

Questions For 2005 ANO UNIT 2 RO/SRO Exam

BANK 0498 **Rev** 0 **Rev Date:** 10/29/2004 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00
Lic Level: RS **Difficulty:** 2 **Taxonomy:** K **Source:** NEW **Originator** COBLE
10CFR55_41: 41.7 **10CFR55_43:** NA **Section:** 3.2 **Type** RCS INVENTORY
System PRESSURIZER LEVEL CONTROL **System** 011 **K/A:** K2.02
RO Tier: 2 **RO Group:** 2 **RO Imp:** 3.1 **SRO Tier:** 2 **SRO Group:** 2 **SRO Imp:** 3.2
Description Knowledge of bus power supplies to the following: PZR Heaters.

Question # 57
Given the following:

* The plant is at full power when a loss of offsite power causes a plant trip
* Electrical Bus 2A1 has a LOCKOUT alarm in.
* Both EDGs start and tie onto their respective ESF buses

Which of the following pressurizer heater banks would be available for RCS pressure control?

A. Both Proportional heater banks.

B. All Backup heater banks.

C. Both Backup heater banks #3 and #4.

D. All Proportional and Backup heater banks.

Answer:

A. Both Proportional heater banks

Notes:

Distracter B is incorrect because BU heaters banks are 480 non vital powered and there is no power to their bus.
Distracter C is incorrect because all BU heater banks are non vital powered.

References

STM 2-03, RCS, Section 2.2.2
A2LP-RO-RCS OBJ 10.c, Describe the following, concerning the RCS pressurizer: Heaters

Historical

This question has not been used on any previous NRC exams. BNC 10/29/2004.

Original question. From ANO-2005

1. 011K2.02 001

Given the following:

- The plant is at full power when a loss of offsite power causes a plant trip.

- Both EDGs start and tie onto their respective ESF buses. All equipment sequences on as expected.

Which **ONE** of the following is **CORRECT** for PRZR heater banks available for RCS pressure control?

A. All Backup Heater Banks only.

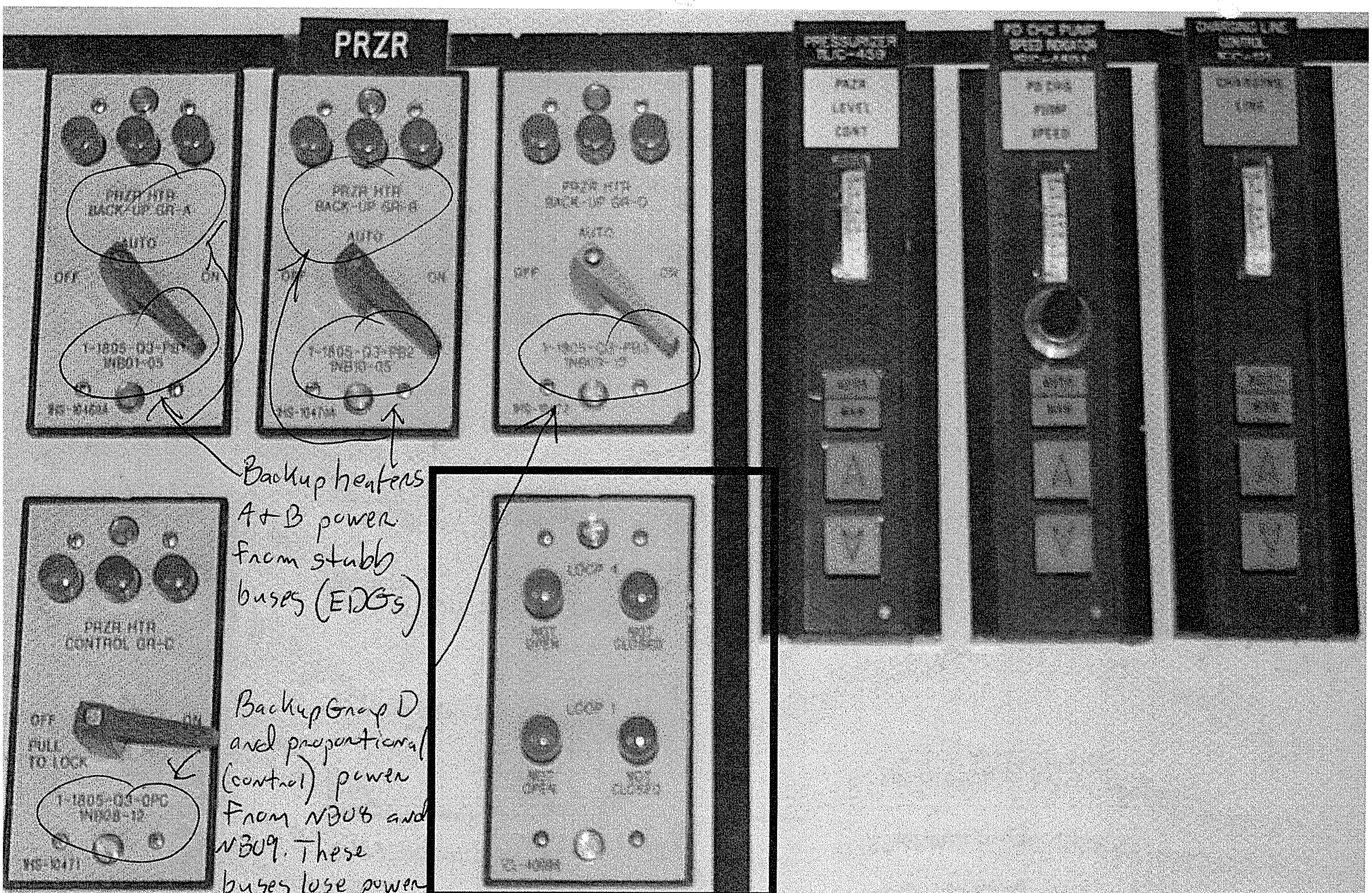
B. Proportional Heater Banks only.

C.✓ Backup Heater Banks A and B only.

D. All Proportional and Backup Heater Banks.

Bank question.

Question used on HL-15 audit in April 2009.

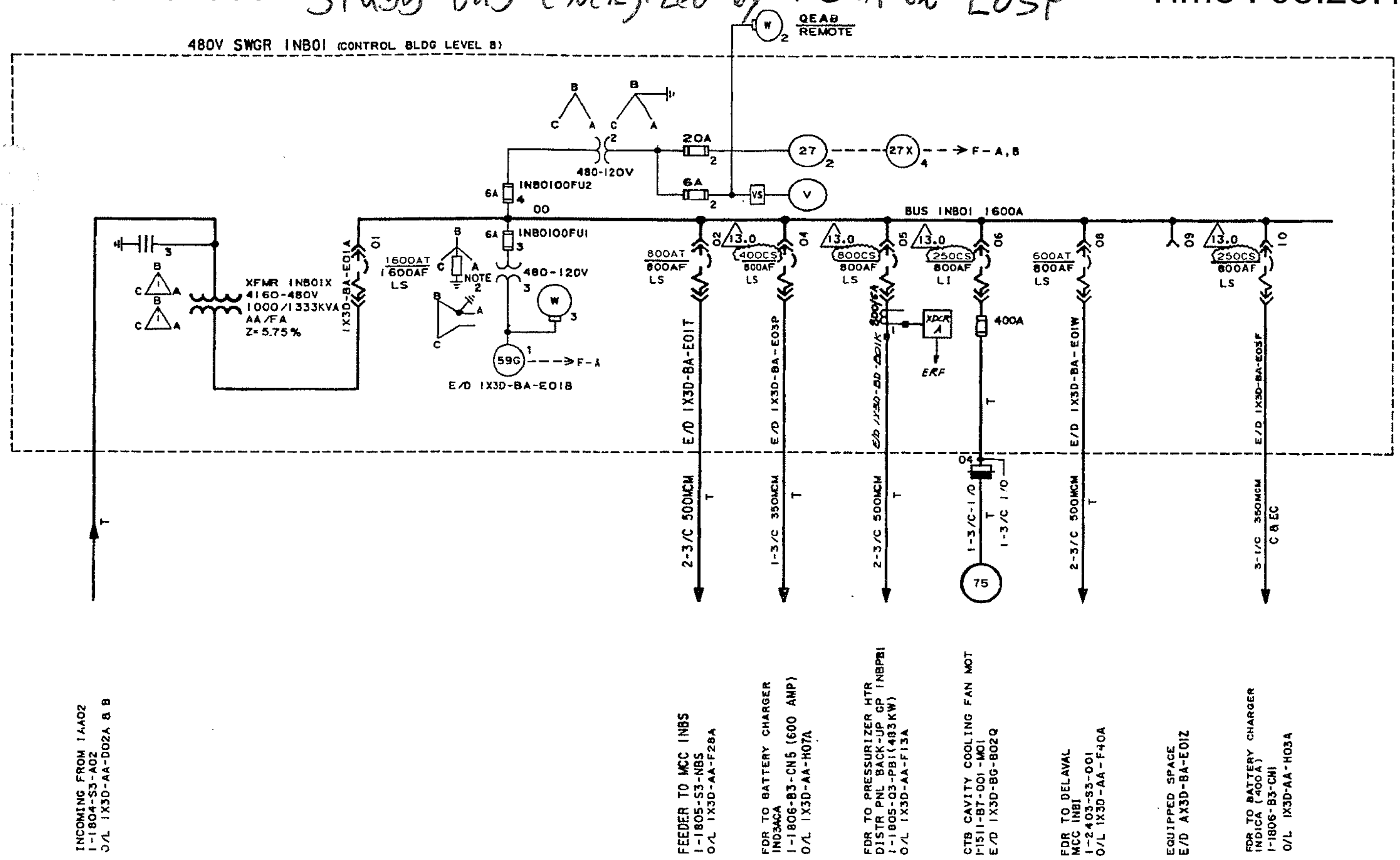


Backup heaters A + B power from stubb buses (EDGs)

Backup Group D and proportional (control) power from NB08 and NB09. These buses lose power on LOSP!

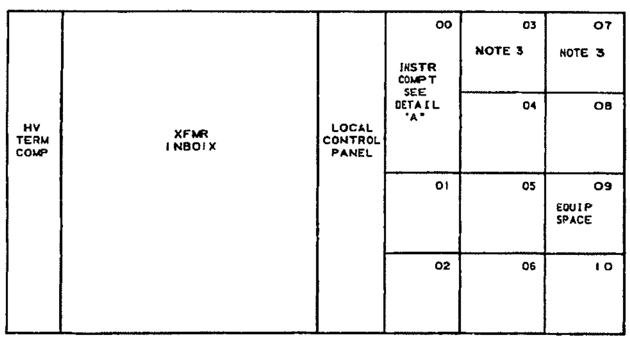
V-LO-PP-16303 slide # 25

Date: 12/17/2009 stubb bus energized by DG1A on LOSP Time : 03:25:17 PM

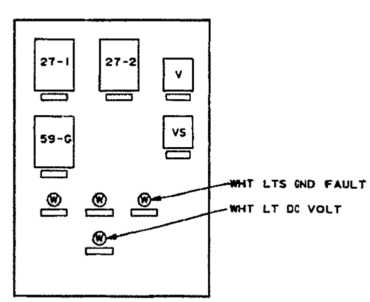


"A"

FUNCT NO
27-1
27-2
27X1-
27X2-
59-G
LI
LS
LI NOTE
LS NOTE



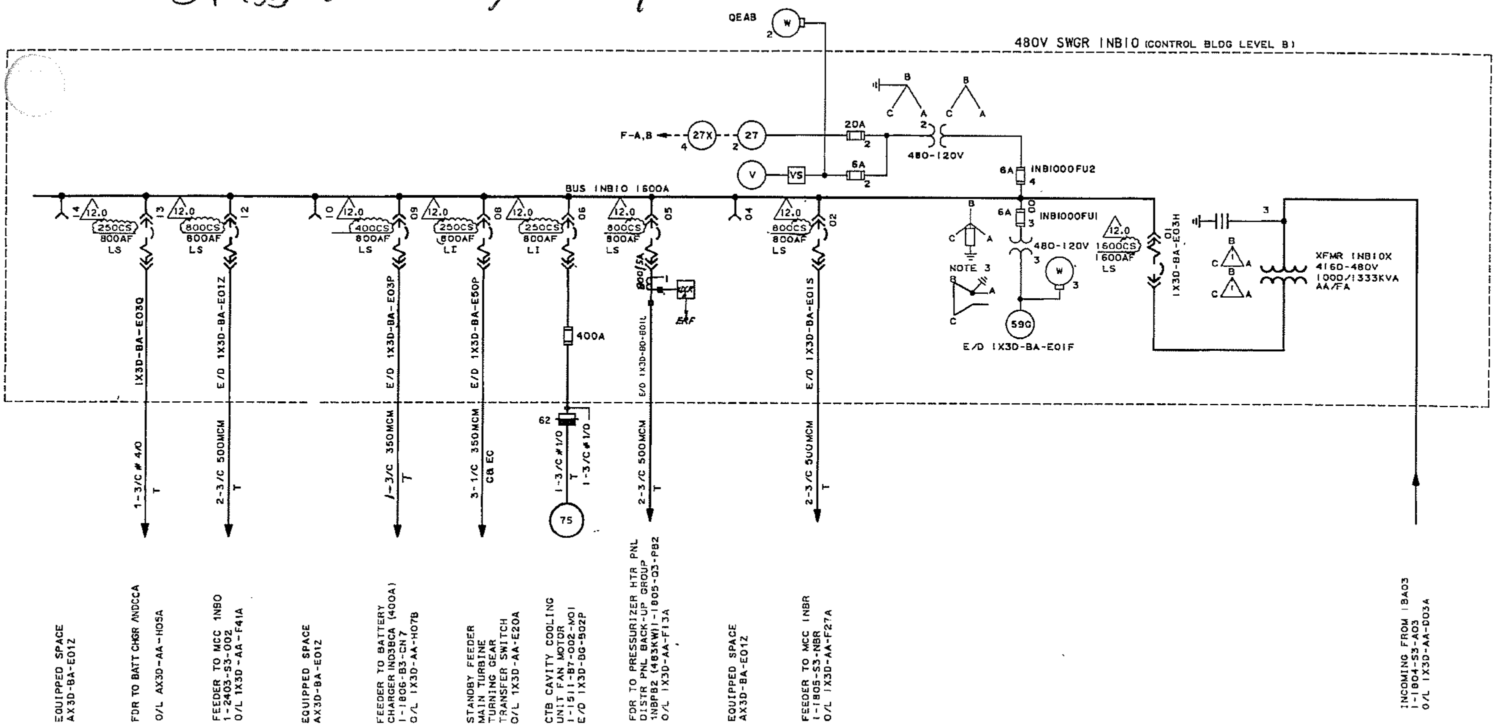
480V SWGR 1NB01
FRONT ELEVATION
(NOT TO SCALE)



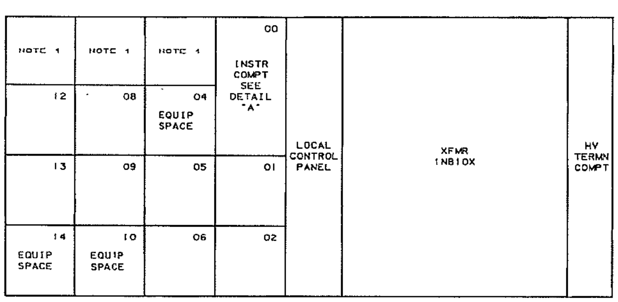
DETAIL "A"

Title: C:\DATA\HL-15 Recovery References\ONE LINES\1X3D-AA-E01A.cal

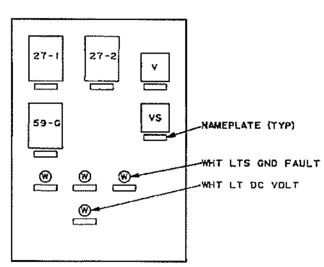
Stubb bus energized by DG-1B on LOSP



B



480V SWGR 1NB10
FRONT ELEVATION
(NOT TO SCALE)



DETAIL "A"

LEGEND

FUNCTION NO	MFR	TYPE	DESCRIPTIO
27-1 27-2	GE	IAV	UNDERVOLTAGE RELAY. I 55-140V OPERATING RA
27X1-1,-2 27X2-1,-2	GE	HFA	AUX TRIP RELAY. 125V
59G	GE	IAV	OVERVOLTAGE RELAY. I 16-64V OPERATING RAN
LI	ABB	PR112	MICROPROCESSOR TRIP LONGTIME & INSTANTAN
LS	ABB	PR112	MICROPROCESSOR TRIP LONGTIME & SHORT TIM
W			INDICATING LIGHT. WH

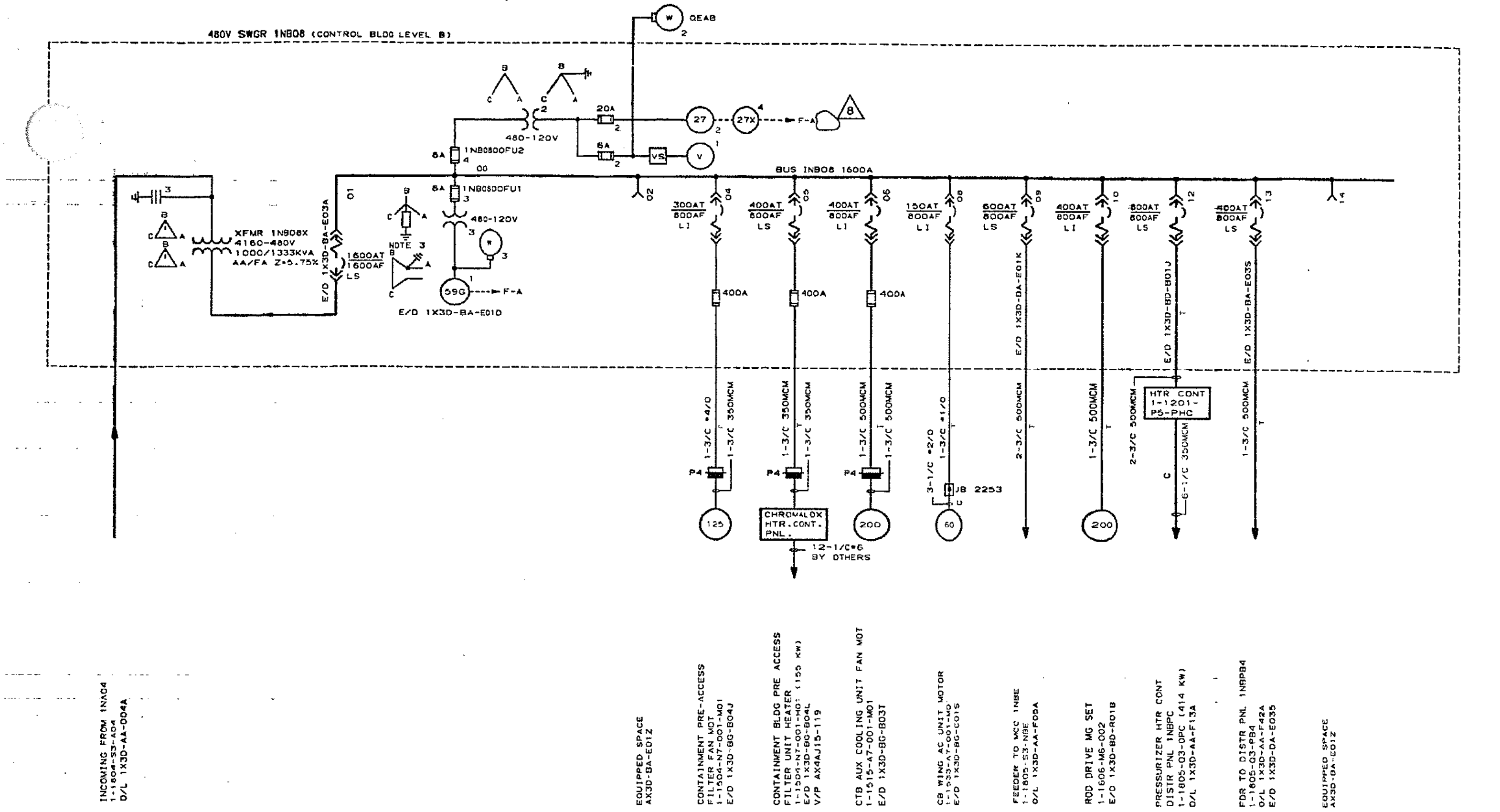
FUNCTIONAL TA

FUNCTION CODE-F	DESCRIPTION FUNCTION
A	ANNUNCIATOR
B	TRIP SELECTE

8 7 6 5 4

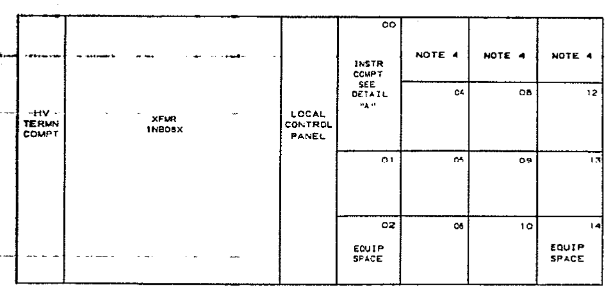
Date: 12/17/2009 Time : 03:26:05 PM

LOSP - bus is dead

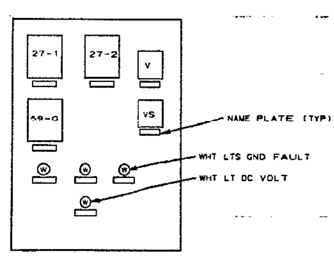


LEGEND

FUNCTION NO.	MFR	TYPE	DESCRIPTION
27-1 27-2	GE	IAV	UNDERVOLTAGE RELAY, 115V 55-140V OPERATING RANGE
27X1-1,-2 27X2-1,-2	GE	HFA	AUX TRIP RELAY, 125V DC
59G	GE	IAV	OVERVOLTAGE RELAY 199V AC 16-64V OPERATING RANGE
LI	GE	ECS	SOLID STATE TRIP DEVICE W LONGTIME & INSTANTANEOUS
LS	GE	ECS	SOLID STATE TRIP DEVICE W LONGTIME & SHORT TIME TRI
W			INDICATING LIGHT, WHITE L



480V SWGR 1NB08
FRONT ELEVATION
(NOT TO SCALE)



DETAIL "A"

FUNCTIONAL TABLE

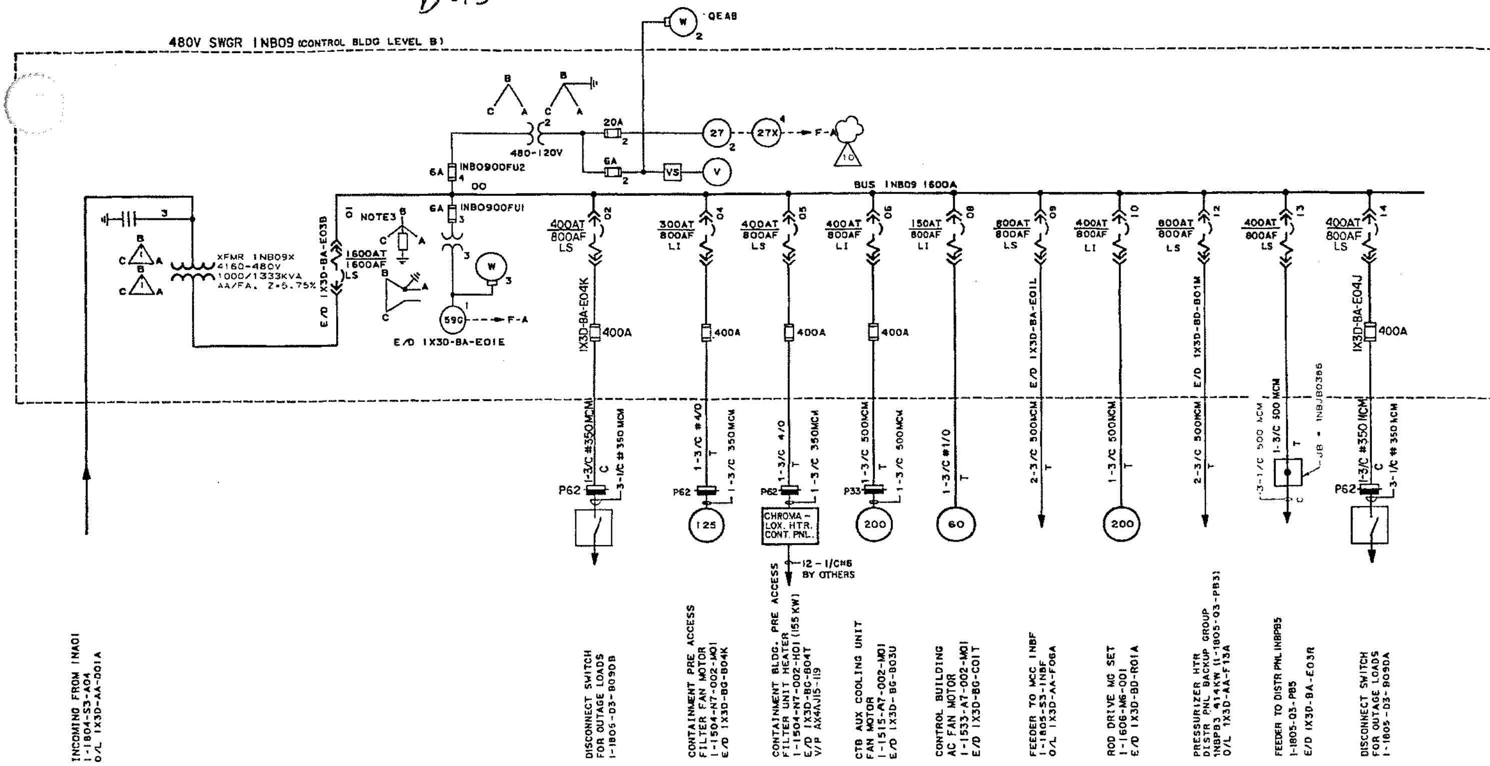
FUNCTION CODE-F	DESCRIPTION OF DEV
A	ANNUNCIATOR
B	TRIP SELECTED BREA

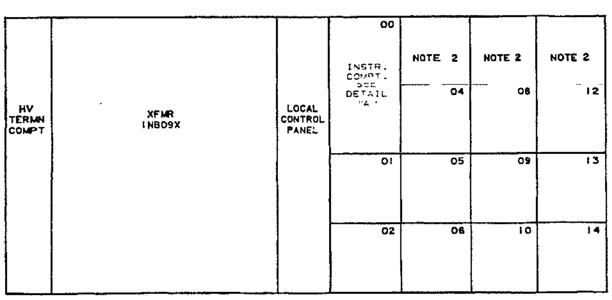
NO.	REVISIONS
	INCORP. ABN-12097
	ISSUED FOR CONSTRUCTION & VENDO

8 7 6 5 4

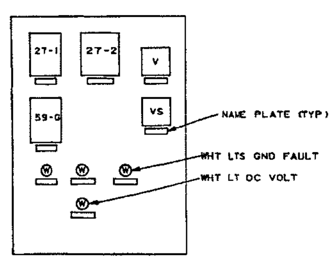
Title: C:\DATA\HL-15 Recovery References\ONE LINES\1X3D-AA-E08A.cal

bus is dead on LOSP





480V SWGR 1NB09
FRONT ELEVATION
(NOT TO SCALE)



DETAIL "A"

LEGEND			
FUNCTION NO.	MFR	TYPE	DESCRIPTION
27-1 27-2	GE	IAV	UNDERVOLTAGE RELAY 55-140V OPERATING
27X1-1,-2 27X2-1,-2	GE	HFA	AUX TRIP RELAY, 12
59G	GE	IAV	OVERVOLTAGE RELAY 16-64V OPERATING R
LI	GE	ECS	SOLID STATE TRIP D LONGTIME & INSTANT
LS	GE	ECS	SOLID STATE TRIP D LONGTIME & SHORT T
(W)			INDICATING LIGHT,

FUNCTION T	
FUNCTION CODE-F	DESCRIPTION
A	ANNUNCIATOR
B	TRIP SELECT

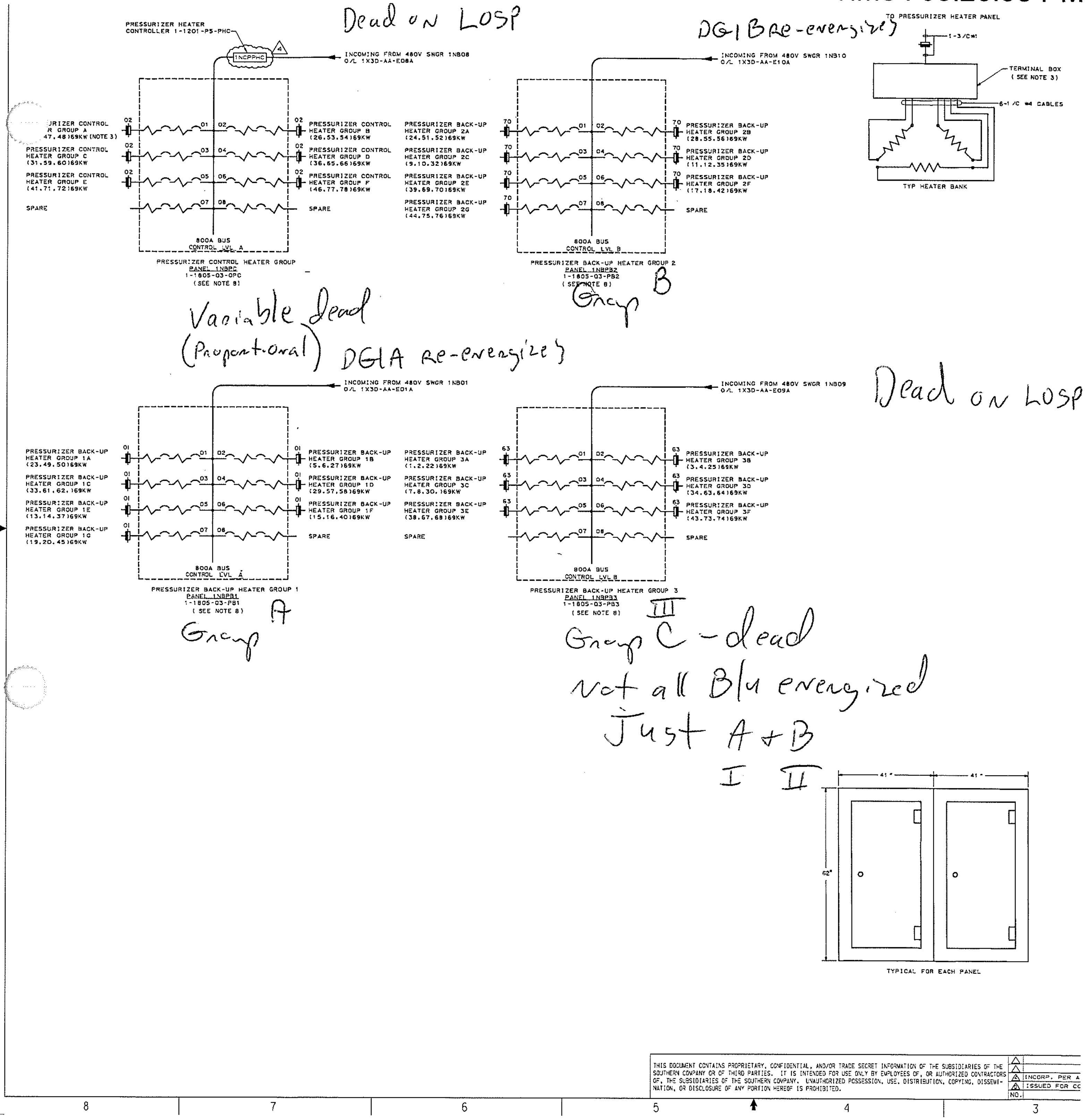
NO.		NO.		NO.	
					INCORP. ABN-... REVISIO
					ISSUED FOR CONSTRUCTIO REVISIO

8 7 6 5 4

Title: C:\DATA\HL-15 Recovery References\ONE LINES\1X3D-AA-E09A.cal

Date: 12/17/2009

Time : 03:26:53 PM



Dead on LOSP

DG1B Re-energizes

Variable dead (Proportional)

DG1A Re-energizes

Group B

Dead on LOSP

Group A

III

Group C - dead

Not all B/U energized

Just A + B

I II

Title: C:\DATA\HL-15 Recovery References\ONE LINES\1X3D-AA-F13A.cal

29. 028K5.01 001/2/2/H2 PURGE-EXPLSVE H2/MEM - 3.4 / 3.9/NEW/HL-15R NRC/RO/DS/TNT

A large RCS LOCA has occurred
The CNMT Hydrogen monitors indicate 5%
Service Air to CNMT is NOT available.

Which one of the following choices correctly desribes the operational implication of the hydrogen monitor readings and methods to reduce hydrogen concentration inside containment?

	Operational Implication	Method used to reduce hydrogen
A.	Combustible atmosphere	Post-LOCA hydrogen recombiners
B.✓	Combustible atmosphere	Post-LOCA hydrogen purge
C.	Embrittlement of cladding	Post-LOCA hydrogen recombiners
D.	Embrittlement of cladding	Post-LOCA hydrogen purge

K/A

028 Hydrogen Recombiner and Purge Control System (HRPS)

K5.01 Knowledge of the operational implications of the following concepts as they apply to the HRPS:

Explosive hydrogen concentration

K/A MATCH ANALYSIS

The question requires the student to correctly identify the implication of a high containment hydrogen concentration and what method is used to remove the hydrogen from containment.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Hydrogen becomes combustible above 4% in air. The Post-LOCA hydrogen recombiners have been retired in place.

B. Correct. Hydrogen becomes combustible above 4% in air. The Post-LOCA purge system is a method used per 13130 to reduce CNMT hydrogen concentration.

C. Incorrect. High hydrogen concentrations will cause embrittlement of steel and Zirconium. However, this is NOT the issue for high hydrogen in containment atmosphere. The Post-LOCA hydrogen recombiners have been retired in place.

CNMT hydrogen concentration.

REFERENCES

V-LO-PP-29101 "CNMT HVAC Systems" presentation, slides 3, 84, and 86

V-LO-LP-35103 "Corrosion" lesson plan, pages 14 and 15

SOP 13130-1, "Post-Accident Hydrogen Control" page 1 and 14.

VEGP learning objectives:

V-LO-LP-36107-02:

State the hazardous concentration ranges of explosive and flammable mixtures of hydrogen in air.

V-LO-LP-36107-03:

State the means available to measure and control the containment hydrogen concentration.

V-LO-PP-29101-18:

State the upper and lower limits for an explosive mixture of hydrogen.

V-LO-PP-29101-20:

List the methods for monitoring and controlling hydrogen inside Containment.

Rev No	Date	Reason for Revision	Author's Initials	Supv's Initials
0.0	1/7/04	Initial Development	EMT	DS
1.0	5/04/04	Incorporated Instructor Feedback Action Item # 2004201028	TLH	DS
2.0	10/07/05	Per AI 2005202977 Delete references to CNMT H2 Recombiners	LPV	DS
3.0	10/19/05	Removed H2 Recombiners from Picture of Containment internals	LPV	DS
4.0	11/02/05	Incorporated MDC 2000.048 per Action Item # 2004201074	TLH	DS
5.0	06/12/06	Revised for 06/13/06 class to enhance and cover the objectives.	TNT	DS

V-LO-PP-29101

HYDROGEN IS FORMED BY:

- Zirconium - H_2O Reaction
- Aluminum/zinc - H_2O
- Radiolysis of Water
- H_2 From the RCS and PRZR

84

H2 flammability (combustible) limit is 4% to 75% in air.

18% – 59% is the explosive range.

V- LO- PP- 29101

Preferred order for Post- LOCA H2 control

1 . Dilution with Service air

- Must stop at CNMT pressure 40psig

2. Post LOCA Purge

- Radiological release to environment

-Requires emergency Director Approval

V-LO-PP-29101

a. In steam generators around the tubes where boiling can cause concentration of hydroxyl ions

b. Nucleate boiling in the core can also lead to a caustic environment

5. The mechanism of caustic stress corrosion is similar to that of chloride stress corrosion

Q. Carbide Precipitation (Sensitization)

1. Type 304 stainless steel is susceptible

2. Sensitization is an increased susceptibility to intergranular attack

3. Sensitization is caused by exposure to heat treatments that precipitate chromium-rich carbides along the grain boundaries

   a. Carbon atom "ties up" many chromium atoms

   b. Chromium cannot exert corrosion-resistant effect

4. These carbides are precipitated during thermal and strain transients such as multiple pass welding

5. High carbon content of alloys contributes to susceptibility to sensitization

6. Susceptibility is decreased by stabilizing with carbine forming elements (boron, titanium, niobium)

R. Hydrogen Embrittlement

LO-TP-35103-021 Objective 7

1. Process by which steel loses ductility and strength due to tiny cracks

   - Result from internal pressure of H_2 or CH4 which forms at grain boundaries

2. Process

   a. Monatomic hydrogen produced in corrosion reaction is absorbed into metal

   b. Hydrogen diffuses along grain boundaries

   c. Hydrogen can combine with carbon, which is alloyed with iron, to form methane gas

d. Gas collects in small voids and builds up enormous pressures which initiate cracks

e. If the hydrogen atoms combine on the surface of the metal, they are released to the environment as H_2 gas

3. Zirconium alloys are susceptible

a. $Zr + 2H_2O \text{ ---> } ZrO_2 + 2H_2$ + heat

b. H2 diffuses through oxide layer to metal

c. $Zircaloy_2$ absorbs as much as 50% of corrosion-produced hydrogen

d. $Zircaloy_4$ absorbs significantly less because of:

1) Lower percentage of nickel

2) Addition of niobium

S. Boric Acid Corrosion

Commitment start GP-12962.000

1. Carbon steel highly susceptible, stainless steel mildly susceptible

2. Carbon steel and stainless steel undergo wastage or general dissolution corrosion

a. Other forms of corrosion; i.e., pitting, stress corrosion cracking, intergranular attack not types of boric acid corrosion

3. Conditions needed for corrosion to occur

a. High boric acid concentration (>1%)

b. Elevated temperatures (approx 200° or more)

c. Susceptible material

d. Aerated atmosphere

4. Corrosion rates

Concentration of boric acid (%)	Condition	Temperature (°F)	Corrosion rate mils/month
25	Aerated	200	400
25	Deaerated	200	250
15	Aerated	200	350-400
15-25	Dripping	210	400

1.0 PURPOSE

This procedure provides instructions for operation of the Containment Hydrogen Monitoring System, the Post-LOCA Cavity Purge System, and the Post-LOCA Containment Hydrogen Purge System during normal and post-LOCA conditions. Instructions are provided in the following sections.

4.1.1 Placing The Containment Hydrogen Monitoring System In Standby

4.1.2 Deleted

4.1.3 Placing The Post-LOCA Cavity Purge And Post-LOCA Containment Hydrogen Purge Systems In Standby

4.2.1 Containment Hydrogen Monitor 1-1513-P5-HMA Operation (Hydrogen Measurement)

4.2.2 Containment Hydrogen Monitor 1-1513-P5-HMB Operation (Hydrogen Measurement)

4.4.1 Deleted – this used to be the CNMT recombiner

4.4.2 Diluting Containment Hydrogen Concentration Using The Service Air System

4.4.3 Post-LOCA Containment Hydrogen Purge System Operation

4.4.4 Changing O_2 Reagent Gas Bottles At The H_2 Monitors

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 Adhere to all applicable radiological controls.

2.1.2 Train A Hydrogen Monitor Supply Valves 1-HV-2792A, 1-HV-2792B, 1-HV-2791B, and Return Valve 1-HV-2793B may be opened in Modes 1, 2, 3, and 4 under administrative control as described in the basis for Technical Specification LCO 3.6.3.

2.1.3 Train B Hydrogen Monitor Supply Valves 1-HV-2790A, 1-HV-2790B, 1-HV-2791A, and Return Valve 1-HV-2793A may be opened in Modes 1, 2, 3, and 4 under administrative control as described in the basis for Technical Specification LCO 3.6.3.

INITIALS

4.4.3 Post-LOCA Containment Hydrogen Purge System Operation

NOTE

If plant conditions warrant, the Emergency Director may waive the Gaseous Release Permit requirement.

No Recombiners
No air
No other means is available.

CAUTIONS

- The Post-LOCA Containment Hydrogen Purge System is to be operated ONLY if the containment hydrogen concentration cannot be maintained below 4% by other means.
- The Post-LOCA Containment Hydrogen Purge System is designed to operate with a maximum pressure of 3 psi downstream of CNMT POST LOCA PURGE EXH DUCT CONTROL VLV 1-FV-2693.

4.4.3.1 **Initiate** a Gaseous Release Permit. ________

4.4.3.2 **Verify** containment atmosphere is sampled and analyzed. ________

4.4.3.3 **Verify** the Service Air System is operating. ________

4.4.3.4 **Verify** compliance with the ODCM Section 3.1.1 Table 3-1 for the gaseous effluent monitoring requirements. ________

4.4.3.5 **Verify** the Auxiliary Building Heating Ventilation And Air Conditioning System is operating. ________

4.4.3.6 **Place** disconnect switch at local Heater Control Panel 1-1508-N7-001-H01 to on. ________

4.4.3.7 **Push** RESET button at local Heater Control Panel 1-1508-N7-001-H01 and **verify** that reset red light is ON. ________

Critical

4.4.3.8 Due to high radiation area potential, **verify** Containment Inside Isolation Valves 1-HV-2624A and 1-HV-2624B are closed and remain closed during the performance of the next step and until personnel have exited the area. ________

____
CV

30. 032AA2.07 001/1/2/LOSS SR NI-CH DISAGR/C/A - 2.8 / 3.4/NEW/HL-15R NRC/RO/TNT/DS

The plant is in Mode 3.

- SR / IR Signal Processor Channel Operational Tests are in progress.
- Background counts for both SR channels are 1000 cps
- The UO records the counts when the HFASA alarm lights for each SR channel.

N31 - 2080 cps N32 - 2340 cps

Which **ONE** of the following is **CORRECT** regarding Technical Specification 3.3.8, High Flux At Shutdown Alarm (HFASA)?

A. The LCO is met for both SR NIS HFASA alarms.

B. LCO entry required due to N31 setpoint too low.

C.✓ LCO entry required due to N32 setpoint too high.

D. LCO entry required for both SR NIS HFASA alarms.

K/A

032 Loss of Source Range Nuclear Instrumentation

AA2.07 Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation.

Maximum allowable channel disagreement.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a SR Signal Processor Channel Operational Test is in progress. The student is given data for the cps where the HFASA alarm illuminates. The student must determine if the LCO for HFASA is met and why.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. The LCO HFASA alarm setpoint is > 2.3 X background. The HFASA LCO is required to be entered.

B. Incorrect. SR N31 is within the allowable range for the HFASA alarm, there is no low alarm setpoint requirement per Tech Specs. Per our procedures, $\geq$2 X background is the normal conservative alarm setpoint. Therefore, N31 alarms close to the expected setpoint per our ARPs.

criteria of the procedure. N31 setpoint is set as expected ($\geq$2 X background).

D. Incorrect. While N32 setpoint is too high and outside the Tech Spec limits and acceptance criteria, N31 setpoint is set as expected ($\geq$2 X background)

REFERENCES

OSP-14423-1, N31/N35 Signal Processor Channel Operational Test, data sheet 1.

OSP-14424-1, N32/N36 Signal Processor Channel Operational Test, data sheet 1.

17010-1, window C01 for Source Range Hi Flux Level At Shutdown.

V-LO-PP-17201, Source and Intermediate Range NIS.

Technical Specification 3.3.8, High Flux At Shutdown Alarm (HFASA).

VEGP learning objectives:

LO-PP-17201-04, Discuss the operation of the High Flux At Shutdown Alarm

LO-PP-17201-05, Discuss all applicable Technical Specifications associated with the Source and Intermediate Range Nuclear Instrumentation (from memory).

a. All LCOs

b. Applicability

c. All 1 hour actions.

DATA SHEET 1 Sheet 2 of 3

SECTION 5.1 N32/N36 SIGNAL PROCESSOR CHANNEL OPERATIONAL TEST BELOW P-6

FUNCTION	ACTION	LOWER LIMIT	**REQUIRED VALUE**	UPPER LIMIT	AS FOUND
SR HF@SD	ALARM*		**≤2.3 x SDM indication**		**
IR P6 PERMISSIVE	TRIP*	1.5 E-5	**2.0 E-5**	2.6 E-5	*
	RESET	7.6 E-6	**1.0 E-5**	1.3 E-5	
SR HI FLUX	TRIP*	7.9 E+4	**1.0 E+5**	1.3 E+5	*
	RESET	3.9 E+4	**5.0 E+4**	6.4 E+4	
IR HI LEVEL ROD STOP	TRIP	1.5 E+1	**2.0 E+1**	2.6 E+1	
	RESET	5.4 E+0	**1.0 E+1**	1.3 E+1	
IR HI FLUX	TRIP*	1.9 E+1	**2.5 E+1**	3.3 E+1	*
	RESET	1.4 E+1	**1.88 E+1**	2.5 E+1	

NOTE 1: **Calculate** HF@SD TRIP setpoint upper limit by multiplying SR count rate recorded in Step 5.1.3.2 by 2.3 and **record** in the Upper Limit box for SR HF@SD.

NOTE 2: **Attach** additional Data Sheet 1 if High Flux At Shutdown Alarm Setpoint Retest is required following calibration.

* Technical Specification limit (ACCEPTANCE CRITERIA).

** Must be less than or equal to 2.3 times background level recorded in Step 5.1.3.2 WHEN performing quarterly surveillance (ACCEPTANCE CRITERIA).

Containment Evacuation Siren Test Completed SAT ________ INIT

P-6 Permissive Verified RESET ________ INIT

P-10 Permissive Verified RESET ________ INIT

DATA SHEET 1 Sheet 2 of 3

SECTION 5.1 N31/N35 SIGNAL PROCESSOR CHANNEL OPERATIONAL TEST BELOW P-6

FUNCTION	ACTION	LOWER LIMIT	**REQUIRED VALUE**	UPPER LIMIT	AS FOUND
SR HF@SD	ALARM*		**≤2.3 x SDM indication**		**
IR P6 PERMISSIVE	TRIP*	1.5 E-5	**2.0 E-5**	2.6 E-5	*
	RESET	7.6 E-6	**1.0 E-5**	1.3 E-5	
SR HI FLUX	TRIP*	7.9 E+4	**1.0 E+5**	1.3 E+5	*
	RESET	3.9 E+4	**5.0 E+4**	6.4 E+4	
IR HI LEVEL ROD STOP	TRIP	1.5 E+1	**2.0 E+1**	2.6 E+1	
	RESET	5.4 E+0	**1.0 E+1**	1.3 E+1	
IR HI FLUX	TRIP*	1.9 E+1	**2.5 E+1**	3.3 E+1	*
	RESET	1.4 E+1	**1.88 E+1**	2.5 E+1	

NOTE 1: **Calculate** HF@SD TRIP setpoint upper limit by multiplying SR count rate recorded in Step 5.1.3.2 by 2.3 and **record** in the upper limit box for SR HF@SD.

NOTE 2: **Attach** additional Data Sheet 1 if High Flux At Shutdown Alarm Setpoint Retest is required following calibration.

* Technical Specification limit (ACCEPTANCE CRITERIA).

** Must be less than or equal to 2.3 times background level recorded in Step 5.1.3.2 when performing quarterly surveillance. (ACCEPTANCE CRITERIA)

Containment Evacuation Siren Test Completed SAT ________ INIT

P-6 Permissive Verified RESET ________ INIT

P-10 Permissive Verified RESET ________ INIT

WINDOW C01

ORIGIN	SETPOINT
1-N31AX* (NIS) 1-N32AX* (NIS)	≥2.0 times Shutdown Monitor Indication
UN0031 (IPC) UN0032 (IPC)	≥2.0 times IPC UN5031/UN5032

SOURCE RANGE
HI FLUX LEVEL
AT SHUTDOWN

NOTES

The High Flux At Shutdown Alarm should be in service:

- In Modes 3, 4 and 5.
- In Mode 6 with the Reactor Makeup Water Valves 1-1208-U4-175, 1-1208-U4-176, 1-1208-U4-177, and 1-1208-U4-183 not closed and secured in position (by mechanical stops).

The High Flux At Shutdown Alarm may be blocked:

- In Modes 1 and 2.
- During fuel movement when the Reactor Makeup Water Valves 1-1208-U4-175, 1-1208-U4-176, 1-1208-U4-177, and 1-1208-U4-183 are closed and secured in position (by mechanical stops) and the SR channels are operable.
- Mode 3 during reactor startup.

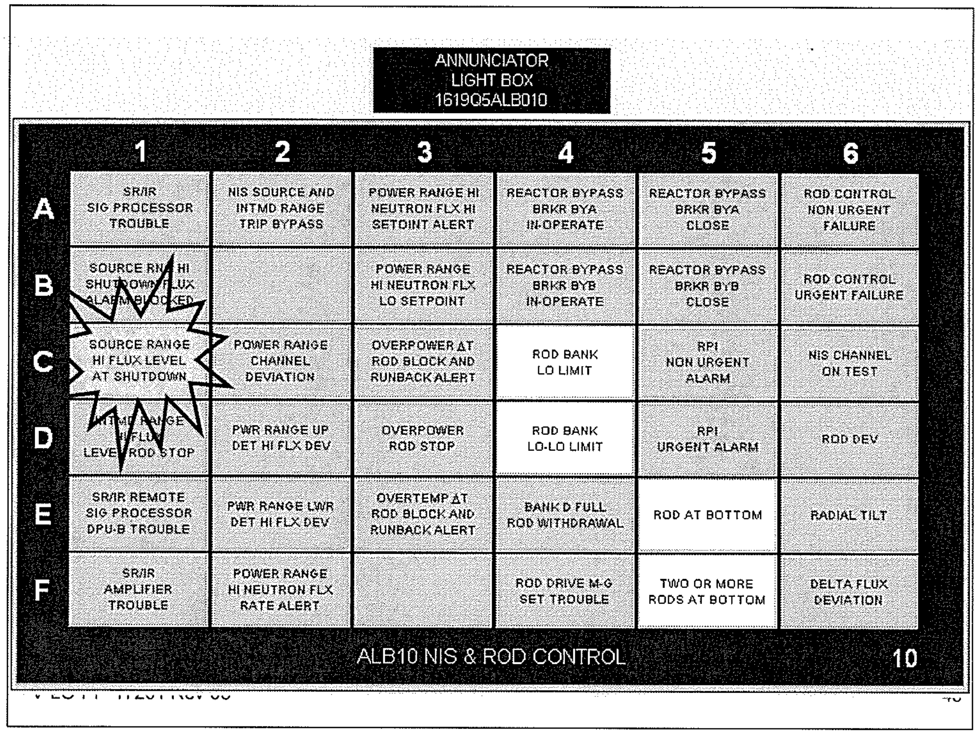
1.0 PROBABLE CAUSE

Reactivity addition caused by any of the following:

1. RCS dilution.
2. RCS cool down.
3. Xenon decay.
4. Rod withdrawal.
5. Refueling activities.

2.0 AUTOMATIC ACTIONS

Containment evacuation horn actuates.



Technical Specification (High Flux At Shutdown)

The primary purpose of the "High Flux AT Shutdown Alarm" (HFASA) is to warn the operator of an unplanned boron dilution event in sufficient time (15 minutes prior to loss of shutdown margin) to allow manual action to terminate the event. In order to comply with this you must have two separate channels of alarms OPERABLE receiving inputs from their respective Source Range Channels. The Technical Specification required setpoint for this Main Control Room alarm is 2.3 times background, however ours is set a 2.0 times background which is more conservative (T.S./Bases for 3.3.8).

This function is required to be OPERABLE in MODES 3, 4, and 5 (The HFASA may be blocked in MODE 3 during Reactor Startup).

In addition if the plant is in MODE 6 for refueling or MODE 5, Loops not filled, all the unborated water sources to the RCS must be secured in the closed position. The Technical Specification does allow for RCS chemical additions using water from the Reactor Makeup Water Storage System (RMWST) through the chemical mixing tank under these conditions. One of the requirements would be to have the HFASA OPERABLE while adding chemicals. Refer to T.S./Bases for 3.4.8 & 3.9.2 for details.

3.3 INSTRUMENTATION

3.3.8 High Flux at Shutdown Alarm (HFASA)

LCO 3.3.8 Two channels of HFASA shall be OPERABLE.

One Does NOT meet acceptance criteria of 14915 (N32). LCO entry required

APPLICABILITY: MODES 3, 4, and 5

---------------------------------------NOTE---------------------------------------
The HFASA may be blocked in MODE 3 during reactor startup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel of HFASA inoperable.	A.1 ------------------NOTE---------------- LCO 3.0.4c is applicable provided Required Actions B.1 and B.2 are met. -- Restore channel to OPERABLE status.	 48 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two channels of HFASA inoperable.	B.1 Perform SR 3.1.1.1 (verify SDM). AND B.2 Perform SR 3.9.2.1 (verify unborated water source isolated).	1 hour AND Once per 12 hours thereafter 4 hours AND Once per 14 days thereafter

B 3.3 INSTRUMENTATION

B 3.3.8 High Flux at Shutdown Alarm (HFASA)

BASES

BACKGROUND	The primary purpose of the HFASA is to warn the operator of an unplanned boron dilution event in sufficient time (15 minutes prior to loss of shutdown margin) to allow manual action to terminate the event. The HFASA is used for this purpose in MODES 3 and 4, and MODE 5 with the loops filled. The HFASA consists of two channels of alarms, with each channel receiving input from one source range channel. An alarm setpoint of ≤2.3 times background provides at least 15 minutes from the time the HFASA occurs to the total loss of shutdown margin due to an unplanned dilution event. This meets the Standard Review Plan criteria for mitigating the consequences of an unplanned dilution event by relying on operator action.
APPLICABLE SAFETY ANALYSES	The analysis presented in Reference 1 identifies credible boron dilution initiators. Time intervals from the HFASA until loss of shutdown margin were calculated. The results demonstrate that sufficient time for operator response is available to terminate an inadvertent dilution event taking credit for one HFASA with a setpoint of ≤2.3 times background. The HFASA satisfied Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	The LCO requires two channels of HFASA to be OPERABLE with input from two source range channels to provide protection against single failure.
APPLICABILITY	The HFASA must be OPERABLE in MODES 3, 4, and 5. The Applicability is modified by a Note which allows the HFASA to be blocked in MODE 3 during reactor startup so that spurious alarms are not generated.

(continued)

BASES

APPLICABILITY (continued)

In MODES 1 and 2, operators are alerted to an unplanned dilution event by a reactor trip on overtemperature delta-T or power range neutron flux high, low setpoint, respectively. As a protective measure in addition to HFASA, in MODE 5 with the loops not filled, unplanned dilution events are precluded by requiring the unborated water source (reactor makeup water storage tank (RMWST)) to be isolated.

ACTIONS

A.1

With one channel of HFASA inoperable, Required Action A.1 requires the inoperable channel to be restored within 48 hours. In this condition, one channel of HFASA remains available to provide protection. The 48 hour Completion Time is consistent with that required for an inoperable source range channel. Required Action A.1 is modified by a Note stating that LCO 3.0.4c is applicable provided that Required Actions B.1 and B.2 are met. When Condition A (and Required Action A.1) are applicable, the Note permits MODE changes provided that Required Action B.1 and B.2 are met. Required Action B.1 is a periodic verification of shutdown margin, and Required Action B.2 ensures that the unborated water source isolation valves are shut, precluding a boron dilution event. With one channel of HFASA inoperable, it is prudent to take the compensatory actions of Required Actions B.1 and B.2 if MODE changes are desired or required.

B.1 and B.2

With the Required Action A.1 and associated Completion Time not met, or with both channels of HFASA inoperable, the appropriate ACTIONS are to verify that the required SDM is present and isolate the unborated water source by performing

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

SR 3.9.2.1. This places the unit in a condition that precludes an unplanned dilution event. The Completion Times of 1 hour and once per 12 hours thereafter for verifying SDM provide timely assurance that no unintended dilution occurred while the HFASA was inoperable and that SDM is maintained. The Completion Times of 4 hours and once per 14 days thereafter for verifying that the unborated source is isolated provide timely assurance that an unplanned dilution event cannot occur while the HFASA is inoperable and that this protection is maintained until the HFASA is restored.

SURVEILLANCE REQUIREMENTS

The HFASA channels are subject to a COT and a CHANNEL CALIBRATION.

SR 3.3.8.1

SR 3.3.8.1 requires the performance of a COT every 184 days to ensure that each channel of the HFASA and its setpoint are OPERABLE. This test shall include verification that the HFASA setpoint is less than or equal to 2.3 times background. The frequency of 184 days is consistent with the requirements for the source range channels. This Surveillance Requirement is modified by a Note that provides a 4-hour delay in the requirement to perform this surveillance for the HFASA instrumentation upon entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without delay for the performance of the surveillance to meet the applicability requirements in MODE 3.

SR 3.3.8.2

SR 3.3.8.2 requires the performance of a CHANNEL CALIBRATION every 18 months. This test verifies that each channel responds to a measured parameter within the necessary range and accuracy. It encompasses the HFASA portion of the instrument loop. The frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. FSAR, Subsection 15.4.6.

31. 033A1.02 001/2/2/SFPCS-RAD MONITORS/MEM - 2.8 / 3.3/NEW/HL-15R NRC/RO/TNT/DS

Which ONE of the following is CORRECT regarding:

1) The minimum Spent Fuel Pool level (elevation) **required** by Tech Specs for adequate shielding and design basis fuel handling events.

2) how the FHB crew would be alerted to High radiation on RE-0008, Fuel Handling Building Area radiation monitor.

Tech Spec level	RE-0008 high rad alarm
A. 214 ft. 6 inch	audible horn and blinking strobe light
B. 214 ft. 6 inch	warble type siren on plant gai-tronics
C.✓ 217 ft. 0 inch	audible horn and blinking strobe light
D. 217 ft. 0 inch	warble type siren on plant gai-tronics

K/A

033 Spent Fuel Pool Cooling System (SFPCS)

A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool System operating the controls including:

Radiation monitoring systems.

K/A MATCH ANALYSIS

The question asks the candidate the Tech Spec level required for adequate shielding in the Spent Fuel Pool and how the crew would be alerted of high radiation.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. 217 ft is the minimum Tech Spec limit (23 ft above fuel), 214 ft is plausible as this is prominently mentioned in the AOP. However, this is the level to stop the SFP cooling pumps and is plausible they may recall this level. 214 elevation is also the level a SFP cooling system line break can drain the pool down to. RE-0008 sounds an audible horn and strobe light to alert workers of high radiation, this part of the choice is correct.

B. Incorrect. 217 ft is the minimum Tech Spec limit (23 ft above fuel), 214 ft is plausible as this is prominently mentioned in the AOP. However, this is the level to stop the SFP cooling pumps and is plausible they may recall this level. RE-0008 sounds an audible horn and strobe light to alert workers of high radiation. A warble type siren is

C. Correct. 217 ft is the Tech Spec minimum required level (23 ft above fuel) and RE-0008 sounds an audible horn and strobe light.

D. Incorrect. 217 ft is the Tech Spec minimum required level (23 ft above fuel) and RE-0008 sounds an aubible alarm and strobe light. A warble type siren is used on gai-tronics for plant events, but not locally on high radiation.

REFERENCES

17100-1, ARP for RE-0008 high radiation (included).

18030, Loss of Spent Fuel Pool Cooling pages 3 and 4 (inlcuded)

Tech Spec 3.7.15, Fuel Storage Pool Level and Bases (Background)

V-LO-PP-25102, Spent Fuel Pool Cooling System, page # 23

VEGP learning objectives:

LO-LP-25102-09, Describe the impacts of the following conditions:

a. Low level in the Spent Fuel Pool.

LO-LP-25102-12, Describe the minimum allowable water level over the spent fuel pool and the basis for this level.

ORIGIN	SETPOINT	
Area Monitor	As determined by Chemistry Department	1-RE-0008 (High)

NOTE

For other than HIGH conditions see Pages 4 and 5.

1.0 **PROBABLE CAUSE**

Increase in radiation level near Unit 1 Spent Fuel Pool in the Fuel Handling Building.

2.0 **AUTOMATIC ACTIONS**

On the south wall of the Fuel Handling Building Spent Fuel Pool Room near the door:

a. Alarm horn on 1-RA-0008 sounds.

b. Strobe light on 1-RA-0008 blinks.

3.0 **INITIAL OPERATOR ACTIONS**

Evacuate the Fuel Handling Building.

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

__1. Initiate the Continuous Actions Page.

NOTE

Local SFP level indicator and FIGURE 1 should be compared to determine actual SFP level.

__*2. **Check SFP level - LESS THAN OR EQUAL TO 217 FT ELEVATION.**

__*2. Go to Step 20.

__3. Place fuel assembly in transit in a safe place.

__4. Suspend all fuel assembly movement.

__5. Suspend crane operation over the spent fuel pool.

Tech Spec actions.

__6. Makeup to restore level by initiating 13719, SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM.

7. Check conditions requiring emergency makeup:

- __ Imminent security threat.
- __ Any other extreme conditions warranting emergency makeup.

__7. Go to Step 9.

8. Perform the following as necessary:

- __ Contact the TSC for temporary repair options.
- __ Initiate ATTACHMENT C for SFP makeup sources.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__*9. **Check SFP level - LESS THAN OR EQUAL TO SFP PUMP SUCTION.** (214 ft. 6 in.)	__*9. Go to Step 20.

Plausible important level.

__10. Stop SFP cooling pump(s).

UNIT 1 (At door to AB-A53 & FHB-A06)

UNIT 2 (At door to AB-A91 & FHB-A05)

__11. Stop SFP skimmer pump(s).

UNIT 1 (AB-A53)

UNIT 2 (AB-A91)

__12. Close SFP cooling pump suction valve(s).

UNIT 1

- 1213-U6-001 (FHB-A06)
- 1213-U6-003 (FHB-A07)

UNIT 2

- 1213-U6-001 (FHB-A05)
- 1213-U6-003 (FHB-A04)

__13. Close SFP skimmer pump(s) suction isolation.

UNIT 1

- 1213-U6-014 (AB-A53)

UNIT 2

- 1213-U6-014 (AB-A91)

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Water Level

LCO 3.7.15 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies in the fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

BASES

BACKGROUND

Design Limits

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Subsection 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Subsection 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Subsection 15.7.4 (Ref. 3).

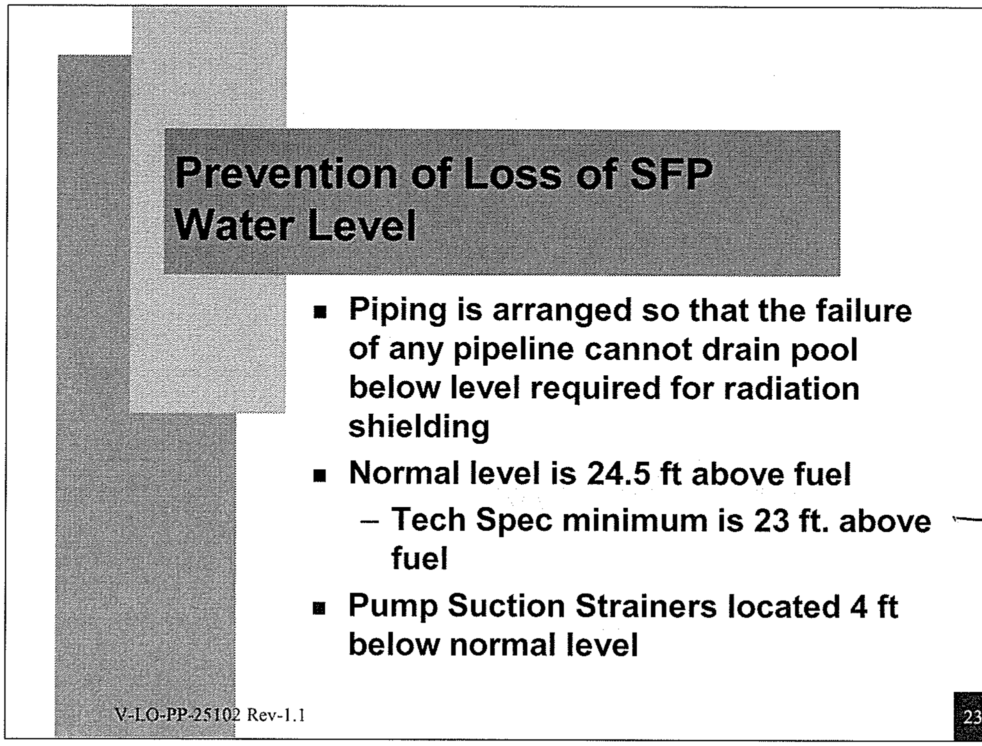
APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)



V-LO-PP-25102-05, Describe how the Spent Fuel Pool is designed to minimize the occurrence and effects of inventory loss.

24.5 feet above fuel is 218'6"

23 feet above fuel is 217' Tech Spec limit

218'6 – 4ft = 214'6" Pump suction strainers (Plausible + in the AOP)

The biggest threat would be a Transfer Canal Gate seal failure with the transfer tube open and the canal and reactor cavity empty.

We have admin controls in the UOP to handle this concern.

32. 035G2.2.40 001/2/2/SG-TECH SPECS/MEM - 3.4 / 4.7/M- LOIT BANK/HL-15R NRC/RO/TNT/DS

SG # 2 ARV PV-3010 has been declared inoperable due to large hydraulic fluid leaks on the valve operator.

Which ONE of the following is CORRECT regarding

1) real LCO entry and

2) **all** the applicable modes the ARVs are **required** to be OPERABLE?

	LCO entry	**Applicable Modes**
A.	required	Modes 1, 2, and 3
B.	required	Modes 1 and 2
C.✓	NOT required	Modes 1, 2, and 3
D.	NOT required	Modes 1 and 2

HL-15R RO NRC Exam

K/A

035 Steam Generator System (S/GS)

G2.2.40 Ability to apply Technical Specifications for a system.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a SG ARV is declared inoperable. The candidate must determine if LCO entry is required and the applicable modes.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. LCO entry is NOT required but plausible, the student may think all ARVs are required to be operable. Only 3 ARVs are required to meet the LCO. ARVs are required in Modes 1 through 3.

B. Incorrect. LCO entry is NOT required but plausible the student may think all ARVs are required to be operable only 3 ARVs are required to meet the LCO. Part of choice about ARVs required in Modes 1 and 2 is incorrect.

C. Correct. 3 ARVs required in Modes 1, 2, and 3. LCO entry NOT required as the minimum requirements are met.

D. Incorrect. LCO entry is NOT required as only 3 ARVs are required to meet the LCO. ARVs are required in Modes 1 through 3. Part of choice about ARVs required in Modes 1 and 2 is incorrect.

REFERENCES

LO-LP-39211-02-006 from LOIT Exam Bank used as base for modification (included).

Tech Spec LCO 3.7.4 for Atmospheric Relief Valves

VEGP learning objectives:

LO-PP-21101-10, Discuss the following concerning the Atmospheric Relief Valve(ARV).

d. Technical Specification requirements for operability.

LO-LP-39211-01, For any LCO in section 3.7 of Tech Specs, be able to:

a. State the LCO.

1. LO-LP-39211-02 006
The following conditions exist on Unit Two.

* Unit Two Reactor Power is at 100%.

* ARVs PV-3000 and PV-3030 have been declared inoperable because of large fluid leaks on the valve operators.

* The hand pump for PV-3010 has been tagged out for two weeks awiating parts to repair the pump

Based on this information, what Technical Specification actions are required?

A. ✓ Restore one required ARV line to operable within 30 days

B. Restore Both ARV lines to operable within 30 days

C. Restore one required ARV line to operable within 24 hours

D. Restore at least two ARV lines to operable within 24 hours

Question used as base for modification

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Three ARV lines shall be OPERABLE. (3 Required)

APPLICABILITY: MODES 1, 2, and 3 (applicability)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ARV line inoperable.	A.1 Restore required ARV line to OPERABLE status.	30 days
B. Two or more required ARV lines inoperable.	B.1 Restore at least two ARV lines to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4	6 hours 18 hours

33. 038EK1.04 001/1/1/SGTR-REFLUX BOIL/MEM - 3.1 / 3.3/NEW/HL-15R NRC/RO/TNT/DS

Heat removal from the core by boiling, the steam flows to the S/G via the top of the hot legs, transfers heat to the secondary side water, condenses in the S/G tubes and returns to the vessel via the bottom of the hot legs.

This process is known as_____(1)_____ and is_____(2)_____ to occur during a Steam Generator Tube Rupture (SGTR) of a single tube.

	(1)	(2)
A.	Reflux Cooling	expected
B.✓	Reflux Cooling	NOT expected
C.	Natural Circulation	expected
D.	Natural Circulation	NOT expected

K/A

038 Steam Generator Tube Rupture (SGTR)

EK1.04 Knowledge of the operational implications of the following concepts as they apply to the SGTR:

Reflux boiling.

K/A MATCH ANALYSIS

The question gives the definition for reflux cooling which the student must recognize as reflux cooling versus natural circulation and whether reflux cooling is expected during a SGTR of a single tube (not).

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Reflux cooling is the correct term, however, it is not expected to occur during a SGTR, it is possible during some small break LOCAs and loss RHR while at midloop.

B. Correct. Reflux cooling is the correct term, it is not expected to occur during a SGTR, it is possible during some small break LOCAs and loss RHR while at midloop.

C. Incorrect. Natural circulation is plausible since this sounds like a circulation flow path, however, natural circulation as described in WOG backgrounds is with loops full. This is also not expected to occur during a SGTR, it is possible during some small break LOCAs and loss RHR while at midloop.

flow path, however, natural circulation as described in WOG backgrounds is with loops full. This is also not expected to occur during a SGTR, it is possible during some small break LOCAs and loss RHR while at midloop.

REFERENCES

V-LO-PP-36101, MCD Core Cooling Mechanisms, slides # 14, 15, 16, 17, 20, 26 & 27.

V-LO-HO-37311-001, Steam Generator Tube Rupture, no mention of Reflux Cooling since on a SGTR of a single tube, RCP trip criteria is not normally met nd SG tubes remain filled.

VEGP learning objectives:

LO-LP-36101-04, State three conditions required to establish natural circulation flow.

LO-LP-36101-14, State, in order of effectiveness, alternate methods of cooling available tot he operator during accident conditions.

LO-LP-36101-15, State the effectiveness of steam cooling versus water cooling during core heat removal.

Natural Circulation

- 7. List and describe three factors that can be used by the operator to enhance natural circulation flow.
  - Keep PZR pressure > 1920 PSIG - keep voids from forming in head.
  - PZR level > 25% - water inventory for pressure control.
  - Steam generator NR level in NR of at least one > 10% maintain heat sink.

Natural Circulation

- 8. List and describe three separate indications of natural circulation flow.
  - RCS ΔT < full load ΔT.
  - RCS or CETC's constant or decreasing.
  - SG pressure constant or decreasing at RCS temperature rate while maintaining level.
  - RCS cold leg temperatures at saturation temperature for SG pressure

Natural Circulation

- **9. State how the formation of noncondensible gases and/or steam can result in degradation of natural circulation flow.**
  - **If gases block flow path, i.e., loops, SG tubes, below vessel nozzles, they can retard/stop NC flow.**
    - **Major concern is during a Large Break LOCA when the Accumulators inject Nitrogen, potential exist for the Nitrogen to accumulate in the SG U-tubes.**

Noncondensible Gases

- **10. List several (at least 6) sources of noncondensible gases in the RCS.**
  - Dissolution of hydrogen
  - Radiolysis
  - PZR vapor space
  - Zr-H_2O reaction
  - Accumulator nitrogen
  - Fission gases/fuel pin helium
  - Gases from injected water

Pool Boiling

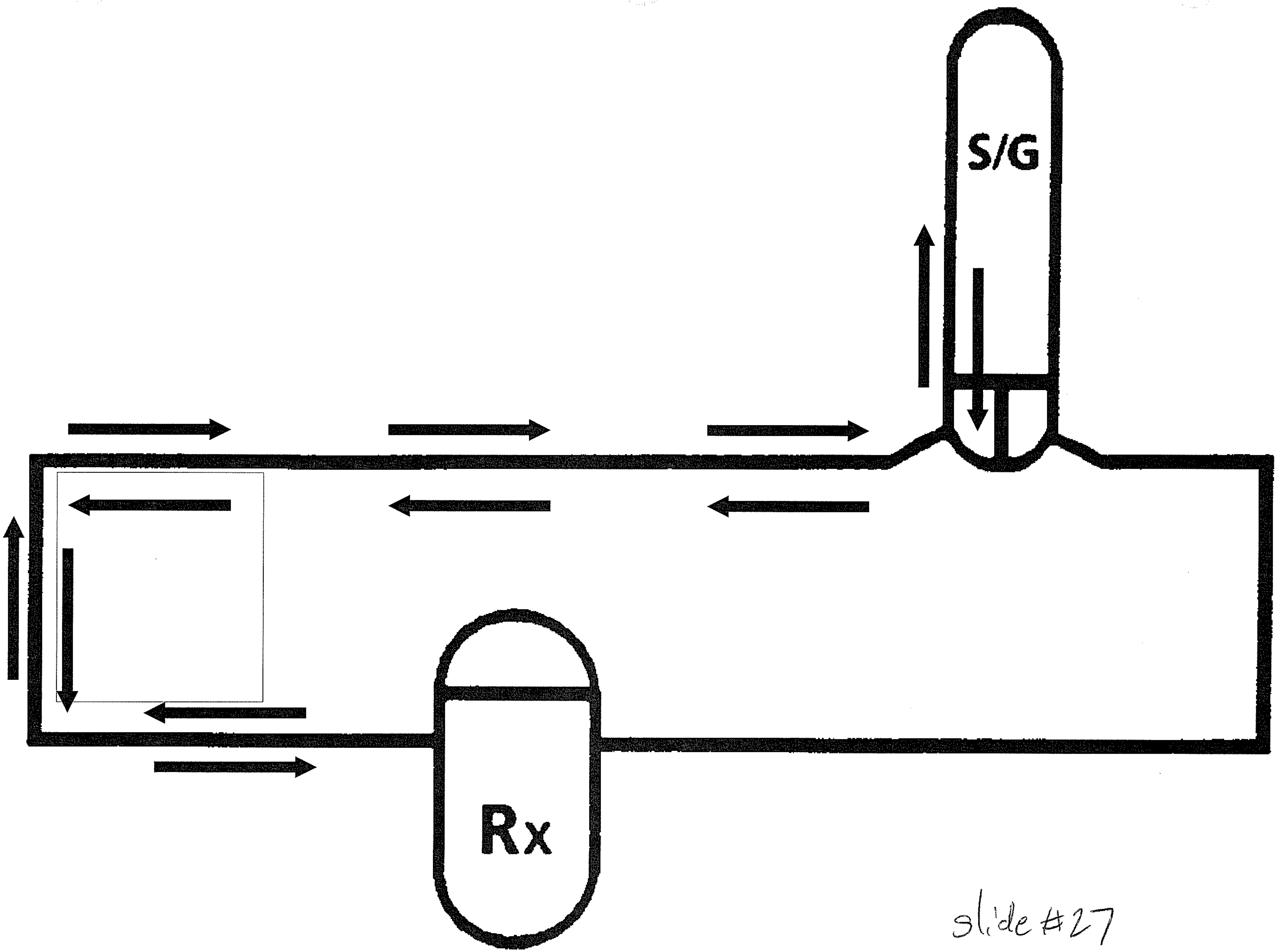
- **11. Define the term, "Pool Boiling."**
  - Stagnant water in the reactor vessel boils and then moves upward carrying away energy.

Reflux Cooling

– Cooling process that can occur during mid-loop following Loss of RHR.

- ♦ Heat is removed from core by boiling RCS water.
- ♦ Steam flows to SG Via top of hot legs.
- ♦ Steam condenses in SG tubes, transfers heat to SG secondary side water.
- ♦ Water from condensed steam returns to Rx vessel via bottom of the same hot

Slide 26



slide #27

34. 039A2.01 001/2/1/MAIN STEAM-LOCA FLOW/C/A - 3.1 / 3.2/NEW/HL-15R NRC/RO/TNT/DS

Given the following conditions:

- An RCS LOCA has occurred.
- Containment pressure is 15.5 psig and stable.
- A loss of offsite power to 13.8 kV switchgear NAA has occurred.
- RCS pressure is 1080 psig and stable.
- The crew is performing 19012-C, "E-1.2 Post LOCA Cooldown and Depressurization".

Which ONE of the following describes the method that will be used to perform the cooldown of the RCS and the rate of the cooldown?

A.✓ S/G ARVs.

$\leq$ 100°F per hour.

B. S/G ARVs.

Maximum rate attainable.

C. Steam Dumps.

$\leq$ 100°F per hour.

D. Steam Dumps.

Maximum rate attainable.

K/A

039 Main and Reheat Steam System (MRSS)

A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations;

HL-15R RO NRC Exam

K/A MATCH ANALYSIS

The question presents a plausible scenario where an RCS LOCA has occurred and the crew is cooling down the plant per 19012-C, Post LOCA Cooldown and Depressurization, the student must choose the correct method from plant conditions and the cooldown rate.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. SL pressure > 14.5 psig would cause an SLI and steam dumps would not be available. S/G ARVs would need to be used and per 19012-C, the RCS cooldown rate is limited to 100°F per hour.

B. Incorrect. SL pressure > 14.5 psig would cause an SLI and steam dumps would not be available. S/G ARVs would need to be used part is correct. However, the RCS cooldown rate is limited to 100°F per hour but this is plausible as other EOPs such as SGTR, Loss of All AC, LOHS, Inadequate Core Cooling use a maximum rate cooldown.

C. Incorrect. Steam dumps would not be available with Hi-2 Ctmt pressure present as this causes an SLI. The RCS cooldown rate limit of 100°F per hour is correct.

D. Incorrect. Steam dumps would not be available with Hi-2 Ctmt pressure present as this causes an SLI. the RCS cooldown rate is limited to 100°F per hour but this is plausible as other EOPs such as SGTR, Loss of All AC, LOHS, Inadequate Core Cooling use a maximum rate cooldown.

REFERENCES

19012-C, Post LOCA Cooldown and Depressurization, pages 9 and 11.

V-LO-PP-28103, Reactor Trip and ESFAS Signals, slide # 140

V-LO-PP-21201, Steam Dumps, slides # 91 (+ blow up of same slide).

VEGP learning objectives:

LO-LP-37112-01, Using EOP 19012 as a guide, briefly describe how each step is accomplished.

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

NOTE

When the low steamline pressure SI/SLI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

*10. **Check if low steamline pressure SI/SLI should be blocked:**

Hi-2 (14.5 psig) ctmt pressure is NOT addressed student has to recall SLI setpoint.

__a. Steam Dumps – AVAILABLE. → __a. Go to Step 12.

plausible he may think dumps are available

__b. PRZR pressure - LESS THAN 2000 PSIG. | __b. WHEN PRZR pressure is less than 2000 psig, and the high steam pressure rate alarms are clear, THEN block low steamline pressure SI/SLI by performing Step 10.d.

Makes C+D plausible.

__ Go to Step 11.

__c. High steam pressure rate alarms – CLEAR.

d. Block low steamline pressure SI/SLI using the following:

__• HS-40068

__• HS-40069

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***12. Initiate RCS cooldown to cold shutdown:**	
__a. Monitor shutdown margin by initiating 14005, SHUTDOWN MARGIN AND KEFF CALCULATIONS.	
__b. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR.	Max Rate is used in some procedures such as SGTR, Loss all AC, Inad Core Cooling, LOHS, (has to recall from given condition)
__c. Use RHR system if in service.	
Hi-2 prevents steam dumps	
d. Dump steam to Condenser from intact SG(s) using Steam Dumps:	__d. Dump steam from intact SG(s) using SG ARV(s).
__1) Place PIC-507 in Manual.	
__2) Match demand on SG Header Pressure Controller PIC-507 and SD demand meter UI-500.	
__3) Transfer Steam Dumps to STM PRESS mode.	
__4) Open available Steam Dumps by slowly raising demand on PIC-507.	
__13. Check RCS subcooling – GREATER THAN 24°F [38°F ADVERSE].	__13. Go to Step 36.

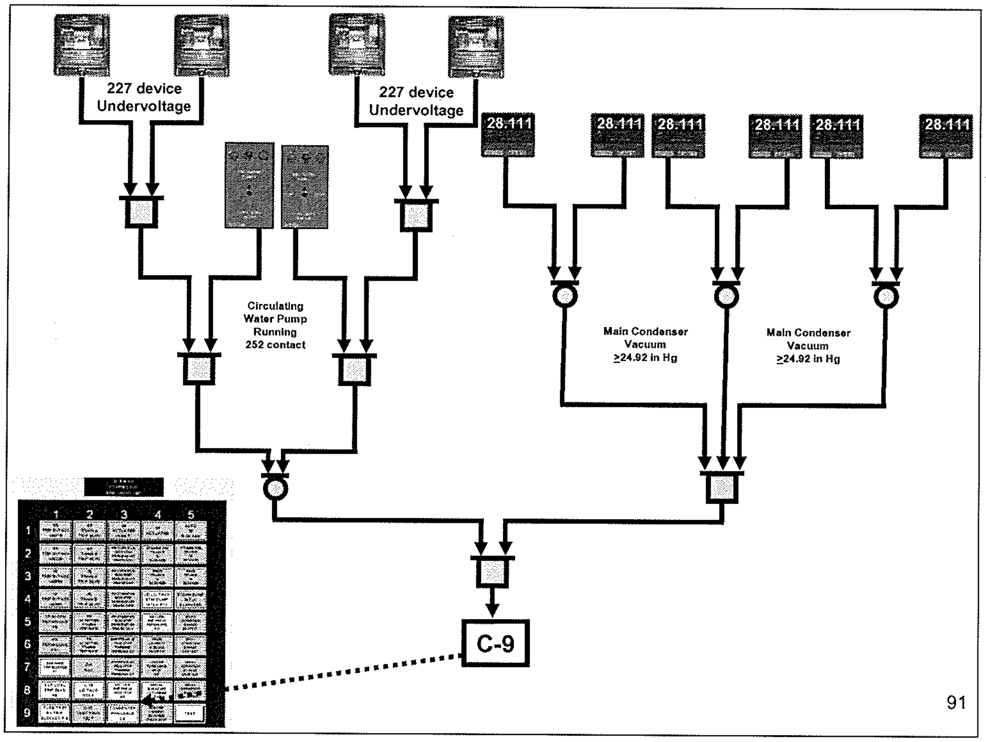
What actuation signals input to SLI?

-Low steam line pressure,
2/3 channels ≤ 585 psig, on 1/4 loops.
(may be manually blocked
below P-11, also rate compensated)

-High-2 containment pressure
2/3 channels ≥ 14.5 psig.

ctmt pressure 15.5 psig in stem cause SLI. Steam dumps NOT available.

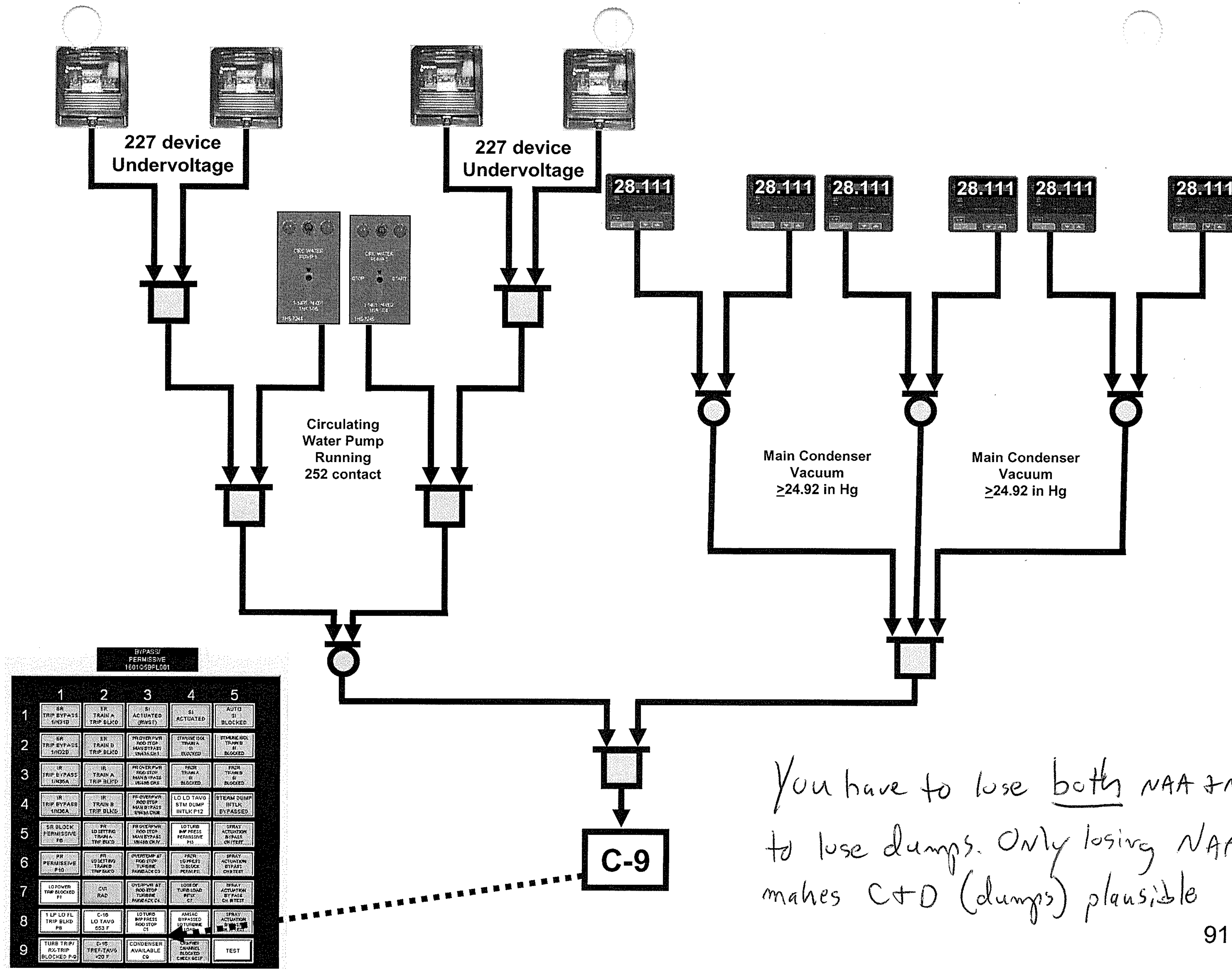
-High steam line negative rate,
2/3 channels on 1/4 loops,
100 psig with a 50 second time constant.
(Requires that P-11 be present and SI/SLI
on low steam line pressure manually blocked).



OBJECTIVE LO-PP-21201-12

If no circulating water pumps are running or insufficient vacuum exists in the main condenser, dumping steam into the condenser can cause an overpressure condition which can damage the condenser. To protect against this, a permissive circuit prevents arming of the steam dumps. As can be seen in the figures below, the permissive circuit is composed of contacts which involve condenser vacuum and the circulating water pump breakers and their associated switchgear voltages (at 0 volts for 5.75 seconds). The condenser vacuum contact will be closed as long as two separate condenser pressure transmitters provide a signal indicating that at least 24.92 inches of mercury vacuum exists in the condenser. Each of the circulating water pump breaker contacts will be closed as long as the breaker for the associated pump is closed and the associated switchgear voltages are present. All of the above contacts are arranged so that the condenser must have at least 24.92 inches of mercury vacuum and at least one circulating water pump breaker closed with voltage applied. This will energize the permissive relay which closes the permissive contact in the arming circuit to permit arming of the steam dump valves. The permissive relay then energizes a **"C-9 Condenser Available"** permissive status light.

With only NAA de-energized. Student may think since NAB and other CW pump available that steam dumps are still available. Makes C+D plausible.



You have to lose both NAA + NAB to lose dumps. Only losing NAA makes C+D (dumps) plausible

35. 051AA1.04 002/1/2/LOSS VACUUM-ROD POST/C/A - 2.5 / 2.5/NEW/HL-15R NRC/RO/DS/TNT

The plant is at 100% power in when the following valid annunciator is received following a circulating water pump trip:

TURB CNDSR LO VAC

- The SS enters AOP 18013-C, "Rapid Power Reduction".

- The Unit Operator rapidly reduces turbine load .

- Tave is 3.8°F higher than Tref.

Wwhich one of the following is correct regarding:

1) AOP-18013-C, power reduction target, and

2) preferred operation of the control rods in accordance with 18013-C?

	Target	Control Rods
A.	low vacuum alarm clear	manually insert at 48 steps per minute
B.✓	low vacuum alarm clear	automatically insert based on Tavg-Tref deviation
C.	20% rated thermal power	manually insert at 48 steps per minute
D.	20% rated thermal power	automatically insert based on Tavg-Tref deviation

K/A

051 Loss of Condenser Vacuum

AA1.04 Ability operate and / or monitor the following as they apply to the Loss of Condenser Vacuum.

Rod position.

K/A MATCH ANALYSIS

The question presents a plausible scenario where a low condenser vacuum alarm is received. The candidate must know the effect of this alarm on steam dump operation when C-7 is received and whether or not control rods are operated in auto versus manual.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Per 18013-C, the target should be a power reduction until the low vacuum plausible if auto does not work and the AOP has been recently changed to make automatic operation the desired method.

B. Correct. The target per 18013-C is to reduce power until low vacuum alarm clears. Rods are desired to be operated in auto per 18013-C which is a recent change.

C. Incorrect. 20% is used as a target since this is the power level 18013-C directs to reduce power and trip the reactor, therefore it is plausible. Manual is plausible if auto does not work and the AOP has been recently changed to make automatic operation the desired method.

D. Incorrect. 20% is used as a target since this is the power level 18013-C directs to reduce power and trip the reactor, therefore it is plausible. Rods are desired to be operated in auto per 18013-C which is a recent change.

REFERENCES

AOP-18013-C, Rapid Power Reduction

V-LO-PP-27101, Rod Control System, slides 40 and 41 (included).

VEGP learning objectives:

LO-PP-60331-01, Describe the entry and exit conditions associated with using the "Rapid Power Reduction" AOP.

LO-PP-60331-03, Describe the type conditions that might warrant the use of the "Rapid Power Reduction" AOP.

V-LO-PP-27101-07, Describe the operation of the Rod Control System with the bank selector switch in the manual or automatic position. Include the following.

a. Rod Speed

c. Input signals

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

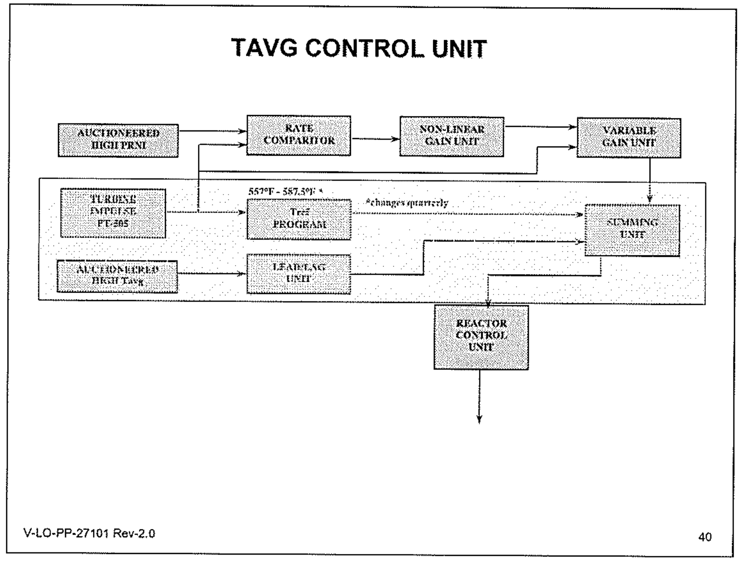
PURPOSE

This procedure provides instructions when plant conditions require a rapid load reduction or plant shutdown in a controlled manner in the judgment of the SS.

Entry	Condition	Target	Approx. Time @ 3-5%/min
17015-D05 17015-E01	MFPT High Vibrations	**<70% RTP**	5-8 minutes
17019-B04 18025-C	Condenser Low Vacuum *or* Circ Water Pump Trip *or* Loss of Utility Water	**Vacuum >22.42" Hg** and STABLE or RISING	
18009-C	SG Tube Leak (≥**75 gpd** with an **ROC ≥30 gpd/hr)**	**<50% RTP** within 1 hour	10-17 minutes
18009-C	SG Tube Leak (≥**5 gpm)**	**20% RTP** within 1 hour & trip reactor	16-27 minutes
18039-C	Confirmed Loose Part	**20% RTP** quickly	16-27 minutes
	SS determination based on plant conditions	As determined by the SS	

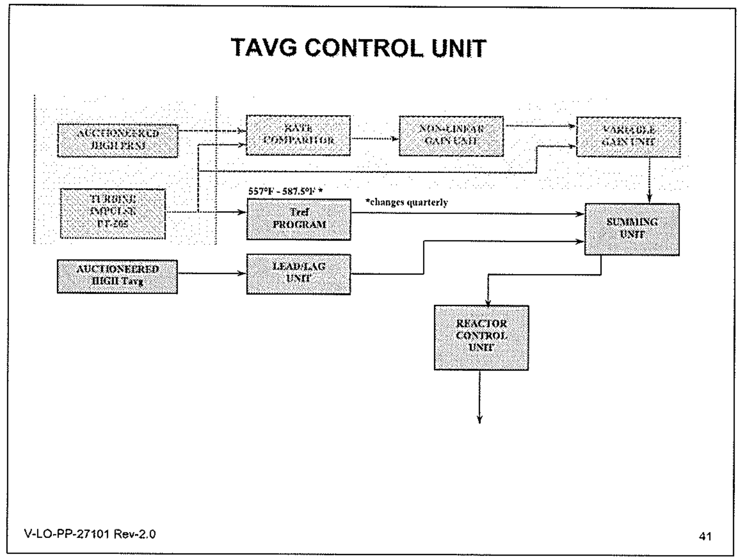
MAJOR ACTIONS

- Perform Pre-job Brief.
- Perform rapid power reduction.



Describes rod motion on Tave Tref mismatch,

The average reactor coolant temperature mismatch channel develops a temperature error signal by comparing actual auctioneered high reactor coolant average temperature, T-avg, to a reference reactor coolant average temperature, T-ref. T-ref represents the reactor coolant average temperature demanded by the turbine. T-ref is developed by sensing turbine first-stage pressure. The pressure signal, which represents the turbine load, is sent to a T-ref programmer. The T-ref programmer generates a linearly increasing T-ref signal of 557 deg F at 0% turbine load up to about 587.5 deg F at 100% turbine load. **Notice that the rod control Tref program is different from the normal Tavg program.** It is adjusted to account for the auctioneered Hi Tavg reading being higher than the average of all loops Tavg. This makes Tref and auctioneered Hi Tavg match when the average of Tavg is on program. The rod control Tref is adjusted as needed to maintain a match with the auctioneered Hi Tavg average temperature signals. The lead-lag unit accomplishes two purposes; the lag portion of the circuit filters noise and the lead portion boosts the rate of change of the temperature signal to compensate for the delay of the RTDs in seeing an actual core temperature change due to the transit time of the RCS coolant as well as the delay associated with RTD response time. This output signal is then sent to the summing unit for comparison to the T-ref signal. The T-avg signal and the T-ref signal are also directed to a two-pen meter on the main control board (QMCB) for indication to the operator and to a comparator (Bistable) for a T-avg-T-ref deviation alarm if their difference is greater than 3 deg F.



Describes Rod motion on Tavg-Tref mismatch,

To improve the T-avg control unit's response during transients, a rate of power mismatch channel also develops a temperature input signal to the summing unit. This circuit compares the rate of change of the difference between reactor power and turbine power. When the difference is constant, the circuit provides no input signal to the summing unit so that the average reactor coolant temperature channel makes the fine adjustments to T-avg. The input signals to this channel are reactor power and turbine load . Reactor power is developed by auctioneering the actual nuclear power signals from the four nuclear power channels so that the highest signal is used as the input to a rate comparator. Turbine power is developed from turbine first-stage pressure. The rates of change between reactor power and turbine power are compared in the rate comparator circuit. When the rates of change differ, an error signal is produced and sent to a nonlinear gain unit.

The nonlinear gain unit converts the power mismatch signal to a temperature signal. When the power mismatch signal is greater than 2 percent, it is multiplied by 1.5 deg F/percent to produce the temperature signal. When the power mismatch signal is equal to or less than 2 percent, it is multiplied by 0.3F deg/percent to produce the temperature signal. The output of this unit is then sent to the variable gain unit.

The variable gain unit compensates for reactor gain. This is needed because the reactivity changes at low power levels have smaller effects on nuclear power than at higher levels. The temperature signal input is multiplied by 2 when turbine load is equal to or less than 50 percent, Between 50 percent and 100 percent, the temperature signal input is multiplied by one divided by percent turbine power (gain is inversely proportional to turbine power). This adjusted temperature signal is then sent to the summing unit.

SHUTDOWN BRIEFING

METHOD

- Auto rod control should be used.
- Reduce Turbine Load at approximately **3%** RTP per minute (approx 36 MW_e) up to **5%** RTP (approx 60 MW_e).
- Borate considering the calculations from the reactivity briefing sheet and BEACON.
- Maintain AFD within the doghouse.
- SS (or SRO designee) - Maintain supervisory oversight.
- <u>All</u> rod <u>withdrawals</u> will be approved by the SS.
- Approval for each reactivity manipulation is not necessary as long as manipulations are made within the boundaries established in this briefing (i.e. turbine load adjustment up to 60 MW_e, etc.).
- A crew update should be performed at approximately every 100 MW_e power change.
- If manpower is available, peer checks should be used for all reactivity changes.

OPERATIONAL LIMITS

- Maintain T_{AVG} within ±6°F of T_{REF}. **If T_{AVG}/T_{REF} mismatch >6°F and *not* trending toward a matched condition <u>or</u> if T_{AVG} ≤551°F, then trip the reactor.**
- *If load reduction due to a loss of vacuum*, every effort should be made to maintain the steam dumps closed (Permissive C-9 ≥24.92" Hg).

INDUSTRY OE

- Shift supervision must maintain **effective oversight** and exercise **conservative decision making**.
- Correction of significant RCS T_{AVG} deviations should only be via secondary plant control manipulations and <u>not</u> primary plant control manipulations (i.e., do <u>not</u> withdraw control rods or dilute).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__1. Perform SHUTDOWN BRIEFING.	
__2. Verify rods in AUTO.	
__3. Reduce Turbine Load at the desired rate up to 5%/min (60 MWE/min).	
__4. Borate as necessary by initiating 13009, CVCS REACTOR MAKEUP CONTROL SYSTEM.	
__5. Initiate the Continuous Actions Page.	
__*6. **Check desired ramp rate - LESS THAN <u>OR</u> EQUAL TO 5%/MIN.**	*6. <u>IF</u> conditions warrant a turbine load rate greater than 5%/min, <u>THEN</u> perform the following: __a. Trip the reactor. __b. Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
*7. **Maintain Tavg within 6°F of Tref:**	
__a. Monitor Tavg/Tref deviation (UT-0495).	
__b. Verify rods inserting as required.	__b. Manual rod control should be used with insertions of up to 5 steps at a time.
__c. Energize Pressurizer back-up heaters as necessary.	

36. 054AK1.01 001/1/1/LOSS FW-MF LINE BRK/C/A - 4.1 / 4.3/NEW/HL-15R NRC/RO/TNT/DS

Given the following plant conditions:

- The unit is at 100% power.

- A Main Feedwater line break occurs at the piping connection to SG # 3.

RCS temperature will _____ prior to the reactor trip and SG # 3 pressure will _____.

	RCS temperature response	SG # 3 pressure response
A.	rise	stabilize when an SLI occurs
B.✓	rise	continue to depressurize after a FWI occurs
C.	lower	stabilize when an SLI occurs
D.	lower	continue to depressurize after a FWI occurs

HL-15R RO NRC Exam

K/A

054 **Loss of Main Feedwater (MFW)**

AK1.01 **Knowledge of the operational implications of the following concepts as they apply to the Loss of Main Feedwater (MFW):**

MFW line break depressurizes the S/G (similar to a steam line break).

K/A MATCH ANALYSIS

The question presents a plausible scenario where a Feedwater line break occurs at the connection to SG # 3. The student must be able to differentiate the RCS temperature response prior to a reactor trip and whether or not a FWI or SLI will isolate the break.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Classic FW line break response is for RCS temperature to be stable or rise in response. An SLI will not isolate the break and SG pressure would continue to depressurize following an SLI as inventory is lost via the feedwater break.

B. Correct. Classic FW line break response is for RCS temperature to be stable or rise in response. A FWI will not isolate the break and SG pressure would continue to depressurize following a FWI as inventory is lost via the feedwater break.

C. Incorrect. RCS temperature should stabilize or rise in response to a FW line break. RCS temperature lowering is the response to a SL break. An SLI will not isolate the break and SG pressure would continue to depressurize following an SLI as inventory is lost via the feedwater break.

D. Incorrect. RCS temperature should stabilize or rise in response to a FW line break. RCS temperature lowering is the response to a SL break. A FWI will not isolate the break and SG pressure would continue to depressurize following a FWI as inventory is lost via the feedwater break.

REFERENCES

V-LO-HO-37121-001, Faulted Steam Generator Isolation, page # 8 (included).

VEGP learning objectives:

LO-LP-317121-05, Describe the plant response to the following conditions.

a. Steam line break versus feed line break.

d. Feed break inside last check valve versus feed break outside last check valve.

Feedline Breaks

For an intermediate feedline break in which the control systems are incapable of compensating for the loss of flow, the secondary side would experience a slowly decreasing steam generator water level in at least one steam generator. A slowly increasing primary average temperature prior to reactor trip may occur due to the loss of main feedwater and degraded steam generator heat transfer. The transient is eventually terminated by manual reactor trip or when the low low level trip setpoint is reached in any one steam generator. This results in a reactor trip and auxiliary feedwater initiation. A subsequent turbine trip occurs due to reactor trip.

If the break occurs downstream of the main feedline check valves, all steam generators continue to experience a reverse blow down through the steam generator associated with the faulted loop until a low steamline pressure setpoint is attained resulting in a safety injection and steamline and feedline isolation. The faulted steam generator will then blow down until atmospheric pressure is reached.

If the break occurs upstream of the feedline check valves, the feedwater spillage is terminated and the auxiliary feedwater system is sufficient to mitigate the consequences of the resultant loss of normal feedwater transient.

The system parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include decreasing water level in at least one steam generator and slowly rising primary system average temperature prior to reactor trip.

For either of the above transients, if the break occurs inside containment, an increasing containment temperature and/or pressure indication could be observed. If the break occurs outside containment, audible or visual indications may assist the operator in diagnosing the transient.

Large Secondary Break

The least likely and most severe of the postulated loss of secondary coolant events is the double ended break.

Main Steamline Break

For the double ended main steamline break, an immediate decrease in pressure in at least one steamline occurs depending upon the location of the break. The low steamline pressure setpoint is reached which

37. 055EG2.4.02 001/1/1/LOSS ALL AC-EOP ENTR/C/A - 4.5 / 4.6/NEW/HL-15R NRC/RO/DS/TNT

Which one of the following choices list conditions which require entry into EOP 19100-C, "ECA-0.0 Loss of All AC Power" with Unit 1 at 100% power?

A.✓ 2 of the 3 white potential lights for both 1AA02 and 1BA03 extinguish.

1AA02 and 1BA03 breaker position indication lights remain lit.

B. 1 of the 3 white potential lights for both 1AA02 and 1BA03 extinguish.

1AA02 and 1BA03 breaker position indication lights extinguish.

C. 2 of the 3 white potential lights for both 1NA01, 1NA04 and 1NA05 extinguish.

1NA01, 1NA04 and 1NA05 breaker position indication lights remain lit.

D. 1 of the 3 white potential lights for 1NA01, 1NA04 and 1NA05 extinguish.

1NA01, 1NA04 and 1NA05 breaker position indication lights extinguish.

<u>**K/A**</u>

055 Station Blackout

EG2.4.02 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions:

<u>**K/A MATCH ANALYSIS**</u>

The question requires the student to interpret the electrical board indications to determine when a station blackout has occurred and entry into ECA-0.0 is required.

<u>**ANSWER / DISTRACTOR ANALYSIS**</u>

A. Correct. 2 of the white lights are fed by the 4160V AC busses through stepdown transformers. The 3rd light indicates DC control power for the switchgear. DC control power also provides breaker position indications. For the indications given, this would be a loss of AC power to both AC emergency busses meeting the ECA-0.0 symptoms.

B. Incorrect. The data given for this choice indicates a loss of DC control power to the emergency busses. This would <u>**not**</u> require entry into ECA-0.0 since all breakers would remain in their previous positions.

C. Incorrect. This choice provides indications for a loss of AC power to the non-1E 4160V AC busses. This does **not** meet ECA-0.0 entry conditions.

D. Incorrect. This choice provides indications of a loss of DC control power to the non-1E 4160V AC switchgear. This does **not** meet ECA-0.0 entry conditions.

REFERENCES

19100-C, "ECA-0.0 Loss of All AC Power" page 1

18034-1, "Loss of Class 1E 125V DC Power" page 13

LO-LP-60329, Loss of Class 1e 125V DC page # 4

One Line 1X3D-AA-D03A, 1BA03 light indications.

VEGP learning objectives:

V-LO-LP-37031-01:

Define "loss of all AC power" condition. Explain its immediate implications for operation of plant equipment.

V-LO-PP-01101-05:

Describe how a failure of DC control power affects the electrical distribution system and its components.

EMERGENCY OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure provides actions to respond to a loss of all AC power. (Applicable in Modes 1, 2, 3, 4)

SYMPTOMS/ENTRY CONDITIONS

The symptoms are:

- Both emergency AC buses are de-energized.

The entry conditions are:

- 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION

MAJOR ACTIONS

- ♦ Check Plant Conditions
- ♦ Restore AC Power
- ♦ Maintain Plant Conditions for Optimal Recovery
- ♦ Evaluate Energized AC Emergency Bus
- ♦ Select Recovery Guideline After AC Power Restoration

LOSS OF 125V DC BUS 1AD1

EQUIPMENT RESPONSE DUE TO LOSS OF TRAIN A 125V DC POWER

NOTE

Feeder Breakers must be locally controlled in the event the transfer to an alternate power supply is required. IF DG1A is not running, it may not be selected as an alternate power source.

- Main Feedwater Isolation, Bypass Feedwater Isolation, and Bypass Feedwater Regulation Valves close resulting in Feedwater Isolation.
- Main Steam Isolation and Bypass Steam Isolation Train A Valves close resulting in steamline isolation.
- Above P10, Reactor and Turbine trip occurs from loss of main feedwater.
- Below P10, Reactor trip occurs from Intermediate Range Instrumentation.
- Control Power is lost to 1AA02, 1AB04, 1AB05, and 1AB15 SWGR Breakers.
- DG1A control power to Generator Control Panel PDG1 and Engine Control Panel PDG2 is lost rendering the DG inoperable; if running, it will fail as is with a loss of electrical protective trips, frequency, and voltage control. Due to loss of power to the Low Speed Relay, the generator space, Engine Lube Oil and Jacket Water Heaters and Lube Oil and Jacket Water Keep-Warm Pumps will come on.
- Loss of Train A DG AUTO sequencer reset.
- Power to Inverters 1AD1I1 and 1AD1I11 is lost causing 120V AC Vital Busses 1AY1A and 1AY2A to de-energize.
- Instrument Air Containment Isolation Valve 1-HV-9378 closes resulting in loss of instrument air inside Containment.
- Power To Isolation Panel 1ACQIP1 is lost rendering the annunciators in Train A inoperable.
- Pressurizer PORV 1-PV-455A fails closed.
- TDAFW Steam Supply 1-HV-3019 fails as is.

I. INTRODUCTION

A. This procedure provides the actions to be followed in the event that power is lost to one of the 125V DC vital busses

B. Present lesson objectives

II. PRESENTATION

A. Present this lesson using the latest revision of AOP 18034

B. Procedure consists of four subsections: — **Objective 4**

1. Loss of 125V DC Bus 1AD1

a. Symptoms

1) Loss of 1AY1A and 1AY2A will be obvious to the operator due to all Channel I bistable lights lite up and sequencer trouble.

2) Loss of control power lights on 1AA02, 1AB04, 1AB05, and 1AB15.

3) Train A SLI

4) Train A FWI

5) Loss of voltage on 1AD1

2. Loss of 125V DC Bus 1BD1

a. Symptoms

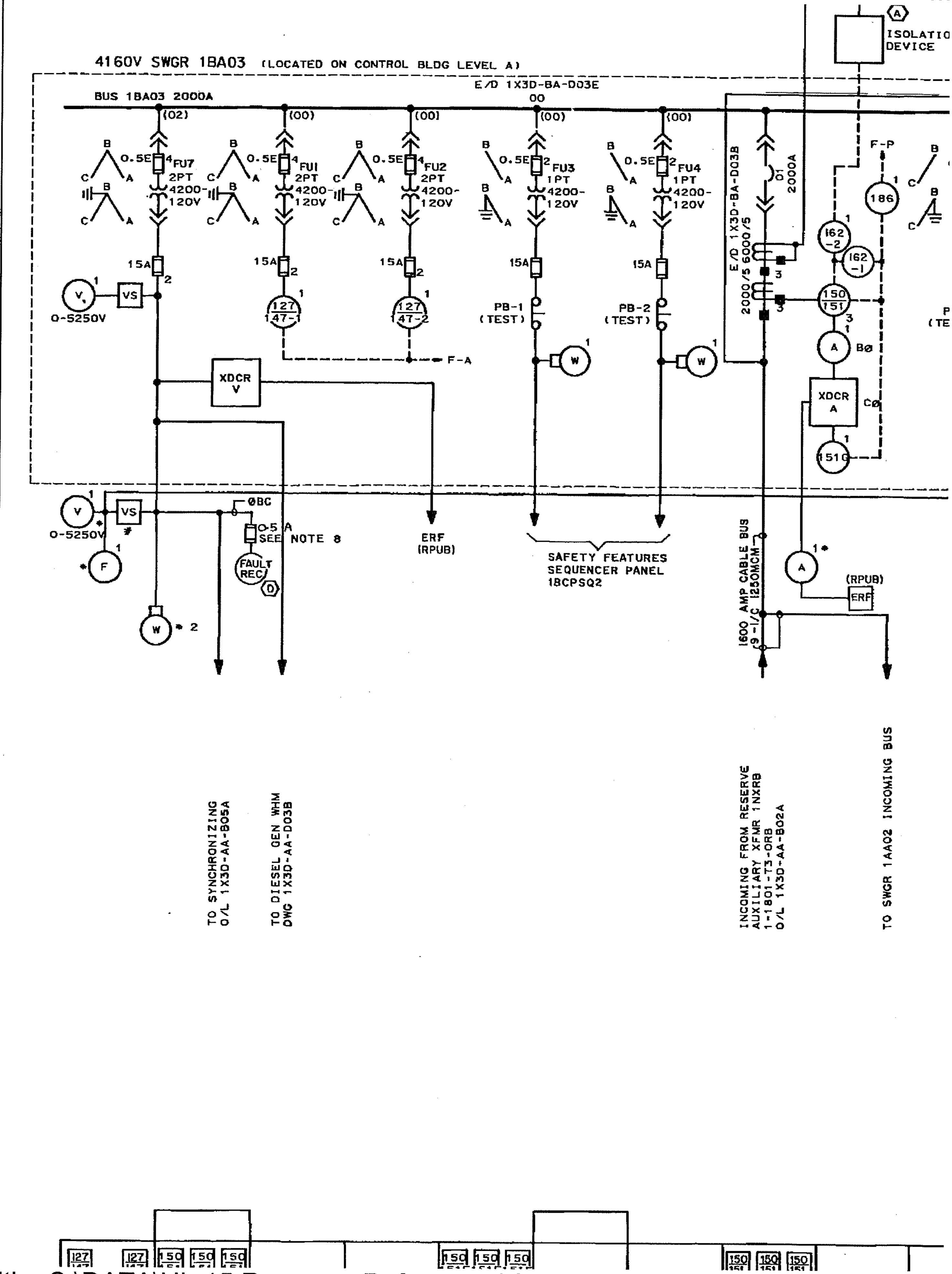
1) Loss of 1BY1B and 1BY2B causing all Channel II bistable lights to lite and sequencer trouble alarm.

2) Loss of control power lights on 1BA03, 1BB06, 1BB07, and 1BB16.

3) Train B SLI

4) Train B FWI

5) Loss of Voltage on IBD1



4160V SWGR 1BA03 (LOCATED ON CONTROL BLDG LEVEL A)
ISOLATIO DEVICE
BUS 1BA03 2000A
E/D 1X3D-BA-D03E
00
(02)
(00)
0.5E
FU7
FU1
FU2
FU3
FU4
2PT
1PT
4200-120V
15A
0-5250V
VS
127/147-1
127/147-2
F-A
F-P
PB-1 (TEST)
PB-2 (TEST)
XDCR V
XDCR A
E/D 1X3D-BA-D03B
2000/5 6000/5
01 2000A
186
162-2
162-1
150/151
151G
ØBC
0.5A SEE NOTE 8
FAULT REC
ERF (RPUB)
SAFETY FEATURES SEQUENCER PANEL 1BCPSQ2
1600 AMP CABLE BUS (9-1/C 1250MCM)
(RPUB)
ERF
TO SYNCHRONIZING O/L 1X3D-AA-B05A
TO DIESEL GEN WHM DWG 1X3D-AA-D03B
INCOMING FROM RESERVE AUXILIARY XFMR 1NXRB 1-1801-T3-0RB O/L 1X3D-AA-B02A
TO SWGR 1AA02 INCOMING BUS

38. 055K3.01 001/2/2/CARS-MAIN CONDENSER/C/A - 2.5 / 2.5/NEW/HL-15R NRC/RO/TNT/DS

Initial conditions:

- The unit is at full power

- The time is 1200.

Current conditions:

- The time is 2400.

- The UO notes Circulating Water temperature has lowered by 6^{o}F since 1200.

Which **ONE** of the following is **CORRECT** regarding the effect of lowering Circulating Water temperature on Main Condenser pressure and Main Turbine MW output?

	Main Condenser Pressure (psia)	Main Turbine MW Output
A.	Rises	Rises
B.✓	Lowers	Rises
C.	Rises	Lowers
D.	Lowers	Lowers

HL-15R RO NRC Exam

K/A

055 **Condenser Air Removal System (CARS):**

K3.01 **Knowledge of the effect that a loss or malfunction of the CARS will have on the following:**

Main Condenser

K/A MATCH ANALYSIS

The question presents a plausible scenario where Circulating Water pump temperature lowers from day shift to night shift. The student must determine the effect on Main Condenser Vacuum in psia and Main Turbine MW output.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Colder CW temperature would cause vacuum to get better (psia lowers) resulting in more MW output. Plausible the student could invert either parameter.

B. Correct. Colder CW temperature would cause vacuum to get better (psia lowers) resulting in more MW output.

C. Incorrect. Colder CW temperature would cause vacuum to get better (psia lowers) resulting in more MW output. Plausible the student could invert either parameter.

D. Incorrect. Colder CW temperature would cause vacuum to get better (psia lowers) resulting in more MW output. Plausible the student could invert either parameter.

REFERENCES

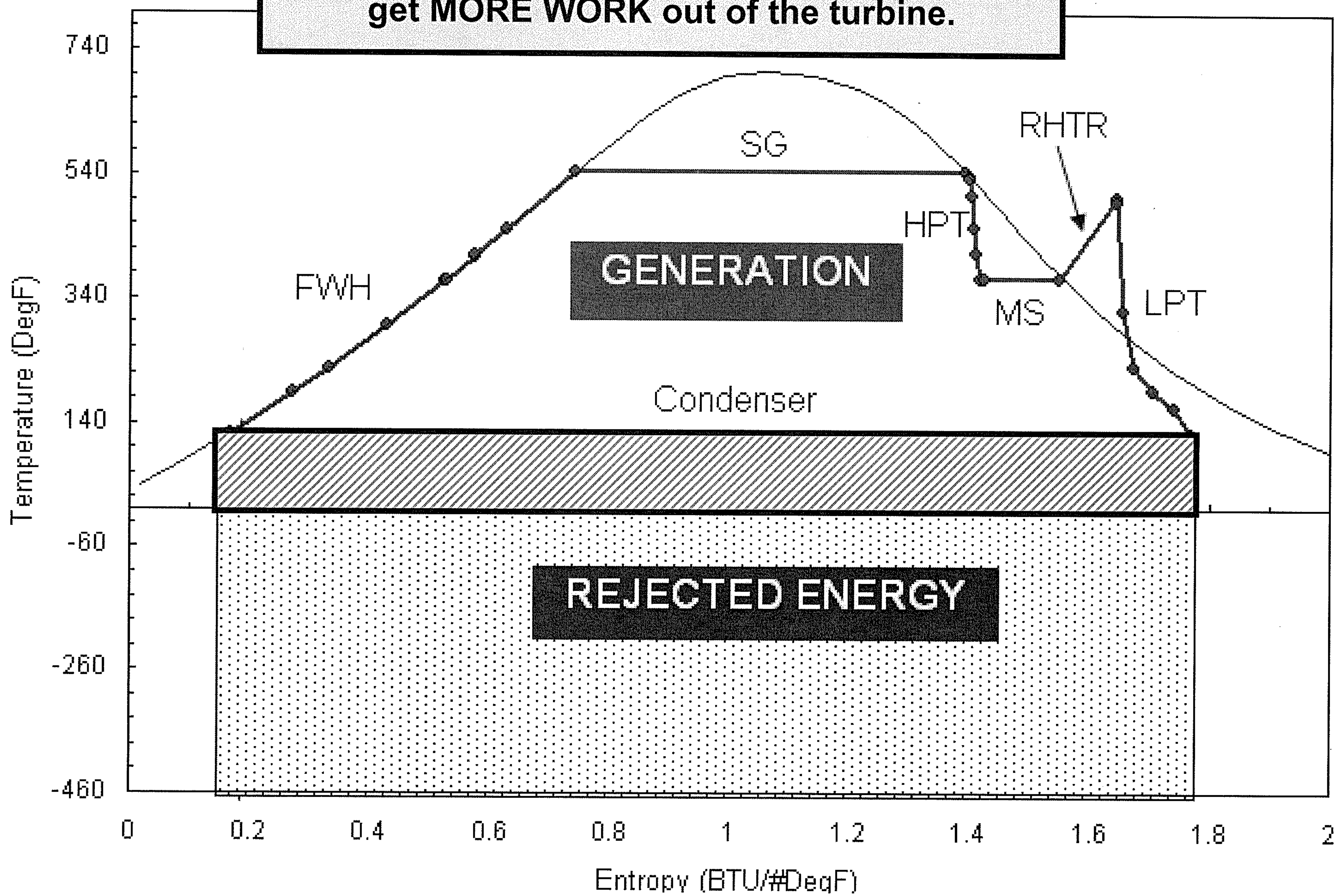
V-LO-PP-18101, Condensate and Feedwater, slide # 12

VEGP learning objectives:

LO-LP-18101-02, Discuss how the following will impact Main Condenser vacuum.

a. Circulating Water pump temperature.

Impact on plant operation
With lower Circulating water temperature;
Main Condenser vacuum increases and we get MORE WORK out of the turbine.



39. 056AK3.01 002/1/1/LOSS OFFSITE PWR-SEQ/C/A - 3.5/3.9/NEW/HL-15R NRC/RO/TNT/DS

Initial conditions:

- ACCW pump # 1 running

A plant event results in the both ESF Sequencers running, final plant conditions are:

- Both ACCW pumps running.
- The last load starts at 50.5 seconds.

Which ONE of the following is the CORRECT initiating event?

A. SI

B.✓ U/V

C. SI followed by a U/V

D. U/V concurrent with an SI

K/A

056 Loss of Offsite Power

AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power:

Order and time to intiation of power for the load sequencer.

K/A MATCH ANALYSIS

The question requires the student to correctly identify from given plant conditions the type of ESF sequence which initiated.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. An SI only sequence does not start an ACCW pump but the pump that is running would remain running. Two ACCW pumps running would rule out the SI sequence. In addition, the SI sequence completes in 30.5 seconds where a UV sequence runs for 50.5 seconds to allow the Containment Cooler starts to be staggered.

B. Correct. U/V sequence runs in 50.5 seconds and a single train UV or dual train UV will end up with both ACCW pumps running.

C. Incorrect. An SI sequence would not start the second ACCW pump, when the subsequent UV occurs, following load shed, the SI sequence prevails and would block start of either or both ACCW pumps whether, no ACCW pumps would be running. The SI sequence would also be the predominant sequence and takes 30.5 seconds to complete.

D. Incorrect. A simultaneous SI and UV sequence occurring would result in the SI sequence running as it is the predominant sequence. no ACCW pumps would be running. The SI sequence also takes 30.5 seconds to complete.

REFERENCES

V-LO-TX-28201, "Sequencer" pages 13, 14, 16, 17, and 18.

V-LO-PP-04101, ACCW slides 15, 17, and 19.

VEGP learning objectives:

LO-PP-28201-03:

Describe sequencer operation, including load shedding, load sequencing, and diesel generator operation under the following conditions:

a. Undervoltage (UV)
b. Safety Injection
c. UV followed by SI
d. SI followed by UV

V-LO-TX-28201

The U/V Schemes are as follows:

First level voltage- (INSTANTANEOUS Trip) <71.5 %(2975 VAC) for >0.8 sec.
Coincidence is 2/4.

Second level voltage- DEGRADED(Trip) <90 % (3746 VAC) for >20 sec.
Coincidence is 2/4.

Third level voltage- (Alarm only) <93.1 % (3873 VAC) for >10 sec.
Coincidence is 2/4.

The operation of the Vogtle Safety Features Sequencer System is automatic. Upon receiving a bus undervoltage (U/V) signal, the sequencer will automatically shed loads from the power bus, provide a start signal to the diesel generator, and, when the diesel generator is on line, sequentially return selected loads to the bus.

On an undervoltage actuation (First or second Level) the following actions are performed by the sequencer: (There is figure 1, Sequencer Manual Test Panel at the end of this section that can be used with the discussions of Sequencer operation)

Time=0 sec

Emergency start signal sent to Diesel Generator.

Load shed occurs:

1 sec trip signal sent to the Normal and Emergency feeder breakers to the 4160 VAC bus.

1 sec trip signal sent to pump breakers to the 4160 VAC bus.

1 sec trip signal sent to 480 VAC secondary side (low side) Switchgear breakers for all 1E and Non 1E loads.

Auto/Manual Block circuit is enabled.

U/V non sequenced load relays are energized to actuate those loads. This same signal generates the UNDERVOLTAGE light on the sequencer panel.

Signal sent to Loss of Power (LOP) monitor circuit.

Reset and stop signal sent to ATI subsystem. ATI step counter is reset to 00 and stopped.

Reset and inhibit signal sent to Manual test circuits.

Lights **U/V SIGNAL** (red), **U/V RELAYS ACTUATED** (red), **BLOCK AUTO/MNL SIG** (red), **SEQ LOGIC FAILURE** (amber), **UNDERVOLTAGE** (amber) generated on Sequencer panel.

Sequencer Trouble and **AAO2 (BA03)SWGR Trouble** alarms received in the control room on the QEAB.

Time=0.5 sec

Sends DG Breaker Auto Closure permissive to DG output breaker closure circuit if not blocked by LOP monitor circuit.

The U/V Schemes are as follows:

First level voltage- (INSTANTANEOUS Trip) <71.5 %(2975 VAC) for >0.8 sec.
Coincidence is 2/4.

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Reset and inhibit signal sent to Manual test circuits.

Lights **U/V SIGNAL** (red), **U/V RELAYS ACTUATED** (red), **BLOCK AUTO/MNL SIG** (red), **SEQ LOGIC FAILURE** (amber), **UNDERVOLTAGE** (amber) generated on Sequencer panel.

Sequencer Trouble and **AA02 (BA03) SWGR Trouble** alarms received in the control room on the QEAB.

Time=0.5 sec

Sends DG Breaker Auto Closure permissive to DG output breaker closure circuit if not blocked by LOP monitor circuit.

Time=6.0 to 11.5 secs	When DG ready to load, DG output breaker closes. **D-G BRKR CLOSED** (red) light generated on Sequencer panel. Sequencer elapsed time display begins running.
Brkr CL +0.5 to 30.5 secs	**SEQ STEPS INDICATION** (red) for steps 1A-9A and 1C-9C will begin flashing in the intervals specified in the list below as the components are sequence on.

UV LOAD SEQUENCE

Train A only (Train B loads are similar)

TIME	LOAD
0.5 secs	CCP A, 480 VAC Secondary side feeder breakers
5.5 sec	NONE
10.5 sec	NB01 (Stub bus secondary side feeder breaker closes)
15.5 sec	ACCW Pump 1
20.5 sec	CCW Pumps 1 and 3 MDAFW Pump A
25.5 sec	NSCW Pumps 1 and 3 CCW Pump 5 (if CCW Pumps 1 or 3 breaker did not close)
30.5 sec	CTMT Cooling Units 5 and 6 (Fast Speed) NSCW pump 5 (if NSCW Pumps 1 or 3 breaker did not close) CTMT Cooling Units 1 and 2 start contact closed (Fast Speed)

+20 sec

50.5 seconds

For all loads to start.

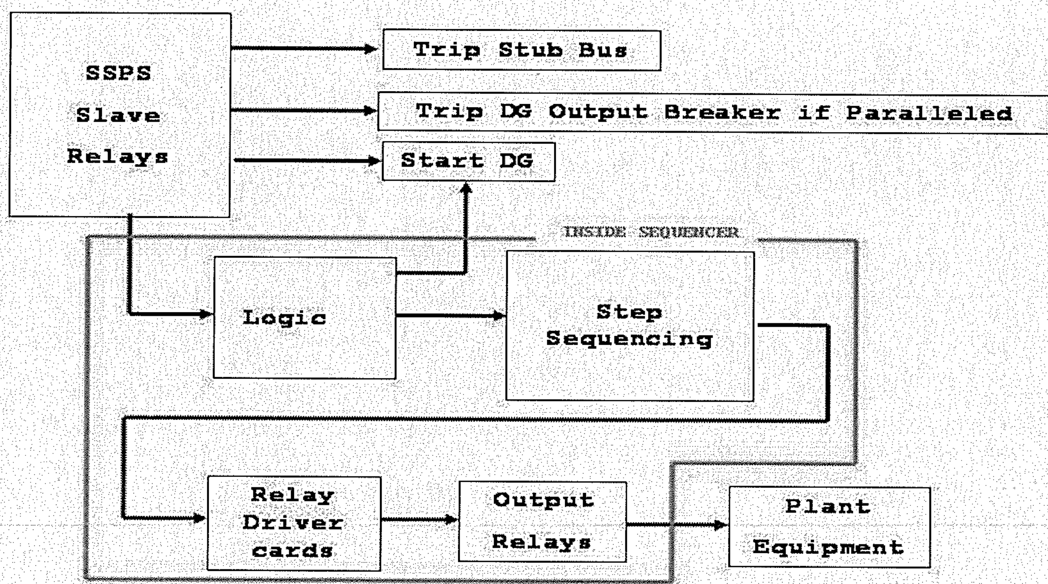
CTMT cooling units must not all be started at the same time to prevent bus voltage transients. Analysis has shown that if all four were allowed to be simultaneously, DG voltage could drop below 80 percent. The sequencer provides all CTMT cooling units with a start signal at the 30.5 second step. Coolers 1, 2, 7, and 8 start at 50.5 seconds due to an additional time delay of 20 seconds by an agastat time delay relay in the auto-start circuit. This is for a UV condition only.

Time=32 secs	**SEQ STEPS INDICATION** (red) flashing lights extinguish. **SAF EQPT FAIL TO START** (amber) light to indicate that Cnmt Coolers 1 and 2 have not started. Audible alarm is sounded on the sequencer panel. Alarm generated on QEAB. ATI stop removed and ATI restarted
Time=36 secs	**BLOCK AUTO/MNL SIG** (red) light extinguishes.
Time=50.5 sec	CTMT Cooling Units 1 and 2 start (Fast Speed) *This is not a sequencer function but internal to the start logic of the cooler high speed motors* **SAF EQPT FAIL TO START** (amber) light extinguishes. QEAB alarm clears.

Sequencer Operation in Response to an SI Condition.

The Sequencer is also awaiting an SI signal. The following diagram shows how the SI signal is processed from the SSPS to the plant loads.

SI Signal Processing Block Diagram



On receipt of an SI signal, the following actions are performed by the sequencer:

Time=0 sec	Emergency start signal sent to Diesel Generator.
	Auto/Manual Block circuit is enabled.
	ATI step counter is reset to 00 and stopped.
	Reset and inhibit signal sent to Manual test circuits.
	Non sequenced SI maintained relays actuated.
	Non sequenced SI Momentary relays actuated for 1 sec
	Lights **SI SIGNAL** (red), **SI MAIN RELAYS ACTUATED** (amber), **BLOCK AUTO/MNL SIG** (red), **SI MOM RELAYS ACTUATED** (amber) generated on Sequencer panel.
Time~0.1 secs	SI sequence timing starts. Sequencer elapsed time display begins running. This is set at 90 millisecs.
Time=1.0 sec	Light **SI MOM RELAYS ACTUATED** (amber) extinguishes.
Time=0.5 to 30.5 secs	Lights **SEQ STEPS INDICATION** (red) will begin flashing in the intervals specified in the list below. Steps 1-9, 1A-9A, 1B-9B, and 1C-9C steps are sequenced on.

SI LOAD SEQUENCE

Train A only (Train B loads are similar)

TIME	LOAD
0.5 sec	CCP A
5.5 sec	SIP A
10.5 sec	RHR pump A
15.5 sec	Containment Spray Pump A (W/ CSAS)
20.5 sec	CCW pumps 1 and 3 MDAFW Pump A
25.5 sec	NSCW pumps 1 and 3 CCW 5 (if CCW Pumps 1 or 3 breaker did not close)
30.5 sec	CTMT Cooling Units 1, 2, 5 and 6 (Slow Speed) NSCW 5 (if NSCW Pumps 1 or 3 breaker did not close)

ACCW pumps NOT ON SI sequence.

Note that the Containment spray pump sequence is unique. At 15.5 secs the sequencer will send a 1 sec start signal. IF the CNMT Spray Actuation slave relay is energized the pump will start. This one sec start signal is sent on all the following steps to start CNMT spray if the actuation signal occurs. Therefore it is possible for the CNMT Spray pump to start at 15.5, 20.5, 25.5, or 30.5 secs depending on when a CNMT Spray actuation signal is received.

Time=6.0 to 11.5 secs	When DG ready to load, **D-G READY FOR LOADING** (red) light generated on Sequencer panel.
Time=32 secs	**SEQ STEPS INDICATION** (red) for 1B-9B and 1C-9C steps flashing lights extinguish. The 1-9 and 1A-9A steps will continue to flash until the SI signal is reset.
Time=36 secs	**BLOCK AUTO/MNL SIG** (red) light extinguishes.

Note that if CCW pumps 5 or 6 or NSCW pumps 5 or 6 are in service before an SI signal occurs, then all three pumps will be running in those trains after the SI sequence is complete.

ATI must be manually reset after the SI signal is reset.

Sequencer Operation in Response to an SI Condition with a U/V Condition.

There are five separate combinations to consider on operation with an SI and an U/V condition. They are:

SI signal and U/V simultaneously
SI signal following U/V (before sequencing is complete)
SI signal following U/V (after sequencing is complete)
U/V following SI signal (before SI is reset)
U/V following SI signal (after SI is reset)

The general rule of operation in these conditions is the sequencer operation will be a combination of the U/V and SI sequences. By understanding priority system of the sequencer, each of the above combinations can be evaluated. The U/V sequence will predominate until the ESF bus is energized. With the bus energized, the SI signal will predominate.

Simultaneous SI/UV

If a SI and U/V signal are received simultaneously, the SI sequence will be initiated after the completion of the load shed and subsequent re-energization of the 1E bus by the EDG.

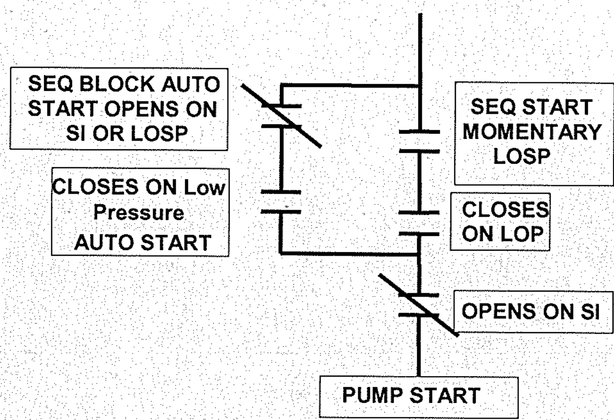
SI followed by U/V

If a U/V signal is received after SI actuation, the sequencer will initiate a load shed and generate the permissive for the EDG to re-energize the 1E bus (the EDG would have previously been started by the initiating SI signal). After the 1E bus is re-energized, the loading sequence will be a function of the status of the SI signal. If the SI signal is still present, the SI sequence will be initiated at step 1.

If the SI signal is no longer present (i.e. SI has been reset) when the EDG re-energizes the 1E bus, then the U/V sequence will be initiated at step 1. After completion of the loading sequence, any SI loads required to be in service that were not started during the U/V sequence will have to be manually placed in service (i.e. SI pumps, RHR pumps, Containment Cooler Low Speed motors, ESF Chilled Water pumps and ESF Chillers).

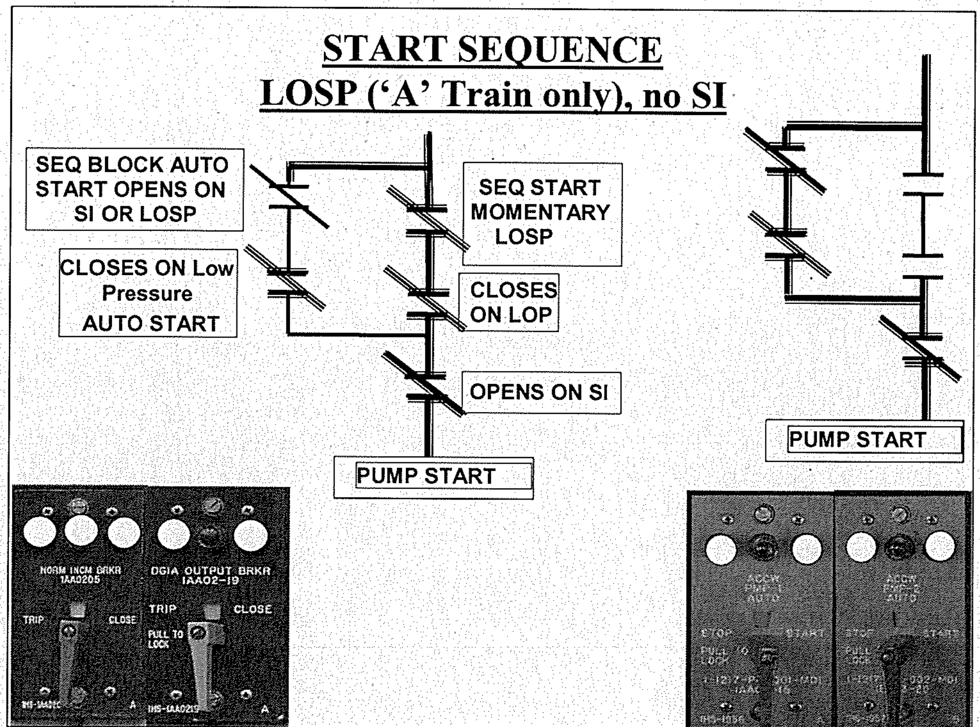
If an SI signal is received after the U/V sequence has been initiated, the sequencer will suspend the U/V sequence upon receipt of the SI signal and restart at step 1 of the SI sequence. If the SI signal is received after completion of the U/V sequence, then the sequencer will begin at step 1 of the SI sequence. For these conditions, since the 1E bus was energized at the time the SI signal was received, no additional load shed will occur. Any U/V loads started from the initiating U/V signal will remain in operation.

AUTO START LOGIC



SI blocks ACCW pump starts,

V-LO-PP-04101, ACCW



ACCW pump response to LOSP ('A' train ONLY), with no SI

Initial conditions: 1AA02 and 1BA03 energized from the RAT's, ACCW pump #1 in service with ACCW pump #2 in standby.

Click #1: shows circuit energized up to first contacts all the time.

Click #2: Loss of 1AA02, pump #1 trips (no amber light).

Click #3:

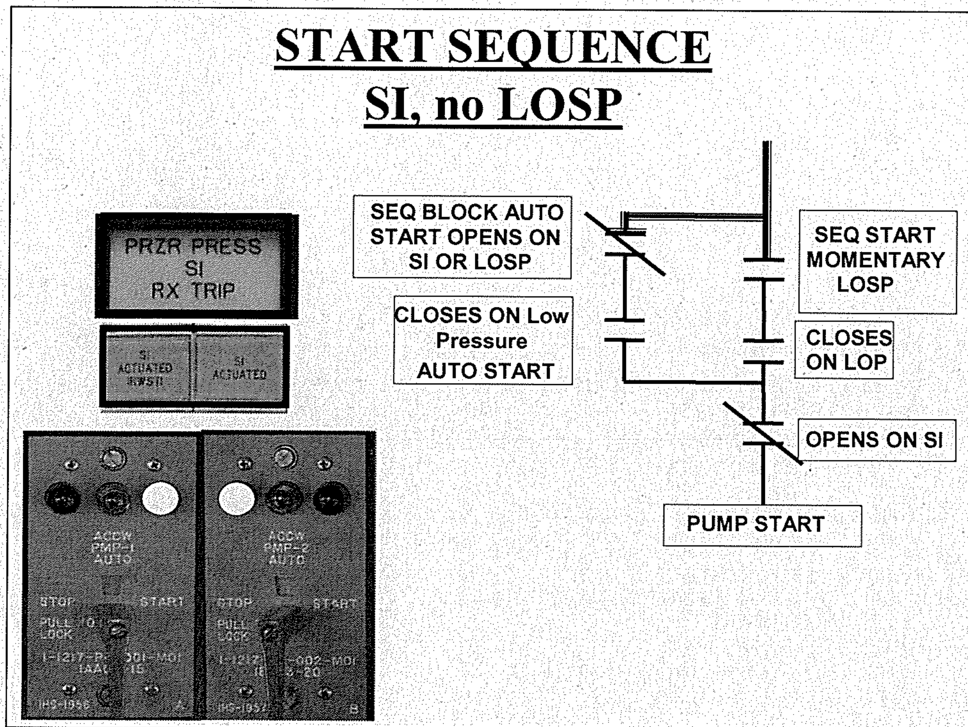
For 'A' Train : Contact for" LOSP" and contact for "Low Header Pressure" closes, contact for "Sequencer Block AUTO Start on SI or LOSP" opens.

For 'B' Train: "Sequencer BLOCK AUTO Start" is normally closed. Upon reaching the setpoint for low header pressure, the contact for low hdr pressure closes thus completing the circuit for a start of the standby pump.

Click #4: DG's start and re-energize 1AA02 and 1BA03. Sequencer closes in the last contact to make up the circuit for a pump start. 'A' pump now running from LOSP and 'B' pump running from low header pressure.

Dual or Single Train LOSP/UV results in both ACCW pumps running,

V-LO-PP-04101, ACCW



→ Prevent ACC pumps starting

ACCW pump response to SI with no LOSP

Initial conditions: ACCW pump #1 in service with ACCW pump #2 in standby.

<u>Click #1</u>: shows circuit energized up to first contacts all the time.

<u>Click #2</u>: Low pressurizer pressure SI signal received

<u>Click #3</u>: Contact for “Sequencer Block AUTO Start on SI or LOSP” opens, SI contact opens, contact for “Sequencer Start Momentary SI or LOSP” closes. Pump #1 remains running, Pump #2 remains in standby. Auto start signals for both pumps BLOCKED by SI signal.

1 pump remains running

V-LO-PP-04101, ACCW

40. 057AA2.05 001/1/1/LOSS VIT AC-SG METER/C/A -3.5/3.8/NEW/HL-15R NRC/RO/DS / TNT

The following indications occur with the unit at full power:

All four SGs channel 1 NR levels go off-scale low
All four SGs channel 1 pressures go off-scale low

This is a loss of __________ and the correct action to take is to...

A.✓ 1AY1A

place all 4 MFRVs and MFPT SPEED CONTROL MASTER in manual and match channel 2 feed flows to channel 2 steam flows while maintaining SG NR levels.

B. 1AY2A

place all 4 MFRVs and MFPT SPEED CONTROL MASTER in manual and match channel 2 feed flows to channel 2 steam flows while maintaining SG NR levels.

C. 1AY1A

verify reactor trip and initiate 19000-C, "E-0 Reactor Trip or Safety Injection".

D. 1AY2A

verify reactor trip and initiate 19000-C, "E-0 Reactor Trip or Safety Injection".

K/A

057 Loss of Vital AC Inst. Bus

AA2.05 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:

S/G pressure and level meters.

K/A MATCH ANALYSIS

The question presents the indications of a loss of vital AC instrument bus 1AY1A for the SG levels and pressure instruments. The student is required to correctly diagnose the failure and take the proper actions to stabilize the plant and prevent a reactor trip matching the K/A topic.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. A loss of 1AY1A will result in all channel 1 SG level and pressure instruments failing low, refer to AOP 18032-1 Attachment A Table for the I&C loads for 1AY1A. The actions to stabilze the plant with power above P10 (10%) are to control SG NR levels between 60% and 70% with the MFRVs in manual and the MFPT SPEED CONTROL MASTER in manual. Since SG channel 1 pressures input into the density compensation circuit the channel 1 steam flow instruments will also be reading down scale requiring the UO to use the channel 2 steam flow indications.

B. Incorrect. A loss AC vital instrument bus 1AY2A will not affect any SG level or pressure instruments, so the there is no need to manually control SG NR levels. 1AY2A is a train A vital instrument bus with different actions contained in a different section (B) of the same abnormal operating procedure (18032-1) used to address the loss of 1AY1A.

C. Incorrect. The diagnosis for the vital AC instrument bus is correct. A loss of 1AY1A will result in all channel 1 SG level and pressure instruments failing low, refer to AOP 18032-1 Attachment A Table for the I&C loads for 1AY1A. The actions for this choice are incorrect since reactor power is above the P10 setpoint of 10%. The actions list for this choice are those specificed by the AOP for a loss of 1AY1A with power below P10 (10% reactor power).

D. Incorrect. A loss AC vital instrument bus 1AY2A will not affect any SG level or pressure instruments, so the there is no need to perfrom the actions specified for section A of AOP 18032-1.

REFERENCES

AOP 18032-1, "Loss of 120V AC Instrument Power"
Section A Loss of 1AY1A, pages 10 and 25
Section B Loss of 1AY2A, pages 28 and 30

VEGP learning objectives:

LO-LP-60324-01:

Given the appropriate plant drawings, logics, and/or procedures, describe how the plant will respond to a loss of the following 120VAC instrument panels:

a. 1AY1A
b. 1AY2A
c. 1BY1B
d. 1BY2B
e. 1CY1A
f. 1DY1B
g. 1NY1N
h. 1NY2N
i. 1NY3N
j. 1NY4N
k. 1NYC2

l. 1NYJ
m. 1NYR
n. 1NYS
o. 1NYRS
p. 1NY01

LO-LP-60324-02:

Given that a loss of 120VAC instrument power has occurred to any of the following panels, and given the appropriate plant procedures, describe the operator actions required and why these actions are taken.

a. 1AY1A
b. 1AY2A
c. 1BY1B
d. 1BY2B
e. 1CY1A
f. 1DY1B
g. 1NY1N
h. 1NY2N
i. 1NY3N
j. 1NY4N
k. 1NYC2
l. 1NYJ
m. 1NYR
n. 1NYS
o. 1NYRS
p. 1NY01

A. LOSS OF VITAL INSTRUMENT PANEL 1AY1A (CB-B52)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
__A1. Check reactor power - GREATER THAN P-10 SETPOINT.	A1. Perform the following: __a. Verify reactor trip. __b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
__A2. Verify ROD BANK SELECTOR SWITCH in manual.	
***A3. Control SG NR levels - BETWEEN 60% AND 70%:** __• MFRVs in manual. __• MFPT SPEED CONTROL MASTER in manual.	
SUBSEQUENT OPERATOR ACTIONS	
__A4. Initiate the Continuous Actions Page.	
***A5. Control charging to:** __• Maintain seal injection flow to all RCPs - 8 TO 13 GPM. __• IF letdown isolated, THEN adjust charging flow to approximately 10 gpm greater than total seal injection flow.	

ATTACHMENT A

Sheet 2 of 4

TABLE 1 – I&C LOADS - PANEL 1AY1A

INST. NO.	DESCRIPTION	ALTERNATE

STEAM GENERATOR INSTRUMENTATION

LR-501-P1	SG 1 WR Level Rec	
LI-501	SG 1 WR Level (I)	
LI-529	SG 2 NR Level (I)	LI-528 (III)
LI-539	SG 3 NR Level (I)	LI-538 (III)
LI-551	SG 1 NR Level (I)	LI-517 (IV)
LI-554	SG 4 NR Level (I)	LI-547 (IV)
PR-514	SG 1 & 2 Press Rec	
PI-514A	SG 1 Press (I)	PI-516A (IV)
PI-524A	SG 2 Press (I)	PI-526A (III)
PI-534A	SG 3 Press (I)	PI-536A (III)
PI-544A	SG 4 Press (I)	PI-546A (IV)

AUXILIARY FEEDWATER INSTRUMENTATION

LI-5111A	CST 1 Level (I)	LI-5101 (II)
LI-5116A	CST 2 Level (I)	LI-5104 (II)
FI-5150A	SG 4 AFW Flow	
FI-5152A	SG 1 AFW Flow	

MISCELLANEOUS SYSTEMS INSTRUMENTATION

LR-990-P1	RWST Level Rec	LR-990-P2
LI-990	RWST Level (I)	LI-991A (II)
LI-102A	Boric Acid Tk Level (I)	LI-104A (IV)
PI-937	CNMT NR Press (I)	PI-936 (II)
PI-505	Turbine Impulse Press (I)	PI-506 (II)
PI-1636	NSCW Train A Supply Press	PI-1637
FT-22425	Train A Essential Chiller Evaporator Flow	
PT-6161	Turbine ETS pressure	
PT-10942	Containment Wide Range Pressure (Computer input)	
QT-2791	Diesel Generator Train A Power Output (Computer input)	
FI-12191	CB Control Room Exhaust Flow	
PDIS-12136	CB Control Room HEPA Filter Differential Pressure	
FI-12542	Aux Build Piping Penetration Room Exhaust Flow	
AFI-12551	FHB Post Accident Exhaust Flow	
PDI-2550	Aux Bldg Piping Penetration Room Differential Pressure	
APDIS-2524	FHB Post Accident HEPA Filter Differential Pressure	
PDIS-2546	Aux Bldg Piping Penetration Room HEPA Filter Differential Pressure	
APDT-12567	FHB Differential Pressure	

B. LOSS OF VITAL INSTRUMENT PANEL 1AY2A (AB-118)

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

NOTES

- Train A ESF sequencer will not operate following loss of Panel 1AY2A.
- Loss of power to Panel 1AY2A will result in a Containment Ventilation Isolation.
- SG-1 and SG-4 ARVs will NOT operate from QMCB following loss of 1AY2A.

B1. Notify Chemistry that the following radiation monitors will be out of service and will need to be reset when power is restored:

- __ 1RE-0002 (CVI)
- __ 1RE-0005
- __ 1RE-2532A (FHBI)
- __ 1RE-2532B (FHBI)
- __ 1RE-12116 (CRI)
- __ 1RE-13119
- __ 1RE-13120

__B2. Dispatch an operator to restore Panel 1AY2A by initiating 13431, 120V AC 1E VITAL INSTRUMENT DISTRIBUTION SYSTEM.

__B3. Refer to ATTACHMENT B to determine affected instrumentation.

__B4. Refer to Technical Specifications and complete any applicable action statements.

ATTACHMENT B

Sheet 1 of 1

TABLE 1 PANEL 1AY2A LOAD LIST

BREAKER	LOAD
03	SAFETY RELATED DISPLAY CONSOLE DRMS
04	DATA MODULE 1RX0005 - CNMT AREA MONITOR
05	DATA MODULE 1RX0002 - CNMT AREA MONITOR
06	DATA MODULE ARX-2532 - FUEL HANDLING BLDG HVAC MONITOR
07	DATA MODULE 1RX-12116 - CR AIR INTAKE MONITOR
08	SEQUENCER BOARD 1-1821-U3-001
09	BOP SAFETY ACTUATION CABINET 11CQESF
10	TRAIN A SYSTEM STATUS MONITORING PANEL
11	PREAMP 1RT-005
12	DATA MODULE 1RX-13119 - MAIN STEAMLINE MONITOR
13	DATA MODULE 1RX-13120 - MAIN STEAMLINE MONITOR
14	DISPLAY PROCESSING UNIT (DPU-A)
15	REMOTE PROCESSING UNIT A1, CHANNEL I
16	REMOTE PROCESSING UNIT A2, CHANNEL I
17	SERVO-AMP FOR ATM DUMP VALVE 1ATPY3000
18	SERVO-AMP FOR ATM DUMP VALVE 1ATPY3030
19	SPARE
20	SPARE
21	SPARE
22	SPARE
23	SPARE
24	SPARE

END OF ATTACHMENT B