1. 001AA2.01 001/BANK/WATTS BAR MAY 09/C/A 4.2/4.2/APE001AA2.01/N///TELL NRC

Unit 1 is operating at 85% power with the following conditions:

- STP-33.0B, Solid State Protection System Train B Operability Test, is in progress.
- The 'B' Reactor Trip Bypass Breaker has been racked in and closed.
- Control Bank D is at 203 steps.
- The Rod Control Bank Selector Switch is in AUTO.
- PS/446Z, FIRST STG IMPULSE PRESS SEL SWITCH, is in the Channel IV / PT447 position.

Subsequently, the following occurs:

- PT-447, TURB FIRST STG PRESS, fails HIGH.

Which one of the following completes the statements below?

The control rods will __(1)__.

If the reactor is manually tripped at this time, the 'B' Reactor Trip Bypass Breaker __(2)__ light will be LIT.

	(1)	(2)
A.	insert	RED
B.	insert	GREEN
C.✔	withdraw	GREEN
D.	withdraw	RED

PT-447 failing low will cause rods to insert continuously and failing high will cause rods to withdraw.

Per FSD-A181007, Figure 2 Sheet 2, a manual reactor trip will open the 'B' Reactor Trip Bypass Breaker.

Distracter Analysis

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant reverses the system response so that the rod control system would insert rods to suppress the indicated power rise due to the failure instead of trying to adjust actual reactor power and Tavg to match the failed indication.

Second part is incorrect (See C.2). Plausible if the applicant believes that during the performance of STP-33.0B, the 'B' Reactor Trip Bypass Breaker will not open since that train is being tested.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See C.2).

C. Correct. First part is correct. Impulse pressure, PT-477 is used to calculate Tref. Since PT-447 failed high, Tref fails to the 100% power Tref and the rod control system will step rods out in order to raise current Tavg to match Tref. Impulse pressure is also used to determine turbine power as compared to reactor power in the rod control circuitry. When PT-447 fails high, the rapid rate of change of impulse power as compared to reactor power will also cause rods to step out while impulse pressure is changing.

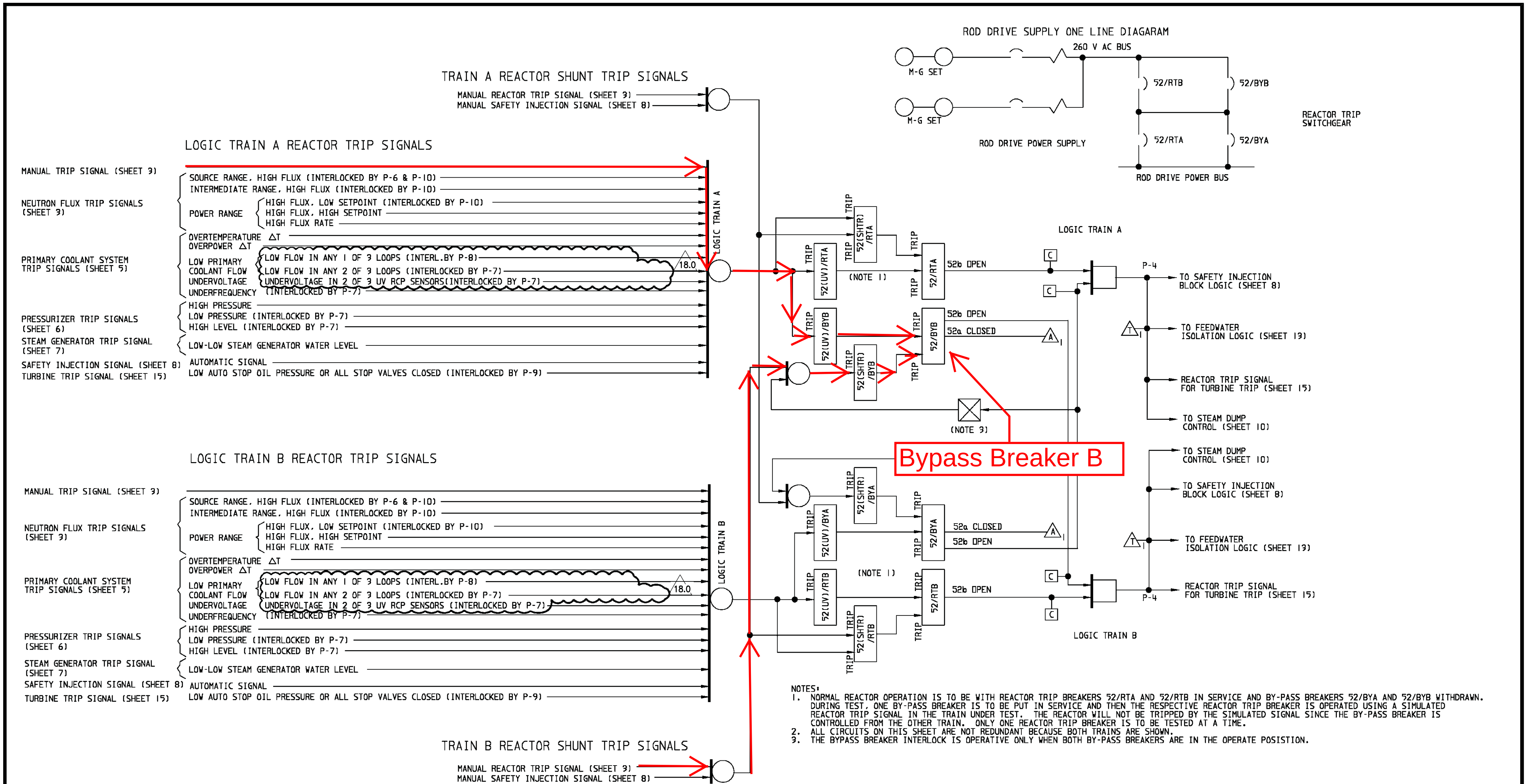
This scenario has been run on desktop simulator and the rods will step out (IC 058).

Second part is correct. FSD A18007, Figure 2 sheet 2, shows that a manual trip actuation will open the 'B' Reactor Trip Bypass Breaker.

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See A.2).

K/A: **001AA2.01**	Continuous Rod Withdrawal - Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal : Reactor tripped breaker indicator
Importance Rating:	4.2 4.2
Technical Reference:	FSD-A181007, Reactor Protection System, Ver 18 FNP-1-EEP-0.0, Reactor Trip or Safety Injection, Ver 44 FNP-1-AOP-100, Instrumentation Malfunction, Ver 12
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if entry into AOP-100, Instrument Malfunction is required. (OPS-52521Q02) ANALYZE plant conditions and DETERMINE the successful completion of any step in (1) EEP-0, Reactor Trip or Safety Injection [...]. (OPS-52530A07)
Question History:	WATTS BAR MAY 09
K/A match:	This question requires the applicant to **determine that a continuous rod withdrawal is occurring** due to the failure of PT-447. After the reactor is tripped, the applicant is then required to **interpret the reactor trip bypass breaker indication as to whether or not it is open.**
SRO justification:	N/A



NOTES:

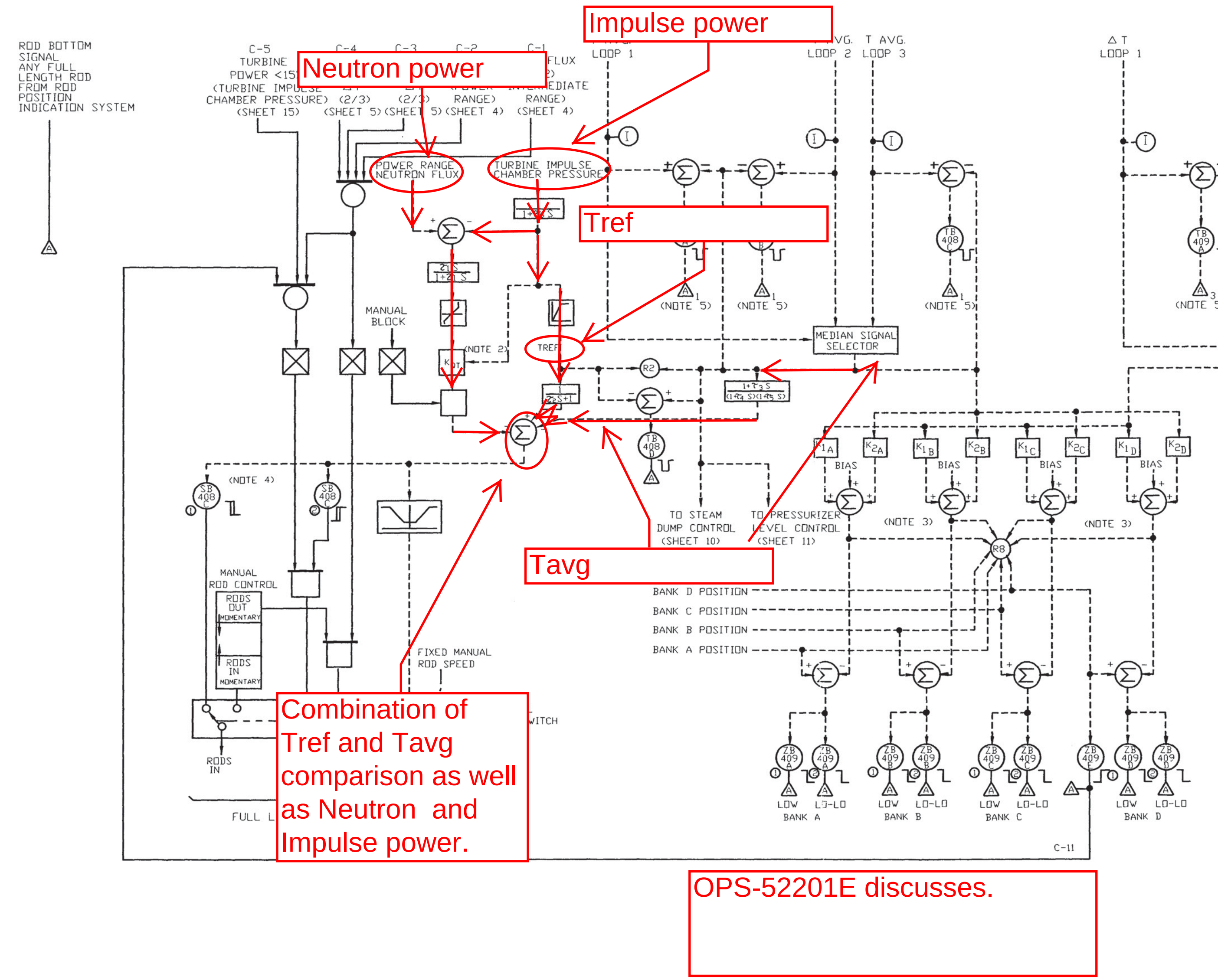
1. NORMAL REACTOR OPERATION IS TO BE WITH REACTOR TRIP BREAKERS 52/RTA AND 52/RTB IN SERVICE AND BY-PASS BREAKERS 52/BYA AND 52/BYB WITHDRAWN. DURING TEST, ONE BY-PASS BREAKER IS TO BE PUT IN SERVICE AND THEN THE RESPECTIVE REACTOR TRIP BREAKER IS OPERATED USING A SIMULATED REACTOR TRIP SIGNAL IN THE TRAIN UNDER TEST. THE REACTOR WILL NOT BE TRIPPED BY THE SIMULATED SIGNAL SINCE THE BY-PASS BREAKER IS CONTROLLED FROM THE OTHER TRAIN. ONLY ONE REACTOR TRIP BREAKER IS TO BE TESTED AT A TIME.
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT BECAUSE BOTH TRAINS ARE SHOWN.
3. THE BYPASS BREAKER INTERLOCK IS OPERATIVE ONLY WHEN BOTH BY-PASS BREAKERS ARE IN THE OPERATE POSISTION.

CAD FSD F-2 S2
ACAD2007 KB-01

REACTOR TRIP

FIGURE F-2 SHEET 2

VER. 18.0



NOTES:
1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. K_{OT} MAY VARY INVERSELY PROPORTIONAL TO LOAD WITH A FIXED LIMIT, OR MAY VARY IN TWO DISCRETE STEPS WITH BREAK POINTS AT 30-50% AND 60 TO 80% TURBINE LOAD.
3. THE SUMMER OUTPUTS HAVE FIXED MANUALLY ADJUSTABLE UPPER LIMITS.
4. THE ROD DIRECTION BISTABLES #SB408C ARE "ENERGIZE TO ACTUATE".
5. ALARM 1 AND ALARM 3 MUST HAVE REFLASH CAPABILITY.

ROD CONTROLS AND BLOCKS

FIGURE F-2 SHEET 9

RPSFSD09.DWG

Step	Action/Expected Response	Response Not Obtained

SECTION 1.3

TURBINE IMPULSE PRESSURE INSTRUMENTATION

NOTE: Steps 1 and 2 are immediate operator actions.

__ **1** **Check no load rejection in progress.**

1 Go to FNP-1-AOP-17.0, TURBINE LOAD REJECTION.

__ **2** **Check no unexpected ROD motion is occurring OR has occurred.**

2 Place Rods in MANUAL.

2.1 IF rods continue to move, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

NOTE: Section 1.3 Figure 1 and 2 are provided as reference material

---

__ **3** **IF the selected channel of Pimp has failed, THEN select the unaffected channel.**

FIRST STG IMPULSE
PRESS SEL SWITCH
PS/446Z

[ ] PT-446 (CH-III) selected

OR

[ ] PT-447 (CH-IV) selected

Check C5 permissive light on.

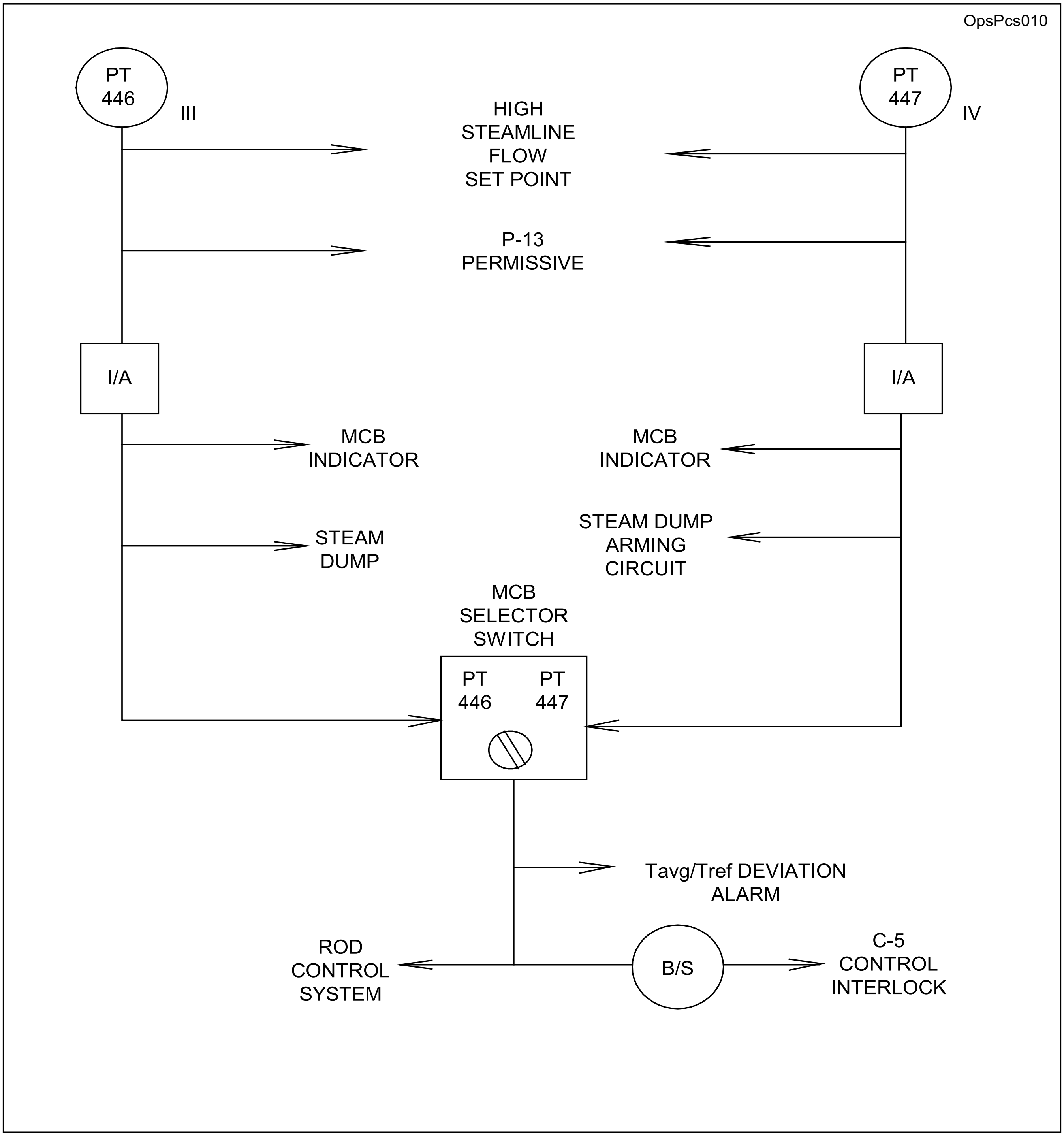
LOW TURB
IMPULSE PRESS
AUTO ROD WDRL

[ ] BLOCKED C-5- lit

__Page Completed

SECTION 1.3

Figure 1



TURBINE FIRST STAGE IMPULSE PRESSURE

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Steps 1 through 4 are IMMEDIATE ACTION steps.
- FOLDOUT PAGE should be monitored continuously.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Check reactor trip.** [] Reactor trip and bypass breakers - OPEN [] Nuclear power - FALLING [] IF DRPI available, THEN rod bottom lights - LIT	1 Perform the following. 1.1 Manually trip reactor. 1.2 IF reactor can NOT be tripped, THEN trip both MG set supply breakers. [] N1C11E005A [] N1C11E005B 1.3 IF reactor will NOT trip, THEN go to FNP-1-FRP-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWT.
___ 2	**Check turbine trip.** [] TSLB2 14-1 - LIT [] TSLB2 14-2 - LIT [] TSLB2 14-3 - LIT [] TSLB2 14-4 - LIT	2 Perform the following. 2.1 Place main turbine emergency trip switch to TRIP for at least 5 seconds. 2.2 IF turbine can NOT be tripped, THEN reduce GV position demand signal to zero from DEH panel. [] TURBINE MANUAL depressed [] GV CLOSE depressed [] FAST ACTION depressed 2.3 IF steam flow to main turbine is NOT secured, THEN close all main steam line isolation and bypass valves.

5.3.2.4 **Reinstall** the BYPASS BREAKER B control power fuses. ______

5.3.2.5 In the B Train REACTOR TRIP SWITCHGEAR cabinet rear terminal panel Q1C11E004A-AB, verify the voltage between TB5-6 and TB5-7 is ~ 0 VDC. IF the voltage is > 0.5 volts, THEN **notify** Maintenance that either the bypass breaker's auxiliary switch contacts or secondary disconnects are not made up properly. **Submit** a CR to investigate and repair the problem. ______ I&C

5.3.2.6 At Reactor Trip BYPASS BREAKER B, **depress** the black close button on the front of the breaker and **verify** closure. ______

5.3.2.7 **Depress** the black TRIP button located on the front of Reactor Trip BYPASS BREAKER B to energize its shunt coil. ______

5.3.2.7.1 **Check** that Reactor Trip BYPASS BREAKER B opens. ______

5.3.2.8 **Depress** the black CLOSE button on the front of Reactor Trip BYPASS BREAKER B. ______

5.3.2.8.1 **Check** closure of Reactor Trip BYPASS BREAKER B. ______

5.3.2.9 **Depress** the BLOCK BYPASS-UV button, located in the A TRAIN SHUNT TRIP TEST CABINET, to deenergize the Reactor Trip BYPASS BREAKER B undervoltage trip coil. ______

5.3.2.10 **Check** the BYPASS BREAKER B open by observing the breaker open indication flag on the front of the breaker. ______

5.3.2.11 **Remove** the BYPASS BREAKER B control power fuses from the top cubicle located above the breaker. (See Figure 3.) ______

5.3.2.12 **Rack** BYPASS BREAKER B to the DISCONNECT position to discharge the spring. ______

5.3.2.13 **Rack** BYPASS BREAKER B to the CONNECT position. ______

5.3.2.14 **Reinstall** BYPASS BREAKER B control power fuses. ______

5.3.2.15 In the B Train REACTOR TRIP SWITCHGEAR cabinet rear terminal panel Q1C11E004A-AB, **verify** the voltage between TB5-6 and TB5-7 is ~ 0 VDC. IF the voltage is > 0.5 volts, THEN **notify** Maintenance that either the bypass breaker's auxiliary switch contacts or secondary disconnects are not made up properly. **Submit** a CR to investigate and repair the problem. ______ I&C

5.3.2.16 In the B Train REACTOR TRIP SWITCHGEAR cabinet rear terminal panel Q1C11E004A-AB, **check** the following voltages:

5.3.2.16.1 Voltage between TB5-1 (+) and TB5-4 (-) is 48 (44 to 52) VDC. ______ I&C

5.3.2.16.2 Voltage between TB4-1 (+) and TB4-4 (-) is 125 (110 to 140) VDC. ______ I&C

5.3.2.17 **Depress** the black close button on the front of Reactor Trip BYPASS BREAKER B. ______

5.3.2.18 **Check** closure of Reactor Trip BYPASS BREAKER B. ______

5.3.2.18.1 **Record** time of BYPASS BREAKER closure ______

____

5.3.2.18.2 **Log** time of BYPASS BREAKER closure in the Control Room Log. ______

5.3.3 In the B Train REACTOR TRIP SWITCHGEAR cabinet rear terminal panel Q1C11E004A-AB, **check** the following (Ref. Figures 1 and 2):

5.3.3.1 Voltage between TB5-1 (+) and TB5-4 (-) is 0 (-1.0 to 2.0) VDC. ________volts ______ I&C

5.3.3.2 Voltage between TB4-1 (+) and TB4-4 (-) is 0 (-1.0 to 2.0) VDC. ________volts ______ I&C

5.3.4 On SOLID-STATE PROTECTION SYSTEM TRAIN-A LOGIC CABINET, **check** the GENERAL WARNING lamp OFF. ______

______ CV

5.3.5 On SOLID-STATE PROTECTION SYSTEM TRAIN-B LOGIC CABINET, **check** the GENERAL WARNING lamp ON. ______

5.3.6 In SOLID-STATE PROTECTION SYSTEM TRAIN-B OUTPUT CABINET, at the OUTPUT RELAY TEST PANEL, **place** the MODE SELECTOR switch in the TEST position. ______

5.3.6.1 **Check** that the OPERATE lamp is OFF. ______

5.3.6.2 **Record** time the MODE SELECTOR switch in the TEST position ____________ ☐

5.3.6.3 **Log** time the MODE SELECTOR switch in the TEST position in the Control Room Log _________ ☐

ROD CONTROL SYSTEM

Both manual and automatic rod control is available. The system is designed such that maximum operating reliability is provided while ensuring reactor safety. Single malfunctions usually allow the use of alternate circuits.

Reactor Control Unit

Refer to Figure 3 & 5. The reactor control unit is that portion of the rod control system that provides for automatic and manual operation of the control rods. The reactor control unit processes the various plant parameters, establishes the T_{avg} program, and determines rod speed and direction requirements. The control unit consists of two error signal channels, the sum of whose outputs is the input of the rod speed programmer. The two channels used to generate the total error signal are (1) the deviation of the actual median primary coolant temperature (T_{avg}) from the programmed average temperature (T_{ref}) and (2) the rate of mismatch between turbine load and nuclear power.

Temperature Mismatch Channel

The temperature mismatch channel receives the median loop T_{avg} from the RCS temperature instrumentation. The signal is electronically conditioned and then compared to T_{ref}. Since the steam pressure in the impulse chamber of the high pressure turbine is linearly proportional to the turbine load, the pressure signal is used to generate the programmed temperature (T_{ref}). The T_{ref} signal is compared to the processed T_{avg} signal. This is the dominant control channel in steady-state conditions.

Power Mismatch Rate Channel

The power mismatch rate channel provides control stability and fast response to load changes. The mismatch between turbine power (P_{imp}) and nuclear power (N-44) is an anticipatory input to the control system. Any mismatch between turbine power and nuclear power will result in a change in T_{avg}. The circuit responds to a rate of change of the mismatch signal. This technique ensures that steady-state errors will have no effect on system control. As a result, the output of the channel is zero at steady-state conditions even though a power mismatch might exist. The power mismatch signal is compared to the temperature mismatch signal to generate rod speed and direction signals.

into the core. The distance the rod drops into the core is dependent upon the requested rod speed and the length of time the rod movement is called for.

OPERATIONS

Instrumentation and Controls

Reactor Control Unit

Refer to Figure 5. The reactor control unit consists of two channels: (1) the power mismatch channel and (2) the temperature mismatch channel. The power mismatch channel provides an error signal whenever there is a rate of change between turbine power and reactor power. (During constant power operation, the error signal will be zero even if turbine power and reactor power are not equal.) The temperature mismatch channel produces an error signal proportional to the deviation between median T_{avg} and P_{imp} generated T_{ref}. (The error signal will be zero only if the difference between T_{avg} and T_{ref} is zero.) This is the normal control channel.

The error signals produced by these channels are summed and routed to a bistable and a function generator. The bistable determines the direction of rod motion, and the function generator determines the rod speed. The resultant output will be sent to the logic cabinets.

Temperature Mismatch Channel

The temperature mismatch channel receives inputs of median T_{avg} and turbine first stage impulse pressure (P_{imp}). Prior to entering a differential amplifier, the T_{avg} signal passes through a lead/lag card for dynamic conditioning. The lead/lag card provides dynamic compensation by producing an output that anticipates the actual plant T_{avg} when T_{avg} is changing.

On a ramp up in T_{avg}, the output of the lead/lag card will be the value of actual T_{avg} at some future point in time. This compensates for the delay between the time when temperature begins to increase in the reactor and the time when the increase will actually be sensed by the resistance temperature detectors (RTDs) in the loops.

The P_{imp} signal, a measure of turbine load, feeds into a function generator. The function generator creates a T_{ref} signal programmed to vary as a function of plant load. Again, for purposes of discussion, the program is 547°F at zero-percent power to 575°F at 100-percent power. The T_{ref} signal passes through a lag circuit for dynamic compensation prior to entering the differential amplifier.

The T_{avg} and T_{ref} inputs connect to the differential amplifier, which performs the following function:

$$T_{error} = (T_{ref} - T_{avg})$$

The T_{error} signal will be summed with the P_{error} signal from the power mismatch channel.

Power Mismatch Channel

The power mismatch channel receives inputs from nuclear power (N-44) and P_{imp}. P_{imp} is conditioned to produce a turbine power signal that may be compared with the nuclear power signal from N-44. When compared in a differential amplifier, nuclear power and turbine power produce an error output signal equivalent to the difference between turbine and reactor power multiplied by a gain.

The output of the differential amplifier supplies a signal to a derivative card. This card produces an output only when the turbine-reactor power deviation is changing. When the deviation is constant, the output of the derivative card, and, therefore, the output of the power mismatch circuit, equals zero.

Any output obtained from the derivative card enters a function generator, which serves as a non-linear gain unit. A small input to the unit will be amplified with a gain of only 0.24, resulting in little rod motion. However, if the input is greater in magnitude, the gain becomes 1.2, lending greater weight to the error signal and resulting in increased rod motion.

The output of the non-linear gain unit enters a variable gain unit. The variable gain unit varies the gain applied to the error signal inversely with turbine power. The variable gain unit compensates for the fact that a step of rod motion produces a greater change in power at high power levels than at low power levels. Therefore, the power error (P_{error}) signal must be reduced as power increases to reduce the rod motion at higher power levels. The variable gain is accomplished by dividing the error signal by the output from the power compensation unit. The power compensation unit generates a function that varies inversely with P_{imp}.

The output of the variable gain unit, a P_{error} signal, inputs to a summing unit to be summed with the T_{error} signal from the temperature mismatch channel. The output of the variable gain unit is provided with a defeat switch, which is located in control cabinet eight of the 7300 cabinets, along with the rest of the reactor control unit. This switch, operated by the I&C department using procedures under their control, allows the power mismatch channel to be isolated from the rest of

the reactor control unit for calibration or maintenance purposes. I&C procedures and the Precautions, Limitations and Setpoint document require the rod control system to be in manual control any time this switch is open. If the rod control system were operated in automatic with the mismatch channel defeat switch open, the rod control system would be without the benefit of the anticipatory response provided by this channel, causing a possible improper response of the system.

The output of the summing unit, which can be either positive or negative, provides an input to the rods in/out bistable and a function generator. The rods in/out bistable provides the signal to direct rod motion (in or out). The polarity of the input signal to the bistable will dictate the direction the rods are to move. If the input signal exceeds the output setpoint in the positive direction, a rods-out command will be generated. The output setpoints equate to ± 1.5°F temperature error. The rod motion command will reset at ± 1°F. This 0.5°F lockup will prevent unnecessary rod motion near the bistable output setpoint.

The function generator determines the rod speed based on the magnitude of the error signal. The rod speed varies from 8 steps per minute (0 to ± 3°F error) to 72 steps per minute (± 5°F error). The rod speed varies linearly from eight steps per minute to 72 steps per minute (± 3°F to ± 5°F error).

Motor Generator (MG) Set Controls

The MG set control panels are located in the non-rad Aux bldg on the 121' elev. (rod control room.) Each control panel has the following components: generator line voltage and current indication (with phase selector switch), voltage regulator potentiometer, field flash pushbutton, overcurrent relay, ground relay, reverse current relays, null meter (A MG panel only), motor supply breaker control switch and indication, generator output breaker control switch and indication, and an automatic synchronizer to parallel the generator output.

Metering and Protective Relay Equipment – Current indication on each phase (meter range 0-500amps, normal operating range 25-100amps); Voltage indication on each phase (meter range 0-400vac, normal operating range 220-300vac); Overcurrent protection; Reverse current protection; Overvoltage protection; Ground detection (alarm only);

1. 003A3.05 002/MOD/VOGTLE 12/MEM 2.7*/2.6/003A3.05/N///

Unit 1 is in Mode 3 and preparing to start the 1C RCP.

Which one of the following completes the statements below per SOP-1.1, Reactor Coolant System?

The 1C RCP oil lift pump handswitch white light indicates __(1)__ .

The 1C RCP breaker closing operation __(2)__ interlocked with oil lift pump pressure.

A. 1) BOTH the oil lift pressure has reached 600 psig AND 2 minutes have elapsed

2) is NOT

B.✓ 1) ONLY that the oil lift pressure has reached 600 psig

2) IS

C. 1) ONLY that the oil lift pressure has reached 600 psig

2) is NOT

D. 1) BOTH the oil lift pressure has reached 600 psig AND 2 minutes have elapsed

2) IS

SOP-1.1
3.6 DO NOT attempt to start a RCP unless its oil lift pump has been delivering oil to the upper thrust shoes for at least two minutes. Observe the oil lift pumps indicating lights to verify correct oil pump motor operation and oil pressure. The oil lift pumps should run at least 1 minute after the RCP's are started. **An interlock will prevent starting a RCP until 600 psig oil pressure is established.**

Note prior to step 4.3.11 - The oil lift pump must be operated for at least 2 minutes prior to starting the RCP.

4.3.18 Verify that the Oil Lift Pump for RCP 1C has run for at least two minutes, and is producing adequate pressure (white light ON).

<u>**Distracter Analysis**</u>

A. Incorrect. First part is incorrect (See B.1). Plausible since the procedure requires both conditions to be met before starting the RCP. The applicant could have the misconception that the light is associated with both the time and oil pressure.

Second part is incorrect (See B.2). Plausible if the applicant does not recall that there is a physical interlock in the RCP circuit breaker associated with the oil lift pressure. This fact is only mentioned in the Precautions and Limitations.

B. Correct. First part is correct. (See 4.3.18 above)

Second part is correct (See 3.6 above).

C. Incorrect. First part is correct (See B.1).

Second part is incorrect (See A.2).

D. Incorrect. First part is incorrect (See A.1).

Second part is correct (B.2).

K/A: **003A3.05**	Reactor Coolant Pump System (RCPS) - Ability to monitor automatic operation of the RCPS, including: RCP lube oil and bearing lift pumps
Importance Rating:	2.7* 2.6
Technical Reference:	FNP-1-SOP-1.1, Reactor Coolant System, Ver 47.2
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Reactor Coolant Pumps, to include the following (OPS-40301D02): • Oil lift system
Question History:	MOD VOGTLE 12
K/A match:	The only AUTO features of the RCP LO and Brg lift pump is the white light will come on when the pressure reaches 600 psig and then the permissive will clear at 600 psig allowing the RCP breaker to be closed. Applicant must be **able to monitor the white indicating light for the RCP oil lift pump** which indicates the discharge pressure of that pump is > 600 psig and have knowledge that when the light comes on the permissive **automatically** allows the RCP circuit breaker being capable of closing.
SRO justification:	N/A

3.0 PRECAUTION AND LIMITATIONS

3.1 The RCS (except the pressurizer) shall be limited to a maximum heatup of 100°F in any one hour period and a maximum cooldown of 100°F in any one hour period at all times.

3.2 The pressurizer temperature shall be limited to a maximum cooldown of 200°F in any one hour period, a maximum heatup of 100°F in any one hour period, and a maximum spray water temperature differential of 320°F at all times.

3.3 A Residual Heat Removal (RHR) pump or a RCP must be operating to provide reactor coolant recirculation and thorough mixing during boron concentration changes, chemical addition or any time the RCS temperature exceeds 140°F.

3.3.1 Verify the desired boron concentration in the RCS and PRZR has been achieved prior to securing the only running RCP.
(SOER 94-02)

3.3.2 At least one RCP must be in operation prior to performing any RCS dilution or boration except as noted in Step 3.3.3 below.
(SOER 94-02)

3.3.3 IF no RCP is in operation and at least one RHR pump is on service providing 3000 gpm flow, THEN chemicals may be added to the RCS provided an evaluation of the effects of a small volume dilution to the RCS with potentially inadequate mixing has been performed and with Shift Supervisor concurrence. The evaluation should consider shutdown margin for present conditions and the affected volume of RCS. Boron samples should be obtained and analyzed to ensure adequate shutdown margin is maintained.

3.4 RCS pressure and temperature are limited to maximum of 375 psig and 350°F respectively when the RHR system is valved into the RCS.

3.5 RCP's shall not be operated continuously until the RCS has been filled and vented in accordance with FNP-1-SOP-1.3, Reactor Coolant System Filling And Venting - Vacuum Method, or FNP-1-SOP-1.11, Reactor Coolant System Filling And Venting - Dynamic Method.

3.6 DO NOT attempt to start a RCP unless its oil lift pump has been delivering oil to the upper thrust shoes for at least two minutes. Observe the oil lift pumps indicating lights to verify correct oil pump motor operation and oil pressure. The oil lift pumps should run at least 1 minute after the RCP's are started. An interlock will prevent starting a RCP until 600 psig oil pressure is established.

3.7 Shift Supervisor's approval must be obtained prior to removing any seal wires or changing the position of any throttle valves.

3.8 RCP seal water injection flow of 6 gpm or CCW to the RCP thermal barrier must be continuously supplied when RCS temperature exceeds 150°F.

NOTES

- Initial startup of a RCP will be performed in accordance with FNP-1-UOP-1.1, Startup Of Unit From Cold Shutdown To Hot Standby, OR FNP-1-SOP-1.3, Reactor Coolant System Filling And Venting. FNP-1-UOP-1.1, Startup Of Unit From Cold Shutdown To Hot Standby, covers the initial startup of a RCP with the reactor coolant system in solid water operation and ensures all pumps are running prior to entry into mode 4. Starting of additional pumps may be performed by this procedure following the start of the first RCP. ☐
- Initial startup of a RCP may also be performed during Plant Shutdown per this procedure (FNP-1-SOP-1.1) when sent here from FNP-1-UOP-2.2, Shutdown Of Unit From Hot Standby To Cold Shutdown, Appendix 5. FNP-1-UOP-2.2, Appendix 5 references this procedure (FNP-1-SOP-1.1) to start a RCP during shutdown, either solid plant or with a bubble in the pressurizer. ☐

CAUTION

Prior to starting a RCP, consideration should be given to raising the 230 kV bus voltage such that the emergency 4160 volt buses are approaching the 4200 volt limit to preclude spurious RCP breaker trip. (AI2010200391) ☐

4.3 1C RCP Startup.

4.3.1 **Verify** RCP seal flow established per FNP-1-SOP-2.1, Chemical And VolumeControl System Plant Startup And Operation. ☐

4.3.2 **Verify** VCT pressure > 18 psig. ☐

4.3.3 IF RHR is aligned to the RCS, THEN **verify** RCS pressure is 325-375 psig. ☐

4.3.4 IF RHR is NOT aligned to the RCS, THEN **verify** RCS pressure $\geq$ 350 psig. ☐

4.3.5 **Verify** that the 1C RCP STANDPIPE LVL LO annunciator DA3 is clear. ☐

4.3.6 **Verify** that the RCP THRM BARR CCW FLOW HI annunciator DD2 is clear. ☐

4.3.7 **Verify** that the CCW FLOW FROM RCP OIL CLRS LO annunciator DD3 is clear. ☐

4.3.8 **Verify** that the RCP 1C BRG UPPER/LOWER OIL RES HI LVL annunciator HG3 is clear. ☐

4.3.9 **Verify** that the RCP 1C BRG UPPER/LOWER OIL RES LO LVL annunciator HH3 is clear. ☐

4.3.10 WHEN one or more of the RCS cold leg temperatures is $\leq$ 325°F, THEN **record** Przr level and steam generator ΔT as per FNP-1-UOP-1.1, Startup Of Unit From Cold Shutdown To Hot Standby. ☐

NOTES

- The oil lift pump must be operated for at least 2 minutes prior to starting the RCP. ☐
- An oil lift pump may need to operate for several minutes (more than 5) before enough pressure is developed to actuate the pressure switch that will illuminate the white light. The length of run time may vary due to oil temperature, secured time and pressure switch sensitivities. (AI12011201221) ☐

4.3.11 **Start** the 1C RCP Oil Lift Pump. Adequate oil pressure is indicated by the white light coming ON. ☐

4.3.12 **Verify** that the 1C RCP SEAL LEAKOFF valve Q1E21HV8141C is OPEN. ☐

4.3.13 **Verify** that 1C RCP No. 1 Seal Leakoff Flow rate is within the limits of Figure 1. ☐

4.3.14 **Verify** that the SEAL WTR INJ FLTR HI ΔP annunciator DC4 is clear. ☐

4.3.15 **Verify** that the RCP SEAL INJ FLOW LO annunciator DD1 is clear. ☐

4.3.16 **Verify** that all RCP No. 1 Seal ΔP's are greater than 210 psid. ☐

4.3.17 **Verify** that the RCP #1 SEAL LO ΔP annunciator DC3 is clear. ☐

4.3.18 **Verify** that the Oil Lift Pump for RCP 1C has run for at least two minutes, and is producing adequate pressure (white light ON). ☐

4.3.19 **Start** 1C RCP. Verify that all loop 1C flow instruments show an increasing flow rate. ☐

4.3.20 **Verify** that RCP 1C amperage decreases to a normal operating range of 900 amps cold and 700 amps hot. ☐

4.3.21 **Verify** that the 1C RCS LOOP FLOW LO annunciator EF3 is clear. ☐

CAUTION

The RCP seal water bypass valve Q1E21HV8142, should only be opened IF No. 1 seal leakoff flow rate is < 1 gpm AND RCS pressure is < 1000 psig. During RCS heatup/pressurization or cooldown/depressurization, the seal water bypass valve, Q1E21HV8142, may be left closed unless pump bearing temperature or No. 1 seal outlet temperature approach their alarm levels. During normal operation, the seal water bypass valve should remain closed. ☐

4.3.22 After at least 1 minute of 1C RCP operation, **stop** the Oil Lift Pump. ☐

1. 015AK2.10 001/1/1/RCP/F-2.8*/2.8/NEW/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

- Unit 1 is in Mode 3.

- All RCPs are in service.

Current conditions:

- RCP # 1 trips due to a faulty relay.

After the plant stabilizes and when starting RCP #1 after repair,

the RCP oil lift pump handswitch blue light ___(1)___ indicate that the lift oil pressure has been greater than **600 psig** for **TWO** minutes.

The interlock associated with this must be met in order to close the RCP ___(2)___ breaker.

	__(1)__	__(2)__
A.	does NOT	1E
B.	does	1E
C.✔	does NOT	non-1E
D.	does	non-1E

1. 004K3.08 003/BANK/SURRY 09 1ST AUDIT/C/A 3.6/3.8/004K3.08/N///

Unit 1 was operating at 100% power when the following occurred:

- The air supply to FCV-122, CHG FLOW REG, actuator has been severed and the valve has repositioned to its failed position.

Which ONE of the following completes the statement below?

FCV-122 is _(1)_ and RCP seal injection flow will _(2)_.

	(1)	(2)
A.	CLOSED	LOWER
B.	CLOSED	RISE
C.✓	OPEN	LOWER
D.	OPEN	RISE

AOP-6.0 Table 1:

Component No.	Name	Failed Position
Q1E21V347 (1-CVC-FCV-122)	CHG FLOW REG	OPEN

ARP-1.4, DC4 - SEAL WTR INJ FLTR HI Δ P
Probable Cause.
2. High seal injection flow rate.
3. Chg Flow Q1E21FCV122. (Also labeled Q1E21V347) failed closed.

Distracter Analysis

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant does recall the fail position of FCV-122.

Second part is correct (See C.2). Logical connection to the first part if the applicant believes that FCV-122 is upstream of the seal injection line and the closure of FCV-122 would stop seal injection flow.

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See C.2). Logical connection to the first part if the applicant thought that FCV-122 failed closed since it would be the correct seal injection response for this condition.

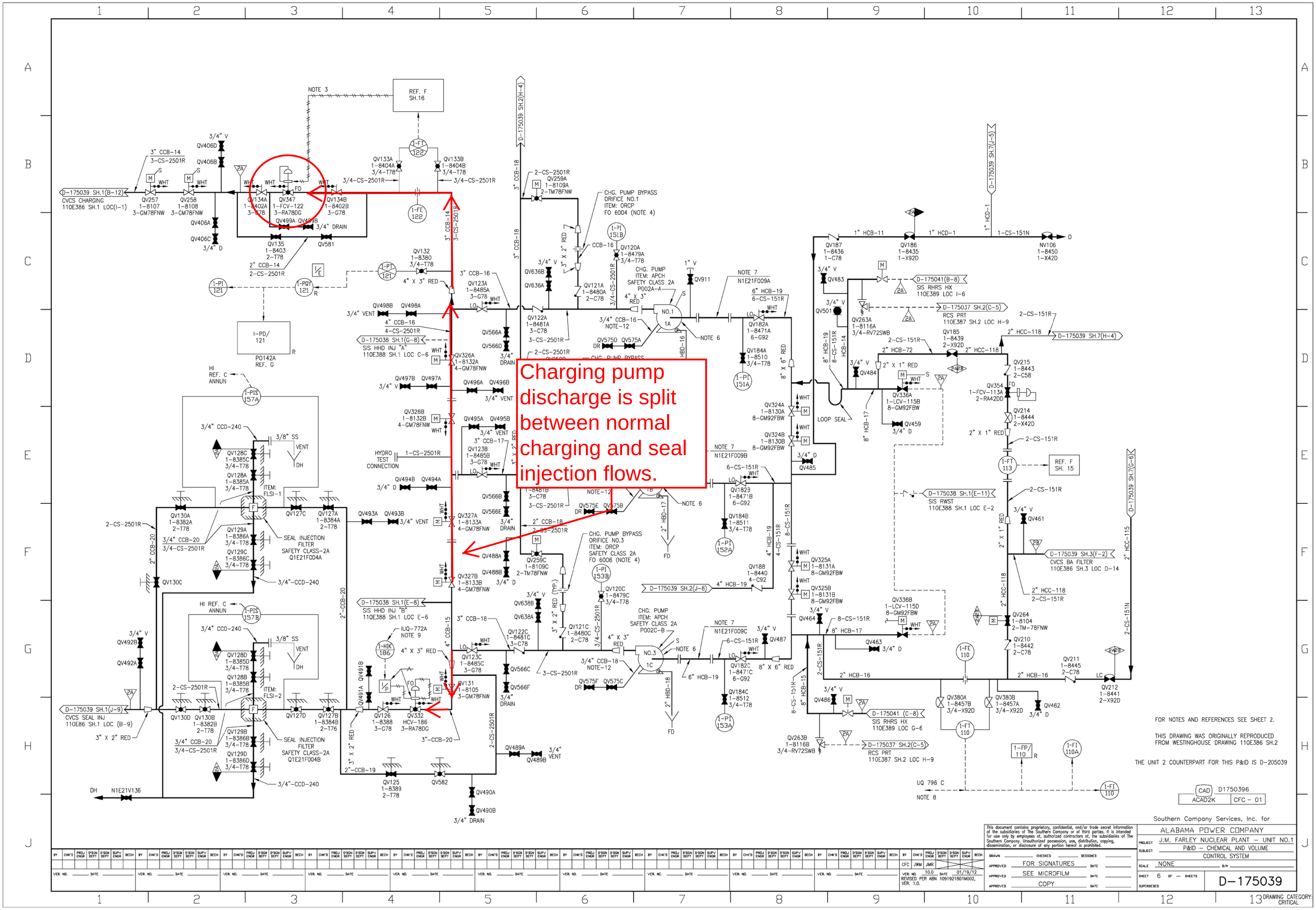
C. Correct First part is correct. FCV-122 fails open.

Second part is correct. The closure of FCV-122 will cause high seal injection flow and thus high filter DP due to all of the charging pump discharge flow being directed to the seal injection filter (See ARP-1.4, DC4 above). Conversely if the air line fails on FCV-122, the valve will fail open and a majority of the charging pump discharge flow will be directed to the normal charging path causing the seal injection flow to go down. (See P&ID D-175039, SH 6, Chemical and Vol Control System for system flow.)

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See C.2). Logical connection to the first part if the applicant assumes that more charging flow equates to more seal injection flow.

K/A: **004K3.08**	Chemical and Volume Control System (CVCS) - Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: RCP seal injection
Importance Rating:	3.6 3.8
Technical Reference:	P&ID D-175039, SH 6, Chemical and Vol Control System Ver 10 FNP-1-AOP-6.0, Loss of Instrument Air, Ver 40 FNP-1-ARP-1.4, DC4, Ver 53
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Chemical and Volume Control System, to include the components found on Figure 3, Chemical and Volume Control System and Figure 4, RCP-Seal Injection System (OPS-40301F02).
Question History:	SURRY 09 1ST AUDIT
K/A match:	The CVCS malfunction is that the air line to FCV-122 has been severed. Applicant must know how this **CVCS malfunction affects seal injection flow**.
SRO justification:	N/A



Charging pump discharge is split between normal charging and seal injection flows.
FOR NOTES AND REFERENCES SEE SHEET 2.
THIS DRAWING WAS ORIGINALLY REPRODUCED FROM WESTINGHOUSE DRAWING 110E386 SH.2
THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205039
CAD D1750396
ACAD2K CFC - 01
Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
P&ID - CHEMICAL AND VOLUME CONTROL SYSTEM
SCALE NONE
SHEET 6
D-175039
FOR SIGNATURES
SEE MICROFILM
COPY
DRAWING CATEGORY: CRITICAL

TABLE 1

COMPONENT NUMBER	NAME	MANUAL OPERATOR	FAILED POSITON	OPERATOR DRAWING
N1E21HV8155 (1-CVC-PCV-8155)	VCT N2 SUPPLY REGULATOR	NO	CLOSED	
N1E21HV8156 (1-CVC-PCV-8156)	VCT H2 SUPPLY REGULATOR	NO	CLOSED	
Q1E21V332 (1-CVC-HCV-186)	SEAL WATER INJECTION FLOW CONTROL	NO	OPEN	U-357413
Q1E21V337 (1-CVC-FCV-113B)	MAKEUP TO CHG PUMP HEADER	NO	CLOSED	U-176715
Q1E21V339 (1-CVC-FCV-114A)	MAKEUP DILUTION TO VCT	NO	CLOSED	U-176715
Q1E21V340 (1-CVC-HV-8547)	BTRS BYPASS CONTROL	NO	OPEN	U-259973
Q1E21V341 (1-CVC-PCV-8157)	VCT HI PRESS VENT BACK PRESS REG	NO	CLOSED	U-259962
Q1E21V345 (1-CVC-FCV-114B)	REAC MAKEUP WATER TO BLENDER	NO	CLOSED	U-166865
Q1E21V347 (1-CVC-FCV-122)	CHG FLOW REG	NO	OPEN	U-357413
Q1E21V350 (1-CVC-TCV-381B)	REHEAT HX BYPASS	NO	OPEN	U-357413
Q1E21V351 (1-CVC-HCV-142)	RHR TO LETDOWN LINE	NO	CLOSED	U-357413
Q1E21V352 (1-CVC-PCV-145)	LETDOWN PCV	NO	OPEN	U-357413
Q1E21V353 (1-CVC-TCV-143)	LETDOWN HI TEMP DIVERT	NO	FLOW TO VCT	U-357413
Q1E21V354 (1-CVC-FCV-113A)	BORIC ACID TO BLENDER	NO	OPEN	U-357413
Q1E21V356 (1-CVC-HV-8101)	VCT TO VENT HEADER ISO	NO	CLOSED	U-176710

LOCATION DC4

SETPOINT: 20 PSID

C4
SEAL WTR INJ FLTR HI ΔP

ORIGIN:
1. Diff. Pressure Switch (N1E21PIS0157A-N)
2. Diff. Pressure Switch (N1E21PIS0157B-N)

PROBABLE CAUSE

1. Seal Water Injection Filter 1A or 1B clogged.
2. High seal injection flow rate.
3. Chg Flow Q1E21FCV122 failed closed.

AUTOMATIC ACTION

NONE

CAUTION: To prevent possible over pressurization and failure of a clogged seal injection filter, no attempt to raise seal injection flow should be made while annunciator is in.

CAUTION: Seal injection must not be isolated if CCW cooling to the thermal barrier is not available.

NOTE: Performance of the following steps may result in isolation of seal injection. This may be necessary to preclude possible collapse of a seal injection filter that may result from excessive ΔP.

OPERATOR ACTION

1. IF seal injection flow rate is high, THEN reduce to normal value.

2. IF alarm NOT cleared by previous step, THEN dispatch appropriate personnel to perform the following:
   - 2.1 observe actual Seal Injection Filter ΔP.
   - 2.2 IF filter ΔP is visibly rising, THEN isolate all seal injection flow by closing the following valves:
     - 1A SEAL WATER INJ FILTER INLET Q1E21V127A AND Q1E21V127C
     - 1A SEAL WATER INJ FILTER OUTLET Q1E21V130A AND Q1E21V130C
     - 1B SEAL WATER INJ FILTER INLET Q1E21V127B AND Q1E21V127D
     - 1B SEAL WATER INJ FILTER OUTLET Q1E21V130B AND Q1E21V130D
   - 2.3 IF ΔP is stable, THEN place the standby seal injection filter in service per FNP-1-SOP-2.1.

1. 005A2.02 004/MOD/ANO 05/C/A 3.5/3.7/005A2.02/N///

The following conditions exist on Unit 1:

- The operating crew is cooling down per UOP-2.2, Shutdown of Unit From Hot Standby to Cold Shutdown.
- MODE 5 has just been entered and the following conditions exist:
  - RCS temperature is 195°F.
  - BOTH 1A AND 1B RHR pumps are running in the cooldown mode.

Subsequently, PT-402, 1C LOOP RCS PRESS, fails HIGH.

Which one of the following describes the **minimum** required action(s), if any, to be performed per AOP-12.0, Residual Heat Removal Malfunction?

A. No actions are required.

B.✔ Secure the 1A RHR pump ONLY.

C. Secure the 1B RHR pump ONLY.

D. Secure BOTH 1A AND 1B RHR pumps.

FSD-181002:
3.4.6.2 - All valves will shut automatically if RCS pressure increases to 700 psig. [...] The pressure inputs for this interlock are from PT402 for 8701A and B and from PT403 for 8702A and B to prevent isolation of both trains of RHR due to a single pressure transmitter failing high.

AOP-12:
Entry Conditions
1.4 Closure of loop suction valve

Step 1. Check RHR loop suction valves OPEN	1. Stop any RHR PUMP with closed loop - suction valve(s)

See Tech Spec 3.4.12
Technical Specification 3.4.12, LTOP (Low Temperature Over Pressure Protection System requires two RHR suction relief valve with setpoints $\leq$ 450 psig when the temperature of one or more RCS cold legs is $\leq$ 325°F. If one or more of the RHR Loop suction valves closed, then this Technical Specification would not be met and Low Temperature Over Pressure Protection would not be satisfied.

__Distracter analysis__

A. Incorrect. See B. Plausible if applicant remembers that the RHR Loop suctions are opened and de-energized at some point but cannot recall that it is when RCS temperature is less than 180°F. If the valves were de-energized they would not shut so no action would

be required. Also, the applicant could remember that PT-402 and 403 provide interlocks to OPEN the RHR loop suctions but NOT remember they also will close the valves on high pressure. This would make this a correct answer.

B. Correct. PT-402 failing high will close MOV-8701A and MOV-8701B which isolates the suction to the 1A RHR pump. AOP-12 requires the 1A RHR pump to be secured.

C. Incorrect. See B. Plausible if the applicant improperly believes that PT-402 affects the 1B RHR pump suction valves instead of 1A RHR pump suction valves. If PT-403 failed high, this would be the correct answer.

D. Incorrect. See B. Plausible since the RHR loop suction valves have interlocks to prevent opening them if certain parameters are not met (See reference material FSD A181002). PT-402 must be less than 402.5 psig in order to open MOV-8701A and MOV-8702A which are on OPPOSITE trains. If the applicant thought that the closing on high pressure works the same way as the opening interlocks this would be a correct answer since they would believe a suction valve in each train will close and both RHR pumps would be required to be secured.

K/A: **005A2.02**	**Residual Heat Removal System (RHRS)** - Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown
Importance Rating:	3.5 3.7
Technical Reference:	FSD-A181002, Residual Heat Removal, Ver 44 FNP-1-AOP-12, Residual Heat Removal Malfunction, Ver 25 FNP Technical Specifications, Ver 190
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if entry into AOP-12.0, RHR System Malfunction and/or STP-18.4, Containment Closure is required. (OPS-52520L02)
Question History:	MOD ANO 05
K/A match:	The applicant has to predict **how the failure of the loop pressure transmitter affects the RHR system** in that it removes one of the required Low Temperature Over Pressure Protection System reliefs from service. The applicant will apply that prediction to **AOP-12 to get to** the **action required to mitigate this malfunction.**
SRO justification:	N/A

B. **Symptoms or Entry Conditions**

1 **This procedure is entered when a malfunction of the RHR system is indicated by any of the following:**

1.1 Trip of any operating RHR pump

1.2 Excessive RHR system leakage

1.3 Evidence of running RHR pump cavitation

1.4 Closure of loop suction valve

1.5 High RCS or core exit T/C temperature

1.6 Procedure could be entered from various annunciator response procedures.

CF3 1A OR 1B RHR PUMP OVERLOAD TRIP

CF4 1A RHR HX OUTLET FLOW LO

CF5 1B RHR HX OUTLET FLOW LO

CG3 1A OR 1B RHR HX CCW DISCH FLOW HI

EA5 1A OR 1B RHR PUMP CAVITATION

EB5 MID-LOOP CORE EXIT TEMP HI

EC5 RCS LVL HI-LO

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS AND the RCS temperature is below 180°F.

****

****

CAUTION: Filling the pressurizer to 100% will cause a loss of nozzle dams due to the head of water.

****

NOTE: RCS to RHR loop suction valves will be deenergized if RCS TAVG is less than 180°F.

___ 1 **Check RHR loop suction valves - OPEN.**

RHR PUMP	1A	1B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSED(IF REQUIRED)	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3

1 Stop any RHR PUMP with closed loop suction valve(s).

1.1 IF required, THEN adjust charging flow to maintain RCS level.

___ 2 **IF the standby RHR train is NOT affected AND plant conditions permit operation, THEN place the standby RHR train in service per FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM.**

2 IF core cooling provided by the SGs, THEN proceed to step 8.

specifications of 10CFR50.55A requiring ASME B&PV Code Section III 1971 Edition plus summer 1971 Addenda. These exemptions were granted based on a hardship due to delay in issuance of a construction permit. (Reference 6.1.28)

3.4.4.2 The valves are designed to ASME Code Class I. (References 6.4.5 and 6.5.14)

3.4.4.3 The valves are considered active consistent with ANSI N18.2. (Reference 6.2.27)

3.4.4.4 Consistent with requirements of ASME Section XI these valves should be included in the plant's Inservice Test Program. (References 6.7.24 and 6.7.25)

3.4.5 Seismic Requirements

These valves and operators shall be designated as Seismic Category I. The valve specification requires that the valves be able to withstand seismic loading equivalent to 3.0 g in the horizontal directions and 2.0 g in the vertical direction (Reference 6.5.10). The seismic loading of 3.0 g in the horizontal directions and 2.0 g in the vertical direction is an industry standard used for procurement to encompass all seismically qualified valves.

3.4.6 I&C Requirements

3.4.6.1 Valves 8701A,B and 8702A,B must be interlocked to prevent inadvertent overpressurization of the RHR and CVCS piping, depressurization of the RCS, or overflow and dilution of the RWST. To prevent alignment of the RHR pump suction to the RCS with a single pressure transmitter failed low, the series isolation valves in each train take their pressure input from separate transmitters. The interlocks operate as follows:

a. Valves 8701B and 8702B cannot be opened unless the following conditions are met (References 6.2.24, 6.2.25 and 6.2.8):

1. RCS pressure is less than 402.5 psig as sensed by RCS pressure transmitter Q1(2)B21PT403.

2. The RWST suction isolation valve (8809A or B), which supplies the same RHR pump suction header, is closed.

3. The charging pump recirculation isolation valve supplied by the same RHR pump (8706A or B) is closed.

b. Valves 8701A and 8702A cannot be opened unless the following conditions are met (References 6.2.24, 6.2.25 and 6.2.8):

1. RCS pressure is less than 402.5 psig as sensed by RCS pressure transmitter Q1(2)B21PT402.

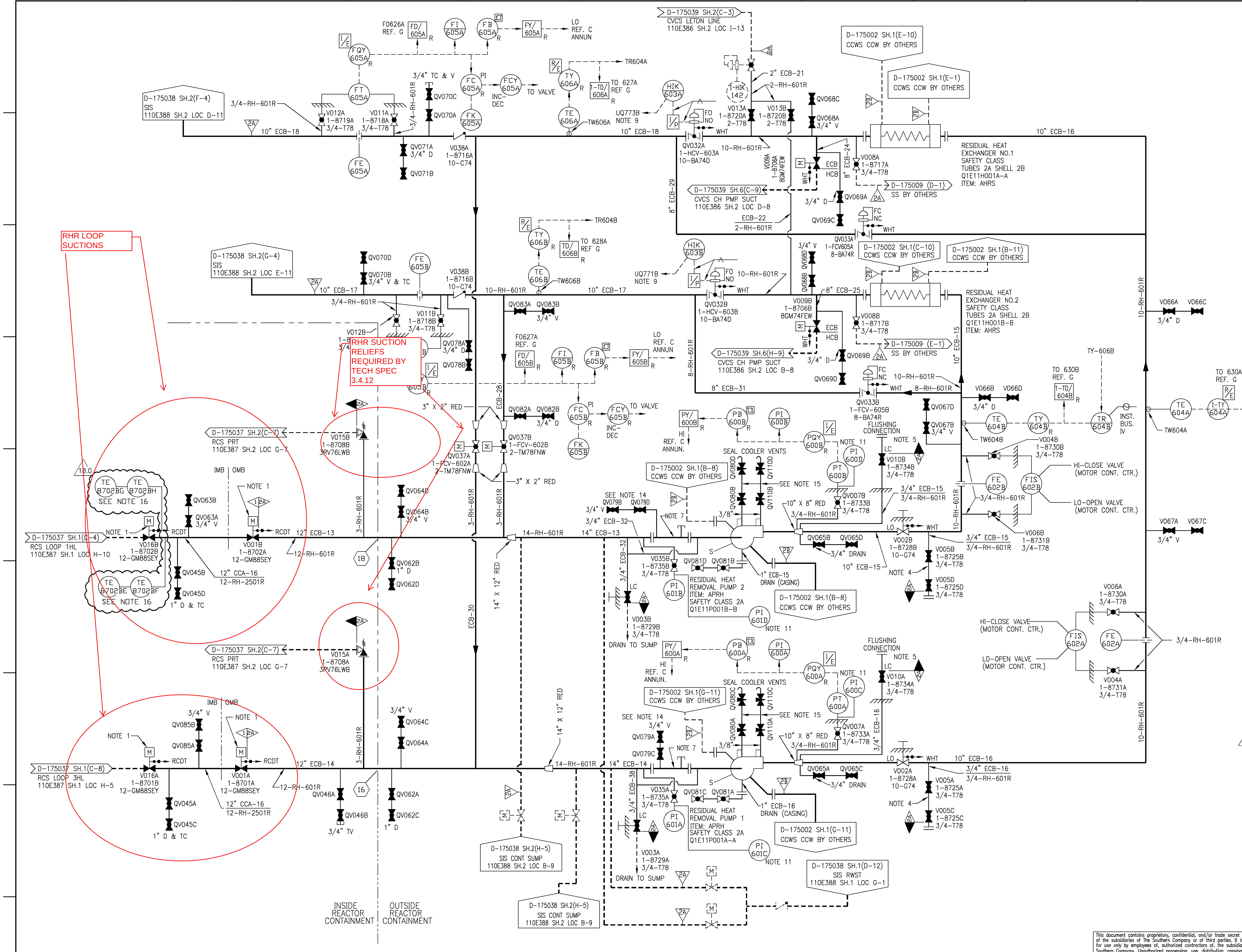
2. The RWST suction isolation valve (8809A or B), which supplies the same RHR pump suction header, is closed.

3. The charging pump recirculation isolation valve supplied by the same RHR pump (8706A or B) is closed.

4. The pressurizer vapor space temperature is less than 475°F on N1(2)B31TE454.

3.4.6.2 All valves will shut automatically if RCS pressure increases to 700 psig. This ensures the valves are closed during a plant startup prior to reaching operating conditions in the event that one valve had been inadvertently left open. Note that this auto-closure interlock occurs at an RCS pressure that is in excess of the RHRS design pressure (600 psig), since this interlock is not to protect the RHRS from overpressurization but is to ensure that a double isolation barrier exists at the reactor coolant pressure boundary. The pressure inputs for this interlock are from PT402 for 8701A and B and from PT403 for 8702A and B to prevent isolation of both trains of RHR due to a single pressure transmitter failing high. (References 6.2.24, 6.2.25 and 6.2.8)

3.4.6.3 Hand switches and position indication must be provided on the main control board to facilitate remote operation of the valves. (References 6.2.8 and 6.7.16)



NOTES:

1. VALVE INTERLOCKED WITH REACTOR SYSTEM PRESSURE SIGNAL.
4. LOCATE VALVE OUTSIDE OF SHIELD WALL SAMPLE. LINE MUST BE AT A LOWER ELEVATION THAN RHR PIPE TO ALLOW GRAVITY FLOW.
5. LOCATE VALVE ABOVE RESIDUAL HEAT REMOVAL PUMP SHIELDING.
6. ALL Ⓦ ITEM NOS. ARE SHOWN WITHOUT PREFIX ALAAC.
7. TEMPORARY STRAINER IS PLACED IN SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START UP.
8. MANUAL AND CHECK VALVES 2 INCHES AND UNDER IN OTHERS SCOPE.
9. MULTIPLE LOOP POWER SUPPLY (CONTROL BOARD) NOT SHOWN.
10. ALL LINE AND VALVE NUMBERS PREFIXED BY Q1E11.
11. INSTRUMENT VALVES FOR PI-600C, D, & PI-601C, D ARE CLOSED AT ALL TIMES EXCEPT DURING SURVEILLANCE TESTING.
12. DRAIN ALL DRAIN CONNECTIONS TO LOCAL EQUIPMENT DRAINS.
13. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175803.
14. VENT PROVIDED ON PERMANENT SPOOL PIECE TO BE USED AFTER STRAINER IS REMOVED.
15. THESE ARE SEAL PIPING VENTS AND ARE COMING OFF THE RHR PUMP SEAL COOLER.
16. TE'S, AS NOTED, ARE CLAMPED ON RTD'S FOR THERMAL CYCLING MONITORING.

REFERENCES:

1. ALPHA REFERENCE ON SHEET 110E390.

THIS DWG. WAS REPRODUCED FROM WESTINGHOUSE DWG. 110E389.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205041

CAD D175041
ACAD KB - 02

Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
P&ID - RESIDUAL HEAT REMOVAL SYSTEM

FOR SIGNATURES SEE MICROFILM COPY

KB GDM JMR
VER. NO. 18.0 DATE 03/02/11
REVISED PER ABN 1090211701E012, VER. 2.0

SCALE NONE B/M N/A
SHEET 1 OF — SHEETS
D-175041

DRAWING CATEGORY: CRITICAL

5.56.5 IF the temperature difference between the pressurizer and the charging water is large (i.e., greater than 20°F), THEN **place** auxiliary spray in operation as follows:

5.56.5.1 **Reduce** charging flow to minimum. ☐

5.56.5.2 **Open** RCS PRZR AUX SPRAY, Q1E21HV8145. ☐

5.56.5.3 **Verify** Charging lines isolated are closed:

- RCS NORM CHG LINE, Q1E21HV8146 ☐
- RCS ALT CHG LINE, Q1E21HV8147 ☐

5.56.6 Slowly **increase** charging flow to allow gradual cool down of the pressurizer aux. spray line. ☐

5.56.7 **Adjust** charging flow as required to control pressurizer cooldown rate. ☐

5.57 WHEN the RCS temperature (indicated by RHR system inlet temperature) is less than 180°F, THEN **perform** the following:

5.57.1 **Verify** RCS loop suction valves are open:

- 1C RCS LOOP TO 1A RHR PUMP Q1E11MOV8701A ☐
- 1C RCS LOOP TO 1A RHR PUMP Q1E11MOV8701B ☐
- 1A RCS LOOP TO 1B RHR PUMP Q1E11MOV8702A ☐
- 1A RCS LOOP TO 1B RHR PUMP Q1E11MOV8702B ______

______/______

NOTE

The power supply breakers for the RCS loop suction valves shall be closed whenever the RCS temperature is greater than or equal 180°F. ☐

5.57.2 **Dispatch** an operator to perform the following

5.57.2.1 **Open** the RCS loop suction valve power supply breakers per Table 4. ☐

5.57.2.2 Caution **tag** the power supply breakers and **place** tags locally at breakers. {CMTs 0003750 & 0004019} ☐

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12

An LTOP System shall be OPERABLE with a maximum of one charging pump capable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two residual heat removal (RHR) suction relief valves with setpoints ≤ 450 psig.

b. The RCS depressurized and an RCS vent of ≥ 2.85 square inches.

APPLICABILITY:

MODE 4 when the temperature of one or more RCS cold legs is ≤ 325°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

---NOTES---

1. The requirement to have only one charging pump capable of injecting into the RCS is only applicable when one or more of the RCS cold legs is ≤ 180°F; however, while in this condition, two charging pumps may be capable of injecting into the RCS during pump swap operations for a period of no more than 15 minutes provided that the RCS is in a non-water solid condition and both RHR relief valves are OPERABLE or the RCS is vented via an opening of no less than 5.7 square inches in area.

2. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

Questions For 2005 ANO UNIT 2 SRO Exam

BANK 0525 **Rev** 0 **Rev Date:** 11/7/2004 **RO Select:** No **SRO Select:** Yes **Points:** 1.00
Lic Level: S **Difficulty:** 3 **Taxonomy:** AP **Source:** NEW **Originator** COBLE
10CFR55_41: 41.5 **10CFR55_43:** 43.5 **Section:** 3.4 **Type** RCS HEAT REMOVAL
System RESIDUAL HEAT REMOVAL **System** 005 **K/A:** A2.02
SRO Tier: 2 **SRO Group:** 1 **SRO Imp:** 3.7
Description Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown.

Question # 87
Given the following:

* The unit is shutdown for a refueling outage
* RCS pressure is 260 psia
* RCS temperature is 215°F
* One train of shutdown cooling is in service.
* 2PT-4623-1 (pressurizer low range pressure transmitter) fails high.

Which of the following Operator actions should be taken?

A. Override all SIT outlet valves due to automatically opening.

B. Stop the running SDC pump due to a loss of suction from the RCS.

C. Place all HPSI and LPSI pumps in PTL due to initiation of a SIAS.

D. Place all Normal Spray Valves to manual and closed due to opening.

Answer:
B. Stop the running SDC pump due to a loss of suction from the RCS.

Notes:
The pressure transmitter failing high will cause SDC suction isolation 2CV-5084-1 to go closed and the AOP for loss of SDC directs securing the running SDC pump.
Distracter A is incorrect because the SIT outlet valve MOV breakers are de-energized after they are closed at ~650 psia during a cooldown.
Distracter C is incorrect because an SIAS is actuated on the Wide Range PZR Pressure Transmitters and the SIAS Low PZR Pressure actuation's have been bypassed on the PPS ROMs.
Distracter D is incorrect because the normal spray valves open on Control Channel PZR Pressure Transmitters which have not failed.

References

STM 2-03, RCS, Section 2.2.5.1, PZR Pressure Instrumentation and Figure on page 55.
STM 2-14, SDC, Section 2.1.1 and Figure on Page 50
AOP 2203.029, Loss of SDC, Step 3.
NOP 2102.010, Plant Cooldown, Step 7.42.3.
A2LP-RO-EAOP OBJ. 22, Discuss the Mitigation strategy, Entry Conditions, applicable industry events, Instructions and Exit Conditions (as per AOP and Tech Guide) of OP 2203.029, Loss Of

Historical
This question has not been used on any previous NRC exams. BNC 11/11/2004.

1. 006K5.06 005/BANK/SUMMER 11/C/A 3.5/3.9/006K5.06/N///

Unit 1 was operating at 100% power when the following conditions occurred:

- A LOCA is in progress.
- The operating crew is performing the actions of EEP-0.0, Reactor Trip or Safety Injection, and is at the step to "Check RCS intact".
- Containment pressure has risen to 8 psig.
- RCS pressure is 475 psig and lowering.

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

	SI Accumulator Level	RHR Injection Flow
A.	Stable and on-scale	Zero
B.✓	Dropping or off-scale low	Zero
C.	Dropping or off-scale low	Rising
D.	Stable and on-scale	Rising

EEP-1
9. Check if LSHI Pumps should be stopped.
9.1 Check RCS pressure - GREATER THAN 275 psig **{435 psig}**

EEB-1
ERP Step Text: Check if LHSI Pumps should be stopped.
Purpose: To stop the low-head SI pumps if RCS pressure is above their shutoff head to prevent damage to the pumps
Basis: Upon safety injection initiation all safeguard pumps are started regardless of the possibility of high RCS pressure with respect to the low-head safety injection pump shutoff head. On low-head systems where the pump recirculates on a small volume circuit there is concern for pump and motor overheating. Shutdown of the pump and placement in the standby mode, when the RCS pressure meets the criteria outlined in this step, allows for future pump operability. If SI has not been previously reset and the low-head SI pumps should be stopped, SI should be reset prior to stopping the pumps. SI can be reset regardless of containment pressure.

FSD - A181009
3.3.1.1 3 Safety injection accumulators shall function as passive safeguards components to rapidly inject [...] whenever the RCS pressure decreases below the tank cover gas pressure of 601 - 649 psig due to a loss of coolant accident

<u>Distracter Analysis</u>

A. Incorrect. First part is incorrect (See B.1). Plausible if the applicant is unfamiliar with the injection pressures of the accumulators. The

applicant may believe that RCS pressure is high enough to prevent accumulator injection.

Second part is correct (See B.2). Logical connection to the first part if the applicant recognizes that RHR injects at a lower pressure than the accumulators.

B. Correct. First part is correct. FSD-A181009, 3.3.13. Safety injection accumulators shall function as passive safeguards components to rapidly inject [...] whenever the RCS pressure decreases below the tank cover gas pressure of 601- 649 psig due to a loss of coolant accident.

Second part is correct. At 500 psig in the RCS, the accumulators will have injected but the RCS pressure will be above the RHR shut off head pressure. EEP-1 uses 435 psig (Adverse, because containment pressure is > 4 psig) as the criteria for RHR pump shut off head. (See EEP-1 and EEB-1 above).

C. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2) Logical connection to the first part if the applicant is unfamiliar with the injection pressures of the RHR pumps. If the applicant knows that the accumulators are(have) injecting(ed), they may also assume that RCS pressure is low enough to allow RHR injection.

D. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See B.2). Logical connection to the first part if the applicant is unfamiliar with the injection pressures of the RHR pumps. The applicant may believe that the RHR pumps inject before the accumulators.

The K/A: **006K5.06**	Emergency Core Cooling System (ECCS) - Knowledge of the operational implications of the following concepts as they apply to ECCS: Relationship between ECCS flow and RCS pressure
Importance Rating:	3.5 3.9
Technical Reference:	FSD-A181009, CVCS/HHSI/Accumulators/RMWS, Ver 39 FNP-1-EEP-1.0, Loss of Reactor or Secondary Coolant, Ver 31 FNP-0-EEB-1.0, Specific Background Document for FNP-1/2-EEP-1.0, Ver 4
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Emergency Core Cooling System, to include the components found on Figure 2, Accumulators, Figure 3, Refueling Water Storage Tank, and Figure 4, Emergency Core Cooling System (OPS-40302C02).
Question History:	SUMMER 11
K/A match:	The applicant is required to know **which ECCS components are injecting into the core (flow) based on RCS pressure.**
SRO justification:	N/A

processing. The tank may be vented locally. (References 6.4.1 and 6.2.18)

3.2.6.3 The BATs shall be supplied with a minimum of 4 wt.% boric acid ($\geq$7000 ppm boron) from either the boric acid batching tank via the boric acid pumps, or the Boron Recycle System via the evaporator package. (Reference 6.4.1)

3.2.6.4 Provisions are provided to allow for a local grab sample of the tank contents. The tank boron concentration must be maintained between 7000 and 7700 ppm to satisfy technical specification requirements. Sampling every 7 days is required. (References 6.4.1, 6.1.18 and 6.7.16)

3.3 ACCUMULATOR TANKS

Q1E21T008A	Q2E21T008A
Q1E21T008B	Q2E21T008B
Q1E21T008C	Q2E21T008C

3.3.1 Basic Functions

3.3.1.1 The three safety injection accumulators shall function as passive safeguards components to rapidly inject borated emergency coolant (equivalent to refueling water chemistry) into the RCS cold leg piping whenever the RCS pressure decreases below the tank cover gas pressure of 601-649 psig due to a loss of coolant accident.

The tanks are of particular importance for intermediate and large break loss of coolant accidents. In the worst of these cases rapid depressurization of the system occurs and the reactor core is completely uncovered. Single failure considerations dictate that the tanks be designed to hold enough liquid such that only two tanks are required to initiate refill of the reactor vessel following the accident. (Reference 6.2.2)

3.3.2 Functional Requirements

3.3.2.1 The accumulator tanks shall have a design pressure of 700 psig. This supports those conditions assumed in the Accident analysis and the Precautions Limitations and Setpoints document. The present accident analysis assumes

Section: **Procedure**

Unit 1 ERP Step: 9 **Unit 2 ERP Step:** 9 ***ERG Step No:*** *8*

ERP StepText: Check if LHSI Pumps should be stopped.

ERG StepText: *Check If Low-Head SI Pumps Should Be Stopped*

Purpose: To stop the low-head SI pumps if RCS pressure is above their shutoff head to prevent damage to the pumps

Basis: Upon safety injection initiation all safeguard pumps are started regardless of the possibility of high RCS pressure with respect to the low-head safety injection pump shutoff head. On low-head systems where the pump recirculates on a small volume circuit there is concern for pump and motor overheating. Shutdown of the pump and placement in the standby mode, when the RCS pressure meets the criteria outlined in this step, allows for future pump operability. If SI has not been previously reset and the low-head SI pumps should be stopped, SI should be reset prior to stopping the pumps. SI can be reset regardless of containment pressure.

Knowledge: This step is a continuous action step.

References: DW-98-015; DW-01-014; DW-07-006

Justification of Differences:

1 Changed to plant specific.

2 Incorporated DW-98-015 and DW-01-014.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: [CA] To ensure proper SI flow to the reactor, the RHR pumps must be manually restarted if they are secured and RCS pressure falls below 275 psig{435 psig}.

****

****

CAUTION: Pump damage may occur if RHR pumps are operated on miniflow for longer than three hours with no CCW supplied to the RHR heat exchangers.

****

___ 9 **[CA] Check if LHSI Pumps should be stopped.**

Action/Expected Response	Response NOT Obtained
9.1 Check RCS pressure - GREATER THAN 275 psig{435 psig} 1C(1A) LOOP RCS NR PRESS [] PI 402B [] PI 403B	9.1 Perform the following. 9.1.1 Establish CCW flow to RHR heat exchangers. CCW TO 1A(1B) RHR HX [] Q1P17MOV3185A open [] Q1P17MOV3185B open 9.1.2 Proceed to Step 11.
9.2 Check RCS pressure - STABLE OR RISING 1C(1A) LOOP RCS NR PRESS [] PI 402B [] PI 403B	9.2 Perform the following. 9.2.1 Establish CCW flow to RHR heat exchangers. CCW TO 1A(1B) RHR HX [] Q1P17MOV3185A open [] Q1P17MOV3185B open 9.2.2 Proceed to Step 10.
9.3 RHR pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST.	9.3 Proceed to step 10

Step 9 continued on next page.

Step	Action/Expected Response		Response NOT Obtained
1	**Monitor RCP criteria.**		
1.1	Greater than 16°F{45°F} subcooled in CETC mode.	1.1	IF HHSI flow greater than 0 gpm, THEN stop all RCPs.
2	**Monitor SI reinitiation criteria.**		
2.1	Greater than 16°F{45°F} subcooled in CETC mode and PRZR level above 13%{43%}.	2.1	Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 6, RE-ESTABLISHING HHSI FLOW.
3	**Monitor FNP-1-EEP-2, FNP-1-EEP-3 & FNP-1-ECP-1.1 branch criteria.**		
3.1	No SG pressure falling in an uncontrolled manner or less than 50 psig.	3.1	IF affected SG NOT previously isolated, THEN go to FNP-1-EEP-2.
3.2	No high secondary radiation or SG level rising uncontrolled.	3.2	Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 6, RE-ESTABLISHING HHSI FLOW. THEN go to FNP-1-EEP-3.
3.3	At least one train ECCS recirculation capability exists.	3.3	Go to FNP-1-ECP-1.1.
4	**Monitor switchover criteria.**		
4.1	RWST level greater than 12.5 ft.	4.1	Go to FNP-1-ESP-1.3.
4.2	CST level greater than 5.3 ft.	4.2	Align AFW pumps suction to SW using FNP-1-SOP-22.0.
5	**Monitor charging miniflow criteria (during SI).**		
5.1	RCS pressure less than 1900 psig.	5.1	Verify miniflow valves open.
5.2	RCS pressure greater than 1300 psig.	5.2	Verify miniflow valves closed.
6	**Monitor adverse containment criteria.**		
6.1	CTMT pressure less than 4 psig and radiation less than 10^5 R/hr.	6.1	Utilize bracketed adverse CTMT condition numbers.

1. 006K6.18 006/NEW//C/A 3.6/3.9/006K6.18/N///

Unit 1 has experienced a Reactor Trip and Safety Injection due to a faulted SG.
The following conditions exist:

- The operating crew is performing EEP-2.0, Faulted Steam Generator Isolation.
- SCMM is in the CETC mode.
- RCS pressure is 1900 psig and rising slowly.
- At the step for verifying SI termination criteria, the crew notes that PT-457, PRZR PRESS, has failed LOW.

Which one of the following completes the statements below?

Subcooling margin calculated by A Train ICCMS will be __(1)__.

Subcooling margin calculated by B Train ICCMS will be __(2)__.

	(1)	(2)
A.	lower	unaffected
B.✔	unaffected	lower
C.	lower	lower
D.	unaffected	unaffected

SOP-68:
3.2 The normal display mode for the SMM is the “CETC” mode. This displays the margin to saturation (°F) using the highest core exit thermocouple (excluding upper head) **and the lowest pressure**. The “RTD” mode displays the margin to saturation (°F) using the hottest reactor coolant system (RCS) RTD (Th or Tc) and the lowest pressure. **The pressure inputs are from PT-402 and 403 and from PT-455 for A-train and PT-457 for B-train.**

Distracter Analysis

A. Incorrect. First part is incorrect (See B.1). Plausible if the applicant thinks that PT-457 inputs to 'A" train SMM.

Second part is incorrect (See B.2). Logical connection to the first part if the applicant recognizes that PT-457 is train related but assumes it inputs to the wrong train.

B. Correct. First part is correct. PT-457 inputs to 'B' Train so the 'A' Train is unaffected.

Second part is correct. PT-457 inputs to 'B' Train and the SMM uses the lowest pressure therefore the subcooling value of the 'B' Train will be lower.

C. Incorrect. First part is incorrect (See B.1). Plausible if the applicant thinks that PT-455 and 457 input to both trains of SMM which is incorrect. If they did input to both trains, this would be a correct answer. PT-402 and 403 input to both trains. This is a common misconception.

Second part is correct (See B.2) A logical connection to the first part if the applicant thinks that PT-455 and 457 input to both trains of SMM instead of PT-402 and 403 which would make this a correct answer.

D. Incorrect. First part is correct (See B.1). Plausible if the applicant thinks that PT-455 inputs to 'A' Train and PT-456 inputs to 'B' Train instead of PT-457 which would make this a correct answer.

Second part is incorrect (See D.1). Logical connection to the first part based on D.1 discussion.

K/A: **006K6.04**	Emergency Core Cooling System (ECCS) - Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Subcooling margin indicators
Importance Rating:	3.6 3.9
Technical Reference:	FNP-1-SOP-68.0, Inadequate Core Cooling Monitoring System, Ver 8.1
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Inadequate Core Cooling Monitor System (OPS-52202E02): • Subcooled Margin Monitor (SMM)
Question History:	NEW
K/A match:	At step 8 of EEP 2, the Shift Supervisor is required to evaluate plant conditions to determine if ECCS flow can be terminated. Part of this determination is evaluating subcooling. The applicant has to **know the effect of the loss of PT-457 on the subcooling margin monitors in order to be able to provide the Shift Supervisor the correct subcooling value.**
SRO justification:	N/A

Step	Action/Expected Response	Response NOT Obtained
___ 8	**Check SI termination criteria.**	
8.1	Check SUB COOLED MARGIN MONITOR indication - GREATER THAN 16°F{45°F} SUBCOOLED IN CETC MODE.	8.1 Proceed to step 9.
8.2	Check secondary heat sink available. • Total feed flow to intact SGs - GREATER THAN 395 gpm. AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229 <u>OR</u> • Narrow range level in at least one intact SG - GREATER THAN 31%{48%}.	8.2 Proceed to step 9.
8.3	Check RCS pressure - STABLE OR RISING. 1C(1A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	8.3 Proceed to step 9.
8.4	Check pressurizer level - GREATER THAN 13%{43%}.	8.4 Proceed to step 9.
8.5	<u>IF</u> all SI termination criteria satisfied, <u>THEN</u> go to FNP-1-ESP-1.1, SI TERMINATION	
___ 9	**Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.**	

-END-

3.2 The normal display mode for the SMM is the "CETC" mode. This displays the margin to saturation (°F) using the highest core exit thermocouple (excluding upperhead) and the lowest pressure. The "RTD" mode displays the margin to saturation (°F) using the hottest reactor coolant system (RCS) RTD (T_h or T_c) and the lowest pressure. The pressure inputs are from PT-402 and 403 and from PT-455 for A-train and PT-457 for B-train. A subcooled margin to saturation is displayed as a positive number and superheat is displayed as a negative number.

3.3 IF any digital display or a REACTOR VESSEL LEVEL mimic LED starts flashing, THEN determine the cause of the alarm per section 4.3.

3.4 Ensure that the Inadequate Core Cooling Monitoring System cabinet cooling fans are operating when the system is in operation.

4.0 Instructions

4.1 System Startup

NOTE:	**Indicate completion of asterisked steps by initialing procedure sign-off list FNP-1-SOP-68.0A.**

*4.1.1 Verify Maintenance has completed FNP-1-STP-300.0, INADEQUATE CORE COOLING MONITORING SYSTEM CALIBRATION (TRAIN A) and FNP-1-STP-301.0, INADEQUATE CORE COOLING MONITORING SYSTEM CALIBRATION (TRAIN B).

*4.1.2 Verify all circuit breakers in back of cabinet are ON and the system has been powered up for at least one hour.

*4.1.3 Verify that the Heated Junction Thermocouple power controllers are producing an output as indicated by the amber light of each controller ON.

*4.1.4 Verify the RUN light on the cabinet front panel is ON.

NOTE:	**In the following step, when the SYSTEM RESET push-button is depressed, the data link is disrupted.**

*4.1.5 Depress the SYSTEM RESET push-button.

1. 007EA2.06 007/MOD/FNP EXAM BANK/C/A 4.3/4.5/EPE007EA2.06/N///

Unit 2 was operating at 2% power with a plant startup in progress per UOP-1.2, Startup of the Unit from Hot Standby to Minimum Load and the following conditions occurred:

At 1000:

- DG-15-2, 2B S/U XFMR TO 2G 4160V Bus, trips open.

At 1005:

- DF-01-2, 2A S/U XFMR TO 2F 4160 V BUS, trips open.

Which one of the following completes the statements below **at 1006** with no operator actions taken?

The Reactor Trip breakers will be __(1)__.

DRPI rod bottom lights __(2)__ be LIT.

	(1)	(2)
A.	OPEN	WILL
B.	OPEN	will NOT
C.✓	CLOSED	WILL
D.	CLOSED	will NOT

FSD-181007
Figure 2 Sheet 2 shows all signals that open the reactor trip breakers and none are present in this scenario.

Unit 2 Load list:
2A CRDM MG Set powered from 600V LC 2D which is power from 4160V 2F.
2B CRDM MG Set powered from 600V LC 2E which is power from 4160V 2G.

Rod Position Indication System (DRPI) has two power sources:
MCC 2D - Normal - is NOT powered from a DG.
MCC 2B - Alternate - IS powered from a DG and is the source which DRPI is NORMALLY aligned.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since the applicant may believe that the loss of power to both trains of safety related power causes the reactor trip breakers to open.

Second part is correct (See D.2). Logical connection to the first part since the rod bottom lights would be lit if the applicant thought the reactor trip breakers opened.

B. Incorrect First part is incorrect (See A.1).

Second part is incorrect (See C.2). Plausible if the applicant believes that Rod Position Indication System (DRPI) is aligned to its normal power supply which would make this a correct answer. DRPI comes off B Train power and will lose power for a time while the 2B DG starts and loads. Then the rod bottom lights will be LIT.

C. Correct. First part is correct. When the loss of the 2F bus occurs, the 2A CRDM MG Set will de-energize causing the rods to fall into the core. No reactor trip setpoints are exceeded at 1006 so the Reactor trip Breakers will not open.

Second part is correct. Rod Position Indication System (DRPI) is normally aligned to its ALTERNATE power supply which is a vital bus. When the DG re-energizes the 2G bus, the rod bottom lights will be LIT.

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See B.2). Logical connection to the first part for two reasons. If the applicant thought there was no trip (rods did not fall into the core) this would be the correct conclusion. If the applicant knew that the trip breakers would not open they could still believe the plausibility of B.2.

K/A: **007EA2.06**	Reactor Trip - Ability to determine or interpret the following as they apply to a reactor trip: Occurrence of a reactor trip
Importance Rating:	4.3 4.5
Technical Reference:	FNP-2-SOP-41.0, Control Rod Drive and Position Indication System, Ver 35.1 A351199, Unit 2 Electrical Load List, Ver 61
References provided:	None
Learning Objective:	RECALL AND DESCRIBE the operation and function of the following reactor trip signals, permissives, control interlocks, and engineered safeguards actuation signals associated with the Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07): • Table 1, Reactor Trip Signals
Question History:	MOD FNP EXAM BANK
K/A match:	The applicant is required to **interpret plant conditions and determine if a reactor trip has occurred**.
SRO justification:	N/A

NOTE

- Normal DRPI power supply is from MCC 2D breaker HDN6L, and the Alternate supply is from MCC 2B breaker HBL7R ☐
- The Alternate supply has a vital source of power and is normally aligned to power DRPI. ☐

4.9 Energizing Control Rod Position Indicating Distribution Panel N2C11L008-N (Aux Bldg 139')

4.9.1 IF **energizing** DRPI from the NORMAL power supply, **perform** the following:

4.9.1.1 **Verify** NORMAL supply breaker, 2D MCC HDN6L CLOSED. ☐

NOTE

Control Rod Position Indicating Distribution Panel N2C11L008-N has a break before make mechanical interlock installed between the distribution panel breakers for the Normal and Alternate supply breakers. ☐

4.9.1.2 At Control Rod Position Indicating Distribution Panel N2C11L008-N, **open** the panel ALTERNATE supply breaker. ☐

4.9.1.3 At Control Rod Position Indicating Distribution Panel N2C11L008-N, **close** the panel NORMAL supply breaker. ☐

4.9.2 IF **energizing** DRPI from the Alternate power supply, **perform** the following:

4.9.2.1 **Verify** Alternate supply breaker, 2B MCC Q2R17BKRHBL7R, CLOSED. ☐

NOTE

Control Rod Position Indicating Distribution Panel N2C11L008-N has a break before make mechanical interlock installed between the distribution panel breakers for the Normal and Alternate supply breakers. ☐

4.9.2.2 At Control Rod Position Indicating Distribution Panel N2C11L008-N, **open** the panel NORMAL supply breaker. ☐

4.9.2.3 At Control Rod Position Indicating Distribution Panel N2C11L008-N, **close** the panel ALTERNATE supply breaker. ☐

CRDM MG Sets power supplies

2F 4160V BUS **AB - 139'** **D-207005**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0006-A	**2F 4160V BUS**	
DF01	**N2R11A0501-N**	**2A STARTUP TRANSFORMER (NORMAL) <<<**	
DF02	**Q2R15A0505-A**	**2K 4160V BUS >>>**	K-1
DF03	**Q2R11B0004-A**	**2D 4160/600V SST >>> ED02 (NORMAL) >>>**	F-2
DF04	Q2P17M0001C-A	2C CCW PUMP	
DF05	Q2P17M0001B-AB	2B CCW PUMP DISC SWITCH Q2R18A0004A-A >>> 2B CCW PUMP (A TRAIN SUPPLY)	
DF06	Q2E21M0001A-A	2A CHARGING/HHSI PUMP	
DF07	Q2E21M0001B-AB	2B CHG PUMP DISC SWITCH Q2R18A0001A-A >>> 2B CHARGING/HHSI PUMP (A TRAIN SUPPLY)	
DF08	**QSR43A0501-A**	**1-2A DIESEL GENERATOR (EMERG) <<<**	
DF09	Q2E11M0001A-A	2A RHR/LHSI PUMP	
DF10	Q2N23M0001A-A	2A AFW PUMP	
DF11	Q2E13M0001A-A	2A CTMT SPRAY PUMP	
DF12	**Q2R16B0008-AB**	**2F 4160/600V SST DISC SW Q2R18A0003A-A >>> 2F 4160/600V SST >>> 2F LOAD CENTER (A TRAIN SUPPLY) >>>**	F-108
DF13	**Q2R15A0503-A**	**2H 4160V BUS >>>**	H-1
DF14	Q2R15BKRDF14	PT COMPARTMENT	
DF15	**N2R11A0502-N**	**2B STARTUP TRANSFORMER (ALT) <<<**	

DF03
2D 600V LOAD CENTER AB - 139' D207010

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R16B0006-A	**2D 600V LOAD CENTER**	
ED01	Q2R16BKRED01	PT COMPARTMENT	
ED02	**Q2R11B0004-A**	**2D 4160/600V SST (NORMAL) <<< DF03**	
ED03	N2C11M0001A-N	2A CRDM MG SET	
ED04	**Q2R42E0001A-A**	**2A BATTERY CHARGER >>> 2A 125VDC SWGR >>>**	F-3
ED05	**QSR17B0006-A**	**1F 600V MCC >>>**	UNIT 1 F-83
ED06	**Q2R17B0098**	**2CC 600V MCC >>>**	F-83
ED07	-----------	SPACE	
ED08	**Q2R16B0002-A**	**2A 600V LC (ALT-EMERG) >>> EA09 >>>**	D-46
ED09	Q2R42E0001C-AB	2C BATTERY CHARGER (A TRAIN SUPPLY)	
ED10	**Q2R17B0001-A**	**2A 600V MCC >>>**	F-91
ED11	N2T47M0001A-A	2A CTMT PENETRATION CABINET Q2R18B022-A (SEE APPENDIX 1) >>> 2A CRDM COOLER FAN	
ED12	**Q2R16B0008-AB**	**2F 600V LC (ALTERNATE) <<< EF06**	
ED13	**Q1R17B0509-A**	**1S 600V MCC >>>**	UNIT 1 F-102
ED14	**Q2R17B0008-A**	**2U 600V MCC >>>**	F-98
ED15	Q2E12M0001A-A	2A CTMT PENETRATION CABINET Q2R18B022-A (SEE APPENDIX 1) >>> 2A CTMT COOLER (EMERG./ LOW SPEED)	
ED16	Q2E12M0001B-A	2A CTMT PENETRATION CABINET Q2R18B022-A (SEE APPENDIX 1) >>> 2B CTMT COOLER (EMERG./LOW SPEED)	
ED17	------------	SPACE	

2G 4160V BUS AB - 121' D-207006

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0007-B	2G 4160V BUS	
DG01	N2R11A0501-N	2A STARTUP TRANSFORMER <<<	
DG02	Q2R15A0506-B	2L 4160V BUS >>>	L-1
DG03	Q2R11B0005-B	2E 4160/600V SST >>> EE02	G-2
DG04	Q2P17M0001A-B	2A CCW PUMP	
DG05	Q2P17M0001B-AB	2B CCW PUMP DISC SWITCH Q2R18A0004B-B >>> 2B CCW PUMP (B TRAIN SUPPLY)	
DG06	Q2E21M0001C-B	2C CHARGING/HHSI PUMP	
DG07	Q2E21M0001B-AB	2B CHG PUMP DISC SWITCH Q2R18A0001B-B >>> 2B CHARGING/HHSI PUMP (B TRAIN SUPPLY)	
DG08	Q2R43A0505-B	2B DIESEL GENERATOR <<<	
DG09	Q2E11M0001B-B	2B RHR/LHSI PUMP	
DG10	Q2N23M0001B-B	2B AFW PUMP	
DG11	Q2E13M0001B-B	2B CTMT SPRAY PUMP	

DG12	Q2R11B0006-AB	2F 4160/600V SST DISC SW Q2R18A0003B-B
		>>> 2F 4160/600V SST
		>>> 2F LOAD CENTER (B TRAIN SUPPLY)>>>
			F-108

DG13	Q2R15A0504-B	2J 4160V BUS >>>	J-1
DG14	Q2R15BKRDG14	PT COMPARTMENT	
DG15	N2R11A0502-N	2B STARTUP TRANSFORMER <<<	

DG03
2E 600V LOAD CENTER AB - 121' D-207011

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R16B0007-B	**2E 600V LOAD CENTER**	
EE01	Q2R16BKREE01	PT COMPARTMENT	
EE02	**Q2R11B0005-B**	**2E 4160/600V SST (NORMAL) <<< DG03**	
EE03	N2C11M0001B-N	2B CRDM MG SET	
EE05	**Q2R42E0001B-B**	**2B BATTERY CHARGER >>> 2B 125VDC SWGR**	G-3
EE06	Q2R42E0001C-AB	2C BATTERY CHARGER (B TRAIN SUPPLY)	
EE07	**Q2R16B0005-B**	**2C 600V LC (ALT-EMERG) >>> EC10**	E-5
EE08	Q2E12M0001C-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2C CTMT COOLER (EMERG. / LOW SPEED)	
EE09	**QSR17B0007-B**	**1G 600V MCC >>>**	UNIT 1 G-88
EE10	**Q2R17B0002-B**	**2B 600/208V MCC >>>**	G-71
EE11	**Q2R17B0099**	**2DD 600/208V MCC >>>**	G-78
EE12	**Q2R16B0008-AB**	**2F 600V LC (ALTERNATE) <<< EF08**	
EE13	N2T47M001B-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2B CRDM COOLER FAN	
EE14	**Q2R17B0510-B**	**2T 600/208V MCC >>>**	G-86
EE15	**Q2R17B0009-B**	**2V 600/208V MCC >>>**	G-93
EE16	Q2E12M0001D-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2D CTMT COOLER (EMERG./ LOW SPEED)	

DRPI Power supplies.

2D 4160V BUS **TB-155'** **D207016**

This is the NORMAL Power Supply and is NOT reenergized from a DG.

BKR	TPNS	DESCRIPTION	SEE PAGE
	N2R15A0501-N	**2D 4160V BUS**	
DD01	**N2R12A0501-N**	**2A UNIT AUX TRANSFORMER (ALTERNATE) <<<**	
DD02	N2R15BKRDD02	PT COMPARTMENT	
DD03	**N2R11A0501-N**	**2A START-UP TRANSFORMER (NORMAL) <<<**	
DD04	**N2R11B0516-N**	**2U 4160/600V SST >>> EU02**	D-2
	N2R11B0514-N	**2W 4160/600V SST >>> EW02**	D-7
	N2R11B0512-N	**2Y 4160/600V SST >>> EY02**	D-15
DD05	**N2R11B0508-N**	**2P 4160/600V SST >>> EP02 (NORMAL)**	D-20
DD06	N2N21M0001B-N	2B CONDENSATE PUMP	
DD07	N2N26M0001A-N	2A HEATER DRAIN PUMP	
DD08	**Q2R11B0001-N**	**2A 4160/600V SST >>> EA02 (NORMAL)**	D-46
DD09	**N2R11B0002-N**	**2B 4160/600V SST >>> EB02 (NORMAL)**	D-82
DD10	**N2R11B0008-N**	**2M 4160/600V SST >>> EM02 (NORMAL)**	D-120
DD11	NSY36M0501B-N	4160/600V TRANS. (NSR12E509A-N)>>>DEEP WELL PUMP #2	
	NSR19L0541B-N	**4160/600V TRANS. (NSR12E509A-N)>>> 600/120V TRANS (NSR12E507B-N) >>> DEEP WELL PUMP #2 DISTR. PNL. >>>**	D-123
	NSY34G0502-N	4160/120-240V TRANS. (NSR12E0512-N) >>> BACK-UP MET TOWER (BY APCO)	
	NSY34L0501-N N1R19L0565-N NSY34L0500-N	4160/120-240V TRANS. (NSR12E0512-N) >>> MICROWAVE BLDG DIST PNL (NSY34L0501) >>> MET ROOM PNL (N1R19L0565) & MICROWAVE ROOM PNL (NSY34L0500)	
DD12	**N2R11B0520-N**	**2I 4160 - 277/480V LTG XFMR >>> UA02 (NORMAL)**	D-124
DD13	N2U41M0506A-N	2A CENTRIFUGAL WATER CHILLER COMPRESSOR	

Ver 61.0

DD09
2B 600V LOAD CENTER AB - 121' D207008

BKR	TPNS	DESCRIPTION	SEE PAGE
	N2R16B0003-N	**2B 600V LOAD CENTER**	
EB01	N2R165BKREB01	PT COMPARTMENT	
EB02	**N2R11B0002-N**	**2B 4160/600V SST <<< DD09 (NORMAL)**	
EB03	N2G12L0001B-N	2B BTRS CHILLER UNIT PANEL	
EB04	**N2R17B0004-N**	**2D 600/208V MCC >>>**	D-83
EB05	Q2E12M0001B-A	2A CTMT PENETRATION CABINET Q2R18B022-A (SEE APPENDIX 1) >>> 2B CONTAINMENT COOLER (NORMAL/HIGH SPEED)	
EB06	Q2E12M0001C-B	2E CTMT PENETRATION CABINET Q2R18B026-B (SEE APPENDIX 1) >>> 2C CONTAINMENT COOLER (NORMAL/HIGH SPEED)	
EB07	**Q2R16B0008-AB**	**2F 600V LOAD CENTER <<< EF09 (ALTERNATE)**	
EB08	**N2V51E0003D-N**	**LIGHTING XFMR 2C >>> 2D, 2G, & 2J LIGHTING PANELS >>>**	D-105, D-108 & D-110
EB09	**N2V51E0003A-N**	**LIGHTING XFMR 2D >>> 2M, 2O*, & 2R* LIGHTING PANELS >>>** ***VIA LIGHTING CONTROL PANELS 2O & 2R**	D-113, D-117 & D-119
EB10	N2E21M0007-N	HYDRO TEST PUMP	
EB11	N2P41M0001B-N	2B AUX BUILDING MAIN EXHAUST FAN	

DD09
EB04
2D 600/208V MCC **AB - 121'** **B207556 Sh. 4-4B**
(CONT'D)

BKR	TPNS	DESCRIPTION	SEE PAGE
FDG5	N2P13M0004-N	MINI PURGE EXHAUST FAN	
FDG6	------------	SPARE	
FDH3	Q2G22M0001A-N	2A WASTE GAS COMPRESSOR PKG PUMP	
FDH4	------------	SPARE	
FDH5	------------	SPARE	
FDH6L	N2V31G0005-N	DISC SWITCH N2R18B014-N >>> FILTER MONORAIL HOIST CABLE REEL	
FDH6R	N2E21K0002-N	BORON CONCENTRATION MEASUREMENT SYSTEM	
FDH7L	------------	SPARE	
FDH7R	N2T31K0005-N	2G CTMT PENETRATION CABINET Q2R18B028-B (SEE APPENDIX 1) >>> CTMT JIB CRANE	
FDJ4L	------------	SPARE	
FDJ4R	**N2R12E0001B-N**	**2DD 600-208Y/120V AUX DIST XFMR >>> 2DD 208/120VAC CONT PWR PANEL**	D-95
FDJ5L	**N2R17E0001-N**	**2D 600/208V MCC TRANSFORMER >>> 2D MCC 208V SECTION**	D-96
FDM2	------------	SPARE	
FDM3	N2G21M0002-N	WASTE EVAP CONDENSATE TANK PUMP	
FDM4	N2V46M0008-N	WASTE GAS AREA FILTRATION UNIT	
FDM5	N2T09M0001B-N	2B TENDON ACCESS GALLERY SUMP PUMP	
FDM6	------------	SPARE	
FDM7L		600V WELDING RECEPTACLES EL. 121' (N2TB120)	
FDM7R		600V WELDING RECEPTACLES EL. 121' (N2TB122)	

DD09
EB04
FDJ5L
2D 208V MCC SECTION AB - 121' B207556 Sh. 4C-4D

BKR	TPNS	DESCRIPTION	SEE PAGE
	N2R17B0004-N	**2D 600/208V MCC (208V SECTION) <<< FDJ5L**	
HDK2L	N2V46C0017-N	SAMPLING ROOM AHU	
HDK2R	N2F15M0005-N	ROD CONT CLUSTER ASSY CHANGE FIXTURE HOIST DRIVE	
HDK3	N2P41M0002-N	CTMT ELEVATOR MACHINE ROOM PROP FAN	
HDK4L	N2V46K0006C-N	2C ELEC PENE ROOM CONDENSING UNIT	
HDK4R	------------	SPARE	
HDK5	N2V47C0023-N	AUX BUILDING LUBE OIL STORAGE AREA SUPPLY FAN	
HDK6	------------	SPARE	
HDK7	------------	SPARE	
HDL2	------------	SPARE	
HDL3	------------	SPARE	
HDL4	------------	SPARE	
HDL5	N2E21M0008A-N	2A CHG/HHSI PUMP AUX LUBE OIL PUMP	
HDL6	------------	SPARE	
HDL7	N2E21M0009-N	BORIC ACID BATCHING TANK AGITATOR	
HDN2	N2V47M0004-N	CABLE SPREADING ROOM AHU	
HDN3	**N2P15L0502-N**	**ION CHROM REG XFMR N2P15E0502-N>>> REG DIST CAB >>>**	D-98
HDN4L	N1P15NFSSS2613B-N	SAMPLE SYSTEM CHILLED WATER UNIT	
HDN4R	N2P15G0001-N	SAMPLE SYSTEM CONDENSATE RETURN UNIT	
HDN5	N2G21MOV3394-N	CTMT SUMP PUMP DISCHARGE MOV	
HDN6L	**N2C11L0008-N**	**ROD POSITION INDICATION DIST PANEL (NORMAL SOURCE) >>>**	D-99
HDN6R	**N2R19L0001G-N**	**2G SPACE HEATERS DIST PANEL >>>**	D-100
HDO2	N2V47M0009-N	2M 600V LOAD CENTER AHU	
HDO3	------------	SPACE HEATER SUPPLY	
HDO4L	**N2R19L0003N-N**	**2N 208/120V AC CONTROL POWER PANEL >>>**	D-101

This is the alternate power supply and IS energized by a DG.

2G 4160V BUS AB - 121' D-207006

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0007-B	**2G 4160V BUS**	
DG01	**N2R11A0501-N**	**2A STARTUP TRANSFORMER <<<**	
DG02	**Q2R15A0506-B**	**2L 4160V BUS >>>**	L-1
DG03	**Q2R11B0005-B**	**2E 4160/600V SST >>> EE02**	G-2
DG04	Q2P17M0001A-B	2A CCW PUMP	
DG05	Q2P17M0001B-AB	2B CCW PUMP DISC SWITCH Q2R18A0004B-B >>> 2B CCW PUMP (B TRAIN SUPPLY)	
DG06	Q2E21M0001C-B	2C CHARGING/HHSI PUMP	
DG07	Q2E21M0001B-AB	2B CHG PUMP DISC SWITCH Q2R18A0001B-B >>> 2B CHARGING/HHSI PUMP (B TRAIN SUPPLY)	
DG08	**Q2R43A0505-B**	**2B DIESEL GENERATOR <<<**	
DG09	Q2E11M0001B-B	2B RHR/LHSI PUMP	
DG10	Q2N23M0001B-B	2B AFW PUMP	
DG11	Q2E13M0001B-B	2B CTMT SPRAY PUMP	

DG12	**Q2R11B0006-AB**	**2F 4160/600V SST DISC SW Q2R18A0003B-B
		>>> 2F 4160/600V SST
		>>> 2F LOAD CENTER (B TRAIN SUPPLY)>>>**
			F-108

DG13	**Q2R15A0504-B**	**2J 4160V BUS >>>**	J-1
DG14	Q2R15BKRDG14	PT COMPARTMENT	
DG15	**N2R11A0502-N**	**2B STARTUP TRANSFORMER <<<**	

DG03
2E 600V LOAD CENTER **AB - 121'** **D-207011**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R16B0007-B	**2E 600V LOAD CENTER**	
EE01	Q2R16BKREE01	PT COMPARTMENT	
EE02	**Q2R11B0005-B**	**2E 4160/600V SST (NORMAL) <<< DG03**	
EE03	N2C11M0001B-N	2B CRDM MG SET	
EE05	**Q2R42E0001B-B**	**2B BATTERY CHARGER >>> 2B 125VDC SWGR**	G-3
EE06	Q2R42E0001C-AB	2C BATTERY CHARGER (B TRAIN SUPPLY)	
EE07	**Q2R16B0005-B**	**2C 600V LC (ALT-EMERG) >>> EC10**	E-5
EE08	Q2E12M0001C-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2C CTMT COOLER (EMERG. / LOW SPEED)	
EE09	**QSR17B0007-B**	**1G 600V MCC >>>**	UNIT 1 G-88
EE10	**Q2R17B0002-B**	**2B 600/208V MCC >>>**	G-71
EE11	**Q2R17B0099**	**2DD 600/208V MCC >>>**	G-78
EE12	**Q2R16B0008-AB**	**2F 600V LC (ALTERNATE) <<< EF08**	
EE13	N2T47M001B-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2B CRDM COOLER FAN	
EE14	**Q2R17B0510-B**	**2T 600/208V MCC >>>**	G-86
EE15	**Q2R17B0009-B**	**2V 600/208V MCC >>>**	G-93
EE16	Q2E12M0001D-B	2D CTMT PENETRATION CABINET Q2R18B025-B (SEE APPENDIX 1) >>> 2D CTMT COOLER (EMERG./ LOW SPEED)	

DG03
EE10
2B 600/208V MCC **AB - 121'** **B207556 Sh. 2-2C**
(CONT'D)

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
FBK2	------------	SPARE	
FBK5L	**------------**	**2B 600/208V MCC TRANSFORMER >>> 2B MCC 208V SECTION**	G-77
FBM2	------------	SPARE	
FBM3	Q2P12M0001B-B	2B REACTOR MAKEUP WATER PUMP	
FBM4L	N2E15K0002A-N	2A PENETRATION FILTER ME ROOM UNIT HEATER	
FBM5	Q2E16M0001B-AB	2B CHARGING/HHSI PUMP ROOM COOLER FAN	
FBO2	N2T49MOV3310B-N	2F CTMT PENETRATION CABINET Q2R18B027-B (SEE APPENDIX 1) >>> REACTOR CAVITY COOL FAN MOV	
FBO3	------------	SPARE	
FBO4	------------	SPARE	

DG03
EE10
FBK5L
2B 208V MCC SECTION AB - 121' B207556 Sh. 2D

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R17B0002-B	**2B 600/208V MCC (208V SECTION) <<< FBK5L**	
HBL2	Q2V47M0016B-B	2B BATTERY ROOM EXHAUST FAN	
HBL3L	Q2D11RE0024B-B	CTMT PURGE MONITOR	
HBL3R	N2D11RE0011-N N2D11RE0012-N	CTMT AIR PARTICLE DET MONITOR	
HBL4L	------------	SPARE	
HBL4R	------------	SPARE	
HBL5L	Q2D11RE0025B-B	SPENT FUEL POOL MONITOR	
HBL6L	------------	SPARE	
HBL7L	------------	SPARE	
HBL7R	**N2C11L0008-N**	**ROD POSITION IND DISTRIBUTION PANEL (ALT. SOURCE) >>>**	D-99
HBN2	Q2E15M0001B-B	2B PENETRATION ROOM EXHAUST FAN	
HBN3		SPACE HEATER SUPPLY	
HBN4	------------	SPARE	
HBN5	------------	SPARE	
HBN6	N2D11M0001-B	CTMT AIR SAMPLE VACUUM PUMP	
HBN7	Q2E22M0001B-B	2F CTMT PENETRATION CABINET Q2R18B027-B (SEE APPENDIX 1) >>> REAC CAVITY H2 DILUTION FAN 2B	

QUESTIONS REPORT

for 007EA2.06 FNP

1. ROD CONT-62201E02 001/HLT/SRO/C/A /3.9/001A2.10////

A Unit 2 plant startup is in progress per FNP-2-UOP-1.2. Reactor power is being raised in preparation for rolling the main turbine to synchronous speed. With reactor power at 8% a loss of power occurs to the 2F 4160V bus. The appropriate Diesel Generators start and align to the 2F 4160V bus.

While the crew is responding to the event, DG15, 2B S/U XFMR to 2G 4160V Bus, trips open. The associated DG trips and does not align to the 2G 4160V bus.

Which ONE of the following will be the status of the reactor trip breakers <u>and</u> the reason after the 2G 4160V bus power is lost?

A. The reactor trip breakers will be open.
All 3 RCPs breaker position indications will be lost and a Loss of Flow Reactor Trip will occur.

B. The reactor trip breakers will be open.
A loss of power to the CRDM MG sets will generate a negative rate reactor trip signal as the rod drop into the core.

C. The reactor trip breakers will NOT be open.
All rods will be on the bottom of the core due to loss of power to the CRDM MG sets but no unblocked automatic reactor trip signal will be present .

D. The reactor trip breakers will NOT be open.
All rods will be in the same position as before the loss of power since no unblocked automatic reactor trip signal will be present.

1. 007K5.02 008/MOD/FNP 12/C/A 3.1/3.4/007K5.02/N///

Unit 1 is in Mode 5 and forming a pressurizer steam space (drawing a bubble) per UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby. The vacuum refill procedure will NOT be performed. The following conditions exist:

- RCS pressure is being maintained at 325-375 psig.
- 1B RCP is running.
- 'A' Train RHR is on service with low pressure letdown aligned.
- RCS is in solid plant pressure control.
- Pressurizer temperature is 178°F and slowly rising.
- All PRZR heaters have been energized.

Which one of the following completes the statements below?

Per UOP-1.1, the pressurizer is at saturation conditions when _(1)_ increases.

During this evolution, PRT level will _(2)_.

	(1)	(2)
A.	charging flow	remain constant
B.✓	letdown flow	remain constant
C.	letdown flow	rise
D.	charging flow	rise

UOP-1.1:
5.11 WHEN pressurizer temperature increases to the saturation temperature for 375 psig (approximately 442°F) as indicated by **increasing RCS pressure or letdown flow**, THEN **establish** a steam space in the pressurizer as follows

5.11.5 WHEN VCT level increases to 81%, THEN **verify** VCT HI LVL DIVERT VLV Q1E21LCV115A in the fully diverted position.

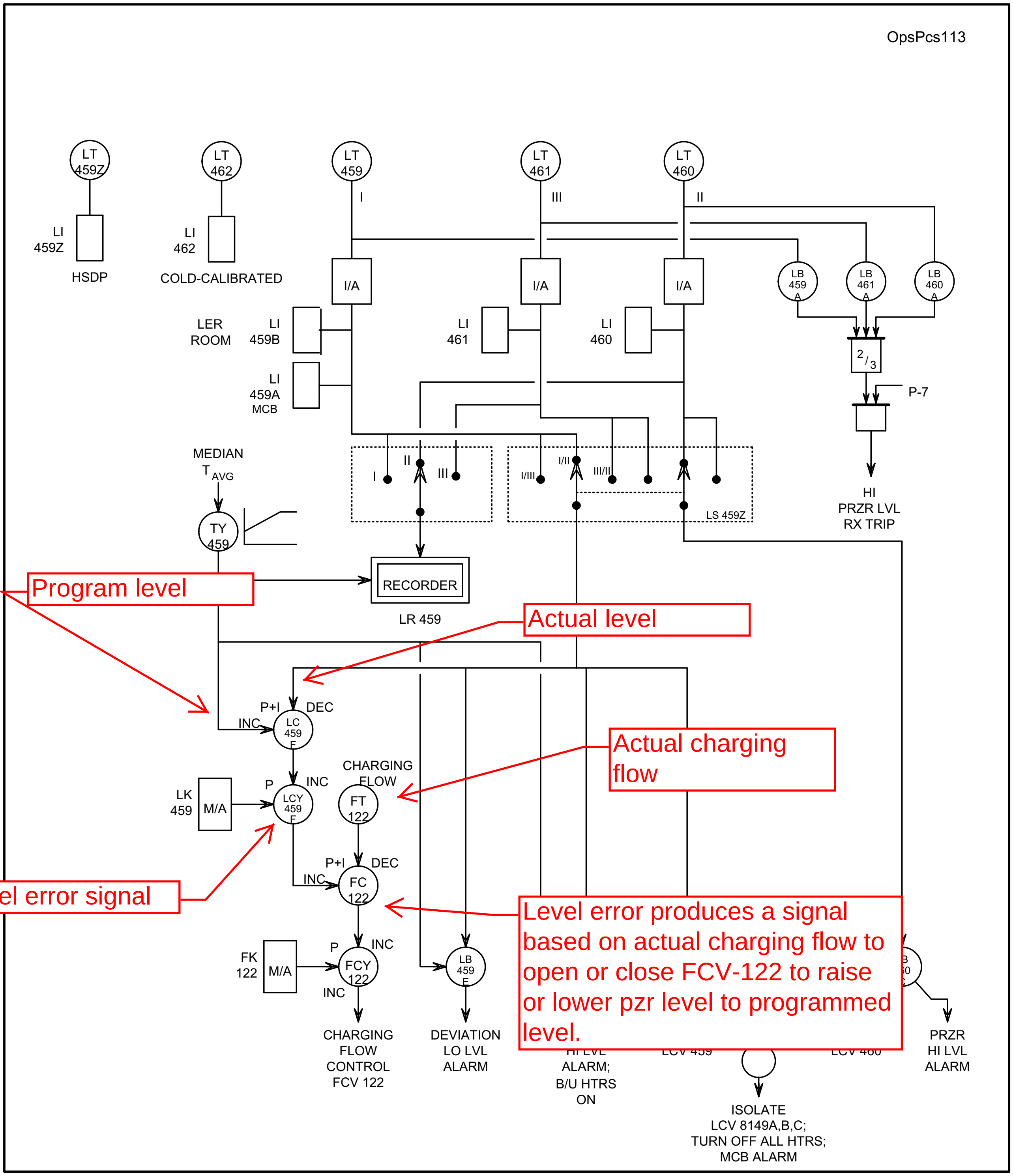
Distracter analysis

A. Incorrect.	First part is incorrect (See B.1). Plausible if the applicant thinks that FCV-122 is in auto and will open to raise charging flow in response to the RCS pressure rise. FCV-122 operates in automatic based on pressurizer level and median Tavg (See AOP-100 Section 1.2 Figure 1 in reference material) Second part is correct (See B.2).
B. Correct.	First part is correct. UOP-1.1: 5.11 WHEN pressurizer temperature increases to the saturation temperature for 375 psig (approximately 442°F) as indicated by **increasing** RCS pressure or **letdown flow**, THEN establish a steam space in the pressurizer as follows: Second part is correct. LCV-115A diverts to the RHT.
C. Incorrect.	First part is correct (See B.1). Second part is incorrect (See above). Plausible if the applicant improperly believes that letdown diverts to the PRT vice RHT.
D. Incorrect.	First part is incorrect (See A.1). Second part is incorrect (See C.2).

K/A: **007K5.02**	Pressurizer Relief Tank/Quench Tank System (PRTS) - Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR
Importance Rating:	3.1 3.4
Technical Reference:	FNP-1-UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby, Ver 94.3
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Pressurizer System components and equipment, to include the following (OPS-40301E07): • Normal Control Methods
Question History:	MOD FNP 12
K/A match:	The applicant has to know the **effect on the PRT level (operational implications) during the formation of a steam bubble in the pressurizer**.
SRO justification:	N/A

SECTION 1.2

Figure 1



PRESSURIZER LEVEL PROTECTION AND CONTROL

4.8 IF RHR will be used to maintain cavity level, THEN manually **throttle** open RHR to RCS Hot Legs Iso, Q1E11MOV8889 (Q1E11V044), approximately 15 – 20 turns. _____

4.9 To **maintain** RCS inventory during the flush using off service train of RHR, **perform** the following while continuing with Step 4.10: _____

4.9.1 IF 1A RHR pump is the off service train, THEN **perform** the following: _____

4.9.1.1 WHEN ready to initiate makeup flow to the RCS, THEN **open** RWST to 1A RHR Pump, Q1E11MOV8809A. _____

4.9.1.2 **Perform** any of the following as necessary to control level:

- Cycle MOV8809A or MOV8887A. _____
- Adjust MOV8889. _____

4.9.2 IF 1B RHR pump is the off service train, THEN **perform** the following: _____

4.9.2.1 WHEN ready to initiate makeup flow to the RCS, THEN **open** RWST to 1B RHR Pump, Q1E11MOV8809B. _____

4.9.2.2 **Perform** any of the following as necessary to control level:

- Cycle MOV8809B or MOV8887B. _____
- Adjust MOV8889. _____

NOTE

Actions performed while letdown is aligned to the RHT's should be performed as expeditiously as possible to minimize loss of RCS inventory to the RHT's. ☐

4.10 **Place** VCT HI LVL DIVERT VLV, Q1E21LCV115A, in the RHT position. _____

4.11 **Throttle open** the following in MANUAL to establish low pressure letdown:

- RHR TO LTDN HX HIK 142 ☐ ☐ ☐ ☐ ☐
- LP LTDN PRESS PK 145 ☐ ☐ ☐ ☐ ☐

4.12 **Place** Q1E21TCV143 in the DEMIN position to establish flow through 1B demineralizer. ☐ ☐ ☐ ☐ ☐

5.2.10 WHEN RCS pressure greater than 100 psig, THEN **open** RCP SEAL LEAKOFF valves:

- Q1E21HV8141A ______
- Q1E21HV8141B ______
- Q1E21HV8141C ______

_______ / _______

> **CAUTION**
> Throttling RHR TO LTDN HX HIK 142 excessively can cause loss of ability to control pressure transients with LP LTDN PRESS PK-145. ☐

5.2.11 **Adjust** the following valves as necessary to maintain RCS pressure 325-375 psig.

- LP LTDN PRESS PK 145 _______
- RHR TO LTDN HX HIK-142 _______

5.2.12 **Perform** the following for the RCP to be started:

5.2.12.1 **Verify** RCP No. 1seal ∆P greater than 210 psid on the following indicators:

5.2.12.2 1A RCP #1SEAL PRESS PI-156A ______

5.2.12.3 1B RCP #1SEAL PRESS PI-155A ______

5.2.12.4 1C RCP #1SEAL PRESS PI-154A ______

_______ / _______

5.2.12.5 **Verify** RCP No. 1 seal leak rate is within the limits of Figure 2.

- RCP SEAL LKOF HIGH RANGE recorder N1E21FR154A ______
- 1A RCP SEAL LEAKOFF LOW RANGE indication FI-156B ______
- 1B RCP SEAL LEAKOFF LOW RANGE indication FI-155B ______
- 1C RCP SEAL LEAKOFF LOW RANGE indication FI-154B ______

_______ / _______

NOTE

Continue with plant heat-up to 190-200°F while performing step 5.11. ☐

5.11 WHEN pressurizer temperature increases to the saturation temperature for 375 psig (approximately 442°F) as indicated by increasing RCS pressure or letdown flow, THEN **establish** a steam space in the pressurizer as follows:

5.11.1 **Decrease** charging flow to minimum. _______

5.11.2 **Adjust** SEAL WTR INJECTION HIK 186 as required to maintain 6-13 gpm seal flow to each RCP. _______

5.11.3 **Operate** pressurizer heaters and/or spray valves to maintain RCS pressure 325-375 psig.

- 1A PRZR HTR GROUP BACKUP ON-AUTO-OFF _______
- 1B PRZR HTR GROUP BACKUP ON-AUTO-OFF _______
- 1C PRZR HTR GROUP VARIABLE ON-AUTO-OFF _______
- 1D PRZR HTR GROUP BACKUP ON-AUTO-OFF _______
- 1E PRZR HTR GROUP BACKUP ON-AUTO-OFF
- 1B LOOP SPRAY VLV PK 444D
- 1A LOOP SPRAY VLV PK 444C

SOP-2.1 (in reference folder) shows that LCV-115A diverts to the RHT

5.11.4 **Adjust** LP LTDN PRESS PK 145 to maintain approximately 120 gpm. _______

5.11.5 WHEN VCT level increases to 81%, THEN **verify** VCT HI LVL DIVERT VLV Q1E21LCV115A in the fully diverted position. ______

______ / ______

5.11.6 IF desired, THEN **place** excess letdown in operation to expedite formation of a steam space in the pressurizer per FNP-1-SOP-2.7, Chemical and Volume Control System Excess Letdown. ______

5.11.7 WHEN pressurizer level reaches 21%, THEN **perform** the following:

5.11.7.1 **Place** CHG FLOW FK-122 in AUTO. ______

______ / ______

5.11.7.2 **Verify** pressurizer level is maintained at approximately 21%. ______

Charging was in manual prior to drawing a bubble.

QUESTIONS REPORT

for 007K5.02 FNP 12

1. 007K5.02 011/FNP BANK/FNP 2007/C/A 3.1/3.4/007K5.02/N///

Unit 1 is solid in Mode 5, preparing to form a pressurizer steam space (drawing a bubble). The following conditions exist:

- UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby, is in progress.
- Vacuum refill will **NOT** be performed.
- RCS pressure is 325-375 psig and ↔.
- 1B RCP is running.
- 'A' Train RHR is in service with low pressure letdown aligned.
- RCS is in solid plant pressure control with pressurizer temperature at 178°F.
- All PRZR heaters have been energized.

Which one of the following completes the statements below per UOP-1.1?

The condition that is monitored to demonstrate the PRZR is saturated is __(1)__ .

As the bubble is being formed, PRT level will __(2)__ .

	(1)	(2)
A.	Letdown flow lowering	remain constant
B.✓	RCS Pressure rising	remain constant
C.	RCS Pressure rising	slowly rise
D.	Letdown flow lowering	slowly rise

1. 008AK1.01 009/MOD/HARRIS 09/C/A 3.2/3.7/APE008AK1.01/Y///

Unit 2 was operating at 100% power when a Reactor Trip occurs and the following conditions exist:

- Q2B13PSV8010A, PZR SAFETY, has failed OPEN.
- Pressurizer pressure is 1020 psig.
- PRT pressure rises to 55 psig.
- Core Exit Thermocouples read 560°F.

Which one of the following completes the statements below?

Temperature on TI-469, SAFETY VLVS, will indicate approximately __(1)__.

Pressurizer level will be __(2)__.

Reference provided

	(1)	(2)
A.	546°F	rising
B.	546°F	lowering
C.	320°F	lowering
D.✓	320°F	rising

WOG Executive Guideline - During situations where a steam vent path is established from the pressurizer vapor space and where RCS subcooling is not indicated, pressurizer level may not be a true indication of RCS inventory. This can result from steam generated in the reactor vessel, passing through the pressurizer surge line and preventing the water inventory of the pressurizer from draining into the RCS loops. This holdup of water can result in a stable or even increasing indicated pressurizer level while RCS water inventory is actually decreasing. Pressurizer level should be relied on only with hot leg or core exit subcooling present. In SI termination steps in the ERGs, pressurizer level is only checked after adequate RCS subcooling is confirmed.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible if applicant believes that the temperature of the steam in the Pressurizer is the same temperature as the steam entering the PRT. 546°F is the approximate saturation temperature for 1035 psia. This was the error made at the TMI accident.

Second part is correct (See D.2).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See D.2). This is initially true but in the scenario given, subcooling is lost in the core and a bubble is formed in the vessel upper head. This will result in the Pzr level rising instead of lowering as one would expect. Plausible since during a LOCA event the normal response is that Pzr level decreases.

C. Incorrect. First part is correct. (See D.1)

Second part is incorrect (See B.2).

D. Correct. First part is correct. Using the steam tables and the following pressures:

1020 psig + 15 = 1035 psia (RCS)
55 psig + 15 = 70 psia (PRT)

~320°F

Second part is correct. Since the break is at the top of the pressurizer, the pressurizer level will be rising. This scenario was run on the desktop simulator. Pressurizer level was rising at 1020 psig.

K/A: **008AK1.01**	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) - Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves
Importance Rating:	3.2 3.7
Technical Reference:	Properties of saturated and superheated steam, 1967 Westinghouse Owners Group, ERG Executive Guideline.
References provided:	Steam tables
Learning Objective:	SELECT AND ASSESS the Pressurizer System instrument/equipment response expected when performing Pressurizer System evolutions, including the Normal Condition, the Failed Condition, Associated Alarms, Associated Trip Setpoints, to include the components found on Figure 3, Pressurizer and Pressurizer Relief Tank (OPS-52101E07)
Question History:	MOD HARRIS 09
K/A match:	Applicant has to **determine the safety valve tailpeice temperature using steam tables (Thermodynamics and flow characteristics)** and **pressurizer level trend (operational implications) during a stuck open PORV condition.**
SRO justification:	N/A

to mitigate a potential Critical Safety Function challenge, reactor vessel level (RVLIS) is used. In all cases within the ERGs, one or the other is used to check for adequate RCS inventory.

Pressurizer level indicating on span including instrument uncertainties is used in E-0, E-1, E-3 and ECA-2.1. For E-0, E-1 and ECA-2.1, if pressurizer level is not on span, but all other SI Termination Criteria are satisfied, then pressurizer spray can be used to stabilize pressure. For E-3, pressurizer level should have already been established through a previous step.

During situations where a steam vent path is established from the pressurizer vapor space and where RCS subcooling is not indicated, pressurizer level may not be a true indication of RCS inventory. This can result from steam generated in the reactor vessel, passing through the pressurizer surge line and preventing the water inventory of the pressurizer from draining into the RCS loops. This holdup of water can result in a stable or even increasing indicated pressurizer level while RCS water inventory is actually decreasing. Pressurizer level should be relied on only with hot leg or core exit subcool-ing present. In SI termination steps in the ERGs, pressurizer level is only checked after adequate RCS subcooling is confirmed.

Reactor vessel level indicating above the top of core with instrument uncertainties is used in ECA-3.3, FR-P.1, ECA-1.1 and FR-H.1 (HP version only). ECA-3.3 addresses a steam generator tube rupture without the capability to depressurize the RCS using normal spray, auxiliary spray or pressurizer PORVs. If pressurizer level is not on span and cannot be reestablished through RCS depressurization, SG tube leakage will continue unless SI is terminated; therefore, reactor vessel level is used to assess RCS inventory. An additional criterion has been added (narrow range level greater than high-high SG level setpoint) to allow time to reestablish required RCS depressurization capability before terminating SI in order to depressurize the RCS. For entry to FR-P.1, a potential pressurized thermal shock situation must exist. Since SI flow can aggravate this potential by adding cold water or pressurizing the RCS, it is important that SI be terminated, as long as core cooling can be ensured through indications of minimum RCS subcooling and reactor vessel level.

2. Given the following plant conditions:
   - A Loss of Offsite Power occurred resulting in a Reactor Trip
   - ONE PRZ PORV is partially OPEN
   - PRZ pressure is 1785 psig
   - PRT pressure is 45 psig
   - Core Exit Thermocouple temperature is 625°F
   - 'A' CSIP is under clearance
   - 'B' CSIP tripped on overcurrent

   Which ONE of the following identifies (1) the temperature indicated on the PRZ PORV Tailpipe Temperature Indicator, TI-463 AND (2) the expected PRZ level trend?

   A. (1) 274°F

      (2) rising

   B. (1) 274°F

      (2) lowering

   C. (1) 293°F

      (2) lowering

   D. (1) 293°F

      (2) rising

1. 008K2.02 010/NEW//MEM 3.0*/3.2*/008K2.02/N///

Unit 2 is operating at 100% power when the following occurs:

- A simultaneous dual Unit LOSP occurs:

Which one of the following completes the statement below?

The __(1)__ CCW pump is being powered by the __(2)__ DG.

	(1)	(2)
A.	2A	1-2A
B.	2A	1C
C.	2C	1-2A
D.✓	2C	1C

2C CCW pump is power from the 2F 4160V bus which is powered from the 1C DG in the above scenario.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since, with the exception of spent fuel pool cooling, every other train related pump with an 'A' designation is an 'A' train pump. The applicant may not recall that the CCW system is "backwards"

Second part is incorrect (See C.2). Plausible because if there was an LOSP on Unit 2 only, the 1-2A DG would be assigned to the Unit 2 'A' train busses. However, there is a DUAL UNIT LOSP and the 1C DG gets assigned to the Unit 2 'A' Train busses. Applicants often get confused as to the assignment of 'A' train DGs during various loss of power scenarios.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

D. Correct. First part is correct. FSD A-181000: 3.1.5.4 [...] Without offsite power available and with or without the presence of SIAS signal, the on-service train CCW pump shall trip, then both train CCW pumps start by the diesel generator loading sequencers (ESS or LOSP).

The 2C CCW pump is the 'A' train pump therefore it will be powered from the 1C DG.

Second part is correct. Under the conditions in the stem, the 1C DG will tie to Unit 2 and supply the 2F, 2K and 2H busses (See FSD A181005 in reference material).

K/A: 008K2.02	Component Cooling Water System (CCWS) - Knowledge of bus power supplies to the following: CCW pump, including emergency backup
Importance Rating:	3.0* 3.2*
Technical Reference:	FSD-A181000, Component Cooling Water, Ver 24 FSD-A181005, Diesel Generators, Ver 44
References provided:	None
Learning Objective:	NAME AND IDENTIFY the Bus power supplies, for those electrical components associated with the CCW System, to include the following: (OPS-40204A04):
Question History:	NEW
K/A match:	Requires the applicant to **know the normal bus power supply to the CCW pumps** in order to **know the correct DG that is its emergency backup power supply.**
SRO justification:	N/A

of a loss of offsite power (LOSP). The CCW pump overload trip shall be alarmed in the MCR to alert the operator (References 6.1.01, 6.4.15, 6.4.16, 6.4.17).

3.1.5.3 If the standby pump is out of service and the on-service pump trips due to a fault, operator action is required to start the pump in the off-service train. The off-service train will not supply the loads of the miscellaneous equipment header without operator action.

3.1.5.4 With offsite power available, the SIAS (train A/B) shall start the off-service train CCW pump to supply CCW to the off-service train RHR pump and charging pumps. Without offsite power available and with or without the presence of SIAS signal, the on-service train CCW pump shall trip, then both train CCW pumps start by the diesel generator loading sequencers (ESS or LOSP) (see Table T-4). (References 6.4.15, 6.4.16, 6.4.17, 6.4.27.)

3.1.5.5 The CCW pump selector switch on the HSP being placed in the "LOCAL" control position shall actuate an alarm in the MCR to alert the operator of control transfer to the HSP (see Section 3.1.5.1) (Reference 6.7.05).

3.1.5.6 The CCW pump and CCW room coolers "ON" status is monitored by train A and train B monitor light boxes (MLB-1 and MLB-4) located on the main control board (MCB). This function is required to be monitored during safety injection (SI) and LOSP (Reference 6.4.19).

3.1.5.7 The CCW pump suction indicators (PI-3308A, B, C) shall cover a range of 0 to 60 psig with an accuracy of $\pm$1 percent. The CCW pump discharge pressure indicators (PI-3035A, B, C) shall cover a range of 0 to 160 psig with an accuracy of $\pm$1 percent. (References 6.7.33, 6.7.34). The requirements are to support inservice testing.

3.1.5.8 In order to provide full manual and automatic control for all three CCW pumps to function in two redundant trains, four sets of hand switches located on the MCB and HSP shall be utilized:

HS-3307CA,CB,CC - CCW Pump C (Train A)
HS-3307AA,AB,AC - CCW Pump A (Train B)
HS-3307BA,BB,BC - CCW Pump B (Train A)
HS-3307DA,DB,DC - CCW Pump B (Train B)

1A CCW pump

TABLE T-6

POWER SUPPLY BOUNDARIES FOR CCW MECHANICAL COMPONENTS

COMPONENT	TPNS NO.	POWER SUPPLY
CCW Pump Train B	QP001A-B	4 kV Bus G, Breaker DG-04
Swing CCW Pump	QP001B-AB	4 kV Bus F, Breaker DF-05 or 4 kV Bus G, Breaker DG-05
CCW Pump Train A	QP001C-A	4 kV Bus F, Breaker DF-04
Valve, CCW Surge Tank Vent	RV-3028 (SV3028A-B) (SV3028B-A)	 125 V dc, Pnl D, Breaker 12 125 V dc, Pnl A, Breaker 15
Valve, CCW to RHR HX A	MOV-3185A	600 V MCC U, Starter FU-E3
Valve, CCW to RHR HX B	MOV-3185B	600 V MCC V, Starter FV-U4
Valve, CCW to SFP HX A	MOV-3094B-A	600 V MCC U, Starter FU-S4
Valve, CCW to SFP HX B	MOV-3094A-B	600 V MCC V, Starter FV-D3
Valve, Makeup to Surge Tank from Reactor Makeup	MOV-3031A	600 V MCC U, Starter FU-D4
Valve, Makeup to Surge Tank from Reactor Makeup	MOV-3031B	600 V MCC V, Starter FV-G5
Valve, Makeup to Surge Tank from Demin. Water	MOV-3030A	600 V MCC U, Starter FU-I3

1C CCW pump

2.0 SYSTEM FUNCTIONAL REQUIREMENTS

2.1 CONFIGURATION REQUIREMENTS

The Diesel Generator System (DGS) for Units 1 and 2 consists of five diesel driven generators which can supply standby power to 4160 volt emergency buses F, G, H, J, K and L of each unit. These busses provide power for Engineered Safety Feature (ESF) loads (Reference 6.1.005).

Diesel generators 1-2A and 1C are designated as train A while diesel generators 1B, 2B and 2C are designated as train B (Reference 6.1.005 and 6.1.027).

Diesel generator 1B is uniquely assigned to Unit 1 while diesel generator 2B is assigned to Unit 2. Diesel generators 1-2A, 1C and 2C are shared between the two units. The design of the onsite emergency power system is such that the plant meets its licensing basis for all design basis events using only four of the diesel generators, namely 1-2A, 1C, 1B and 2B. Therefore, these four diesel generators are dedicated for use during the design basis events. (Reference 6.1.005, 6.1.027, 6.7.027 and 6.7.080).

Diesel generator 2C is dedicated as the altenate AC (AAC) power source for use during Station Blackout (SBO) events as discussed later in section A.5.0. The SBO event does not assume an accident condition in either unit, therefore the loading of EDG 2C will be identical, for all practical purposes to the LOSP loading of EDG 1B or EDG 2B (Reference 6.7.080).

Each of the five diesel generators shall be independent including their associated equipment, except for the 125 Vdc control power. Service water independence is achieved through train/unit orientation. Fuel oil transfer piping is interconnected and independence is maintained through valve lineup (Reference 6.1.003, 6.4.125, 6.4.252, 6.4.269, 6.7.027 and 6.7.039).

The design basis diesel generators shall have sufficient redundancy and testability to perform their safety functions assuming a single failure (Reference 6.1.004, 6.7.027 and 6.7.039).

Diesel generator 2C is the AAC for SBO events and is not considered a candidate for the design basis single failure. However, diesel generator 2C meets all

applicable safety-related criteria and thus, it will be available through operator action for use during a design basis event if diesel generator 1B or 2B fails (Reference 6.7.080).

A minimum of two separate and independent generator sets shall be operable during plant modes 1,2,3 and 4.

Unit 1 sets - 1-2A and 1C (Train A) and1B (Train B)
Unit 2 sets - 1-2A and 1C (Train A) and 2B (Train B)

A.3.1.1 LOSP on both units

For LOSP on both units, the alignment of the diesel generators will be as follows:

1-2A	Unit 1	Buses 1F, 1K and 1H
1C	Unit 2	Buses 2F, 2K and 2H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

The design decision to align the diesel generator 1-2A to Unit 1 and diesel generator 1C to Unit 2 for this scenario was arbitrary since both diesels are capable of energizing the LOSP loads for either unit.

A.3.1.2 LOSP on both units and LOCA on Unit 1

For LOSP on both units and LOCA on Unit 1, the alignment of the diesel generators will be as follows:

1-2A	Unit 1	Buses 1F and 1K
1C	Unit 2	Buses 2F, 2K and 2H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

Diesel generator 1-2A must be aligned to Unit 1 because diesel generator 1C does not have the capacity to energize a full train of LOCA loads; therefore, diesel generator 1C must align to the non-LOCA unit (Unit 2).

A.3.1.3 LOSP on both units and LOCA on Unit 2

For LOSP on both units and LOCA on Unit 2, the alignment of the diesel generators will be as follows:

1-2A	Unit 2	Buses 2F and 2K
1C	Unit 1	Buses 1F, 1K and 1H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

1. 009EK2.03 011/BANK/VOGTLE 10/MEM 3.0/3.3.*/EPE009EK2.03/N///

The following conditions exist on Unit 1:

- An RCS leak is in progress.
- RCS pressure is 1600 psig and stable.
- Containment pressure is 3.1 psig and slowly rising.
- The crew has just transitioned to ESP-1.2, Post LOCA Cooldown and Depressurization.

Which one of the following completes the statement below?

The **minimum** SG narrow range water level must be greater than _(1)_ to _(2)_.

A. 1) 31%

2) ensure SG tubes are covered to promote reflux boiling

B. 1) 48%

2) ensure SG tubes are covered to promote reflux boiling

C✓ 1) 31%

2) ensure adequate SG inventory to provide a secondary heat sink

D. 1) 48%

2) ensures adequate SG inventory to provide a secondary heat sink

ESB-1.2
ERP Step Text - Check intact SG levels
Purpose: To ensure adequate feed flow or SG inventory for secondary heat sink requirements

ESP-1.2
8. Check any intact SG narrow range level - GREATER THAN 31% {48%}

Distracter analysis

A. Incorrect. First part is correct (See C.1).

Second part is incorrect (See C.2). Plausible if the applicant does not understand the mechanism of reflux boiling and improperly assumes this leak rate is sufficient to allow reflux boiling to be the method of heat removal at some time during the event. Reflux boiling is mentioned in ECP-1.1, Loss of Emergency Cooling Recirculation as a method of cooling if the RCS is NOT full.

B. Incorrect. First part is incorrect (See C.1). Plausible if the applicant applies adverse containment numbers as this would be the correct level.

Second part is incorrect (See A.2).

C. Correct. First part is correct. With containment pressure < 4 psig, adverse numbers are not warranted. The required SGWL is >31% (See Step 8 of ESP-1.2 above).

Second part is correct. This is the correct reason for maintaining SGWL above 31% (See ESB-1.2 above). Additionally, at this leak rate, the RCS will remain full as the HHSI pump flow exceeds break flow and reflux cooling will not occur.

(Ran on desktop simulator - IC 073, 200 gpm leak rate, trip and SI at 2000 psig Pzr pressure -- SI flow rate ~230 gpm at 2200 psig)

D. Incorrect. First part is incorrect (See B.1).

Second part is correct (See C.2).

K/A: **009EK2.03**	Small Break LOCA - Knowledge of the interrelations between the small break LOCA and the following: S/Gs
Importance Rating:	3.0 3.3*
Technical Reference:	FNP-1-ESP-1.2, Post LOCA Cooldown and Depressurization., Ver 24. FNP-1-ESB-1.2, Specific Background Document for FNP-1/2-ESP-1.2, Ver 2.1
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with ESP-1.2, Post LOCA Cooldown and Depressurization. (OPS-52531F03)
Question History:	VOGTLE 10
K/A match:	The applicant must know how the **SGs interrelate to the RCS during a Small Break LOCA in that they are required to provide a secondary heat sink for the RCS.**
SRO justification:	N/A

Section: **Procedure**

Unit 1 ERP Step: 42 **Unit 2 ERP Step:** 42 ***ERG Step No:*** *36*

ERP StepText: Maintain RCS heat removal.

ERG StepText: *Maintain RCS Heat Removal*

Purpose: To ensure RCS heat removal

Basis: This step instructs the operator to maintain RCS heat removal either by continued RHR System operation (if available) or by dumping steam, since at this time no SI flow is being provided to the RCS. If no intact SGs are available for dumping steam and the RHR System is not in service, the operator is instructed to use a faulted SG to maintain RCS heat removal.

Knowledge: If the RCS is not full of liquid at this time, it is especially important to keep the secondary system adequately full of water to promote reflux cooling. Reflux cooling is the mechanism by which steam that is generated in the RCS enters the SG tubes and is condensed by the cold water on the SGs secondary side. This liquid then remains in the primary system and promotes cooling.

References:

Justification of Differences:

1 Changed to make plant specific.

Section: **Procedure**

Unit 1 ERP Step: 8 **Unit 2 ERP Step:** 8 ***ERG Step No:*** *7*

ERP StepText: Check intact SG level.

ERG StepText: *Check Intact SG Levels*

Purpose: To ensure adequate feed flow or SG inventory for secondary heat sink requirements

Basis: The minimum feed flow requirement satisfies the feed flow requirement of the Heat Sink Status Tree until level in at least one SG is restored into the narrow range. Narrow range level is reestablished in all SGs to maintain symmetric cooling of the RCS. The control range ensures adequate inventory with level readings on span. This step also provides for monitoring level in the steam generators to detect tube failures. In the case of steam generator level increasing in an uncontrolled manner, the operator is directed to stop any RCS cooldown in progress and transition to E_3, STEAM GENERATOR TUBE RUPTURE, to isolate the affected steam generator. Note that E-3 and its associated network deals with minimizing primary-to-secondary leakage whether it be due to a single event or multiple events, such as a SGTR coincident with a LOCA

Knowledge: 1. "Level increase in an uncontrolled manner" means that the operator cannot control level using available equipment, i.e., level continues to rise even when all feed flow valves to that SG are fully closed. 2. This step is a continuous action step. 3. If a steam generator tube rupture is identified during the recovery actions of ES-1.2, the operator is directed to E-3, Step 1 to take actions to isolate the ruptured steam generator. This action is necessary in order to isolate the steam generator in a timely manner to allow for a higher RCS pressure at which the rupture is isolated. It will also decrease the likelihood of overfilling the ruptured steam generator. If RCS cooldown is already in progress when the SGTR is diagnosed, the operator should stop the cooldown until the ruptured steam generator is isolated.

References: DW-01-018

Step	Action/Expected Response	Response NOT Obtained
___ 8	**Check intact SG level.**	
8.1	Check any intact SG narrow range level - GREATER THAN 31%{48%}.	8.1 Verify total AFW flow to intact SGs greater than 395 gpm. AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229
8.2	[CA] <u>WHEN</u> intact SG narrow range level greater than 31%{48%}, <u>THEN</u> maintain intact SG narrow range level 31%-65%{48%-65%}.	8.2 [CA] <u>IF</u> any SG narrow range level rising in an uncontrolled manner, <u>THEN</u> stop RCS cooldown and go to FNP-1-EEP-3, STEAM GENERATOR TUBE RUPTURE.

8.2.1 Control MDAFWP flow.

MDAFWP FCV 3227 RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO 1A/1B/1C SG B TRN
[] FCV 3227 in MOD

Intact SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3227A in MOD	[]3227B in MOD	[]3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

Step 8 continued on next page.

1. 010A2.01 012/BANK/FNP EXAM BANK/C/A 3.3/3.6/010A2.01/N///TELL NRC

Unit 1 was operating at 100% power when a Reactor Trip and LOSP occurred.
The following conditions exist:

- The 1A PZR HTR GROUP BACKUP handswitch is in AUTO.
- RCS pressure is 2000 psig.

Which one of the following correctly describes Pressurizer Heater operation per ESP-0.1, Reactor Trip Response?

A. The 1A PZR Heaters will have power available and **NO** other actions are required to energize them.

B.✔ The 1A PZR Heaters will have power available **AND** manual actions on the MCB are required to energize them.

C. The 1A PZR Heaters will **NOT** have power available. Manual actions are required to align power to them on the EPB but **NO** other actions are required to energize them.

D. The 1A PZR Heaters will **NOT** have power available. Manual actions are required to align power to them on the EPB **AND** manual actions on the MCB are required to energize them.

ESP-0.1
Attachment 3
1.10.4 WHEN pressurizer heater group 1A operation is desired,
THEN place HTR GRP 1A BLOCKING BYPASS SW to BYPASS.

1.10.5 IF required,
THEN manually energize pressurizer heater group 1A.

Distracter analysis

A. Incorrect. See B. Plausible if the applicant fails to recall that by procedure, the heater switch is taken to off, then the blocking bypass switch is taken to BYPASS and the heater control switch placed in ON to energize the heaters.

B. Correct. The BIF LOSP sequencer reenergizes the emergency section of 600v LC A on an LOSP at step 6. By procedure, the heater switch is taken to off, then the blocking bypass switch is taken to BYPASS and the heater control switch placed in ON to energize the heaters.

C. Incorrect. See B. Plausible if the if the applicant confuses the 1A with the 1B PZR heaters which require EPB alignment and fails to recall that by procedure, the heater switch is taken to off, then the blocking bypass switch is taken to BYPASS and the heater control switch is placed in ON to energize the heaters.

D. Incorrect. See B. Plausible if the if the applicant confuses the 1A with the 1B PZR heaters which require EPB alignment but recognizes the heater switch is taken to off, then the blocking bypass switch is taken to BYPASS and the heater control switch placed in ON to energize the heaters.

K/A: **010A2.01**	Pressurizer Pressure Control System (PZR PCS) - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures
Importance Rating:	3.3 3.6
Technical Reference:	FNP-1-ESP-0.1, Reactor Trip Response, Ver 32 FNP-1-EEP-0.0, Reactor Trip or Safety Injection, Ver 44
References provided:	None
Learning Objective:	NAME AND IDENTIFY the Bus power supplies, for those electrical components associated with the Pressurizer Pressure and Level Control System, to include those items in Table 4- Power Supplies (OPS-52201H04).
Question History:	FNP EXAM BANK
K/A match:	**The LOSP causes the pressurizer heaters to become unavailable for use (failed) until operator action is taken to mitigate their loss**. The applicant must know how to re-energize the PRZR heaters when they are lost during an LOSP.
SRO justification:	N/A

ATTACHMENT 10

ESS AND LOSP LOAD SEQUENCE

SEQUENCERS	B1F		B1G	
COMPONENTS	ESS STEP	LOSP STEP	ESS STEP	LOSP STEP
1A CHG PUMP	1	1		
1B CHG PUMP	1	1	1	1
1C CHG PUMP			1	1
1A RHR PUMP	2			
1B RHR PUMP			2	
1A CS PUMP	2			
1B CS PUMP			2	
1A SW PUMP	3	2		
1B SW PUMP	3	3		
1D SW PUMP			3	2
1E SW PUMP			3	3
1A CCW PUMP			4	4
1B CCW PUMP	4	4	4	4
1C CCW PUMP	4	4		
A TRN CTMT CLR FAN	4	4		
B TRN CTMT CLR FAN			4	4
1A MDAFWP	5	5		
1B MDAFWP			5	5
1A BATT CHARGER	6	6		
1B BATT CHARGER			6	6
1A CRDM CLG FAN				2
1B CRDM CLG FAN		2		
1A RX CAV CLG FAN		1		
1B RX CAV CLG FAN				1
1A RX CAV H2 DILUTION FAN	5			
1B RX CAV H2 DILUTION FAN			5	
1A LC EMERG SUPPLY ED08/ EA09 / 1C AIR COMP. EA15	6	6		

-END-

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 3

1.5	Verify BKR DG02 (1G 4160 V bus tie to 1L 4160 V bus) - CLOSED.	1.5 IF diesel generator cooling NOT supplied, THEN secure 1B diesel generator using ATTACHMENT 4, SHUTDOWN OF A DIESEL GENERATOR.
1.6	Verify all RCP busses - ENERGIZED. [] 1A 4160 V bus [] 1B 4160 V bus [] 1C 4160 V bus	
1.7	Check 1E 4160 V bus - ENERGIZED.	1.7 Establish power to 1C 600 V LC emergency section loads. 1.7.1 Place handswitch for pressurizer heater group 1B in OFF. 1.7.2 Open BKR EC08-1. 1.7.3 Close BKRs EE07-1 and EC10-1.

This is the method to energize the 1B Przr heaters. The 1C LC must be manually aligned at the EPB.

****

CAUTION: To prevent diesel generator overloading, at least 0.3 MW of diesel generator capacity must be available prior to energizing a group of pressurizer heaters.

****

		1.7.4 Energize pressurizer heater group 1B as required.
1.8	Check 1D 4160 V bus - ENERGIZED.	1.8 Proceed to step 1.10.
1.9	IF 1D 4160 V bus energized, THEN proceed to step 1.14.	

Step 1 continued on next page.

ATTACHMENT 3

1.10 Establish power to 1A 600 V LC emergency section loads.

1.10.1 Place handswitch for pressurizer heater group 1A in OFF.

1.10.2 Verify open BKR EA08-1.

1.10.3 Verify closed BKRs ED08-1 and EA09-1.

LC A is sequenced on by the LOSP sequencer. Steps 1.10.2 and 1.10.3 are to ensure the bus was sequenced on. Steps 1.10.4 and 1.10.5 are the steps to energize the heaters.

****

CAUTION: To prevent diesel generator overloading, at least 0.3 MW of diesel generator capacity must be available prior to energizing a group of pressurizer heaters.

****

NOTE: The BYPASS position allows manual energization of pressurizer heater group 1A from the MCB handswitch, and automatic energization based on either pressurizer pressure ≤ 2210 psig or pressurizer level 5% above program.

1.10.4 WHEN pressurizer heater group 1A operation is desired,
THEN place HTR GRP 1A BLOCKING BYPASS SW to BYPASS.

1.10.5 IF required,
THEN manually energize pressurizer heater group 1A.

Step 1 continued on next page.

1. 011EK2.02 013/BANK/FNP EXAM BANK/C/A 2.6*/2.7*/EPE011EK2.02/N///

Unit 1 has experienced a Loss of Off-site Power and a Large Break LOCA. The following conditions exist:

- ESP-1.3, Transfer to Cold Leg Recirculation, has been completed.

Subsequently, the Shift Supervisor directs the OATC to perform ESP-1.4, Transfer to Simultaneous Cold and Hot Leg Recirculation and the following occurs:

- Power is lost to the 1G 4160V Bus and will not be restored for 18 hours.

Which one of the following completes the statement below?

At the completion of ESP-1.4, the running LHSI pump will be aligned for _(1)_ leg recirculation and the running HHSI pump will be aligned for _(2)_ leg recirculation.

	(1)	(2)
A.	HOT	COLD
B.	COLD	COLD
C.	HOT	HOT
D.✔	COLD	HOT

ESP-1.4 intends to align LHSI to HOT leg and leave HHSI aligned for Cold leg recirculation. However, during realignment, if any portion of the LHSI system cannot be reconfigured (Step 1), then the system is returned to its original lineup of Cold leg recirc and the available HHSI train is aligned for HOT leg recirculation. Step 4 has the operator assess the re-alignment and if the final requirement is not met, they are directed to Step 1 and contacting the Technical Support Center for guidance

We have recently developed a JPM that causes the alignment to be in a cold/cold or hot/hot alignment since some failures can lead you there. That is precisely the reason the procedure will direct you to the TSC staff if a final alignment other than cold/hot or hot/cold is reached by the end of the procedure.

The final alignment of LHSI and HHSI can be confusing when coupled with power losses and/or equipment failures.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since the applicant may believe the procedure allows only one train of LHSI to be aligned to the Hot Leg. This would be correct if it were HHSI.

Second part is incorrect (See B.2). Plausible if the applicant believes that if one train cannot be realigned then neither will be aligned. This would be correct if it were LHSI.

B. Incorrect. First part is correct (See D.1)

Second part is incorrect (See A.2).

C. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2). Plausible since when power is lost or equipment malfunctions, there are allowances to come back to steps in the procedure and perform them when power is restored and/or equipment repaired such as in ESP-1.3 when the charging suction and discharge header MOVs are aligned. Any alignment is possible in this procedure once malfunctions occur.

D. Correct. First part is correct. Per ESP-1.4 Step 1, if both trains of LHSI cannot be aligned to Hot Leg recirc then both trains are left aligned to cold leg recirc.

Second part is correct. ESP-1.4 Step 2 will align the A train HHSI to Hot Leg recirc and Step 3 will leave B train in its original alignment.

K/A: **011EK2.02**	Large Break LOCA - Knowledge of the interrelations between the Large Break LOCA and the following: Pumps
Importance Rating:	2.6* 2.7*
Technical Reference:	FNP-1-ESP-1.4, Transfer To Simultaneous Cold and Hot Leg Recirculation, Ver 16.
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing [...] (2) ESP-1.4, Transfer to Simultaneous Cold Leg and Hot Leg Recirculation. (OPS-52531G06)
Question History:	FNP EXAM BANK
K/A match:	The applicant is required to know the **interrelation between the RHR/Charging Pumps and the RCS during a Large Break LOCA. Based on the scenario given, the applicant must determine the final pump alignment.**
SRO justification:	N/A

DG03

B train
power

EE15

1V 600/208V MCC **AB - 139'** **B177556-20**

BKR	TPNS	DESCRIPTION
	Q1R17B0009-B	**1V 600/208V MCC (600V SECTION) <<< EE15**
FVA4	Q1E21MOV8803B-AB	HHSI TO RCS CL ISO MOV
FVA5	Q1E11MOV8888B-B	LHSI TO RCS COLD LEG MOV
FVA6	Q1E21MOV8109A-B	1A CHARGING PUMP MINIFLOW ISO
FVB2	Q1E11MOV8889-B	DISC SWITCH Q1R18B036-B >>> LHSI TO RCS HOT LEG MOV
FVB3	Q1E11MOV8809B-B	1B RHR PUMP INLET MOV
FVB4	--------------	SPARE
FVB5	Q1E11MOV8811B-B	CTMT SUMP OUTLET MOV
FVC3	Q1P17MOV3046-B	RCP COMP COOL MOV
FVC4	Q1P16MOV3019C-B	CTMT COOLER SERVICE WATER INLET MOV
FVC5	Q1P16MOV3019D-B	CTMT COOLER SERVICE WATER INLET MOV
FVD2	Q1P17MOV3047-B	CCW HX COOL DISCHARGE MOV
FVD3	Q1P17MOV3094A-B	1A SPENT FUEL POOL HX INLET MOV
FVD4	Q1P16MOV3130A-B	1A COMP COOLING WATER HX INLET MOV
FVD5	Q1P16MOV3130B-B	1B COMP COOLING WATER HX INLET MOV
FVE2	Q1P16MOV3024D-B	CTMT COOLER SERVICE WATER WT DISCHARGE MOV
FVE3	Q1E21MOV8108-B	CHARGING PUMP TO REGENERATIVE MOV
FVE4	Q1E11MOV8706B-B	RESIDUAL HX DISCHARGE MOV
FVE5	Q1E21MOV8100-B	RCP DISCHARGE SEAL INJECTION RETURN MOV
FVF2	----------------	SPARE
FVF3	----------------	SPARE
FVF4	Q1E13MOV8826B-B	CTMT SUMP OUTLET MOV
FVF5	Q1E13MOV8827B-B	CTMT SUMP OUTLET MOV

DG03
EE15
1V 600/208V MCC Via G 4160 Bus (B train) **AB - 139'** **B177556-20**
(CONT'D)

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
FVN2	Q1E22MOV3872B-B	REACTOR CAVITY H2 DILUTION FAN MOV	
FVN3	------------	SPARE	
FVN4	Q1P13MOV2788B-B	CNMT PURGE EXHAUST MOV	
FVN5	Q1E11MOV8885-B	HHSI TO RCS COLD LEG MOV	
FVO2	Q1P16MOV3135-B	RCP PUMP MOTOR COOLS INLET MOV	
FVO3	Q1E15MOV3361B-B	PENETRATION ROOM RECIRC FAN TO PENETRATION RM MOV	
FVO4	Q1E15MOV3362B-B	PENETRATION ROOM TO PENETRATION RM FILTER MOV	
FVP4L	**Q1R17B0009-B**	**1V 600/208V MCC XFMR >>> 1V MCC 208V SECTION >>>**	G-113
FVR2	Q1N23MOV3764E-B	AUX FEEDWATER TO STEAM GENERATOR MOV	
FVS2	Q1E11MOV8808B-B	DISC SWITCH Q1R18B035-B >>> ACCUMULATOR TANK DISCHARGE MOV	
FVS4L	**N1R19L0012-N**	**1WW SPDS 208/120V AC DIST PNL >>>**	G-117
FVS4R	Q1E23AIT02703B-B	POST ACCIDENT H2 ANALYYZER	
FVT2	Q1E21MOV8884-B	DISC SWITCH Q1R18B033-B >>> HHSI TO RCS HOT LEG MOV	
FVT3	Q1E21MOV8109B-B	1B CHARGING PUMP MINIFLOW ISO	
FVT4	Q1E11MOV8812B-B	CTMT SUMP OUTLET MOV	
FVT5	Q1E21LCV00115D-B	RWST TO CHARGING PUMP MOV	
FVU2	Q1P16MOV3023C-B	CTMT COOLER SERVICE WATER DISCHARGE MOV	
FVU3	Q1P16MOV3023D-B	CTMT COOLER SERVICE WATER DISCHARGE MOV	
FVU4	Q1P17MOV3185B-B	RESIDUAL HX INTAKE MOV	
FVU5	Q1P16MOV3024C-B	CTMT COOLER SERVICE WATER WT DISCHARGE MOV	
FVV2	Q1E11MOV8701B-B	RHR PUMP INLET MOV	

Step	Action/Expected Response	Response NOT Obtained
7.20	IF 1A RHR PUMP started, THEN align charging pump suction header isolation valves based on 1B charging pump status.	7.20 IF 1A RHR THEN perf 7.20.1 Verify PUMP s 7.20.2 Procee

If these steps could not be done due to not being powered up when the step was ready to be performed, they would be repositioned once the valves were powered up which could likely be after the completion of ESP-1.3. The valves are directed to be powered up in EEP-1.

1B Charging Pump Status	Aligned As A Train pump	Aligned As B Train pump	Not Available
CHG PUMP SUCTION HDR ISO Q1E21MOV	[] 8130A open [] 8130B open [] 8131A closed [] 8131B closed	[] 8130A closed [] 8130B closed [] 8131A open [] 8131B open	[] 8130A closed [] 8130B closed [] 8131A closed [] 8131B closed

Step	Action/Expected Response	Response NOT Obtained
7.21	Open RHR supply to A train charging pump suction. 1A RHR HX TO CHG PUMP SUCT [] Q1E11MOV8706A	7.21 Perform the following. 7.21.1 Stop the running A train CHG PUMP. 7.21.2 Proceed to step 7.25.
7.22	Verify A train CHG PUMP - started.	
7.23	Verify VCT level - GREATER THAN 5%.	
7.24	Close A train RWST to charging pump header valve. RWST TO CHG PUMP [] Q1E21LCV115B	7.24 Perform the following. 7.24.1 Stop the running A train CHG PUMP. 7.24.2 Close RHR supply to A train charging pump suction. 1A RHR HX TO CHG PUMP SUCT [] Q1E11MOV8706A

Step 7 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

7.25 IF 1B RHR PUMP started, THEN align charging pump suction header isolation valves based on 1B charging pump status.

1B Charging Pump Status	Aligned As A Train pump	Aligned As B Train pump	Not Available
CHG PUMP SUCTION HDR ISO Q1E21MOV	[] 8130A open [] 8130B open [] 8131A closed [] 8131B closed	[] 8130A closed [] 8130B closed [] 8131A open [] 8131B open	[] 8130A closed [] 8130B closed [] 8131A closed [] 8131B closed

7.25 IF 1B RHR PUMP NOT started, THEN perform the following.

7.25.1 Verify the B train CHG PUMP stopped.

7.25.2 Proceed to step 7.30.

7.26 Open RHR supply to B train charging pump suction.

1B RHR HX
TO CHG PUMP SUCT
[] Q1E11MOV8706B

7.26 Perform the following.

7.26.1 Stop the running B train CHG PUMP.

7.26.2 Proceed to step 7.30.

7.27 Verify B train CHG PUMP - started.

7.28 Verify VCT level - GREATER THAN 5%.

Step 7 continued on next page.

FARLEY NUCLEAR PLANT

EVENT SPECIFIC PROCEDURE

FNP-1-ESP-1.4

TRANSFER TO SIMULTANEOUS COLD AND HOT LEG RECIRCULATION

PROCEDURE USAGE REQUIREMENTS-per NMP-AP-003	SECTIONS
Continuous Use	**ALL**
Reference Use	
Information Use	

S
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Y

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T
E
D

Approved:

David L Reed (for)

Operations Manager

Date Issued: 08/27/12

Table of Contents

A. Purpose

This procedure provides the necessary instructions for transferring the safety injection system to simultaneous cold and hot leg recirculation.

The intent of achieving simultaneous cold and hot leg recirculation is to have EITHER at least one train of RHR aligned to the hot legs with at least one train of HHSI aligned to the cold legs OR at least one train of RHR aligned to the cold legs with at least one train of HHSI aligned to the hot legs.

B. Symptoms or Entry Conditions

I. This procedure is entered when the specified time intervals for transfer to simultaneous cold and hot leg recirculation have elapsed; from the following:

   a. FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 19

II. When a decision is made, based upon the recommendation of the TSC Staff, that transfer to hot-leg recirculation is required. Transfer to hot-leg recirculation might be required eventually, after transferring to cold-leg recirculation during the implementation of:

   a. FNP-1-ESP-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

   b. FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.

   c. FNP-1-ECP-3.2, SGTR WITH LOSS OF REACTOR COOLANT SATURATED RECOVERY DESIRED.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Align LHSI for hot leg recirculation.**	1 Align LHSI for cold leg recirculation.
	1.1 Align RHR pumps discharge for hot leg recirculation. 1A(1B) RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A closed [] Q1E11MOV8888B closed RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A open [] Q1E11MOV8887B open RHR TO RCS HOT LEGS ISO [] Q1E11MOV8889 open	a) Align RHR pumps discharge for cold leg recirculation. RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A closed [] Q1E11MOV8887B closed RHR TO RCS HOT LEGS ISO [] Q1E11MOV8889 closed 1A(1B) RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A open [] Q1E11MOV8888B open
	1.2 Proceed to Step 4.	b) Proceed to Step 2.

This valve has lost power and will not close.
The RNO column will re-open, MOV-8888A and both trains of LHSI will still be aligned to Cold Leg Recirc.

Step	Action/Expected Response	Response NOT Obtained
___ 2	**Align Train A HHSI for hot leg recirculation.**	2 Align Train A HHSI for cold leg recirculation.
	2.1 Stop any running A train CHG PUMP.	a) Stop any running A train CHG PUMP.
	2.2 IF power NOT available to Q1E21MOV8803B, THEN align alternate(A train) power. TRANSFER SWITCH PANEL FOR Q1E21MOV8803B-AB(139' Non-Rad - Room 343, Key # CAT-50) [] Q1R18B037-AB selected to "Train A"	b) Close charging pump recirculation to RCS hot legs valve. CHG PUMP RECIRC TO RCS HOT LEGS [] Q1E21MOV8886
	2.3 Close HHSI isolation valves. HHSI TO RCS CL ISO [] Q1E21MOV8803A [] Q1E21MOV8803B (A Train) OR [] Q1E21MOV8803B (B Train)	c) Open HHSI isolation valve. HHSI TO RCS CL ISO [] Q1E21MOV8803A (preferred) OR [] Q1E21MOV8803B d) Start only one A train CHG PUMP. e) Proceed to Step 3.
	2.4 Open charging pump recirculation to RCS hot legs valve. CHG PUMP RECIRC TO RCS HOT LEGS [] Q1E21MOV8886	
	2.5 Start only one A train CHG PUMP.	

A train power is available so this will align A train HHSI for HOT Leg recirc

MOV-8803B has an alternate A train power supply

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Align Train B HHSI for hot leg recirculation.**	3 Align Train B HHSI for cold leg recirculation.
	3.1 Stop any running B train CHG PUMP.	a) Stop any running B train CHG PUMP.
	3.2 Close charging pump recirculation to RCS cold legs valve. CHG PUMP RECIRC TO RCS COLD LEGS [] Q1E21MOV8885	b) Close charging pump recirculation to RCS hot legs valve. CHG PUMP RECIRC TO RCS HOT LEGS [] Q1E21MOV8884
	3.3 Open charging pump recirculation to RCS hot legs valve. CHG PUMP RECIRC TO RCS HOT LEGS [] Q1E21MOV8884	c) Open charging pump recirculation to RCS cold legs valve. CHG PUMP RECIRC TO RCS COLD LEGS [] Q1E21MOV8885
	3.4 Start only one B train CHG PUMP.	d) Start only one B train CHG PUMP. e) Proceed to Step 4.
___ 4	**Check simultaneous cold and hot leg recirculation in progress by one of the following:** • At least one train of LHSI aligned to the hot legs <u>AND</u> at least one train HHSI aligned to cold legs. <u>OR</u> • At least one train LHSI aligned to cold legs <u>AND</u> at least one train HHSI aligned to hot legs.	4 <u>IF</u> simultaneous cold and hot leg recirculation <u>NOT</u> established, <u>THEN</u> consult TSC <u>AND</u> return to step 1 of this procedure.

B train power

No B train power means this alignment cannot be made

Final alignment:
A and B train LHSI still on Cold leg recirc and
A train HHSI on Hot leg recirc.

Step	Action/Expected Response	Response NOT Obtained
___ 5	**Verify SI flow to RCS cold <u>AND</u> hot legs - STABLE.** A TRN HHSI FLOW [] FI 943 HHSI B TRN RECIRC FLOW [] FI 940 1A(1B) RHR HDR FLOW [] FI 605A [] FI 605B	5 <u>IF</u> neither train of simultaneous cold and hot leg recirculation established, <u>THEN</u> consult TSC <u>AND</u> return to step 1 of this procedure.
___ 6	**Go to procedure and step in effect.**	

-END-

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. 011K2.02 014/NEW//MEM 3.1/3.2/011K2.02/N///

Unit 1 is operating at 100% power when the 1E 4160V bus becomes de-energized due to an electrical fault.

Which one of the following completes the statement below?

Pressurizer heater groups ____ have lost their normal power supply.

A. 1C and 1D ONLY

B. 1A, 1C and 1D ONLY

C.✓ 1B and 1E ONLY

D. 1B, 1D and 1E ONLY

Load
Pressurizer Heater Group A 600V LC A (Normal) 4160V D
Pressurizer Heater Group B 600V LC C (Normal) 4160V E
Pressurizer Heater Group C 600V LC M 4160V D
Pressurizer Heater Group D 600V LC M 4160V D
Pressurizer Heater Group E 600V LC N (Unit 1 and 2) 4160V E

Distracter analysis

A. Incorrect.	See C. Plausible since these heaters are powered from the same LC. The applicant may believe that this is the only LC affected by the power loss. Although 1A heaters are also powered from the same 4160V Bus as these heaters, the applicant may believe the 1A heaters are powered from the Emergency Bus (1F) since they are sequenced on after an LOSP.
B. Incorrect.	See C. Plausible since this would be the impact for the loss of 1D 4160V bus and the applicant may think theses heaters are supplied by 1E 4160V bus.
C. Correct.	Per Unit 1 Electrical Load List: 1E 4160V Bus supplies 1C Load Center (LC) and 1N LC. 1C LC - 1B pressurizer heaters. 1N LC - 1E pressurizer heaters
D. Incorrect.	See C. Plausible if the applicant knows that 2 sets of heaters are powered from the same LC but cannot correctly recall which ones. The 1B heaters is a partially correct answer and would be included if the applicant thinks these heaters are powered from the same 4160V bus as the 1B heaters.

K/A: **011K2.02**	Pressurizer Level Control System (PZR LCS) - Knowledge of bus power supplies to the following: PZR heaters
Importance Rating:	3.1 3.2
Technical Reference:	A506250, Unit 1 Electrical Load List, Ver 74.0
References provided:	None
Learning Objective:	NAME AND IDENTIFY the Bus power supplies, for those electrical components associated with the Pressurizer Pressure and Level Control System, to include those items in Table 4- Power Supplies (OPS-52201H04).
Question History:	NEW
K/A match:	Applicant is required to **know the power supplies to the pressurizer heaters** in order to determine which ones have lost power. The power supply has to go back to the 4160V bus so the applicant also has to know the LC supplies as well.
SRO justification:	N/A

1E 4160V BUS

SECTION E
TABLE OF CONTENTS
SORT BY PAGE # and 4KV BREAKER

DE07

1C 600V LOAD CENTER **AB - 121'** **C177009**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R16B0005-B	**1C 600V LOAD CENTER**	
EC01	Q1R16BKREC01	PT COMPARTMENT	
EC02	**Q1R11B0003-N**	**1C 4160/600V SST <<< DE07 (NORMAL)**	
EC03	N1G12L0001A-N	1A BTRS CHILLER UNIT PANEL	
EC04	**N1R17B0003-N**	**1C 600/208V MCC >>>**	**E-6**
EC05	N1P41M0001A-N	1A AUX BUILDING MAIN EXHAUST FAN	
EC06	N1V46M0001-N	RADWASTE AIR HANDLING UNIT	
EC07	**Q1R16B0008-AB**	**1F 600V LOAD CENTER <<< EF07 (ALTERNATE)**	
EC08	Q1R16BKREC08	1C 600V LOAD CENTER TIE BKR (NORMAL - EMERG)	
EC09	Q1G31M0002A-B	1A SPENT FUEL POOL PUMP	
EC10	**Q1R16B0007-B**	**1E 600V LOAD CENTER <<< EE07 (ALTERNATE - EMERG)**	
EC11	**Q1B31L0001B-B**	**1B PRESSURIZER HEATER DIST PANEL >>>**	**E-19**
EC12	Q1E12M0001D-B	1D CONTAINMENT COOLER (NORMAL/HIGH SPEED)	
EC13	**N1V51E0003B-N**	**1E 600-480/227V NORMAL LIGHTING TRANSFORMER >>> LTG PNL LP-1L, LP-1P & LP-1Q**	**E-20 E-23 E-24**
EC14	**N1V51E0003E-N**	**1F 600-480/227V NORMAL LIGHTING TRANSFORMER >>> LTG PNL LP-1C, LP-1F & LP-1I**	**E-25 E-27 E-29**

DE08
1N 600V LOAD CENTER AB - 155' C177029

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
	N1R16B0015-N	**1N 600V LOAD CENTER**	
EN01	N1R165BKREN01	PT COMPARTMENT	
EN02	**NSR11B0009-N**	**1N 4160/600V SST <<< 1-DE08 (NORMAL) OR 2-DE08 (ALTERNATE)**	
EN03	---------------	SPARE	
EN04	**Q1R16B0008-AB**	**1F 600V LOAD CENTER <<< EF09 (ALTERNATE)**	
EN05	**N1B31L0001E-N**	**1E PRESSURIZER HEATER CABINET GROUP DISTRIBUTION PANEL**	E-33
EN06	**NSR17B0012-N**	**1II 600/208V MCC >>>**	E-34
EN07	**N2B31L0001E-N**	**2E PRESSURIZER HEATER CABINET GROUP**	E-38
EN08	QSV31G0001-N	DISC SWITCH NSR18B0004-N >>> SPENT FUEL CASK CRANE DISTRIBUTION PANEL	

1. 012A4.06 015/BANK/FNP 10/C/A 4.3/4.3/012A4.06/N///TELL NRC

The following conditions exist on Unit 1:

- A loss of 'A' Train Auxiliary Building 125V DC Bus has occurred.

Which one of the following completes the statement below?

Placing the MCB Reactor Trip handswitch in TRIP would ____ if they were closed.

A.✔ open **ALL** reactor trip and bypass breakers

B. **ONLY** open the 'B' reactor trip breaker and the 'B' reactor trip bypass breaker

C. **ONLY** open the 'B' reactor trip breaker and the 'A' reactor trip bypass breaker

D. open **BOTH** reactor trip breakers but **NOT** open either reactor trip bypass breaker

FSD-A181007:
3.3.2 pg 3-10
The **first method** of tripping the breaker (i.e., reactor trip or bypass breakers) is by a loss or drop of rated voltage to the **Undervoltage Relay (UV**). **The relay is normally energized from the 48 volt DC from the RPS.** When the voltage is removed by an automatic reactor trip signal, the relay is de-energized and releases the UV trip lever, which actuates the trip shaft, causing the breaker to unlatch from the closed position.

The **second method** of tripping the trip shaft is by the shunt trip lever when the normally de-energized shunt trip (SHTR) coil is energized. When energized, the **SHTR coil is powered from the 125 volt DC system** used to close the reactor trip and bypass breaker closing circuits.

<u>Distracter analysis</u>

A. Correct.	Without 'A' train DC, the UV coils from the "A' Train Reactor Protection System (RPS) will still open 'A' Trip and 'B' Bypass breakers. 'B' train RPS deenergizes the UV coils for 'B' Trip and 'A' Bypass breakers. 'B' Train Aux Building DC will open the 'B' Trip breaker.
B. Incorrect.	See A. Plausible if the applicant did not recall that the UV coils from RPS will trip ALL Trip and Bypass breakers. Since both listed breakers are 'B' breakers, this adds to plausibility due to the applicant thinking the 'B' train is unaffected and still would cause a reactor trip if the system worked this way.
C. Incorrect.	See A. Plausible since this is how the RPS opens the Trip and Bypass breakers. The applicant may recall that this is how the RPS works but not realize that the loss of 'A' Train DC has no effect on the RPS.
D. Incorrect.	See A. Plausible if the applicant thinks that the Trip breakers are tripped by RPS and the Bypass breakers from Aux Building DC. Since the Shunt trip coils on the Bypass breakers can ONLY be operated locally, the applicant may think that without DC the Bypass breakers will not open.

K/A: **012A4.06**	Reactor Protection System (RPS) - Ability to manually operate and/or monitor in the control room: Reactor trip breakers
Importance Rating:	4.3 4.3
Technical Reference:	FSD-A181007 Reactor Protection System, Ver 18 D-177198, Sheet 2, Ver 3
References provided:	None
Learning Objective:	RECALL AND DESCRIBE the operation and function of the following reactor trip signals, permissives, control interlocks, and engineered safeguards actuation signals associated with the Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07): • Table 1, Reactor Trip Signals
Question History:	FNP 10
K/A match:	Requires the applicant to **monitor the effect on the Reactor Trip and Bypass Breaker Positions due to a loss of DC when they are manually tripped (operated).**
SRO justification:	N/A

3.2.7 Interface Requirements

The STC shall interface with the SSPS and shall be supplied by qualified Class 1E power from the 120 Vac vital power cabinets. (References 6.7.014, 6.4.059, 6.4.060, 6.4.084)

3.3 REACTOR TRIP SWITCHGEAR

TPNS Nos.	Service
QC11E004A-AB	(RTA, BYB)
QC11E004B-AB	(RTB, BYA)

3.3.1 Basic Functions

The reactor trip switchgear functions to switch power to or remove power from the control rod positioning equipment. The switchgear opens the reactor trip and bypass breakers A and B on reactor trip causing the control rods to fall by gravity into the reactor core.

3.3.2 Functional Requirements

The switchgear assembly shall consist of two low voltage metal enclosed switchgear sections. One section will contain two series connected reactor trip circuit breakers. The second will contain two bypass circuit breakers connected so that a bypass breaker parallels each reactor trip breaker. The bypass circuit breaker is used to bypass the reactor trip breaker for on-line testing of the latter with the reactor in operation.

The system also includes two 260 volt line to line identical three phase Motor-Generator sets rated at 400 KVA, reverse current relay, generator output circuit breaker, a synchronizer, and a common ground relay.

Each circuit breaker shall have provisions for locking it in the "Test" and "Disconnected" draw-out positions.

The circuit breaker also includes positions for "Connected" and "Remove." (Reference 6.4.077)

Interposing relays shall be used to isolate Train A from Train B wiring where it is necessary to parallel these circuits into a single output.

Each circuit breaker shall be equipped with a 48 volt DC instantaneous undervoltage trip device and a 125 Vdc shunt trip device. (Reference

6.4.086) The Shunt Trip Attachment coil shall operate on 125 Vdc and function as a backup for the undervoltage trip device.

The first method of tripping the breaker (i.e., reactor trip or bypass breakers) is by a loss or drop of rated voltage to the Undervoltage Relay (UV). The relay is normally energized from the 48 volt DC from the RPS. When the voltage is removed by an automatic reactor trip signal, the relay is de-energized and releases the UV trip lever, which actuates the trip shaft, causing the breaker to unlatch from the closed position. The second method of tripping the trip shaft is by the shunt trip lever when the normally de-energized shunt trip (SHTR) coil is energized. When energized, the SHTR coil is powered from the 125 volt DC system used to close the reactor trip and bypass breaker closing circuits.

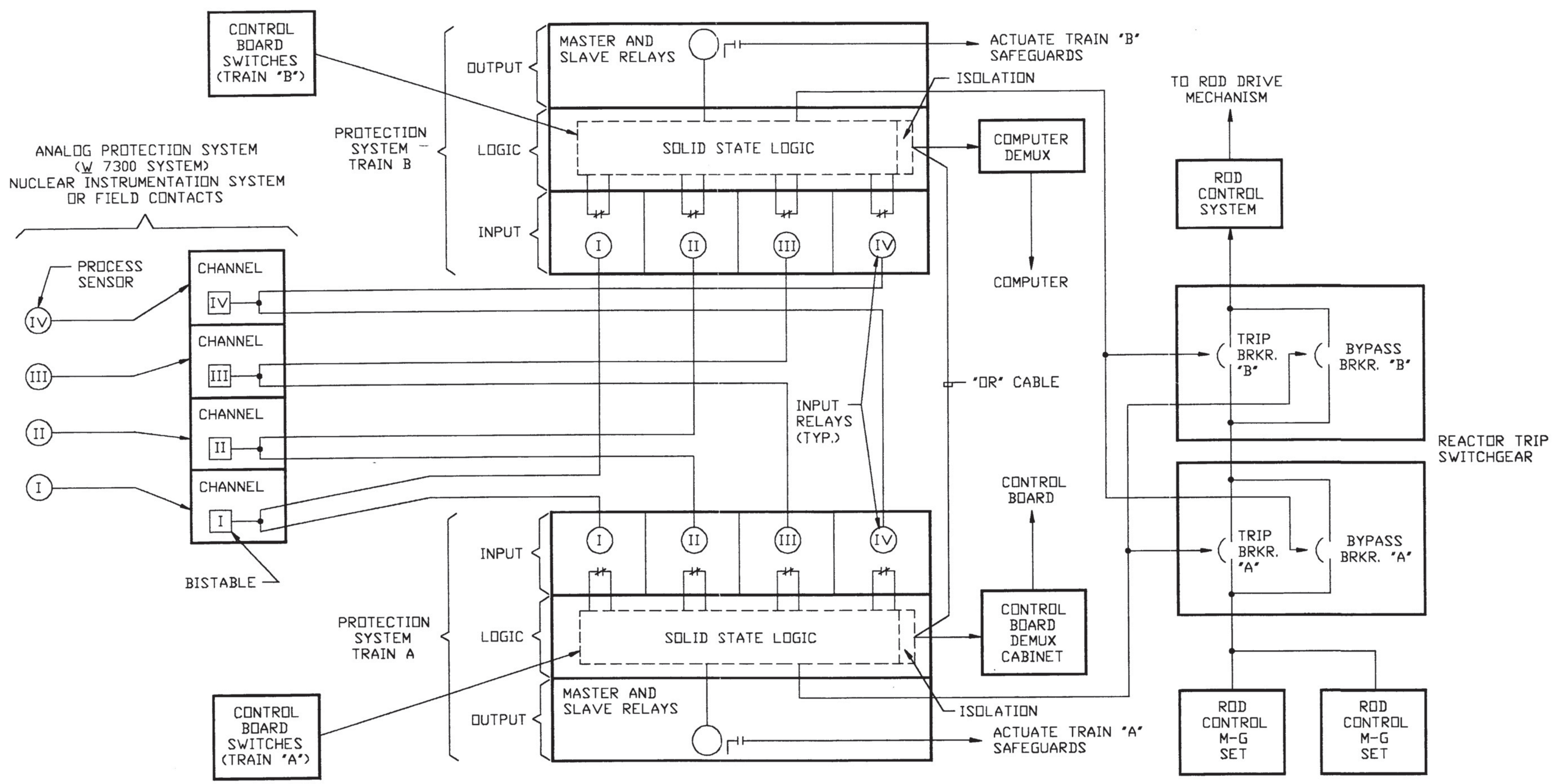
For the reactor trip bypass breaker, the SHTR relay is energized only by a manual pushbutton. After the reactor trip bypass breaker is opened, then a contact in series with the SHTR relay opens to de-energize the coil. Thus, the SHTR relay is only momentarily energized.

For the reactor trip breaker, the SHTR relay is energized by the closing of a contact associated with a shunt trip attachment relay (STA for 52/RTA and STB for 52/RTB). STA (STB) is energized from the RPS voltage to the UV trip coil of the 52/RTA (52/RTB). When the voltage is removed by an automatic reactor trip signal, the relay will de-energize, closing its contact to energize the shunt trip coil of 52/RTA (52/RTB). After the reactor trip breaker is opened, then a contact in series with the SHTR relay opens to de-energize the coil. Thus, the SHTR relay is only momentarily energized.

3.3.3 Design Transients

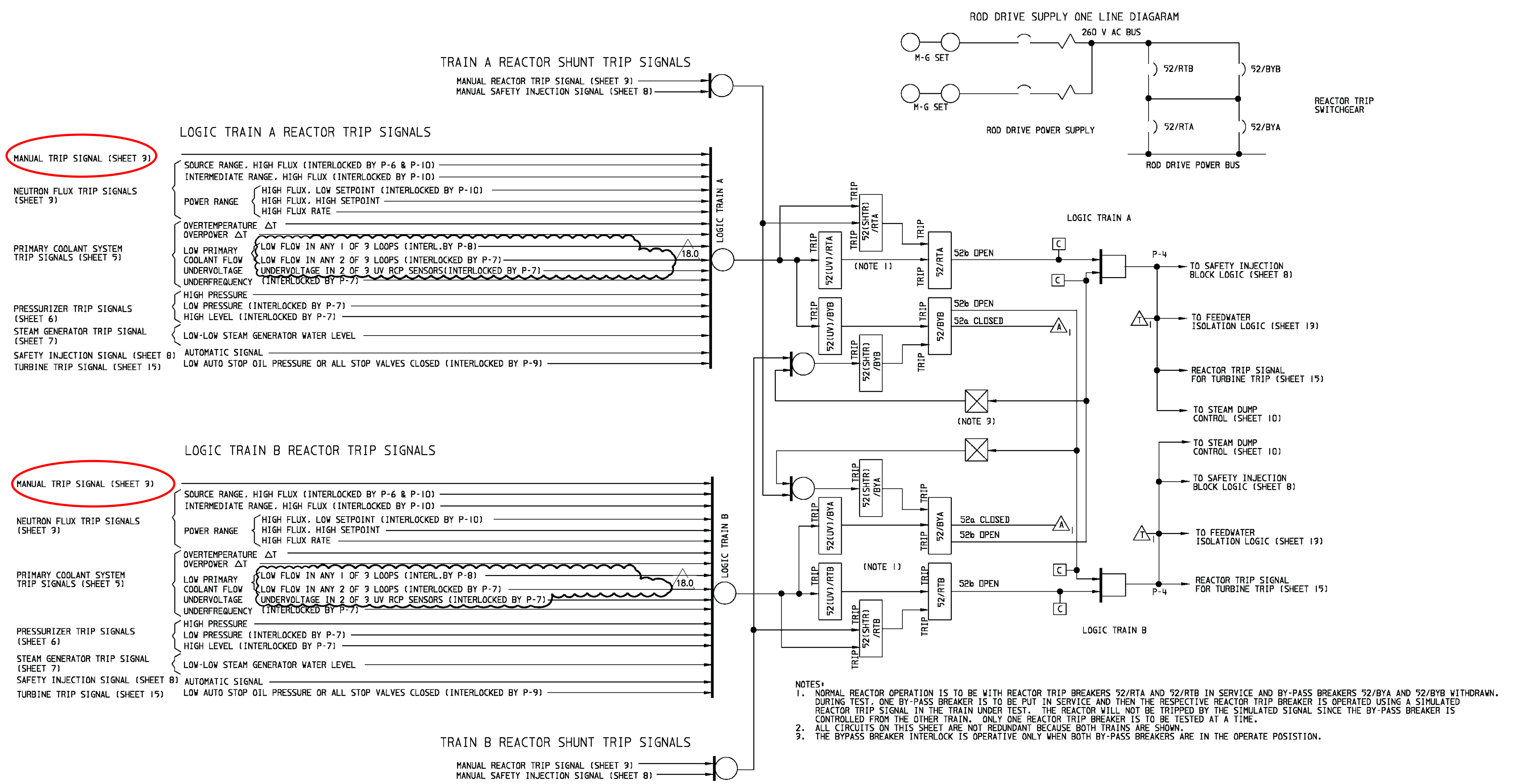
The ambient design conditions are: 95% relative humidity and 40 deg. F to 120 deg. F temperature. (Reference 6.4.090)

Also see Protection Features 3.3.7.



REACTOR PROTECTION SYSTEM BOUNDARIES

FIGURE F-1



NOTES:
1. NORMAL REACTOR OPERATION IS TO BE WITH REACTOR TRIP BREAKERS 52/RTA AND 52/RTB IN SERVICE AND BY-PASS BREAKERS 52/BYA AND 52/BYB WITHDRAWN. DURING TEST, ONE BY-PASS BREAKER IS TO BE PUT IN SERVICE AND THEN THE RESPECTIVE REACTOR TRIP BREAKER IS OPERATED USING A SIMULATED REACTOR TRIP SIGNAL IN THE TRAIN UNDER TEST. THE REACTOR WILL NOT BE TRIPPED BY THE SIMULATED SIGNAL SINCE THE BY-PASS BREAKER IS CONTROLLED FROM THE OTHER TRAIN. ONLY ONE REACTOR TRIP BREAKER IS TO BE TESTED AT A TIME.
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT BECAUSE BOTH TRAINS ARE SHOWN.
3. THE BYPASS BREAKER INTERLOCK IS OPERATIVE ONLY WHEN BOTH BY-PASS BREAKERS ARE IN THE OPERATE POSISTION.

CAD FSD F-2 S2
ACAD2007 KB-01

REACTOR TRIP

FIGURE F-2 SHEET 2

VER. 18.0

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Southern Company Services, Inc.
for

ALABAMA POWER COMPANY

PROJECT JOSEPH M. FARLEY NUCLEAR PLANT UNIT NO.1
SUBJECT ELEM. DIAG. REACTOR TRIP SWITCHGEAR

DRAWN CADD/DGD CHECKED IS DESIGNED
APPROVED JESSE E. LOVE FOR K.C. GANDHI DATE 6/24/86
APPROVED DATE
APPROVED DATE

SCALE NONE B/M
SHEET 2 OF 3 SHEETS
SUPERSEDES

D-177198

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH
RVL	MH/ JBL	DEW				
REV. NO. 3 DATE 2/19/98 REVISED PER ABN 97-0-1151, REV. 0						

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH
JMR RVL	BEA and JBL	MA for NJS				
REV. NO. 2 DATE 4/29/96 REDRAWN PER ABN 95-1-0737, REV. 0						

1. 013G2.1.19 016/NEW//C/A 3.9/3.8/013G2.1.19/N///TELL NRC

Unit 1 has experienced a reactor trip and the following conditions exist:

- The operating crew is verifying the immediate operator actions per EEP-0.0, Reactor Trip or Safety Injection.
- MLB-1, 1-1 and 11-1, SAFETY INJECTION, are NOT LIT.

The STA reports the following indications on the Plant Computer:

- PT0455 PRESSURIZER PRESSURE CHAN 1 is 1841 psig.
- PT0456 PRESSURIZER PRESSURE CHAN 2 is 1855 psig.
- PT0457 PRESSURIZER PRESSURE CHAN 3 is 1845 psig.
- PT0444A PRESSURIZER PRESSURE CHAN 4 is 1857 psig.
- PT0445A PRESSURIZER PRESSURE CHAN 5 is 1855 psig.
- MSIV-3369A, B and C indicate RED on the IPC computer screen.

Which one of the following completes the statements below?

A Safety Injection _(1)_ required.

MSIV-3369A, B and C _(2)_ OPEN.

	(1)	(2)
A.	IS	are NOT
B.✓	IS	ARE
C.	is NOT	are NOT
D.	is NOT	ARE

EEP-0.0 -
III. The following are symptoms that require safety injection, if one has not occurred:

SI Signal	Instrumentation	Setpoint	Coinc
1. Pressurizer pressure low	PT 455, 456, 457	1850psig	2/3

FSD-A181007 - Pg 2-26
The Main Steam Line Isolation is initiated by the following:
b. Low steam pressure; = 585 psig on 2/3 S.G.

Distracter Analysis

A. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2). Plausible because a common misconception is that a Safety Injection (SI) causes the MSIV's to go shut. If the applicant believed this, they could be convinced that the MSIV's were shut regardless of the IPC indication. Low steam line pressure of 585 psig causes BOTH and SI and MSIAS.

B. Correct. First part is correct.

SI Signal	Instrumentation	Setpoint	Coinc
1. Pressurizer pressure low	PT 455, 456, 457	1850psig	2/3

Second part is correct. There are NO MSIAS signals present in the stem and a RED indication on the IPC indicates that a valve is open.

C. Incorrect. First part is incorrect (See B.1). Plausible if the applicant has the misconception that the control channels, PT 444A and 445A are used to evaluate pressure instead of the protection channels PT-455, 456 and 457.

Second part is incorrect (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is correct (See B.2).

K/A: **013G2.1.19**	Engineered Safety Features Actuation System (ESFAS) - Ability to use plant computers to evaluate system or component status
Importance Rating:	3.9 3.8
Technical Reference:	FNP-1-EEP-0.0, Reactor Trip or Safety Injection, Ver 44 FSD - A181007, Reactor Protection System, Ver 18
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Emergency Core Cooling System components and equipment, to include the following (OPS-40302C07): [...] • Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level) [...] • Actions needed to mitigate the consequence of the abnormality
Question History:	NEW
K/A match:	Applicant must **evaluate a set of data from the plant computer and based on that determine if ESFAS system actuation is necessary.**
SRO justification:	N/A

for the required engineered safety features lines. Phase B isolation is initiated by containment pressure High-3 (27 psig) or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

The Containment Ventilation Isolation isolates the containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on the completion of the SI logic, high radioactivity levels in the purge exhaust, or by manual initiation of either Phase A Containment Isolation or Phase B Isolation/Containment Spray Actuation. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.080)

3. Main Steam Line Isolation

Isolation of the Main Steam lines limits the effects of an uncontrolled release of steam either inside or outside the containment. For a break upstream of the isolation valves (MSIV) in the steamlines, valves closure will limit the release to the blowdown of the one affected steam generator. A break downstream of the valves is limited to the depressurization of the pipe volume downstream of the valves. This results in a rapid termination of the event and significantly reduces the mass lost from the secondary.

The Main Steam Line Isolation is initiated by the following:

a. High steam line flow with low-low T_{avg} , 1/2 steam flow channels above setpoint (40% of full steam flow between 0-20% load and increasing linearly to 110% at full load) on 2/3 steam lines with $T_{avg} \leq$ P-12

b. Low steam pressure; ≤ 585 psig on 2/3 S.G.

c. High-2 containment pressure; ≥ 16.2 psig on 2/3

d. Manual. By closing each MSIV by operating individual hand switches. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main Feed line Isolation:

III. The following are symptoms that require safety injection, if one has not occurred:

SI Signal	Instrumentation (TSLB)	Setpoint	Coincidence
1. Pressurizer pressure low (If not blocked)	PI-455,456,457 (TSLB2 17-1,17-2,17-3)	1850 psig	2/3
2. Steam Line Differential pressure	PI-474,484,494, PI-475,485,495, PI-476,486,496 (TSLB4 10-2,10-3,10-4, 11-2,11-3,11-4, 12-2,12-3,12-4, 13-2,13-3,13-4, 14-2,14-3,14-4, 15-2,15-3,15-4)	100 psid	1 steam line 100 psig less than other two on 2/3 protection sets
3. Low Steam Line pressure (If not blocked)	PI-474,485,496 (TSLB4 19-2,19-3,19-4)	585 psig (rate compensated)	2/3
4. Containment pressure high	PI-951,952,953 (TSLB1 1-2,1-3,1-4)	4 psig	2/3
5. Manual	N/A	N/A	1/2

IV. The following are symptoms of a safety injection:

a. Any SI annunciator lit.

b. BYP & PERMISSIVE SAFETY INJECTION ACTUATED status light lit

c. MLB-1 1-1 or MLB-1 11-1 lit

d. HHSI flow greater than 0 gpm.

4.2.5 Slowly open main steam isolation bypass warmup valves:

- 1A MS BYP WARMUP VLV, N1N11V019A
- 1B MS BYP WARMUP VLV, N1N11V019B
- 1C MS BYP WARMUP VLV, N1N11V019C

NOTE: The following note is N/A when Appendix 5, DEFEATING THE AUTO-CLOSURE OF MSIV BYPASS VALVES is in effect.

NOTE: When an MSIV is opened, its associated bypass valve will automatically close. The upstream and downstream MSIV for at least one loop must be opened before proceeding to the other loops to maintain the bypass flowpath.

4.2.6 WHEN steam header pressure AND individual steam generator pressures are approximately equal, THEN perform the following:

4.2.6.1 Verify that ALL main steam lines have been drained per section 4.1.

4.2.6.2 Verify open all main steam isolation bypass warmup valves not previously opened in step 4.2.5.

- 1A MS BYP WARMUP VLV, N1N11V019A
- 1B MS BYP WARMUP VLV, N1N11V019B
- 1C MS BYP WARMUP VLV, N1N11V019C

4.2.6.3 OPEN all the MSIVs.

- 1A SG MSIV, Q1N11HV3369A
- 1A SG MSIV, Q1N11HV3370A
- 1B SG MSIV, Q1N11HV3369B
- 1B SG MSIV, Q1N11HV3370B
- 1C SG MSIV, Q1N11HV3369C
- 1C SG MSIV, Q1N11HV3370C

NOTE: The following step is N/A when Appendix 5, DEFEATING THE AUTO-CLOSURE OF MSIV BYPASS VALVES is in effect.

4.2.7 WHEN the MSIVs reach the fully open position, THEN verify that the associated bypass valves automatically close.

1. 014K5.01 017/BANK/FNP 01/MEM 2.7/3.0/014K5.01/N///

Unit 1 is performing a reactor startup per UOP-1.2, Startup Of Unit From Hot Standby To Minimum Load, when the following conditions occurred:

- The OATC pulled Control Bank D to 100 Steps by Step Demand Counter.
- Rod B8 was noted to be indicating 54 Steps by DRPI.

Which one of the following completes the statement below?

Rod B8's position is __(1)__ .

Per Tech Spec Bases 3.1.7, Rod Position Indication, __(2)__ is(are) the most reliable indication.

A. 1) exactly 100 steps

2) the group step counters

B. 1) approximately 100 steps

2) the group step counters

C. 1) exactly 54 steps

2) DRPI

D.✓ 1) approximately 54 steps

2) DRPI

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ? inch). **If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.**

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 4 steps (all coils operable and 1 step added for manufacturing and temperature tolerances), and the maximum uncertainty is ± 10 steps (only one data system A or B coils operable). With an indicated deviation of 12 steps between the

group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 22 steps.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible if the applicant does not recall that the Step Demand Counter only counts electrical impulses and thinks it actually measures rod location.

Second part incorrect (See D.2). Plausible if the applicant thinks that since the group step counters are the most precise indication, they are the most reliable.

B. Incorrect. First part is incorrect (See D.1). Plausible if the applicant does not recall that the Step Demand Counter only counts electrical impulses. Each bank has 2 Step Demand Counters that step in 1 step increments. Control Bank D Group 1 will move 1 step then Control Bank D Group 2 will move one step. If bank 1 moves 1 step and the rod control switch is released before bank 2 moves, bank 1 would be at 100 steps and bank 2 would be at 99 steps. This is commonly know as $99^{1/2}$ steps. The applicant may reason that this is a potential reason to call rods by step counter as approximate.

Second part is incorrect (See A.2).

C. Incorrect First part is incorrect (See D.1). Plausible since the DRPI lights change only every 6 steps and 54 steps is a DRPI display light location. The applicant may think that since a DRPI display light is lit, the rod is exactly at that position.

Second part is correct (See D.2).

D. Correct. First part is correct. Since DRPI measures actual rod position based on the location of the rod in reference to the measurement coils and the step counter only counts electrical pulses, the rod is at ~54 steps. Also, the accuracy of DRPI is $\pm$ 4 steps so the rod height is approximate.

Second part is correct. DRPI is the most reliable because it actually senses the location of the rod using coils.

K/A: **014K5.01**	Rod Position Indication System (RPIS) - Knowledge of the operational implications of the following concepts as they apply to the RPIS: Reasons for differences between RPIS and step counter
Importance Rating:	2.7 3.0
Technical Reference:	FNP Technical Specifications Bases, Ver 58
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Digital Rod Position Indication System (OPS-52201F02): • Rod Position Detectors
Question History:	FNP 01
K/A match:	This question r**equires the applicant to determine the location of rod B8 (operational implication)** based on their **knowledge of the differences, based on design, of how rod heights are measured between rod control (step counters) and DRPI in that DRPI is the most reliable indication.**
SRO justification:	N/A

TABLE 3
FULL ACCURACY INDICATION

Indication from Data A and Data B						
Actual Rod Position In Steps	Number of Actual Coils Penetrated	Number of Coils Assumed to Be Penetrated	Height of Highest Actual Coil Penetrated	Height of Highest Assumed Coil Indicated Penetrated	Indicated Position	Accuracy*
35	6	6	33	33	36	-1
36	6	6	33	33	36	0
37	6	6	33	33	36	+1
38	6	6	33	33	36	+2
39	6/7	6/7	33/39	33/39	36/42	+3/-3
40	7	7	39	39	42	-2
41	7	7	39	39	42	-1
42	7	7	39	39	42	0
43	7	7	39	39	42	+1
44	7	7	39	39	42	+2
45	7/8	7/8	39/45	39/45	42/48	+3/-3
46	8	8	45	45	48	-2
47	8	8	45	45	48	-1
48	8	8	45	45	48	0
49	8	8	45	45	48	+1
50	8	8	45	45	48	+2
51	8/9	8/9	45/51	45/51	48/54	+3/-3
52	9	9	51	51	54	-2
53	9	9	51	51	54	-1
54	9	9	51	51	54	0
55	9	9	51	51	54	+1
56	9	9	51	51	54	+2
57	9/10	9/10	51/57	51/57	54/60	+3/-3
58	10	10	57	57	60	-2
59	10	10	57	57	60	-1
60	10	10	57	57	60	0
61	10	10	57	57	60	+1
62	10	10	57	57	60	+2
63	10/11	10/11	57/63	57/63	60/66	+3/-3
64	11	11	63	63	66	-2
65	11	11	63	63	66	-1
66	11	11	63	63	66	0
67	11	11	63	63	66	+1
68	11	11	63	63	66	+2
69	11/12	11/12	63/69	63/69	66/72	+3/-3

*Actual Rod Position with Respect to Indicated Position

BASES

BACKGROUND (continued)

The axial positions of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ? inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 4 steps (all coils operable and 1 step added for manufacturing and temperature tolerances), and the maximum uncertainty is ± 10 steps (only one data system A or B coils operable). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 22 steps.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the assumed group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must

(continued)

1. 015/17AA2.02 018/NEW//MEM 2.8/3.0/APE015/017AA2.02/N///

Unit 1 is operating at 100% power.

The following occurs:

- MOV-3052, CCW TO RCP CLRS, closes.
- DD3, CCW FLOW FROM RCP OIL CLRS LO, comes in to alarm.

Which one of the following completes the statements below?

The most limiting components for this event are the RCP __(1)__.

The RCPs will be required to be stopped within approximately __(2)__.

A.✓ 1) Motor Bearings

2) 2 minutes

B. 1) Motor Bearings

2) 60 minutes

C. 1) Pump Lower Radial Bearings

2) 2 minutes

D. 1) Pump Lower Radial Bearings

2) 60 minutes

DD1 - CAUTION: RCP's with #1 Seal Leakoff less than 2.5 gpm may develop lower bearing and seal temperatures that exceed 225°F within 1 to 2 hours following a loss of seal injection.

DD3 - On a complete Loss of CCW Flow to RCP Motor Bearing Oil Coolers, the bearing temperature will exceed 195°F in approximately 2 minutes.

4. IF any RCP Motor Bearing Temperature exceeds 195°F, THEN:

A. IF the Reactor is critical, THEN trip the reactor.
B. Stop the RCP.
C. Perform the actions required by FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
D. Perform action of FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW as time allows.

MOV-3052 isolate CCW flow to the RCP oil coolers and the RCP thermal barrier heat exchanger which functions to cool the lower radial bearing on a loss of RCP seal

injection flow.

Distracter analysis

A. Correct. First part is correct. The RCP motor bearings are the most limiting components for this scenario since the CCW flow is lost and RCP motor bearing temperatures will increase rapidly.

Second part is correct. The RCP motor bearing temperatures will increase to 195°F with 2 minutes. The ARP has the operator trip the reactor and secure all RCPs for this failure.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2) Plausible since 60 minutes is the time the lower radial bearing temperature will rise in 1-2 hours on a loss of RCP's with #1 Seal Leakoff less than 2.5 gpm following a loss of seal injection. Plausible since this is a time requirement for a RCP malfunction on the same Annunciator panel as DD3.

C. Incorrect. First part is incorrect (See A.1) RCP lower radial bearings are cooled from two sources. A loss of the CCW will not cause the lower radial bearing temperatures to rise. Plausible since CCW is normal cooling to components and this is one of a few components with 2 cooling sources.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2).

K/A: **015AA2.02**	Reactor Coolant Pump (RCP) Malfunctions - Ability to determine and interpret **Abnormalities in RCP** air vent flow paths and/or **oil cooling system** as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):
Importance Rating:	2.8 3.0
Technical Reference:	FNP-1-ARP-1.4, DD1 and DD3, Ver 53
References provided:	None
Learning Objective:	LIST AND DESCRIBE the sequence of major actions associated with AOP-9.0, Loss of Component Cooling Water. (OPS-52520I04). EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing AOP-9.0, Loss of Component Cooling Water. (OPS-52520I06).
Question History:	NEW
K/A match:	The malfunction of the RCP is the closure of MOV-3052. The applicant must determine/interpret that a loss of CCW to the RCP oil coolers **and** lower radial bearings has resulted, then must interpret how this malfunction affects the RCP components (oil coolers and seal) and the time required for action to be taken.
SRO justification:	N/A

OPERATOR ACTION CONT'D

6. IF operation of SEAL WTR INJECTION HIK-186 is erratic and operation via the bypass is desired, THEN refer TO FNP-1-SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION, Section 4.16 for guidance.

7. Check the on-service Seal Injection Filter ΔP.

CAUTION:	**RCP's with #1 Seal Leakoff less than 2.5 gpm may develop lower bearing and seal temperatures that exceed 225°F within 1 to 2 hours following a loss of seal injection.**

CAUTION:	**Long term operation of the RCP under loss of seal injection conditions increases the risk of a loss of all seal cooling incident if the CCW system supply to the RCP thermal barrier should also fail.**

CAUTION:	**RCS water being cooled by the RCP thermal barrier contains particulate matter, since it is not filtered, which can have an impact on the long term operability of the RCP seals.**

CAUTION:	**Particulate matter which will deposit in the seal package as the result of a loss of seal injection will be radioactive and will impact dose to received by workers during repair efforts.**

8. IF a Loss of Seal Injection Flow has occurred, THEN;
   a) Ensure that RCP, Component Cooling Water inlet temperature remains below 105°F.
   b) Ensure that at least one Charging Pump is running.
   c) Carefully re-establish the Injection Water Flow, reducing the RCP Lower Bearing Temperatures at a maximum rate of 1°F per minute, using SEAL WTR INJECTION HIK 186, or by establishing flow via HI-186 bypass using FNP-1-SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION, Section 4.16 for guidance.

LOCATION DD3

SETPOINT: 100 + 10 GPM
- 0

ORIGIN:
1. Flow Switch (Q1P17FISL3048A-N)
2. Flow Switch (Q1P17FISL3048B-N)
3. Flow Switch (Q1P17FISL3048C-N)

D3
CCW FLOW FROM RCP OIL CLRS LO

PROBABLE CAUSE

NOTE:	**Following entry into Mode 6 during a refueling outage, it is common to receive alarm DD3 due to low discharge pressure on the O/S pump when aligned to the SFP and RHR HXs, and the RHR seal and charging pump oil coolers.** **AI 2009203964**

1. Loss of Component Cooling Water.
2. Loss of Component Cooling Water Flow to the RCP's due to Phase "B" isolation signal.
3. Improper valve lineup.

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. Determine the cause of the alarm.

2. IF a loss of Component Cooling Water has occurred, THEN perform the actions required by FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.

3. Closely monitor the RCP's Motor Bearing Temperatures.

NOTE:	**On a complete Loss of CCW Flow to RCP Motor Bearing Oil Coolers, the bearing temperature will exceed 195°F in approximately 2 minutes.**

4. IF any RCP Motor Bearing Temperature exceeds 195°F, THEN:
   A. IF the Reactor is critical, THEN trip the reactor.
   B. Stop the RCP.
   C. Perform the actions required by FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
   D. Perform action of FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW as time allows.

5. Correct the cause of the alarm and return flow to normal.

References: A-177100, Sh. 198; B-175968, Pg. 6 & 7; D-175002, Sh. 2; U-258242

1. 017K6.01 019/NEW//C/A 2.7/3.0/017K6.01/N///

Unit 1 has experienced a Reactor Trip and SI due to a LOCA and the following conditions exist:

- The operators have transitioned to EEP-1.0, Loss of Reactor or Secondary Coolant.
- The Core Exit Thermocouples (CETCs) are reading as follows:
  - TWO CETCs are indicating a SHORT circuit.
  - THREE CETCs are 1204°F and rising.
  - All other CETCs are reading between 950°F and 1150°F and rising.

Which one of the following completes the statements below?

The indication for the SHORT circuited CETCs fail __(1)__.

The crew is required to enter __(2)__.

Procedures:
• FRP-C.1, Response To Inadequate Core Cooling.
• FRP-C.2, Response To Degraded Core Cooling.

	(1)	(2)
A.	HIGH	FRP-C.1
B.	HIGH	FRP-C.2
C.	LOW	FRP-C.1
D.✓	LOW	FRP-C.2

CSF-0.2
5th hottest CETC <1200 °F? **NO** → Go to FRP-C.1

↓ **YES**

RCS SUBCOOLING from CETC > 16°F {45°F}? **NO** → 5th hottest CETC <700 °F? **NO** → Go To FRP-C.2

↓ **YES**

CSF - SAT

CSB-0.0 Core Cooling Block Decision
Analyses of inadequate core cooling scenarios (References 1 and 2) show that core exit temperature greater than 1200°F is a satisfactory criterion for basing extreme operator action. **At least 5 thermocouples should be reading greater than 1200°F.**

Five has been chosen to allow for thermocouples failing high. This temperature indicates that most liquid inventory has already been removed from the RCS and that core decay heat is superheating steam in the core. An extreme challenge to the fuel matrix/clad barrier is imminent and a RED priority is warranted. The appropriate guideline for functional response is FR C.1, RESPONSE TO INADEQUATE CORE COOLING. If core exit thermocouples are less than 1200°F, then subsequent blocks check for other extreme, severe, not satisfied or satisfied conditions for the safety function.

U263686 pg *3-5*
The signal conditioning panel contains the open thermocouple detection circuitry, noise filtering capacitors, and the cold reference junction compensation circuitry. Cold junction compensation is accomplished by measuring the barrier temperature utilizing a semiconductor temperature sensor located on the signal conditioning panel. The temperature sensor circuit produces an output voltage, that is equivalent to the temperature Of the barrier strip. This output voltage is read in through one of the channels on the Analog Input Boards (DT1748 and DT1748-24EX boards'. The thermocouple signals are then compensated in the software by adding the value of the cold reference junction to the thermocouple signals. **If any of the thermocouples are open or shorted the signal conditioning panels open thermocouple detection circuitry will cause the input to be driven down to 0V.**

<u>Distracter analysis</u>

A. Incorrect. First part is incorrect (See D.1). Plausible if the applicant does not recall if a thermocouple fails high or low when shorted. An RTD that experiences an open circuit will cause a high temperature reading. The applicant could confuse RTD and thermocouple operating theory.

Second part is incorrect (See D.2). Plausible since this is the correct answer if the shorted thermocouple failed high.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2). This is a logical connection to the first part if the applicant discounts the "failed high" CETCs when evaluating FRP entry requirements. The background documentation uses the fifth hottest CETC because of potential failed high CETCs.

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See D.2). Plausible if the applicant improperly believes that any CETC above 1200°F meets entry requirements for FRP-C.1.

D. Correct. First part is correct. Thermocouples that are shorted fail low.

Second part is correct. With the 2 CETCs failed low due to a short circuit, the fifth hottest CETC is > 700°F and < 1200°F and FRP-C.2 entry is required.

K/A: **017K6.01**	In-Core Temperature Monitor System (ITM) - Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors
Importance Rating:	2.7 3.0
Technical Reference:	FNP-1-CSF-0.2 Core Cooling, Ver 17 FNP-0-CSB-0.0, Specific Background Document For FNP-1/2-CSF-0, Critical Safety Function Status Trees, Ver 1 U-263686, ICCMS Tech Manual Vol II, Ver 2 OPS-31701G, Sensors and detectors, Ver 4
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Inadequate Core Cooling Monitor System (OPS-52202E02): • CETC Monitor
Question History:	NEW
K/A match:	The applicant is required to **have a knowledge of the effect of a shorted CETC on the incore temperature monitoring system.**
SRO justification:	N/A

Section: **Core Cooling Block Decision**

Unit 1 ERP Step: 1 **Unit 2 ERP Step:** 1 ***ERG Step No:***

ERP StepText: FIFTH HOTTEST CORE EXIT TC LESS THAN 1200°F

ERG StepText: *Core Exit TCs Less Than 1200°F*

Purpose: To determine if inadequate core cooling has been reached.

Basis: Analyses of inadequate core cooling scenarios (References 1 and 2) show that core exit temperature greater than 1200°F is a satisfactory criterion for basing extreme operator action. At least 5 thermocouples should be reading greater than 1200°F. Five has been chosen to allow for thermocouples failing high. This temperature indicates that most liquid inventory has already been removed from the RCS and that core decay heat is superheating steam in the core. An extreme challenge to the fuel matrix/clad barrier is imminent and a RED priority is warranted. The appropriate guideline for functional response is FR C.1, RESPONSE TO INADEQUATE CORE COOLING. If core exit thermocouples are less than 1200°F, then subsequent blocks check for other extreme, severe, not satisfied or satisfied conditions for the safety function.

Knowledge:

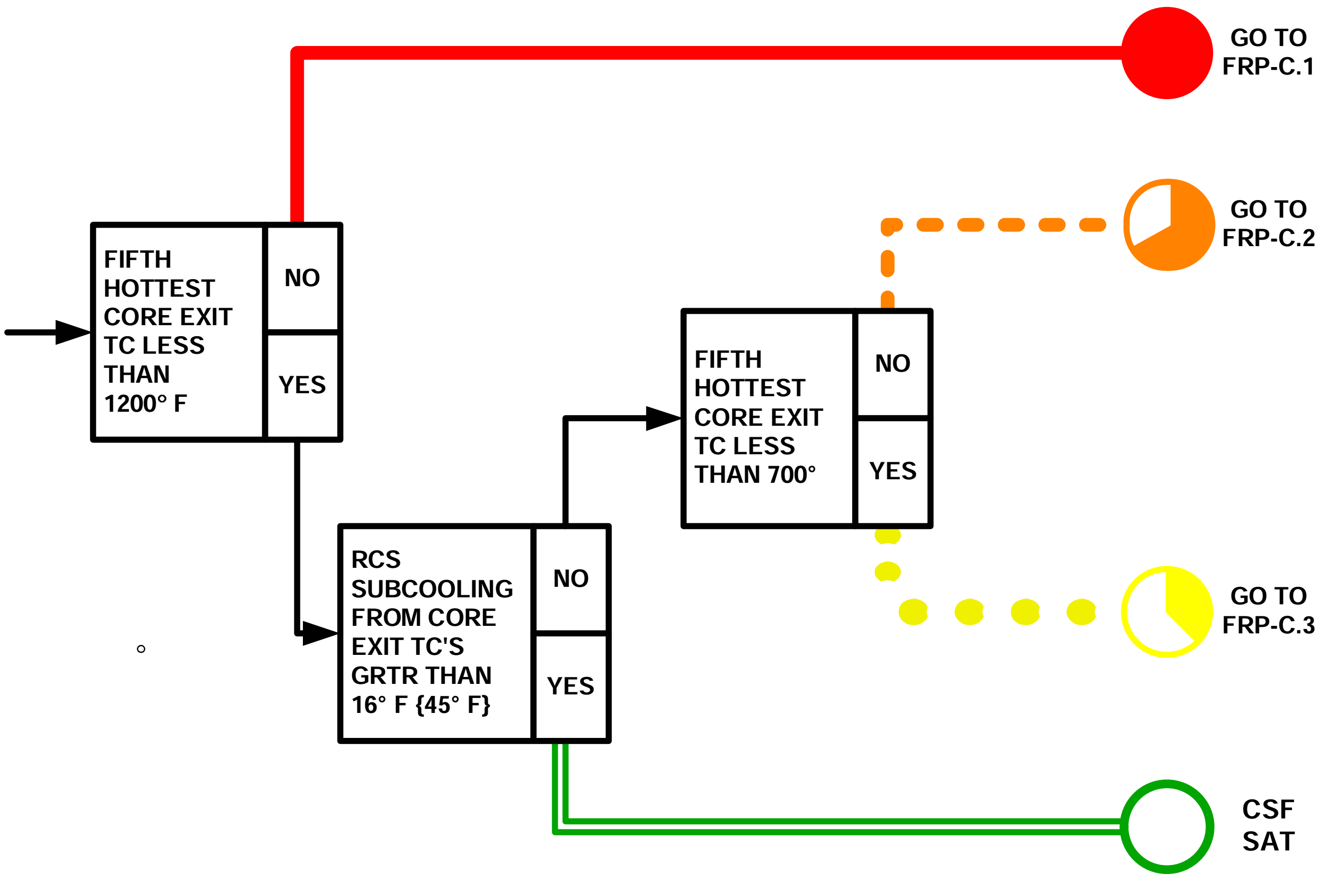
References: N/A

Justification of Differences:

1 Changed to make plant specific.

2 Changed to specify fifth hottest core exit thermocouple. This is consistent with the basis which states that at least five thermocouples should be indicating above the setpoint to allow for possible thermocouple failures.

3 FNP has installed a CE HJTCS RVLIS instead of a Westinghouse RVLIS. Since Westinghouse recommendations for incorporation of the CE HJTCS RVLIS did not include reference to the CSF status trees, FNP utilized the Core Cooling status tree without RVLIS.


FIFTH HOTTEST CORE EXIT TC LESS THAN 1200° F
NO
YES
GO TO FRP-C.1
RCS SUBCOOLING FROM CORE EXIT TC'S GRTR THAN 16° F {45° F}
NO
YES
FIFTH HOTTEST CORE EXIT TC LESS THAN 700°
NO
YES
GO TO FRP-C.2
GO TO FRP-C.3
CSF SAT

FAILURE INDICATIONS

Although the temperature measurement devices just discussed are accurate and reliable over a wide range of temperatures, the operator needs to be sensitive to instrument failures. Mechanical failures in liquid-in-glass thermometers, filled-system thermometers, and bimetallic strip thermometers are easy to determine. Failures in the filled-system are primarily due to a leak in the filled-system tubing causing the instrument to fail low. Failures of the bimetallic strip thermometers are primarily due to a break between the connections of the dissimilar metals, also causing the instrument to fail low.

Failures in the thermocouple and RTD devices are typically indicated electrically either as an open or a short in the circuit. The thermocouple operates on the principle that a voltage is developed when two dissimilar metals are joined and there is a temperature difference between the sensing junction and the reference junction. The voltage created causes current to flow. If an open develops, a path for current flow is no longer available, and therefore the output fails to a low temperature indication. A short circuit causes no voltage to be developed, and the thermocouple indicator fails low.

If a RTD fails open, the Wheatstone bridge sees a large resistance, which is comparable to a high temperature. Therefore, a maximum temperature is indicated. Conversely, if a RTD develops a short circuit, the bridge network sees a low resistance and indicates a minimum temperature.

COMPARISON OF TEMPERATURE SENSORS

For remote reading temperature indications the choice between using a thermocouple and a resistance temperature detector can be based on several considerations. The advantages of a RTD over a thermocouple are:

- RTDs are better suited for small temperature bands.
- RTDs have a higher output voltage which means less auxiliary equipment is required to boost output signals.
- RTDs do not require reference junctions.
- RTDs’ circuitry is more tolerant to electrical noise.
- RTDs have an increased sensitivity to small changes in temperature.
- RTDs are more accurate than thermocouples.

The primary advantages of thermocouples are:

- Thermocouples are more rugged than RTDs.
- Thermocouples are well suited for large temperature bands.
- Thermocouple sensing wires can be drawn very thin, giving a very fast response to temperature changes.

3.3.1 Thermocouple Signals

Type K thermocouples are used to transmit the HJTC and CETC signals to the ICCMS Cabinet. At the Cabinet these field wires are brought to type K Terminal blocks. The Type K thermocouple wire is continued from the terminal blocks to the ICCMS Monitoring Panel (disconnect panel) and then to the Signal Conditioning Panels located on the back of the microprocessor chassis. The signals are then read in by the Analog Input board, converted to digital signals, and stored in an input buffer for use in reactor vessel level and core exit temperature calculations.

The signal conditioning panel contains the open thermocouple detection circuitry, noise filtering capacitors, and the cold reference junction compensation circuitry. Cold junction compensation is accomplished by measuring the barrier temperature utilizing a semiconductor temperature sensor located on the signal conditioning panel. The temperature sensor circuit produces an output voltage, that is equivalent to the temperature of the barrier strip. This output voltage is read in through one of the channels on the Analog Input Boards (DT1748 and DT1748-24EX boards). The thermocouple signals are then compensated in the software by adding the value of the cold reference junction to the thermocouple signals. If any of the thermocouples are open or shorted the signal conditioning panels open thermocouple detection circuitry will cause the input to be driven down to 0v.

3.3.2 SMM Input Signals

The SMM input signals are transmitted from the RCS loops in containment (CTMT) to the ICCMS cabinet using copper wire. At the cabinet these field wires are brought to copper terminal blocks. The copper wire is continued from the terminal blocks to the ICCMS monitoring panel and then to the 52 pin MS connector (J4) located on

1. 022A1.03 020/NEW//C/A 3.1/3.4/022A1.03/N///

Unit 1 is operating at 100% power when a steam break in Containment occurs and the following conditions exist:

At 1000:

- Containment Pressure is 18 psig.
- Containment temperature is 145°F.

At 1010:

- Containment pressure is 25 psig.

Which one of the following completes the statements below?

At 1000, the Containment Cooler discharge will be through the __(1)__ .

At 1010, the Containment Cooler fans will be drawing __(2)__ amps than at 1000.

	(1)	(2)
A.	ductwork	MORE
B.✔	dropout plate	MORE
C.	ductwork	LESS
D.	dropout plate	LESS

FSD-A181013
3.1.2.5 A 125 hp motor is provided for fan operation to meet the design brake horsepower requirement of 105 hp during low-speed operation following a LOCA. During normal operation, the design brake horsepower of the fan in high-speed operation is 80 hp. The fan motors are rated at 550 VAC, 3-phase, 60 Hz. See Section 3.1.6, for cumulative radiation dose (Reference 6.5.002).

3.1.2.2 For post-LOCA operation, the fans are reduced to low speed due to changes in air density and develop a minimum design flow rate of 40,000 cfm at an external static pressure of 1 in. w.g. (References 6.3.010, 6.5.002). Containment pressure and temperature analysis is based on the operation of one containment cooler with a degraded service water flow of 600 gpm (Reference 6.3.002).

3.7.1.2 During a LOCA or MSLB, a rise in containment temperature, humidity, and density occurs. T**his density increase results in an increase in fan brake horsepower because the air is heavier and the friction losses in the ductwork are higher.** To prevent the fans from overloading and to provide increased air cooling flow, the fusible link on the dropout plates melts and the plates fall open, bypassing the plenums, and “dumping” the cooler air into containment at the discharge of the coolers

3.7.2.1 [...] fusible links that are designed to melt at approximately 135 °F,

Distracter analysis

A. Incorrect. First part is incorrect (See B.1). When containment temperature reaches ~135°F, the dropout plates fall open. Plausible if the applicant believes that the links melt at 160°F to 170°F which is when the fire damper fusable links melt.

Second part is correct (See B.2).

B. Correct. First part is correct. The dropout plates open at ~ 135°F.

Second part is correct. The increase in humidity and friction in Containment due to the steam break will cause the fan motors to overload or draw more current.

C. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See B.2). Plausible if the applicant thought that the increased temperature caused the density of the air to lower and did not take into consideration the rise in density due to the steam. They may believe that this would produce less air friction and therefore less current flow.

D. Incorrect. First part is correct (See B.1).

Second part is incorrect (See C.2). Plausible if the applicant thought the dropout plate was closed at 1000.

K/A: **022A1.03**	Containment Cooling System (CCS) - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment humidity
Importance Rating:	3.1 3.4
Technical Reference:	FSD-A181013, Containment Ventilation System, Ver 14.
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Containment Spray and Cooling System components and equipment, to include the following (OPS-40302D07): • Normal Control Methods • Abnormal and Emergency Control Methods • Automatic actuation including setpoint (example SI, Phase-B, LOSP) and the effect of selecting the containment cooler control to local.
Question History:	NEW
K/A match:	Requires the applicant to **predict changes in containment humidity and based on this know that the containment cooler fans will draw more current. The dropout plate will actuate at 135°F to ensure that cooler fans do not exceed their design current and trip on overload.**
SRO justification:	N/A

3.0 CRITICAL COMPONENT FUNCTIONAL DESIGN REQUIREMENTS

This section contains functional design requirements for all major components within the CVS system that are considered to be critical to safety-related system functions (i.e., failure of the component could lead to loss or impairment of safety-related system functions). The functions of each component, in addition to performance and interface requirements, are discussed.

3.1 CONTAINMENT COOLERS

TPNS Nos. QE12H001A, B, C, D

3.1.1 Basic Function

3.1.1.1 The containment cooling system is designed to provide containment atmosphere mixing and cooling (Reference 6.5.002).

3.1.1.2 The containment coolers are designed to maintain a maximum bulk average containment ambient temperature of 120 °F during normal operation. This 120 °F temperature is a design input assumption used in the calculation of containment peak temperature and pressure following a LOCA. Following an accident, the containment coolers provide for long-term containment heat removal (Reference 6.3.002).

3.1.1.3 The design basis for the containment pressure and temperature accident analysis models the operation of one cooler for long-term heat removal at a degraded service water flow of 600 gpm. Containment cooler delay time is modeled as 115 seconds (Reference 6.3.002).

3.1.2 Functional Requirements

3.1.2.1 Each cooler was originally designed to handle 1/3 of the normal plant operating containment heat load. Sensible heat gain for the three units combined is 5.25 x 106 Btu/hr based on low reactor coolant leakage (or 1.75 x 106 Btu/hr per unit). High reactor coolant leakage includes an additional latent heat load of 1.8 x 106 Btu/hr, for a total of 7.05 x 106 Btu/hr (or 2.35 x 106 Btu/hr per cooler). Fans were sized for a nominal 80,000 cfm at an external static pressure of 3 in. w.g. The specified limits of the cooler components is designed to be compatible with a cumulative radiation dose of 2 x 108 rads (References 6.3.010, 6.5.002).

3.1.2.2 For post-LOCA operation, the fans are reduced to low speed due to changes in air density and develop a minimum design flow rate of 40,000 cfm at an external static pressure of 1 in. w.g. (References 6.3.010, 6.5.002). Containment pressure and temperature analysis is based on the operation of one containment cooler with a degraded service water flow of 600 gpm (Reference 6.3.002).

3.1.2.3 Sizing of the containment coolers on the water side was based on a maximum inlet service water temperature of 95 °F with a nominal flow rate of 800 gpm/cooler for normal plant operation and 2000 gpm/cooler for post-LOCA operation (Reference 6.3.010). A total of 69 circuits (equivalent of 2 coils plus 5 circuits) of any containment cooler can be removed from service and not degrade the containment cooling performance to a point that the cooler can not adequately remove the containment analysis heat loads. This analysis was performed using the design basis fouling factor of 0.003 (References 6.3.020, 6.3.021) Coolers have been reconfigured with a waterbox design to allow for individual tube plugging, and cooler coils have been replaced with erosion/corrosion resistant materials (Reference 6.7.052).

Water hammer analysis has identified that the region of the service water piping that is most susceptible to water hammer is the containment cooler return piping. However, this analysis showed that no water hammer will occur in this piping. Due to the elevation differences between the service water pond, the containment coolers, and the river discharge, a substantial flow of service water through the containment coolers will continue in the event of a LOSP. The maximum temperature of the water achievable in the water hammer region of the service water return piping prior to Service Water Pump restart is 119 °F. This temperature, being less than the 164°F required to form a vapor cavity in the return piping, denies the possibility of water hammer when the system is repressurized due to pump restart after a LOSP (Reference 6.3.023, 6.7.053).

3.1.2.4 During post-accident conditions, the air density changes from a nominal value of 0.07 lb/ft3 to 0.19 lb/ft3. This change necessitates a larger motor to have the ability to operate at 80,000 scfm (Δhorsepower is proportional to Δdensity). Therefore, to reduce the power demand on the emergency diesels, the fan speeds are reduced. However, in order to provide additional cooling required during post-accident, the amount of cooling water is increased from a nominal 800 gpm to 2000 gpm (Reference 6.3.010).

3.1.2.5 A 125 hp motor is provided for fan operation to meet the design brake horsepower requirement of 105 hp during low-speed operation following a LOCA. During normal operation, the design brake horsepower of the fan in high-speed operation is 80 hp. The fan motors are rated at 550 VAC, 3-phase, 60 Hz. See Section 3.1.6, for cumulative radiation dose (Reference 6.5.002).

3.1.2.6 Performance of the containment coolers during various degraded conditions is assessed in calculation SM-94-0711-002 (Reference 6.3.022).

3.1.3 Code Requirements

3.1.3.1 Various codes and standards are referenced in the original specification for the coolers. The primary code of reference is the 1971 edition of ASHRAE with the containment cooler coils governed by ASME Boiler and Pressure Vessel Code, Section VIII, 1971 edition since the coils are a pressure boundary component (Reference 6.5.002).

3.1.3.2 The fan motors are Category IE2 and shall meet the requirements of NEMA MG1, 1967 (Reference 6.5.002).

3.1.4 Seismic Qualification Requirements

The fans and coolers shall be designated as Seismic Category I. That is, they are designed to withstand, without exceeding normal allowable working stresses and without loss of function, the OBE requirements of 0.05 g horizontal and 0.033 g vertical accelerations. They are also designed to withstand, without loss of function, an SSE of 0.1 g horizontal and 0.067 g vertical (Reference 6.5.002).

3.1.5 I & C Requirements

Service	TPNS No.
Fan Start/Stop - Hi Speed	HS-3186AA, AD, BA, BD, CA, CD, DA, DD
Fan Start/Stop - Lo Speed	HS-3186AB, AE, BB, BE, CB, CE, DB, DE
Local/Remote Selector	HS-3186AC, BC, CC, DC
Sequencer Fan Selector	HS-3186G, H
Outlet Damper	MOV-3186A, B, C, D
Damper Position	ZS-3186A, B, C, D
Cooler Outlet Temperature	TE-3187A, B, C, D (Noncritical) TE-3192A, B, C, D (Noncritical)

(spring return to AUTO) with indicating lights (green for closed and red for open). The indicating lights shall be actuated by damper limit switches (Reference 6.4.046).

3.6.5.2 Upon receipt of a Phase A containment isolation signal, the valves shall automatically close. The configuration of the control circuit shall be such that holding the control switch in the open position will override the automatic closure signal (Reference 6.4.046).

3.6.6 EQ Requirements

The reactor cavity cooling isolation damper solenoid valves and limit switches shall be environmentally qualified as detailed in the Master List of Environmentally Qualified Equipment and EQ Packages 2, 25A, and 25C. Maintenance specified in the EQ packages shall be performed to maintain qualified life (References 6.4.018, 6.4.019, 6.7.013).

3.6.7 Interface Requirements

3.6.7.1 The instrument air system shall supply clean, dry air at a range of 80 to 100 psig for the air operators (Reference 6.5.009).

3.6.7.2 For interfaces with the electrical distribution system, refer to Table T-1.

3.7 POST-LOCA FUSIBLE LINK DROPOUT PLATES

TPNS Nos. Q1E12XV001, 2, 3, 4
Q2E12XV005, 6, 7, 8

3.7.1 Basic Function

3.7.1.1 During normal plant operation, the dropout plates are designed to remain closed to direct containment cooling flow to the containment air cooling header/plenum (Reference 6.4.053).

3.7.1.2 During a LOCA or MSLB, a rise in containment temperature, humidity, and density occurs. During a LOCA or MSLB, the air density is expected to increase from a normal range of 0.070 - 0.075 lb/ft3 to 0.19 lb/ft3. This density increase results in an increase in fan brake horsepower because the air is heavier and the friction losses in the ductwork are higher. To prevent the fans from overloading and to provide increased air cooling flow, the fusible link on the dropout plates melts and the plates fall open, bypassing the plenums, and “dumping” the cooler air into containment at the discharge of the coolers (Reference 6.5.002).

3.7.2 Functional Requirements

3.7.2.1 Each containment cooler shall have a plate held in place by fusible links that provides approximately 20 ft2 of area when open. Each plate has 8 pairs of fusible links that are designed to melt at approximately 135 °F, allowing the plate to open (Reference 6.4.047).

3.7.2.2 The fusible links must separate and the dropout plate must open within 20 seconds of exposure to an air stream of 200 °F. This provides an increased air cooling flow and prevents the fans from overloading (Reference 6.5.017).

3.7.2.3 A nylon bushing is located between the fusible link and hook portion of the eyebolt to reduce heat transfer to the eyebolt and to ensure the fusible links will open within 20 seconds (References 6.7.041, 6.7.042).

3.7.3 Code Requirements

Fusible links are required to meet Underwriters' Laboratories Standards 555-1970 and 33-1968 (References 6.5.002, 6.7.028).

3.7.4 Seismic Qualification Requirements

The fusible link plates have been qualified as Seismic Class I as part of the complete containment cooler/fan unit (Reference 6.5.002).

3.8 FIRE DAMPERS

TPNS Nos. Q1P13XV001
Q1P13XV002
Q1P13XV003
Q1P13XV004
N2P13XV005
N2P13XV006
N2P13XV007
N2P13XV008

3.8.1 Basic Function

Redundant fire dampers are provided in the containment purge supply and exhaust ductwork which penetrates the penetration room. Fire dampers are provided to prevent the ductwork penetration from providing a path for a fire

event to spread from one fire zone or area to another (References 6.4.002, 6.4.044).

3.8.2 Functional Requirements

3.8.2.1 Fire barriers shall maintain the fire boundary integrity during a postulated fire on either side of the boundary. These fire dampers are rated for a 3 hour fire with a fusible link having a melting temperature rating in the range of 160 to 175 °F (References 6.4.048, 6.4.049).

3.8.2.2 Fire dampers Q1P13XV001, Q1P13XV004, and N2P13XV005 are trap door type, whereas Q1P13XV002, Q1P13XV003, N2P13XV006, N2P13XV007, and N2P13XV008 are curtain type (References 6.4.048, 6.4.049).

3.8.3 Code Requirements

The fire dampers shall conform to the requirements of National Fire Protection Association (NFPA) 90A and shall carry the UL label (Reference 6.5.001).

3.8.4 Seismic Requirements

Of the above listed fire dampers, all are Seismic Category I except for N2P13XV005 and N2P13XV006, which are designated as Seismic Category II.

The ductwork specification requires that the fire dampers be able to withstand an inertial load of 3.0 g in the horizontal and vertical directions simultaneously in addition to normal operating loads. Furthermore, the extended parts of the dampers shall have a vibration frequency greater than 20 cps. Electrical switches or other activating mechanisms shall withstand the inertial load without changing position and accidentally causing a change of position of the damper blades (Reference 6.5.001).

3.9 CONTAINMENT PURGE RADIATION MONITORING

TPNS No. QD11RE0024A, B
QD11RI0024A, B
QD11RISH0024A, B
QD11RSHH0024A, B
QD11RAH0024

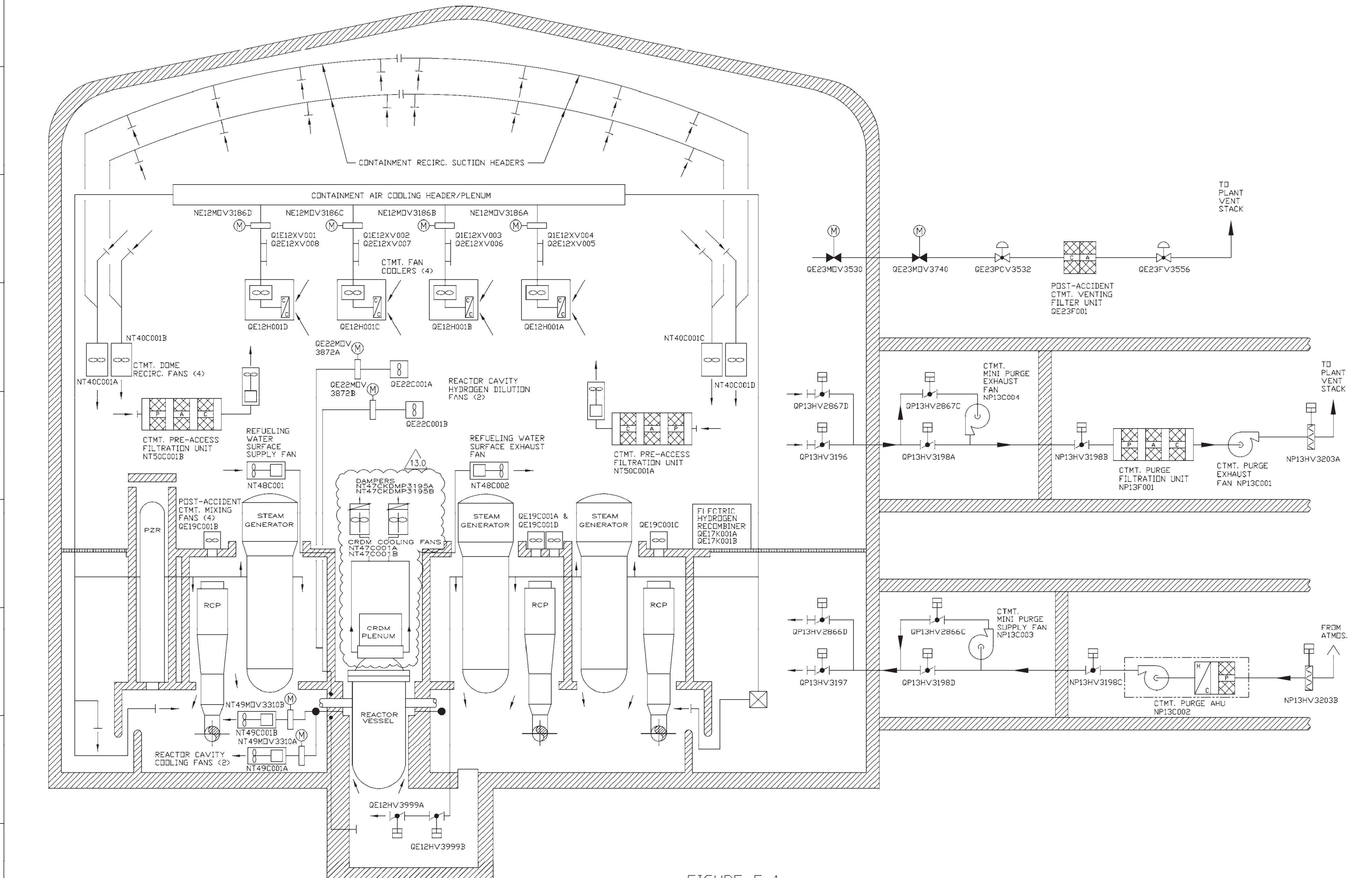


FIGURE F-1
CONTAINMENT VENTILATION SYSTEMS (NORMAL OPERATION)
SIMPLIFIED SYSTEM BOUNDARY DIAGRAM

CAD CVSFSDF1.DWG
AUTOCAD MH -02
VERSION 13.0

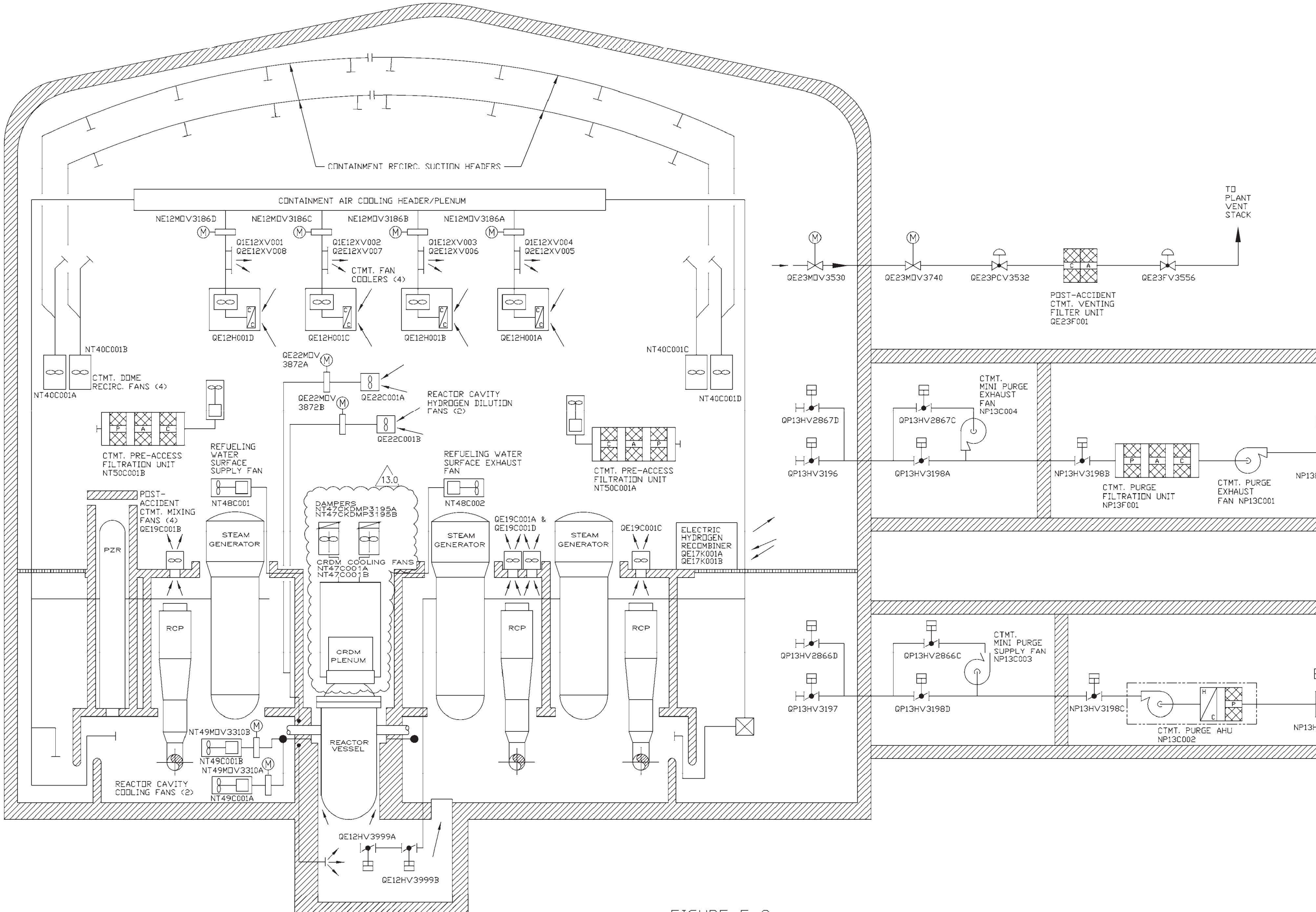


FIGURE F-2
CONTAINMENT VENTILATION SYSTEMS (POST-ACCIDENT)
SIMPLIFIED SYSTEM BOUNDARY DIAGRAM

CAD CVSFSDF2.DWG
AUTOCAD MH -02
VERSION 13.0

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.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

1. 022AK1.04 021/NEW//C/A 2.9/3.0/APE022AK1.04/N///

Unit 1 is operating at 100% power and the following conditions exist:

- AOP-16.0, CVCS Malfunction, has just been exited after a charging flow controller failure.
- FK-122, CHG FLOW, is in MANUAL and has been repaired.

Subsequently, FK-122 is placed in AUTOMATIC and the following conditions exist:

- One 60 gpm orifice is on service.
- Charging flow is stable at 62 gpm.

Which one of the following completes the statement below?

If FK-122 were to go to minimum demand, charging flow would decrease to a **minimum** flow rate of __(1)__ , which is designed to prevent __(2)__.

A.✓ 1) 18 gpm

2) flashing downstream of the letdown orifices

B. 1) 18 gpm

2) flashing downstream of the letdown orifices AND overheating of the charging pumps

C. 1) 40 gpm

2) flashing downstream of the letdown orifices

D. 1) 40 gpm

2) flashing downstream of the letdown orifices AND overheating of the charging pumps

3.3 In auto, CHG FLOW FK 122 minimum demand corresponds to 18 gpm charging flow. This ensures adequate cooling to the regenerative heat exchanger to prevent flashing downstream of the letdown orifices with one 60 gpm orifice on service. With two orifices on service, approximately 40 gpm charging flow is required for regenerative heat exchanger cooling.

Distracter analysis

A. Correct. First part is correct. Minimum charging flow in AUTOMATIC is 18 gpm.

Second part is correct. Per P&L 3.3, 18 gpm ensures adequate cooling to the regenerative heat exchanger to prevent flashing downstream of the letdown orifices with one 60 gpm orifice on service.

B. Incorrect. First part is correct. (See A.1)

Second part is incorrect (See A.2). Plausible if the applicant thinks that reduced charging flow would equate to reduced mini-flow flow which is incorrect. Also, the charging miniflow goes through the seal water return HX and would be unaffected by changing charging flows. Coupled with the correct reason for having a minimum charging flow it makes this a plausible but incorrect answer.

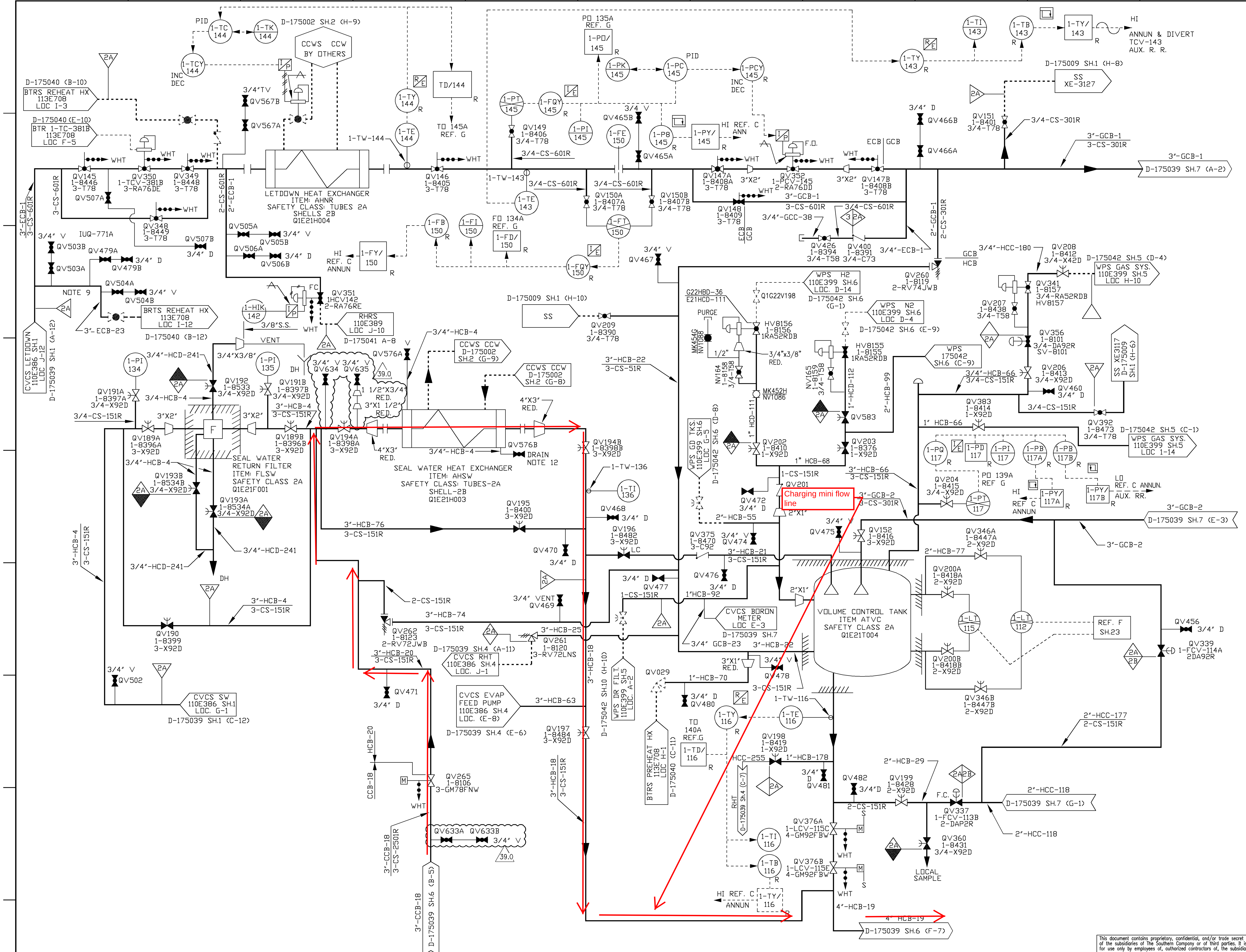
C. Incorrect. First part is incorrect (See A.1). Plausible since Figure 1 of SOP-2.1, re-establishing LTDN after isolation with no equipment malfunction, has the operator establish 40 gpm flow rate in step 1 when placing one orifice on service.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2).

K/A: **022AK1.04**	Loss of Reactor Coolant Makeup - Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Reason for changing from manual to automatic control of charging flow valve controller.
Importance Rating:	2.9 3.0
Technical Reference:	FNP-1-SOP-2.1, Chemical and Volume Control System Plant Startup and Operation, Ver 131
References provided:	None
Learning Objective:	RECALL AND DISCUSS the Precautions and Limitations (P&L), Notes and Cautions (applicable to the "Reactor Operator") found in the following Procedures (OPS-52101F08). • SOP-2.1, CVCS Plant Startup and Operation. [...]
Question History:	NEW
K/A match:	**There has been a loss of CVCS flow** due to a controller failure. FK-122 has been placed in manual and is now being placed in AUTO. **A reason for placing FK-122 in AUTO and not leaving it in MANUAL is to ensure adequate cooling to the regenerative heat exchanger to prevent flashing downstream of the letdown orifices with one 60 gpm orifice on service should the controller fail to minimum demand based on current plant conditions. The operational implication would be that flashing would occur if Chg flow were to fall to <18 gpm and cause damage to the orifices and piping due to water hammer and the flashing of water to steam. A loss of letdown would be the result.** This question meets the KA in that it asks the minimum flow rate for being in auto if a controller were to fail and the reason.
SRO justification:	N/A



NOTES:

1. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
2. DELETED
3. FLOW SET POINT ADJUSTED FROM PRESSURIZER-LEVEL CONTROLLER.
4. PROVIDE MINIMUM OF TWO FEET STRAIGHT RUN AFTER MINIMUM FLOW ORIFICE. MINI-FLOW ORIFICE SUPPLIED BY PUMP MANUFACTURER.
5. ALL Ⓦ NUMBERS IN CVCS ARE SHOWN WITHOUT PREFIX ALACS.
6. CCW REQUIRED FOR PUMP OIL COOLERS ONLY.
7. TEMPORARY STRAINER WAS REPLACED BY SPOOL PIECE AFTER INITIAL FLUSHING OPERATIONS. TPNS NUMBER IS SHOWN FOR REFERENCE ONLY.
8. MULTIPLE LOOP POWER (CABINET MTD.) NOT SHOWN.
9. MULTIPLE LOOP POWER SUPPLY (CONTROL BOARD) SHOWN.
10. DELETED
11. FOR COMPLETE INSTRUMENT AND ASSOCIATED CONTROL VALVE NUMBERS, REFER TO INSTRUMENT INDEX B-175803.
12. DRAIN FROM VALVE TO BE PIPED TO THE NEAREST EQUIPMENT DRAIN.
13. NSE21V001 IS A 4"-125# KNIFE GATE VALVE.
14. PORTIONS OF THE BORON RECYCLE SYSTEM WHICH WERE ORIGINALLY DESIGNED TO ASME CLASS 3 SEISMIC CATEGORY I REQUIREMENTS HAVE BEEN DOWNGRADED TO NNS SEISMIC CATEGORY II BY FSAR AMENDMENT 73. VALVE, EQUIPMENT, AND LINE NUMBERS HAVE NOT BEEN CHANGED. FLAG (△) DESIGNATES NNS CLASSIFICATION.
15. LINES HCC-39, 72, 127, 129, 180, AND 255 ARE DOWNGRADED TO NNS PER NOTE 14.

REFERENCE:

FOR ALPHA REFERENCES SEE DRAWING D-175044

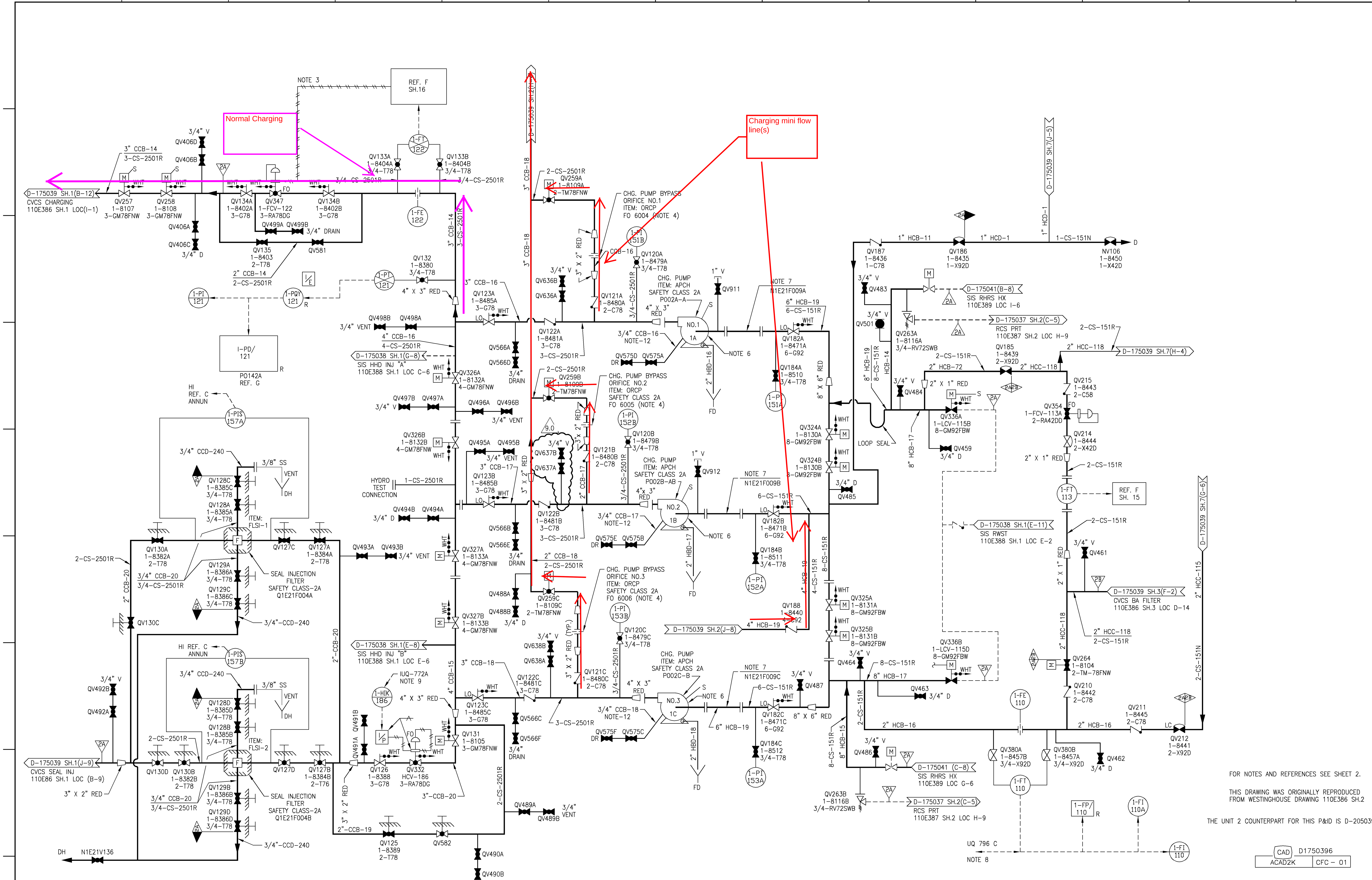
CAD D1750392
ACAD2K KB – 01

Southern Company Services, Inc. for
ALABAMA POWER COMPANY

PROJECT	J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
SUBJECT	P&I DIAGRAM – CHEMICAL AND VOLUME CONTROL SYSTEM
DRAWN	EJS
CHECKED	RLH
APPROVED	M. MALCOM
DATE	11-2-72
SCALE	NONE
SHEET	2 OF – SHEETS
DRAWING	D-175039

VER. NO. 39.0 DATE 05/05/09 KB JWM JMR
REVISED PER ABN 1082199501M002, VER. 1.0

DRAWING CATEGORY: CRITICAL



FOR NOTES AND REFERENCES SEE SHEET 2.

THIS DRAWING WAS ORIGINALLY REPRODUCED FROM WESTINGHOUSE DRAWING 110E386 SH.2

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205039

CAD D1750396
ACAD2K CFC - 01

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

J.M. FARLEY NUCLEAR PLANT - UNIT NO.1

P&ID - CHEMICAL AND VOLUME CONTROL SYSTEM

FOR SIGNATURES SEE MICROFILM COPY

CFC JWM JMR

VER. NO. 10.0 DATE 01/19/12

REVISED PER ABN 10919218011M002, VER. 1.0.

SCALE NONE

SHEET 6

D-175039

DRAWING CATEGORY: CRITICAL

2.14 The RHR system is capable of supplying LTDN flow to the CVCS system per FNP-1-SOP-7.0, Residual Heat Removal System.

2.15 The boron recycle system is capable of receiving CVCS effluent per FNP-1-SOP-2.4, Chemical And Volume Control System Boron Recycle System.

2.16 The auxiliary oil pumps for the available CHG pumps are in AUTO at the HSDP and are running as indicated by the associated white lights being illuminated on the Main Control Board (MCB).

2.17 Charging pump room coolers are in service per FNP-1-SOP-58.0, Auxiliary Building HVAC System.

3.0 Precautions and Limitations

3.1 When a RCP is operating, a minimum back pressure of 15 psig on No. 1 seal must be maintained by maintaining at least 18 psig in the VCT.

3.2 Explosive mixtures of oxygen and hydrogen in the VCT must be avoided at all times. Oxygen content in the VCT must NOT exceed 2% by volume with hydrogen present.

3.3 In auto, CHG FLOW FK 122 minimum demand corresponds to 18 gpm charging flow. This ensures adequate cooling to the regenerative heat exchanger to prevent flashing downstream of the letdown orifices with one 60 gpm orifice on service. With two orifices on service, approximately 40 gpm charging flow is required for regenerative heat exchanger cooling.

3.4 To avoid thermal shock of RCS piping during normal operation the following guidance applies:

3.4.1 When reactor coolant temperature is greater than 350°F, letdown flow should not be stopped without also stopping charging flow.

3.4.2 When operating at minimum charging flow, letdown flow must be monitored to ensure it does NOT rise above 380°F. Opening an additional letdown orifice flow path will increase both charging flow and letdown flow to maintain letdown flow below 380°F.

3.5 Water temperature exiting the LTDN heat exchanger should be less than 115°F. If temperature reaches 135°F, LTDN HI TEMP DIVERT VLV, Q1E21TCV143, diverts flow to the VCT to bypass the demineralizers.

3.6 If letdown flow diverts to the VCT, bypassing the demineralizers for an extended period of time, the boron meter sample point should be realigned.

3.7 Letdown flow should not exceed 135 gpm to prevent exceeding system design flow.

3.8 Maximum flow through the cation bed demineralizer is 60 gpm.

3.9 Demineralizers must be bypassed when using hydrazine for oxygen removal to prevent the demineralizers from removing hydrazine.

FIGURE 1

RE-ESTABLISHING LTDN AFTER ISOLATION WITH NO EQUIPMENT MALFUNCTION

1. Place FK-122 in MANUAL and adjust to greater than or equal to 40 gpm.
2. Place PK-145 in MANUAL and adjust demand to less than or equal to 50%.
3. Verify open the following:
    - Ltdn Line ISO, HV8152
    - Ltdn Line ISO, LCV459
    - Ltdn Line ISO, LCV460
4. Open LTDN ORIF ISO 60 GPM, HV8149B OR HV8149C.
5. Establish desired LTDN pressure and return PK-145 to AUTO. (260-450 psig)
6. Restore FK-122 to AUTO when desired.
7. Refer To SOP-2.1 when time permits.

REMOVING LTDN FROM SERVICE

1. Place PK-145 in MANUAL and adjust demand to less than or equal to 50%.
2. Close LTDN ORIF ISO 45 GPM, Q1E21HV8149A AND LTDN ORIF ISO 60 GPM, HV8149B OR HV8149C, as applicable.
3. Close LTDN LINE ISO, Q1E21LCV459 and Q1E21LCV460
4. Place FK-122 in MANUAL and adjust to 0% (closed).
5. Verify SEAL WTR INJECTION HIK 186 adjusted.
6. Refer To SOP-2.1 when time permits.

Ref: FNP-1-SOP-2.1

Ensure operator aid is updated if this figure is revised.

1. 022K4.03 022/MOD/SUMMER 11/MEM 3.6*/4.0/022K4.03/N///

Unit 1 is operating at 100% power when a Steam Break occurs on 1B SG and the following conditions exist:

- EE5, CTMT ISO PH B, is in alarm.
- All Phase B automatic actions have occurred.

Which one of the following completes the statements below?

CCW to the RCP Thermal Barrier Heat Exchanger _(1)_ isolated.

Seal Injection _(2)_ isolated.

	(1)	(2)
A.✓	IS	is NOT
B.	is NOT	is NOT
C.	IS	IS
D.	is NOT	IS

EE5
Automatic Action:
1. The following valves close
c) CCW FROM RCP THRM BARR Q1P17HV3045
f) CCW FROM RCP THRM BARR Q1P17HV3184

FSD A181003
3.3.1.2 The seal water injection lines to the RCP are considered as open flow paths post-LOCA. The high pressure inflow through these lines during the injection and recirculation phases precludes any containment to atmosphere leakage. In the event of a loss of seal water flow through these lines, a water seal in the charging pump suction and discharge piping precludes containment to atmosphere leakage.

<u>Distracter analysis</u>

A. Correct. First part is correct. Phase 'B" isolates CCW cooling to the RCP Thermal Barrier Heat Exchanger.

Second part is correct. Seal injection is NOT isolated by SI, Phase A or Phase B.

B. Incorrect. First part is incorrect (See A.1). Plausible if the applicant does not recall all the components isolated on a Phase B. Since CCW is water solid and cools the thermal barrier hx, they may believe it is not isolated on a phase B.

Second part is correct (See A.2)..

C. Incorrect. First part is correct (See A.1).

Second part is incorrect (See B.2). Plausible if the applicant believes that Phase B isolates RCP seal injection lines. Seal return is isolated on an SI and the applicant could confuse the two.

This is a plausible combination if the applicant reasons that the shutdown seal will actuate and seal injection and CCW to the thermal barrier hx are no longer needed

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See C.2).

K/A: 022K4.03	Containment Cooling System (CCS) - Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Automatic containment isolation.
Importance Rating:	3.6* 4.0
Technical Reference:	FNP-1-ARP-1.5, EE5, CTMT ISO PH B, Ver 58.0 FSD-A181009, CVCS/HHSI/ACCUM/RMWS, Ver 38 FSD-181003, Containment Isolation System, Ver 26.
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Containment Structure and Isolation System components and equipment, to include the following (OPS-40302B07): • […] • […] • Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
Question History:	MOD SUMMER 11
K/A match:	Requires the applicant to know the **design feature of the Phase 'B' Containment Isolation that isolates CCW cooling to the RCP Thermal Barrier Heat Echanger.**
SRO justification:	N/A

QE21V079A,B,C QP19V004
QE21V115A,B,C

3.3.1 Basic Functions

3.3.1.1 The CIS check valves function as automatic isolation valves inside containment to provide isolation for piping systems penetrating containment (References 6.7.010, 6.7.011, 6.7.012).

3.3.1.2 The seal water injection lines to the RCP are considered as open flow paths post-LOCA. The high pressure inflow through these lines during the injection and recirculation phases precludes any containment to atmosphere leakage. In the event of a loss of seal water flow through these lines, a water seal in the charging pump suction and discharge piping precludes containment to atmosphere leakage. In the event that maintenance requires interrupting flow through these lines, isolation can be achieved by closure of the manually operated 1" needle valves (QE21V116A,B,C) located just outside containment (References 6.1.003, 6.4.018).

3.3.2 Functional Requirements

3.3.2.1 A simple check valve may not be used as an automatic isolation valve outside containment, but may be used inside containment (References 6.7.010, 6.7.011).

3.3.2.2 Containment isolation valves QE11V042A,B; QE13V002A,B; QE21V062A,B,C; QE21V066A,B,C; QE21V076A,B; QE21V078A,B,C; QE21V079A,B,C; QE21V115A,B,C; and QP16V206A,B,C,D are located in systems that represent open in-flow paths during post-accident conditions. Therefore, the respective penetrations shall be subject to Type A testing to meet the requirements of 10 CFR 50, Appendix J. The remaining containment isolation valves are located in systems which do not represent open in-flow paths or closed systems during post-accident conditions; therefore, the valves shall be subject to Type C testing to meet the requirements of 10 CFR 50, Appendix J (References 6.7.005, 6.7.016).

5.30.4 Seismic Requirements

5.30.4.1 The blender is designated Seismic Category I and shall be designed to withstand seismic loads without loss of function. Refer to the referenced specifications and FSAR Sections for additional details. (References 6.5.27, 6.1.2 and 6.1.15)

5.30.5 I&C Requirements

5.30.5.1 Flow measurement (FT-168) shall be provided downstream of the blender to indicate total makeup flow to the charging header and as input to the Reactor Makeup Control System. Refer to section 5.78 for additional discussion. (References 6.2.1 and 6.4.1)

5.30.5.2 Upstream boric acid flow measurement (FT-113) shall be provided to serve as input to the Reactor Makeup Control System. Refer to section 5.77 for additional discussion. (References 6.2.1 and 6.4.1)

5.30.6 Interface Requirements

5.30.6.1 The RMWS shall be capable of supplying 120 gpm flow to the boric acid blender at 95 psig. (References 6.2.46 and 6.1.41)

5.31 RMWST

QP12T503

5.31.1 Basic Functions

5.31.1.1 The RMWST provides a source of makeup water to the CCW surge tank in the event of a low level in the tank resulting from a leak and in the event that the demineralized water makeup system is unavailable. (References 6.4.1.f, 6.4.6 and 6.5.51.b)

5.31.1.2 The RMWST provides a source of recycled demineralized water, which is used as makeup, for the following components:

- boric acid blenders,
- chemical mixing tanks,
- PRTs,
- RCP standpipes (Reference 6.4.1.f)

The equipment supplied was designed for 250°F. The difference between this and the value in Table T-12 represents margin above design. (References 6.2.9 and 6.5.29.b)

5.77.3 Interface Requirements

5.77.3.1 The channel must be powered from a 120 Volt-AC regulated instrument power system. This regulated AC power shall be 118 volts + 2%, 60 cps + 2% nominal, 3% maximum harmonic distortion (normal), 5% maximum. (Reference 6.5.21)

5.78 BORIC ACID FLOW TO BLENDER

FRCA-168

5.78.1 Basic Functions

5.78.1.1 This differential pressure transmitter (and associated orifice flow element) shall provide measurement of the total makeup flow from the boric acid blender. It shall also provide input to the Reactor Makeup Control System for regulation of RMW flow and shall alert the operator of a deviation from the selected flow setpoint. (References 6.2.1, 6.2.9 and 6.4.1)

5.78.2 Functional Requirements

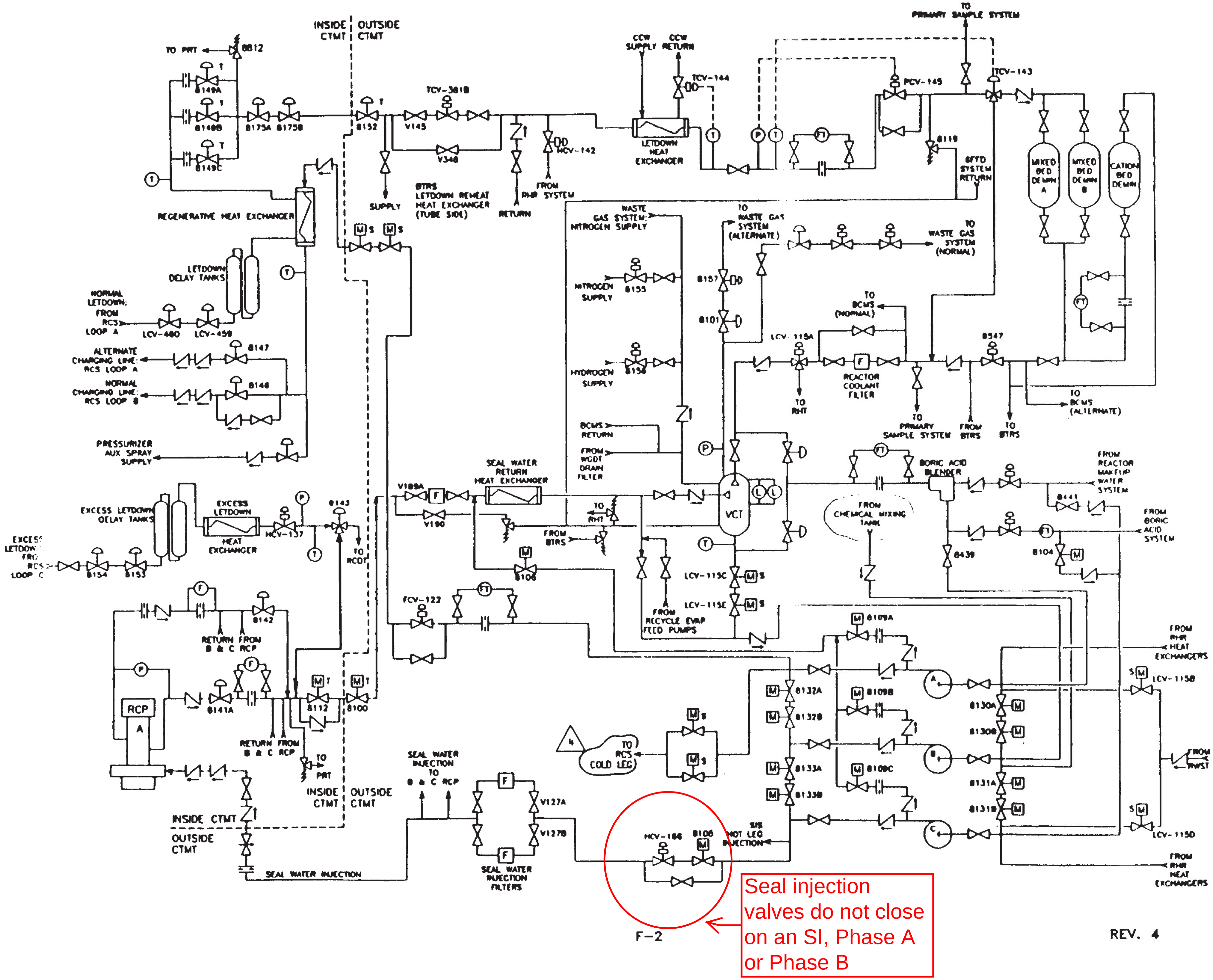
5.78.2.1 The flow rate shall be recorded in the main control room. The total volume of boric acid added shall be indicated on a batch counter. (Reference 6.2.9)

5.78.2.2 The instrument signal shall provide input to the Reactor Makeup Control System for use in controlling the operation of valve FCV-114A (QV399). Visual and audible alarm shall be provided when the flow rate deviates a present amount from the total makeup water flow setpoint. The alarm deviation setpoint (Refer to References 6.2.7 and 6.2.8 for the instrument alarm setpoint) is based on previous operating experience. (References 6.2.1, 6.2.9, 6.4.1, 6.2.7 and 6.2.8)

5.78.2.2 Refer to Table T-13 for instrumentation design requirements including instrumentation type, design pressure, temperature, range, orifice I.D and readout location. (References 6.2.9, 6.5.29.b and 6.5.29.m)

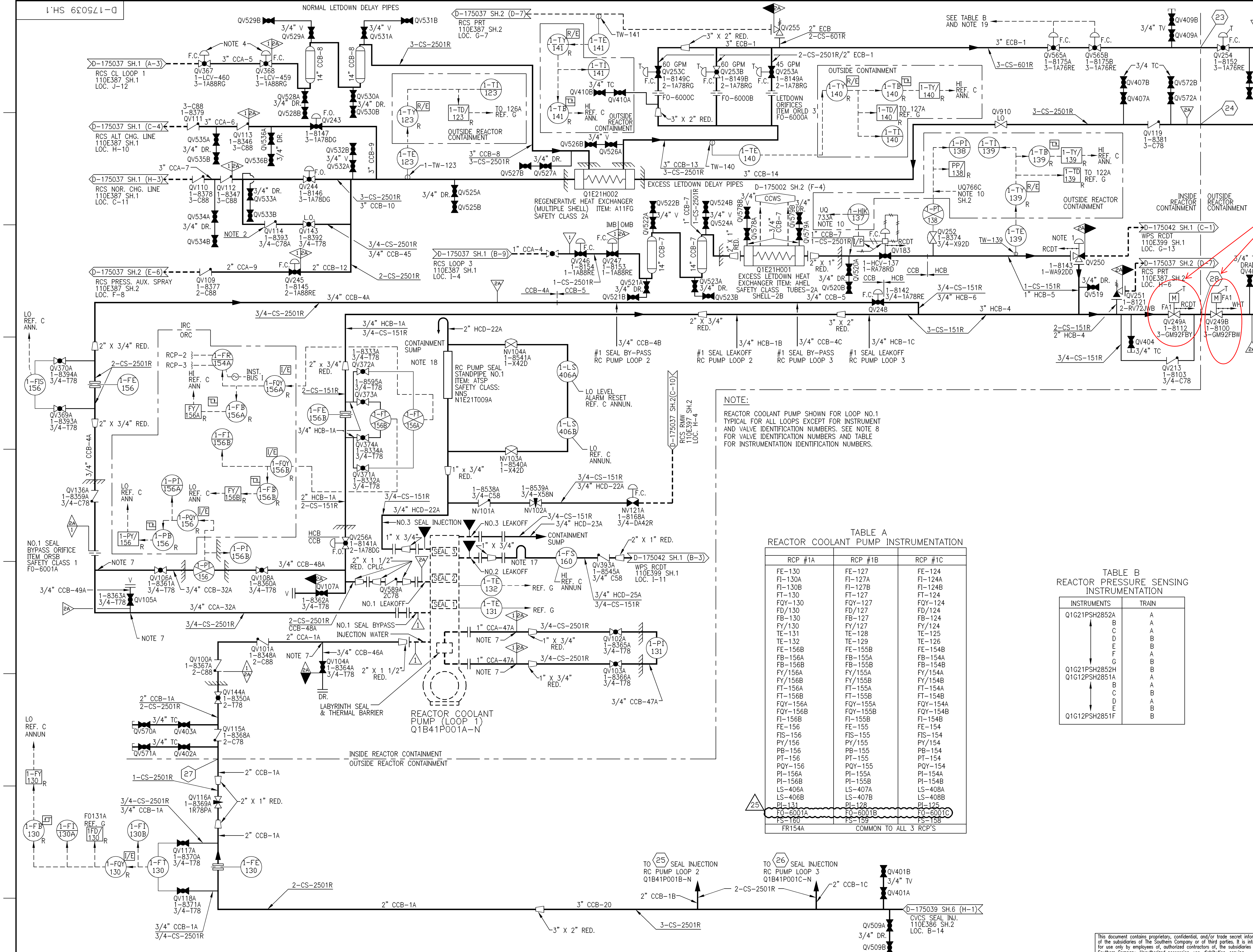
FIGURE 2

CVCS



F-2

REV. 4



NOTE:

REACTOR COOLANT PUMP SHOWN FOR LOOP NO.1 TYPICAL FOR ALL LOOPS EXCEPT FOR INSTRUMENT AND VALVE IDENTIFICATION NUMBERS. SEE NOTE 8 FOR VALVE IDENTIFICATION NUMBERS AND TABLE FOR INSTRUMENTATION IDENTIFICATION NUMBERS.

TABLE A
REACTOR COOLANT PUMP INSTRUMENTATION

RCP #1A	RCP #1B	RCP #1C
FE-130	FE-127	FE-124
FI-130A	FI-127A	FI-124A
FI-130B	FI-127B	FI-124B
FT-130	FT-127	FT-124
FQY-130	FQY-127	FQY-124
FD/130	FD/127	FD/124
FB-130	FB-127	FB-124
FY/130	FY/127	FY/124
TE-131	TE-128	TE-125
TE-132	TE-129	TE-126
FE-156B	FE-155B	FE-154B
FB-156A	FB-155A	FB-154A
FB-156B	FB-155B	FB-154B
FY/156A	FY/155A	FY/154A
FY/156B	FY/155B	FY/154B
FT-156A	FT-155A	FT-154A
FT-156B	FT-155B	FT-154B
FQY-156A	FQY-155A	FQY-154A
FQY-156B	FQY-155B	FQY-154B
FI-156B	FI-155B	FI-154B
FE-156	FE-155	FE-154
FIS-156	FIS-155	FIS-154
PY/156	PY/155	PY/154
PB-156	PB-155	PB-154
PT-156	PT-155	PT-154
PQY-156	PQY-155	PQY-154
PI-156A	PI-155A	PI-154A
PI-156B	PI-155B	PI-154B
LS-406A	LS-407A	LS-408A
LS-406B	LS-407B	LS-408B
PI-131	PI-128	PI-125
FO-6001A	FO-6001B	FO-6001C
FS-160	FS-159	FS-158
FR154A	COMMON TO ALL 3 RCP'S	

TABLE B
REACTOR PRESSURE SENSING INSTRUMENTATION

INSTRUMENTS	TRAIN
Q1G21PSH2852A	A
B	A
C	A
D	B
E	B
F	A
G	B
Q1G21PSH2852H	B
Q1G12PSH2851A	A
B	A
C	B
D	A
E	B
Q1G12PSH2851F	B

NOTES:

1. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
2. SPECIAL SPRING LOADED CHECK VALVE.
3. (DELETED)
4. PRESSURIZER LOW LEVEL SIGNAL CLOSES THESE VALVES.
5. ALL Ⓦ ITEM NO'S. IN CVCS ARE SHOWN WITHOUT PREFIX ALACS.
6. FOR LEGEND SEE DWG. D-175044.
7. A 3/8" FLOW RESTRICTOR IS REQUIRED AS NOTED ON LEGEND.
8. ALL VALVES & LINE IDENTIFICATIONS IN PUMP CIRCUITS FOR PUMPS 1B & 1C ARE NUMBERED THE SAME AS PUMP 1A (SHOWN). SUFFIX A ON VALVE OR LINE NUMBER SIGNIFIES PUMP 1A CIRCUITRY. SUFFIX B FOR PUMP 1B AND SUFFIX C FOR PUMP 1C.
9. ALL VALVE NUMBERS IN THIS SYSTEM ARE PREFIXED BY (Q OR N, 1E21).
10. MULTIPLE LOOP PWR. SUPPLY (CONTROL BOARD) NOT SHOWN.
11. FOR VALVE INFORMATION SEE MASTER VALVE LIST (SPEC. SS-1102-39).
12. FOR EQUIPMENT LIST SEE DWG. D-175070.
13. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX B-175803.
14. (DELETED)
15. FOR PIPING CLASS SUMMARY SHEETS SEE (SPEC. SS-1109-2).
16. (DELETED)
17. RUN PIPING HORIZONTAL OR UPWARD TO LOOP RISING 6" ABOVE RCP NO. 2 SEAL LEAKOFF CONNECTION. AFTER LOOP, RUN PIPING HORIZONTAL OR DOWNWARD THROUGH RUN TO RCDT.
18. LOCATE BOTTOM OF STANDPIPE 28'-4 1/8" ± 0.5 FEET ABOVE CONN. TO RCP #3 SEAL.
19. SENSOR LOCATIONS ARE SHOWN ON D-175143, D-175146, & D-175147.

REFERENCES

1. FOR ALPHA REFERENCES SEE DWG. D-175044.

THIS DRAWING WAS ORIGINALLY REPRODUCED FROM WESTINGHOUSE DWG. 110E386 SH.1 (U-165760)

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205039

CAD D1750391.DWG
ACAD2000 | TJS - 02

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH
TJS	AJC	ASK				

REV. NO. 25 DATE 9-17-03
REVISED PER ABN 03-1-2673, REV. 0

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH
JAB	LFC	DEW				

REV. NO. 24 DATE 7-17-02
REDRAWN PER ABN 02-1-2384, REV. 0

DRAWN ____ CHECKED ____ DESIGNED ____
APPROVED FOR SIGNATURES DATE ____
APPROVED SEE MICROFILM DATE ____
APPROVED COPY DATE ____

Southern Company Services, INC. for
ALABAMA POWER COMPANY
PROJECT J. M. FARLEY NUCLEAR PLANT - UNIT 1
SUBJECT P&ID - CHEM. & VOL. CONTROL SYSTEM
SCALE NONE B/M N/A
SHEET 1 OF — SHEETS
SUPERSEDES
D-175039

OPERATIONS CRITICAL

E5
CTMT ISO PH B

SETPOINT: Pressure 27 PSIG

ORIGIN:
1. Manual
2. Pressure Transmitter (Q1E13PT950)
3. Pressure Transmitter (Q1E13PT951)
4. Pressure Transmitter (Q1E13PT952)
5. Pressure Transmitter (Q1E13PT953)

PROBABLE CAUSE

1. 2/4 Phase B CTMT ISO CS ACTUATION Switches turned to the Actuate position at the same time.
2. 2/4 pressure transmitters above 27 PSIG, due to Reactor Coolant, Steam, or Feed Water rupture in Containment.

AUTOMATIC ACTION

1. The following valves close:
   a) IA TO CTMT Q1P19HV3611
   b) CCW TO RCP CLRS Q1P17HV3052
   c) CCW FROM RCP THRM BARR Q1P17HV3045
   d) CCW FROM RCP OIL CLRS Q1P17HV3182
   e) CCW FROM RCP OIL CLRS Q1P17HV3046
   f) CCW FROM RCP THRM BARR Q1P17HV3184
2. The following valves and dampers open:
   a) 1A and 1B PRF SUCTION DAMPER Q1E15MOV3362A & B
   b) 1A and 1B CS PUMP TO SPRAY HDR ISO Q1E13MOV8820A & B
3. The following equipment starts:
   a) 1A and 1B CS PUMPs
   b) 1A and 1B PRF EXH FANs
   c) 1A and 1B PRF RECIRC FANs

OPERATOR ACTIONS

1. **Trip** the reactor.
2. **Trip** the reactor coolant pumps.
3. **Perform** the action required by FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

References: A-177100, Sh. 255; U-170155; U-211024; U-260610; PLS Document

51. Given the following plant conditions:

- 100% power.
- A main steam line break occurs.
- XCP-612 pt 4-2, PHASE B ISOL, alarms.
- All Phase B automatic actions have occurred.

Based on the current conditions, which ONE (1) of the following states how RCPs should be operated in accordance with ARP-001-XCP-612 pt 4-2, PHASE B ISOL **and** what is the status of RCP seal injection?

A. Ensure the Reactor is tripped, then stop all RCPs.

Seal injection still exists.

B. Ensure the Reactor is tripped, then stop all RCPs.

All seal injection has been lost.

C. RCP operation may continue as long as parameters remain within limits as displayed on IPCS screen ZZRCPBRG.

Seal injection still exists.

D. RCP operation may continue as long as parameters remain within limits as displayed on IPCS screen ZZRCPBRG.

All seal injection has been lost.

1. 025AA1.01 023/MOD/FNP EXAM BANK/C/A 3.6/3.7/APE025AA1.01/N///

Unit 1 is in Mode 5 with the following conditions:

- 1B RHR pump is tagged out.
- All SG Wide Range levels are 84%.
- Pzr level is being maintained at 21% on LI-462, PRZR LVL.
- RCS temperature is 155°F.
- RCS pressure is 325 psig.
- All RCP's are secured.
- 1A RHR pump is running in the cooldown lineup.

Subsequently, the following occurs:

- 1A RHR pump trips on overcurrent and cannot be restarted.
- RCS temperature is 175°F and slowly rising.

Which one of the following completes the statements below?

Per AOP-12.0, Residual Heat Removal System Malfunction, the preferred method to re-establish core cooling is to establish __(1)__.

Core cooling is monitored using __(2)__.

	(1)	(2)
A.	feed and bleed	RCS cold leg temperatures
B.	a secondary heat sink	RCS cold leg temperatures
C.	feed and bleed	CETCs
D.✔	a secondary heat sink	CETCs

ARG-1

If the RCS is intact and the loops are not isolated with SG nozzle dams or loop isolation valves, a secondary heat sink using half or more SGs will be an effective alternate mode of decay heat removal that will last for several hours or longer. Since there would be no significant fluid inventory losses for this case, makeup requirements can easily be met with a minimum amount of charging flow or possibly RWST (or VCT) gravity feed if initiated early enough. For this situation, it should also be possible to refill and pressurize the RCS and then operate the RCPs to sweep the noncondensibles from the loops and thereby improve the primary-to-secondary heat transfer.

AOP-12:

24. Check SGs available.

• Check SG primary nozzle dams
- REMOVED.

• Check SG primary manways -
INSTALLED.

• Check SG secondary handhole
covers - INSTALLED.

NOTE: Establishing a secondary heat sink will reduce RCS heat up and pressurization rate to provide more time for recovery actions.

25. Verify secondary heat sink established.

25.1 Maintain wide range level in all available SGs greater than 75% using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.

25.2 IF SG steam space intact, THEN open atmospheric relief valves to prevent SG pressurization.

1A(1B,1C) MS ATMOS
REL VLV
PC 3371A adjusted
PC 3371B adjusted
PC 3371C adjusted

25.3 IF SGBD system available, AND AFW system available, THEN establish blowdown from available SGs using FNP-1-SOP-16.3, STEAM GENERATOR FILLING AND DRAINING.

Feed and Bleed or Feed and Spill would be established if both of these conditions were met.

29.1 Check RCS level LESS than 121 ft 11 in AND core exit T/Cs GREATER than 200°F.

<u>**Distracter analysis**</u>

A. Incorrect. First part is incorrect (See D.1). Plausible if the applicant believes that establishing a secondary heat sink is not correct because RCS temperature is less than 200°F and so steaming the SG would not

be an option. Feed and Bleed is only used when RCS level is < 121 ft 11 in and RCS temp is > 200°F.

Second part is incorrect (See D.2). Plausible since Tcold is used in other procedures (ESP-0.2) to evaluate cooldown and the applicant could believe that it is used here. Also, Tcold would not give an accurate indication of core temperature.

B. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

C. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

D. Correct. First part is correct. Since the RCS is filled and intact, establishing a secondary heat sink is the correct action per AOP-12.

Second part is correct. AOP-12 directs the use of CETCs

K/A: 025AA1.01 Loss of Residual Heat Removal System (RHRS) - Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS/RHRS cooldown rate

Importance Rating: 3.6 3.7

Technical Reference: Background Information for WOG Abnormal Response Guideline ARG-1 Loss of RHR While Operating at Mid-Loop Conditions, Ver 2
FNP-1-AOP-12.0, RHR System Malfunction, Ver 25

References provided: None

Learning Objective: LIST AND DESCRIBE the sequence of major actions associated with AOP-12.0, RHR System Malfunction and/or STP-18.4, Containment Closure. (OPS-52520L04)

Question History: MOD FNP EXAM BANK

K/A match: Requires the applicant to **know how the RCS is operated to establish a cooldown rate on a loss of RHR and the method which temperature is monitored.**

SRO justification: N/A

3.2 Key Utility Decision Points

The key utility decision point in this guideline is when RHR cooling cannot be restored and alternate means of core cooling should be established. Available options are discussed below:

If the RCS is intact and the loops are not isolated with SG nozzle dams or loop isolation valves, a secondary heat sink using half or more SGs will be an effective alternate mode of decay heat removal that will last for several hours or longer. Since there would be no significant fluid inventory losses for this case, makeup requirements can easily be met with a minimum amount of charging flow or possibly RWST (or VCT) gravity feed if initiated early enough. For this situation, it should also be possible to refill and pressurize the RCS and then operate the RCPs to sweep the noncondensibles from the loops and thereby improve the primary-to-secondary heat transfer. This latter mode of recovery, although not studied in the original spectrum of analyses presented in WCAP-11916, is a reasonable extension to the SG condensation studies of that report. Note that this method requires that an accurate level indication be available after the RCS starts to boil.

If the RCS remains intact but the SGs cannot be made available for cooling, a bleed and feed mode of recovery similar to that used in FR-H.1 can be used. Studies indicate that a reduced set of equipment (one pressurizer PORV and one high-pressure SI pump) will be adequate for removing decay heat and maintaining RCS inventory. For a high decay heat case, RCS level would stabilize well above mid-loop and RCS pressure would stabilize between 100 and 400°psig. Thus, RCS conditions would permit RHR cooling to be reestablished once other support conditions are achieved. Some other bleed and feed scenarios for lower decay heat conditions are evaluated in Section 2.5.5 of ARG-1, Revision 0 (see Appendix C of WCAP-14089, Revision 1, Supplement 1). Note that this method requires that an accurate level indication be available after the RCS starts to boil.

If the RCS is not intact and core exit temperatures reach 200°F, the operator establishes a high makeup rate (to the hot leg or cold leg or both) in order to suppress boiling, raise RCS level, and allow RHR cooling to be reestablished. Once the RCS level goes above the top of the hot legs, heat removal via SG reflux cooling will be affected. The step description tables discuss some guideline variations which would provide flexibility to the utility. If a utility chooses to control makeup to match boiloff, an accurate level indication needs to be available to the operators.

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-12.0

RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION

S A F E T Y R E L A T E D

PROCEDURE USAGE REQUIREMENTS per NMP-AP-003	SECTIONS
Continuous Use	ALL
Reference Use	
Information Use	

Approved:

David L Reed (for)

Operations Manager

Date Issued: 01/28/13

TABLE OF CONTENTS

A. Purpose

This procedure provides actions for response to a malfunction of the RHR system.

Actions in this procedure for restoring RHR PUMPs assume electrical power is available. During loss of electrical power conditions, FNP-1-AOP-5.0, LOSS OF A OR B TRAIN ELECTRICAL POWER, provides actions for restoration of electrical power which should be performed in addition to continuing with this procedure.

The first part of this procedure deals with the protection of any running RHR pump and isolation of any leakage. If a running train is maintained the procedure is exited. Credit may be taken for RCS Loops providing core cooling in place of a running train of RHR. The next portion deals with restoring a train of RHR while monitoring core temperatures. If a train cannot be restored actions are taken for protection of personnel, establishing containment closure, and provides alternate methods of decay heat removal while trying to restore a train of RHR. Alternate cooling methods include: establishing a secondary heat sink if steam generators are available; feed and bleed cooling and feed and spill cooling.

The intent of feed and bleed cooling is to regain pressurizer level and allow steaming through a bleed path to provide core cooling. This requires that the RCS be in a configuration that will allow a level in the pressurizer.

The intent of feed and spill cooling is to allow spillage from the RCS and locally throttle injection flow to provide core cooling. This method is used when the reactor vessel head is blocked or RCS loop openings exist.

This procedure is applicable in modes 4, 5 and 6.

Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS AND the RCS temperature is below 180°F.

B. Symptoms or Entry Conditions

1 This procedure is entered when a malfunction of the RHR system is indicated by any of the following:

1.1 Trip of any operating RHR pump

1.2 Excessive RHR system leakage

1.3 Evidence of running RHR pump cavitation

1.4 Closure of loop suction valve

1.5 High RCS or core exit T/C temperature

1.6 Procedure could be entered from various annunciator response procedures.

CF3 1A OR 1B RHR PUMP OVERLOAD TRIP

CF4 1A RHR HX OUTLET FLOW LO

CF5 1B RHR HX OUTLET FLOW LO

CG3 1A OR 1B RHR HX CCW DISCH FLOW HI

EA5 1A OR 1B RHR PUMP CAVITATION

EB5 MID-LOOP CORE EXIT TEMP HI

EC5 RCS LVL HI-LO

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS AND the RCS temperature is below 180°F.

****

****

CAUTION: Filling the pressurizer to 100% will cause a loss of nozzle dams due to the head of water.

****

NOTE: RCS to RHR loop suction valves will be deenergized if RCS TAVG is less than 180°F.

___ 1 **Check RHR loop suction valves - OPEN.**

RHR PUMP	1A	1B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSED(IF REQUIRED)	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3

1 Stop any RHR PUMP with closed loop suction valve(s).

1.1 IF required, THEN adjust charging flow to maintain RCS level.

___ 2 **IF the standby RHR train is NOT affected AND plant conditions permit operation, THEN place the standby RHR train in service per FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM.**

2 IF core cooling provided by the SGs, THEN proceed to step 8.

Step	Action/Expected Response	Response NOT Obtained
	NOTE: Rapid flow adjustments may cause more severe pump cavitation.	
___ 3	**Check RHR PUMPs - NOT CAVITATING.** The following parameters should be stable and within normal ranges. [] RHR flow rate within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing. [] Discharge pressure [] Suction pressure [] RHR motor ammeter readings [] No unusual pump noise	3 Perform the following: 3.1 Slowly reduce RHR flow rate to eliminate cavitation. 3.2 <u>IF</u> cavitation CANNOT be eliminated, <u>THEN</u> stop the affected RHR pump(s).
___ 4	**Check any RHR PUMP - RUNNING**	4 Proceed to step 13.
___ 5	**Verify RHR flow > 3000 gpm.** 1A(1B) RHR HDR FLOW [] FI 605A [] FI 605B	5 Refer to Technical Specifications 3.9.4 and 3.9.5 for applicability.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Indicated RCS level will rise approximately 1 ft for every 0.5 psi rise in RCS pressure if the indication is not pressure compensated.

****

****

CAUTION: Only borated water should be added to the RCS to maintain adequate shutdown margin.

****

___ 6 **Check RCS level ADEQUATE**

6.1 Compare any available level indications.

- [] LT 2965A&B/level hose
- [] LI-2384 1B LOOP RCS NR LVL
- [] LI-2385 1C LOOP RCS NR LVL
- [] Temporary remote level indicator off of a RCS FT on A or C loop

6.2 Check RCS level within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing.

Response NOT Obtained:

6.2 Raise RCS level.

6.2.1 Notify personnel in containment that RCS level will be raised.

6.2.2 Align Technical Requirements Manual boration flow path.

6.2.3 Raise RCS level to within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing for the existing RHR flow.

Step	Action/Expected Response	Response NOT Obtained
___ 7	**Maintain RCS level within the following limits:** [] Maintain RCS level to within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing for the existing RHR flow. [] Maintain RCS level less than 123 ft 4 in if personnel are in the channel heads without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if primary manways are removed without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if seal injection is not established and RCPs are not backseated. [] Maintain RCS level less than 124 ft if safety injection check valves are disassembled.	7 Verify RHR PUMP(s) stopped AND proceed to step 13.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: IF the leaking RHR train can NOT be identified, THEN both trains should be assumed leaking.

****

___ 8 **Check RHR system - INTACT**

- [] Stable RCS level.
- [] No unexpected rise in containment sump level.
- [] No RHR HX room sump level rising.
- [] No RHR pump room sump level rising.
- [] No waste gas processing room sump level rising
- [] No rising area radiation monitor
- [] No unexplained rise in PRT level or temperature.

8 Isolate RHR leakage.

8.1 Isolate affected RHR train(s) from RCS.

8.1.1 Stop affected RHR pump(s).

8.1.2 Verify closed affected RHR train valves.

Affected RHR Train	A	B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSED	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3
1A(1B) RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B
1A(1B) RHR TO RCS HOT LEGS XCON Q1E11MOV	[] 8887A	[] 8887B

8.2 Isolate source of any RHR/RCS leakage.

___ 9 **Check core cooling provided by RHR or SGs.**

9 Proceed to step 13.

___10 **Check RCS temperature stable or lowering.**

10 Proceed to step 13.

Step	Action/Expected Response	Response NOT Obtained
___11	**Verify low pressure letdown aligned to operating RHR train:**	
11.1	Determine RHR train that low pressure letdown is aligned.	
11.2	IF required, THEN align low pressure letdown to the operating RHR train using FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM	
___12	**Go to procedure and step in effect.**	

****

CAUTION: Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS and the RCS temperature is below 180 F.

****

Step	Action/Expected Response	Response NOT Obtained
___13	**Begin establishing containment closure using FNP-1-STP-18.4, CONTAINMENT MID-LOOP AND/OR REFUELING INTEGRITY VERIFICATION AND CONTAINMENT CLOSURE.**	13 IF in mode 6, THEN refer to Technical Specifications 3.9.4 and 3.9.5 for other containment isolation requirements.

Steps 13 through 19 include monitoring time to saturation, attempt to establish an RCS level, isolate containment per STP-18.4 and attempt to start an RHR pump.

Step	Action/Expected Response	Response NOT Obtained
___14	**Monitor time to core saturation.**	
14.1	Check time to core saturation from the current Shutdown Safety Assessment.	14.1 Determine time to core saturation: • Use ATTACHMENT 3, Time to Core Saturation OR • Monitor any available core exit thermocouples for a heat up trend.
14.2	Monitor RCS temperature trend during the performance of this procedure.	
14.2.1	Check vacuum degas system NOT in service.	14.2.1 IF vacuum refill in progress maintaining a vacuum on the RCS, THEN break vacuum on the RCS using FNP-0-SOP-74.0, OPERATION OF THE RCVRS SKID. (155' CTMT)

NOTE: Step 14.2.2 is a continuing action step.

Step	Action/Expected Response	Response NOT Obtained
14.2.2	IF RCS level decreases to less than 121 ft 11 in AND core exit T/Cs are greater than 200°F, THEN proceed to step 21.	
14.3	IF applicable, THEN review the current shutdown safety assessment of FNP-0-UOP-4.0 for applicability of other outage Abnormal Operating Procedures.	
___15	**Begin venting any RHR trains which have experienced evidence of cavitation using ATTACHMENT 1, RHR PUMP VENTING.**	

Step	Action/Expected Response	Response NOT Obtained
___16	**Suspend any boron dilution in progress. (IN 91-54)**	
___17	**IF the charging system is still in service, THEN align the RWST to the running Charging pump.**	

Operable CHG PUMP	1A	1B(A TRN)	1B(B TRN)	1C
RWST TO CHG PUMP Q1E21LCV	[] 115B	[] 115B	[] 115D	[] 115D

****

CAUTION: The RCS tygon level hose and LT 2965A&B utilize the same level tap. These are not independent indications.

****

Step	Action/Expected Response	Response NOT Obtained
18	**Check for two independent RCS level indications.**	
___ 18.1	Compare available level indications. [] LT 2965A&B/level hose [] LI-2384 1B LOOP RCS NR LVL [] LI-2385 1C LOOP RCS NR LVL [] Temporary remote level indicator off of a RCS FT on A or C loop	
18.2	Check RCS level greater than 123 ft 3 in.	18.2 Raise RCS level. 18.2.1 Notify personnel in containment that RCS level will be raised. 18.2.2 Align Technical Requirements Manual boration flow path. 18.2.3 Raise RCS level to greater than 123 ft 3 in.

Step 18 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
18.3	Maintain RCS level within the following limits: [] Maintain RCS level less than 123 ft 4 in if personnel are in the channel heads without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if primary manways are removed without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if seal injection is not established and RCPs are not backseated. [] Maintain RCS level less than 124 ft if safety injection check valves are disassembled.	

****

CAUTION: The standby RHR train may be lost due to cavitation if it is placed in service without adequate RCS level.

****

****

CAUTION: Starting an RHR PUMP may cause RCS level to fall due to shrink or void collapse.

****

NOTE: The term "standby RHR train" refers to the train most readily available to restore RHR cooling.

Step	Action/Expected Response	Response NOT Obtained
19	**WHEN RCS level greater than 123 ft 3 in, THEN place standby RHR train in service.**	19 IF unable to establish at least one train of RHR, THEN proceed to step 21 while continuing efforts to restore at least one train of RHR.
19.1	Verify CCW PUMP in standby train - STARTED.	

Step 19 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

19.2 Verify CCW - ALIGNED TO STANDBY RHR HEAT EXCHANGER.

Standby RHR Train	A	B
CCW TO 1A(1B) RHR HX Q1P17MOV	[] 3185A	[] 3185B

19.3 Verify the following conditions satisfied.

19.3.1 RWST TO 1A(1B) RHR PUMP Q1E11MOV8809A and B closed.

19.3.2 1A(1B) RHR HX TO CHG PUMP SUCT Q1E11MOV8706A and B closed.

19.3.3 RCS pressure less than 402.5 psig.

19.3.4 PRZR vapor space temperature less than 475°F.

Step 19 continued on next page.

Step | **Action/Expected Response** | **Response NOT Obtained**

NOTE: RCS to RHR loop suction valves will be deenergized if RCS TAVG is less than 180°F.

19.4 Verify standby RHR train loop suction valves - OPEN.

Standby RHR Train	A	B
1C(1A) RCS LOOP to 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSE(IF REQUIRED)	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3

Step 19 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

19.5 Check standby RHR train discharge flow path available.

19.5.1 Verify standby RHR train - ALIGNED TO RCS COLD LEGS.

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV——OPEN	[] 8888A	[] 8888B

NOTE: The RHR HX bypass valves will fail closed and the RHR HX discharge valves will fail open upon loss of air to the AUX BLDG.

19.5.2 Verify standby RHR train HX BYP FLOW - ADJUSTED TO 15% OPEN.

Standby RHR Train	A	B
1A(1B) RHR HX BYP FLOW FK	[] 605A	[] 605B

19.5.3 Verify standby RHR train HX discharge valve - ADJUSTED CLOSED.

Standby RHR Train	A	B
1A(1B) RHR HX TO RCS DISCH VLV HIK	[] 603A	[] 603B

Response NOT Obtained:

19.5.3 Close standby RHR train - TO RCS COLD LEGS ISO valves. (121 ft, AUX BLDG piping penetration room)

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B

Step 19 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

19.6 Verify standby RHR train pump miniflow valve - OPEN.

Standby RHR Train	A	B
1A(1B) RHR PUMP MINIFLOW Q1E11FCV	[] 602A	[] 602B

19.7 Start RHR PUMP in standby train.

19.8 Control standby RHR train RHR HX bypass valve to obtain desired flow.

Standby RHR Train	A	B
1A(1B) RHR HX BYP FLOW FK	[] 605A	[] 605B

RNO 19.8 IF unable to control standby RHR train flow with RHR HX bypass valve,
THEN locally control RHR HX TO RCS COLD LEGS ISO valves.
(121 ft, AUX BLDG piping penetration room)

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B

___20 **IF RHR restored,**
THEN go to procedure and step in effect.

RNO 20 Continue efforts to restore at least one RHR train while continuing with this procedure.

Step	Action/Expected Response	Response NOT Obtained
___21	**Initiate protective measures for personnel in containment.**	
21.1	Evacuate all nonessential personnel from containment.	
21.2	Ensure HP monitors essential personnel remaining in containment for the following: [] Changing containment conditions which could require evacuation of all personnel. [] Use of extra protective clothing if needed. [] Use of respirators if needed.	
21.3	Monitor containment radiation monitors for changing conditions. [] R-2 CTMT 155 ft [] R-7 SEAL TABLE [] R-27A CTMT HIGH RANGE (BOP) [] R-27B CTMT HIGH RANGE (BOP)	

Step	Action/Expected Response	Response NOT Obtained
___22	**Start all available containment coolers**	
___ 22.1	Determine which containment coolers have Service Water aligned. [] Q1E12H001A [] Q1E12H001B [] Q1E12H001C [] Q1E12H001D	
22.2	Start Containment coolers with service water aligned and with power available in FAST speed. [] 1A CTMT CLR FAN FAST SPEED Q1E12H001A to START (BKR EA10) [] 1B CTMT CLR FAN FAST SPEED Q1E12H001B to START (BKR EB05) [] 1C CTMT CLR FAN FAST SPEED Q1E12H001C to START (BKR EB06) [] 1D CTMT CLR FAN FAST SPEED Q1E12H001C to START (BKR EC12)	22.2 Start Containment coolers with service water aligned and with power available in SLOW speed. [] 1A CTMT CLR FAN SLOW SPEED Q1E12H001A to START (BKR ED15) [] 1B CTMT CLR FAN SLOW SPEED Q1E12H001B to START (BKR ED16) [] 1C CTMT CLR FAN SLOW SPEED Q1E12H001C to START (BKR EE08) [] 1D CTMT CLR FAN SLOW SPEED Q1E12H001D to START (BKR EE16)
22.3	Check discharge damper open on any started containment cooler. [] CTMT CLR 1A DISCH 3186A indicates OPEN. [] CTMT CLR 1B DISCH 3186B indicates OPEN. [] CTMT CLR 1C DISCH 3186C indicates OPEN. [] CTMT CLR 1D DISCH 3186d indicates OPEN.	22.3 STOP any containment cooler whose discharge damper fails to indicate OPEN.
___23	**IF not previously started, THEN begin venting any RHR train(s) which have experienced evidence of cavitation using ATTACHMENT 1, RHR PUMP VENTING.**	

Step	Action/Expected Response	Response NOT Obtained

NOTE: Steps 24 and 25 should be performed in conjunction with the remainder of this procedure.

___24	**Check SGs available.** • Check SG primary nozzle dams - REMOVED. • Check SG primary manways - INSTALLED. • Check SG secondary handhole covers - INSTALLED.	24 Proceed to step 26.

NOTE: Establishing a secondary heat sink will reduce RCS heat up and pressurization rate to provide more time for recovery actions.

___25 **Verify secondary heat sink established.**

25.1 Maintain wide range level in all available SGs greater than 75% using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.

25.2 IF SG steam space intact, THEN open atmospheric relief valves to prevent SG pressurization.

1A(1B,1C) MS ATMOS REL VLV
[] PC 3371A adjusted
[] PC 3371B adjusted
[] PC 3371C adjusted

25.3 IF SGBD system available, AND AFW system available, THEN establish blowdown from available SGs using FNP-1-SOP-16.3, STEAM GENERATOR FILLING AND DRAINING.

Step	Action/Expected Response	Response NOT Obtained
___26	**Evaluate event classification and notification requirements using NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION AND INITIAL ACTION, NMP-EP-111, EMERGENCY NOTIFICATIONS, and FNP-0-EIP-8, NON-EMERGENCY NOTIFICATIONS.**	
___27	**Verify RCS isolated.**	
27.1	Close RHR TO LTDN HX HIK 142.	
27.2	Close LTDN LINE ISO Q1E21LCV459 and Q1E21LCV460.	
27.3	Close EXC LTDN LINE ISO VLV Q1E21HV8153 and Q1E21HV8154.	
27.4	Dispatch personnel to isolate all known RCS drain paths.	
27.5	Dispatch personnel to isolate any RCS leakage.	
___28	**Dispatch personnel to close hot leg recirculation valve disconnects. (139 ft, AUX BLDG rad-side)** CHG PUMP TO RCS HOT LEGS Q1E21MOV8886(8884) [] Q1R18B029-A (Master Z key) [] Q1R18B033-B (Master Z key)	
___29	**Check core cooling.**	If these conditions are met. Feed and Bleed is at Step 37. Core cooling is monitored here.
29.1	Check RCS level LESS than 121 ft 11 in <u>AND</u> core exit T/Cs GREATER than 200°F.	29.1 Return to step 1.0.

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Maintaining RCS level is the primary concern. RCS makeup should be restored as soon as possible through any available makeup path.
- RCS makeup flow requirements can exceed 90 gpm due to boil off if an adequate hot leg vent is established.

___30 **WHEN RHR flow restored,**
THEN proceed to step 40.

___31 **Check any CHG PUMP - AVAILABLE.**

31 Establish RWST gravity drain using ATTACHMENT 2, RWST TO RCS GRAVITY FEED.

31.1 WHEN gravity drain established,
THEN proceed to step 37.

___32 **Verify operable CHG PUMP miniflow valves - OPEN.**

1A(1B,1C) CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8109A
[] Q1E21MOV8109B
[] Q1E21MOV8109C

___33 **Verify CHG PUMP miniflow isolation valve - OPEN.**

CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8106

___34 **Verify RWST to CHG PUMP valve for operable CHG PUMP - OPEN.**

Operable CHG PUMP	1A	1B(A TRN)	1B(B TRN)	1C
RWST TO CHG PUMP Q1E21LCV	[] 115B	[] 115B	[] 115D	[] 115D

___35 **Verify operable CHG PUMP - STARTED.**

Step	Action/Expected Response	Response NOT Obtained
___36	**Verify required injection path isolation valve - OPEN.**	

Q1E21MOV8803A	HHSI TO RCS CL ISO
Q1E21MOV8803B	HHSI TO RCS CL ISO
Q1E21MOV8885	CHG PUMP RECIRC TO RCS COLD LEGS
Q1E21MOV8884	CHG PUMP RECIRC TO RCS HOT LEGS
Q1E21MOV8886	CHG PUMP RECIRC TO RCS HOT LEGS

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Reactor vessel level may be much lower than indicated if no hot leg vent path is available.

****

****

CAUTION: RCS pressurization may cause SG nozzle dam failure. This will cause a rapid loss of RCS inventory and the creation of a RCS spill pathway.

****

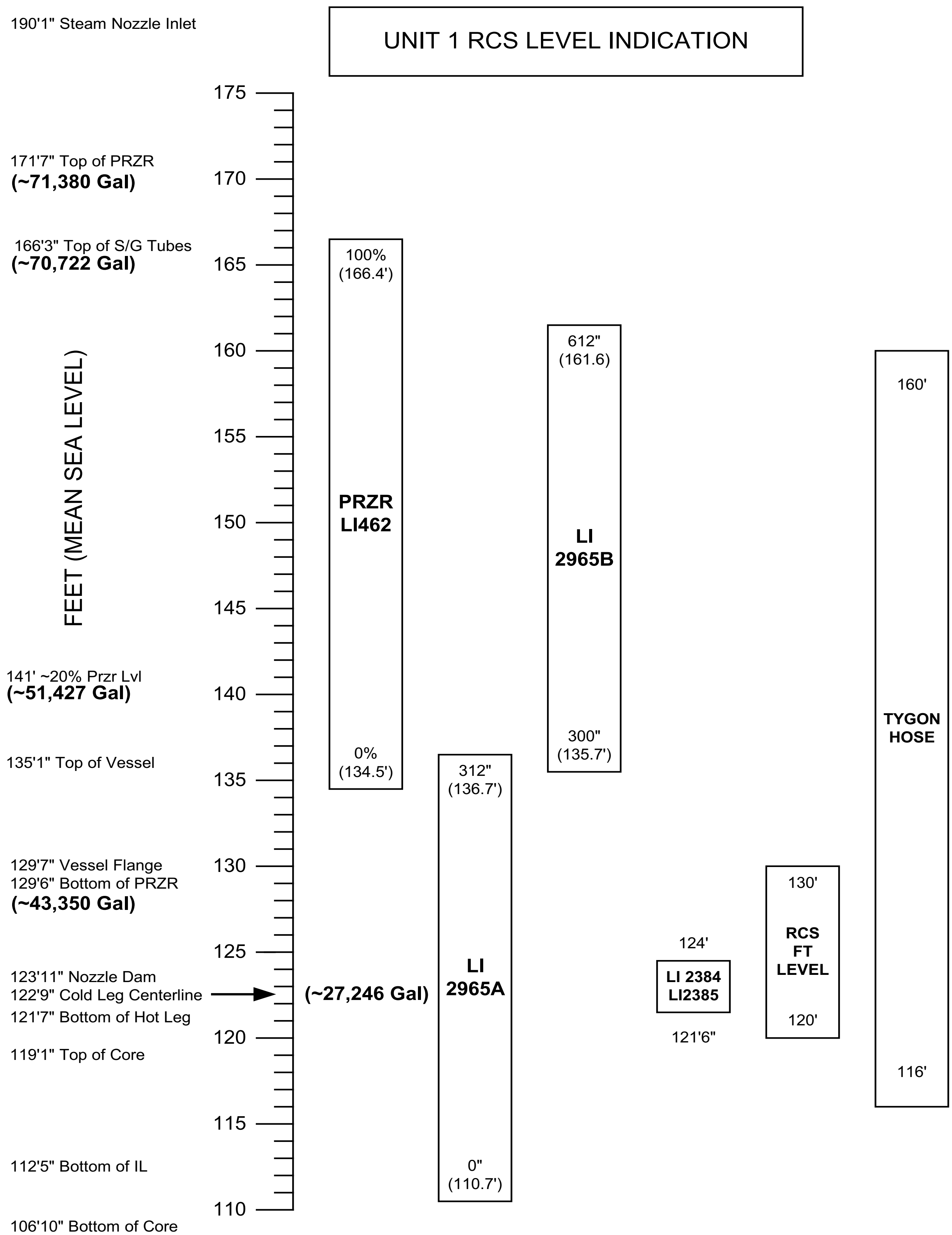
Step	Action/Expected Response	Response NOT Obtained
37	**IF RCS configuration will allow a level in the pressurizer, THEN establish feed and bleed cooling.**	37 IF RCS configuration will NOT allow a level in the pressurizer, THEN establish feed and spill cooling as follows.
37.1	Verify RCS bleed path available as follows. • Verify all pressurizer safety valves - REMOVED. OR • Verify pressurizer manway - REMOVED. OR • Verify both PRZR PORVs and PRZR PORV ISOs - OPEN.	a) Locally control required injection path isolation valve to maintain core exit T/Cs less than 200°F. b) Proceed to step 38.

Step 37 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
		4.1.2 Maintain cooldown rate in RCS cold legs less than maximum allowable limits of FIGURE 5.
		4.1.3 Go to step 4.3.
4.2	[CA] Maintain RCS cold leg cooldown rate - LESS THAN 25°F/hr. RCS COLD LEG TEMP [] TR 410	

Step 4 continued on next page.

Figure 2



UNIT 1 RCS LEVEL INDICATION
190'1" Steam Nozzle Inlet
171'7" Top of PRZR
(~71,380 Gal)
166'3" Top of S/G Tubes
(~70,722 Gal)
FEET (MEAN SEA LEVEL)
141' ~20% Przr Lvl
(~51,427 Gal)
135'1" Top of Vessel
129'7" Vessel Flange
129'6" Bottom of PRZR
(~43,350 Gal)
123'11" Nozzle Dam
122'9" Cold Leg Centerline
121'7" Bottom of Hot Leg
119'1" Top of Core
112'5" Bottom of IL
106'10" Bottom of Core
175
170
165
160
155
150
145
140
135
130
125
120
115
110
(~27,246 Gal)
100%
(166.4')
PRZR
LI462
0%
(134.5')
312"
(136.7')
LI
2965A
0"
(110.7')
612"
(161.6)
LI
2965B
300"
(135.7')
124'
LI 2384
LI2385
121'6"
130'
RCS
FT
LEVEL
120'
160'
TYGON
HOSE
116'

1. AOP-12.0-52520L04 009/HLT/LOCT//C/A 4.4/4.7/G2.1.7///LOCT/

Unit 1 is in Mode 5 with the follow ing conditions:

- The RCS is filled and vented.
- All SG water levels are 84% wide range.
- RCS temperature is 175°F.
- RCS pressure is 325 psig.
- Train A RHR in service, Train B RHR is tagged out for repairs.

Which one of the following is the method of core cooling in the event a loss of RHR shutdown cooling occurs?

A.✓ Establish a secondary heat sink by opening the atmospheric relief valves.

B. Normal charging to RCS, spill through the Pressurizer PORVs.

C. Actuate Safety Injection, spill through the Pressurizer PORVs.

D. Reflux cooling via any SG with > 10% NR level.

1. 026AG2.4.50 024/NEW//MEM 4.2/4.0/APE026G2.4.50/N///

Unit 1 is operating at 100% power when the following occurs:

- A leak develops in the CCW system.
- CCW Surge Tank level is slowly lowering.
- AA4 and AB4, CCW SRG TK LVL A(B) TRN HI-LO, are in alarm.
- AA5, CCW SRG TK LVL A TRN LO-LO, has come into alarm.

Which one of the following completes the statements below?

CCW system automatic isolations are designed to occur at _(1)_ in the CCW Surge Tank.

The _(2)_ is the NORMAL source of makeup water to the CCW Surge Tank.

	(1)	(2)
A.	35 inches	Reactor Makeup Water Storage Tank
B.	20 inches	Reactor Makeup Water Storage Tank
C.	35 inches	Demin Water Storage Tank
D.✓	20 inches	Demin Water Storage Tank

ARP AA5: 20 inches
Automatic Action
1. Closes CCW Valves (Q1P17HV3096A&B) to isolate CCW to/from Evaporator Packages and H2 Recombiners. (Q1P17LSLL3027CD-A)

2. Trips closed Q1P17HV2229, CCW to Sample Cooler (Q1P17LSLL3027CD-A).

Operator Action
4.1 Attempt to fill CCW surge tank using Normal Make-up to maintain level above the lo level alarm point as follows;

4.2. IF unable to fill the CCW Surge Tank per the Normal Make-up method, THEN attempt to fill CCW surge tank using Emergency Make-up to maintain level above the lo level alarm point as follows;

5. IF a loss of CCW cooling has occurred, THEN refer FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since this is the Surge tank LO Level alarm setpoint.

Second part is incorrect (See D.2). Plausible if candidate cannot recall which of the two makeup sources is the NORMAL source.

B. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

C. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

D. Correct. First part is correct. AA5 Setpoint is 20 inches which causes the automatic closure of HV3096A&B and HV2229.

Second part is correct. Demin water storage tank is the normal source for makeup to the CCW surge tank.

K/A: **026AG2.4.50**	Loss of Component Cooling Water (CCW) - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
Importance Rating:	4.2 4.0
Technical Reference:	FNP-1-ARP-1.1 - AA5, CCW SRG TK LVL A TRN LO-LO, Ver 53.1
References provided:	None
Learning Objective:	SELECT AND ASSESS the following instrument/equipment response expected when performing CCW System evolutions including the fail condition, alarms, and trip setpoints (OPS-52102G07). • Surge Tank Level
Question History:	NEW
K/A match:	Requires the applicant to **determine at which level the automatic isolations of the CCW system occur (verify system alarm setpoints which is when these valves close) and know what source of water is used to fill the surge tank (operate controls identified in the ARP to raise the CCW Surge Tank level).**
SRO justification:	N/A

LOCATION AA4

A4
CCW SRG TK LVL A TRN HI-LO

SETPOINT:
1. HI: 50 ± 0.3 inches
2. LO: 35 ± 0.3 inches

ORIGIN:
1. Level Switch (N1P17LSH3027A-A).
2. Level Switch (N1P17LSL3027A-A).

PROBABLE CAUSE

1. HI - In Leakage of Reactor Coolant, Service Water, or through a Makeup Water Valve
   - Letdown heat exchanger (if letdown on service)
   - RCP thermal barriers
   - RHR heat exchanger (if on service)
   - Reactor makeup system (if normally closed, valves leaking by)
   - Demineralized water system (if normally closed, valves leaking by)
   - SW system (if SW discharge pressure higher than CCW discharge pressure)
   - RCDT heat exchanger (if at least one RCDT pump running

2. LO - Rupture or Leakage of A Train CCW components or piping:
   - Spent fuel pool heat exchanger
   - Charging pump oil coolers
   - RHR heat exchanger (if normally closed, MOV 3185 A or B are open)
   - RCP oil coolers
   - Excess letdown heat exchanger (if excess letdown secured)
   - Sample coolers (if sampling not in progress)
   - Seal water heat exchanger
   - Evaporator packages
   - Hydrogen recombiner
   - Waste gas compressors
   - Floor drain tank via CCW relief valves
   - SW system if CCW discharge pressure is higher than SW discharge pressure
   - Primary and secondary sample coolers (if sampling in progress)
   - GFFD sampling assembly

LOCATION AA5

SETPOINT: 20 ± .15"

ORIGIN: Level Switch (Q1P17LSLL3027CA-A)

A5	
	CCW SRG TK LVL A TRN LO-LO

PROBABLE CAUSE

Rupture or leakage of an A Train CCW component or pipe.

AUTOMATIC ACTION

1. Closes CCW Valves (Q1P17HV3096A&B) to isolate CCW to/from Evaporator Packages and H_2 Recombiners. (Q1P17LSLL3027CD-A)

2. Trips closed Q1P17HV2229, CCW to Sample Cooler (Q1P17LSLL3027CD-A).

OPERATOR ACTION

1. Ensure that the automatic actions have occurred.

2. Dispatch personnel to locate and isolate the source of leakage.

3. Notify chemistry to secure the sample system. (2008108346)

4. Perform the following:

   4.1 Attempt to fill CCW surge tank using Normal Make-up to maintain level above the lo level alarm point as follows;

      4.1.1 Notify Shift Chemist that the CCW surge tank is to be made up to.

      4.1.2 Verify open CCW SRG TK VT valves: (MCB)

      - Q1P17SV3028A
      - Q1P17SV3028B

      4.1.3 Monitor CCW surge tank level indications.

      - LI-3027A
      - LI-3027B

4.1.4 Maintain level between 35 inches and 50 inches.

4.1.5 IF desired, THEN open MKUP TO CCW FROM DW STOR TK Q1P17MOV3030A to add makeup to A portion of CCW surge tank.

4.1.6 IF desired, THEN open MKUP TO CCW FROM DW STOR TK Q1P17MOV3030B to add makeup to B portion of CCW surge tank.

4.1.7 WHEN makeup addition is completed, THEN close appropriate valve(s).

- MKUP TO CCW FROM DW STOR TK Q1P17MOV3030A.
- MKUP TO CCW FROM DW STOR TK Q1P17MOV3030B.

4.2 IF unable to fill the CCW Surge Tank per the Normal Make-up method, THEN attempt to fill CCW surge tank using Emergency Make-up to maintain level above the lo level alarm point as follows;

CAUTION:	**Reactor makeup water should only be used as an emergency source of makeup water to the CCW surge tank.**

4.2.1 Close CCW SRG TK DEMIN INLET ISO, N1P11V045.

4.2.2 Verify open CCW SRG TK VT valves: (MCB)

- Q1P17SV3028A
- Q1P17SV3028B

4.2.3 Monitor CCW surge tank level indications.

- LI-3027A
- LI-3027B

4.2.4 Maintain level between 35 inches and 50 inches.

4.2.5 IF desired, THEN open MKUP TO CCW FROM RMW Q1P17MOV3031A to add makeup to the A portion of the CCW surge tank.

4.2.6 IF desired, THEN open MKUP TO CCW FROM RMW Q1P17MOV3031B to add makeup to the B portion of the CCW surge tank.

4.2.7 WHEN makeup addition is completed, THEN close appropriate valve(s).

- MKUP TO CCW FROM RMW Q1P17MOV3031A
- MKUP TO CCW FROM RMW Q1P17MOV3031B

4.2.8 Open CCW SRG TK DEMIN INLET ISO, N1P11V045.

4.3 IF the off-service train CHG pump is running, THEN perform the following to reduce CCW pressure:

4.3.1 Swap running CHG pumps per FNP-1-SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION.

4.3.2 Secure the off-service CCW train per FNP-1-SOP-23.0, COMPONENT COOLING WATER SYSTEM.

4.4 Control SW pressure to minimize DP across CCW HX tubes:

- HIC 3009A 1A CCW HX DISCH FCV
- HIC 3009B 1B CCW HX DISCH FCV
- HIC 3009C 1C CCW HX DISCH FCV
- Q1P16V558 A TRN SW DIL BYP ISO
- Q1P16V557 B TRN SW DIL BYP ISO

NOTE: If CCW to SW leakage is suspected, expeditious actions, consistent with safe plant operation, to secure the pump in the leaking CCW train should be taken. This will minimize environmental release of chromated and potentially radioactive water.

4.5 IF CCW HX tube leak suspected, THEN notify the following groups to assess the environmental impacts, notification requirements and corrective actions for potential chromated water and radiological contamination of the service water system:

- Chemistry
- Environmental
- Health Physics

5. IF a loss of CCW cooling has occurred, THEN refer FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.

6. Refer to Technical Specification 3.7.7 for LCO requirements with a loss of the on service train of component cooling water.

References A-177100, Sh. 55; D-175002, Sh. 1 & 2; B-175968, Sh. 6; D-177183; D-277185; D-177092; D-177670; D-177853; B-175810, Sh. 9, 22, 23 & 101; Technical Specification 3.7.7; PCN B91-1-7431

1. 026K2.01 025/NEW//MEM 3.4*/3.6/026K2.01/N///

Unit 2 is operating at 50% power when a simultaneous Dual Unit LOSP occurs.

- 4160V Bus 2G remains de-energized due to the DG not starting for that emergency bus.

Three (3) minutes after the LOSP, a Large Break LOCA occurs on Unit 2.

- Containment pressure peaked at 29 psig and is trending down.

Which one of the following completes the statement below?

The __(1)__ Containment Spray pump is currently running and is powered from the __(2)__ DG.

	(1)	(2)
A.	2A	1C
B.	2B	1C
C.✓	2A	1-2A
D.	2B	1-2A

In the LOSP the 1-2A DG will tie onto the unit 1 A Train busses. Then when the SI and subsequent phase B signal comes in the 1-2A DG and 1C DG will load shed, and then 1-2A DG will align to Unit 2 and the 2A CS pump will start at step 2.

FSD-A181008: 3.1.5.2
Without offsite power available, the CSS pumps shall start by the diesel generator ESS loading sequencer. Starting will occur at step two of the sequence if the "P" signal is present at that time. If the "P" signal occurs between the completion of step two and step six of the ESS sequence, then starting will occur at the completion of step six of the loading sequence. If the "P" signal occurs after the completion of step six, starting will
take place immediately.

Pg 2-1 CSS initiation is automatic upon a containment pressure hi-3 signal ("P" signal)

FSD-A181005
LOSP on both units and LOCA on Unit 2:
For LOSP on both units and LOCA on Unit 2, the alignment of the diesel generators will be as follows:

1-2A Unit 2 Buses 2F and 2K
1C Unit 1 Buses 1F, 1K and 1H

1B Unit 1 Buses 1G, 1L and 1J
2B Unit 2 Buses 2G, 2L and 2J

Distracter analysis

A. Incorrect. First part is correct (See C.1).

Second part is incorrect (See C.2). Plausible since the the 1C and 1-2A DGs align to either Unit's 'A' Train depending on the scenario and the applicant may not recall the proper DG alignment for this scenario.

B. Incorrect. First part is incorrect (See C.1). Plausible if the applicant does not recall the CS pump power supplies.

Second part is incorrect (See A.2)

C. Correct. First part is Correct. The normal power supply to 2A CS pump is 2F 4160V AC bus.

Second part is correct. For LOSP on both units and LOCA on Unit 2, the alignment of the diesel generators will be as follows:

- 1-2A Unit 2 Buses 2F and 2K

D. Incorrect. First part is incorrect (See B.1).

Second part is correct (See C.1).

K/A: **026K2.01**	Containment Spray System (CSS) - Knowledge of bus power supplies to the following: Containment spray pumps.
Importance Rating:	3.4* 3.6
Technical Reference:	FSD-A181008, Containment Spray System, Ver 24. A-351199, Unit 2 Load List, Ver 61 FSD-A181005, Diesel Generators, Ver 44. FSD-A181007, Reactor Protection System, Ver 18
References provided:	None
Learning Objective:	NAME AND IDENTIFY the Bus power supplies, for those electrical components associated with the Containment Spray and Cooling System, to include those items in Table 3- Power Supplies (OPS-40302D04).
Question History:	NEW
K/A match:	Requires the applicant to **know the normal power supply to the 2B CS pump** and the 1-2A DG alignment and power supply to the 2A CS pump upon an LOSP with a subsequent SI.
SRO justification:	N/A

A.3.1.1 LOSP on both units

For LOSP on both units, the alignment of the diesel generators will be as follows:

1-2A	Unit 1	Buses 1F, 1K and 1H
1C	Unit 2	Buses 2F, 2K and 2H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

The design decision to align the diesel generator 1-2A to Unit 1 and diesel generator 1C to Unit 2 for this scenario was arbitrary since both diesels are capable of energizing the LOSP loads for either unit.

A.3.1.2 LOSP on both units and LOCA on Unit 1

For LOSP on both units and LOCA on Unit 1, the alignment of the diesel generators will be as follows:

1-2A	Unit 1	Buses 1F and 1K
1C	Unit 2	Buses 2F, 2K and 2H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

Diesel generator 1-2A must be aligned to Unit 1 because diesel generator 1C does not have the capacity to energize a full train of LOCA loads; therefore, diesel generator 1C must align to the non-LOCA unit (Unit 2).

A.3.1.3 LOSP on both units and LOCA on Unit 2

For LOSP on both units and LOCA on Unit 2, the alignment of the diesel generators will be as follows:

1-2A	Unit 2	Buses 2F and 2K
1C	Unit 1	Buses 1F, 1K and 1H
1B	Unit 1	Buses 1G, 1L and 1J
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

As mentioned in section A.3.1.2, diesel generator 1C does not have the capacity to energize all of theLOCA loads; therefore, diesel generator 1-2A must be aligned to the LOCA unit (Unit 2) and diesel generator 1C will be aligned to the non-LOCA unit (Unit 1).

A.3.1.4 LOSP on Unit 1

For LOSP on Unit 1 only, the diesel generator alignment will be as follows:

1-2A	Unit 1	Buses 1F, 1K and 1H
1B	Unit 1	Buses 1G, 1L and 1J

(Reference 6.7.08)

The use of diesel generator 1-2A as opposed to diesel generator 1C was an arbitrary design choice because both diesels have sufficient capacity to energize a full train of LOSP loads. However, diesel generator 1C will automatically start and idle but it will not connect to any bus.

A.3.1.5 LOSP on Unit 2

For LOSP on Unit 2 only, the diesel generator alignment will be as follows:

1-2A	Unit 2	Buses 2F, 2K and 2H
2B	Unit 2	Buses 2G, 2L and 2J

(Reference 6.7.080)

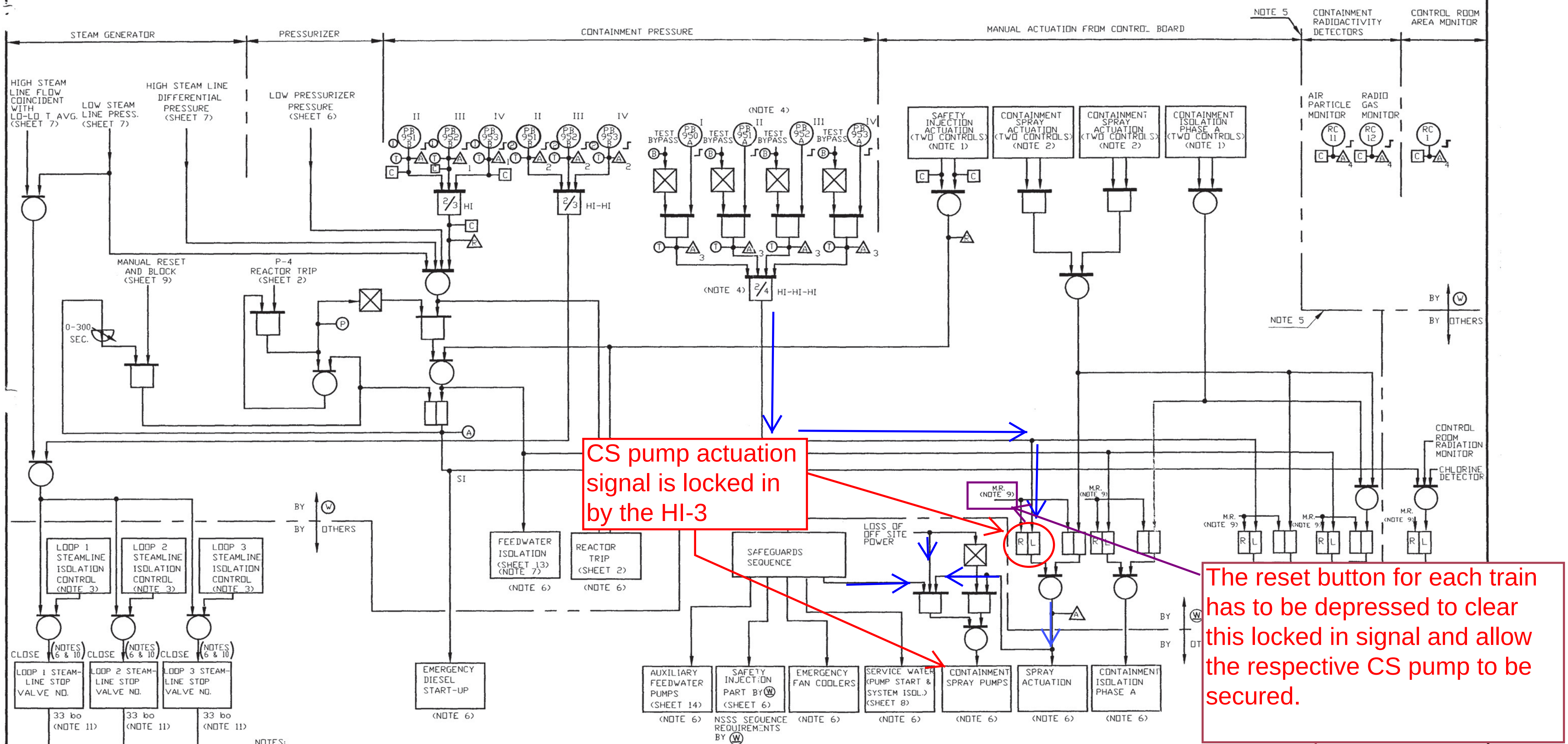
As mentioned in section A.3.1.4, the use of diesel generator 1-2A as opposed to diesel generator 1C was an arbitrary design choice. As in the previous scenario, diesel generator 1C will start and idle but it will not connect to any bus.

A.3.1.6 LOSP and LOCA on Unit 1

For LOSP and LOCA on Unit 1 only, the diesel generator alignment will be as follows:

1-2A	Unit 1	Buses 1F and 1K
1B	Unit 1	Buses 1G, 1L and 1J

(Reference 6.7.080)



NOTES:

1. TWO MOMENTARY CONTROLS: OPERATING EITHER CONTROL WILL ACTUATE.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE MOMENTARY CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE TO ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. ENCLOSED CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM AND IS NOT REDUNDANT.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. FEEDWATER ISOLATION INCLUDES THE TRIPPING OF ALL MAIN FEEDWATER PUMPS.
8. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
9. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
10. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
11. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAM LINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.

SAFEGUARDS ACTUATION SIGNALS

FIGURE F-2 SHEET 8

RPSFSD08.DWG

2.0 SYSTEM FUNCTIONAL REQUIREMENTS

The CSS consists of two independent trains for each unit. The CSS for each unit consists of two pumps, two eductors, trisodium phosphate filled baskets, spray ring headers with spray nozzles, valves, piping, and instrumentation (References 6.2.001, 6.7.033, 6.7.034).

The safety-related function of the CSS is to reduce the containment building pressure and temperature following a LOCA or high-energy line rupture and to reduce airborne fission products in the containment atmosphere following a LOCA.

During the injection phase, the CSS pumps are aligned to take suction off the RWST. When the RWST reaches low-low level, the spray pumps operate in the recirculation mode from the containment sump. Operator action to perform realignment of the CSS pumps to sump recirculation must be completed within 130 seconds of reaching the RWST low-low level setpoint. Completion of this operator action in 130 seconds ensures sufficient volume remains in the RWST to ensure adequate pump NPSH is available and to prevent vortexing in the RWST (References 6.3.020, 6.7.039). Trisodium phosphate (TSP) filled baskets in the recirculation area of containment provide iodine absorption and retention in the containment sump solution (References 6.2.001, 6.3.001, 6.7.001).

As the RCS inventory combined with ECCS solution accumulates in the recirculation sump, the rising water level dissolves the TSP crystals in the baskets (References 6.7.033 and 6.7.034).

CSS initiation is automatic upon a containment pressure hi-hi-hi signal ("P" signal) (References 6.2.001, 6.4.001).

For maximum containment pressure, the CSS is required to deliver a minimum flow rate of 2,480 gpm to the spray rings during injection and 2,290 gpm during recirculation as assumed for the minimum ESF flow in the containment pressure and temperature analysis (References 6.1.005, 6.3.019).

Minimum calculated flow rate to the spray rings is based on single CSS pump operation, a 5% degraded pump curve, containment design pressure, and minimum RWST level (References 6.3.003, 6.7.002).

The maximum calculated flow rate during the recirculation mode with one pump operating is 3,400 gpm per pump, with 3,257 gpm directed to the spray rings and 143 gpm recirculated (Reference 6.3.015). For minimum containment pressure, maximum containment spray system flow rate assumed in the accident analysis to minimize containment backpressure is 6,800 with both CSS pumps operating (References 6.1.006, 6.7.035, 6.7.036).

insulation. Operation at temperatures above 104 °F will result in reduced motor life (Reference 6.7.004).

Operation at temperatures greater than 104°F may be required under certain accident conditions as described in Section 2.1.1 (References 6.1.004, 6.7.037).

3.1.3 Code Requirements

3.1.3.1 The CSS pumps shall meet the requirements of the Draft ASME Code for Pumps and Valves, 1968 Edition (Reference 6.2.002).

3.1.3.2 The pumps are classified Safety Class 2A (References 6.7.005, 6.4.004).

3.1.3.3 The CSS pump motors are category IE1 and shall meet the requirements of NEMA MG1-14.38, "Standards for Motors and Generators" (Reference 6.7.004).

3.1.4 Seismic Qualification Requirements

The CSS pumps are designated as Seismic Class I. The pump specification requires that the pumps be able to withstand the forces caused by a horizontal ground acceleration of 1.0 g and a vertical ground acceleration of 0.67 g applied simultaneously at the center of gravity (Reference 6.2.002).

3.1.5 I&C Requirements

Service	TPNS No.
MCB Control	HS-6505A, HS-6505B
CS Reset	HS-2113A, HS-2113B

3.1.5.1 Manual and automatic control of the CSS pumps shall be provided in the main control room (MCR) to support the CSS pump functions (Reference 6.4.006).

3.1.5.2 With offsite power available, the "P" signal shall start both CSS pumps. Without offsite power available, the CSS pumps shall start by the diesel generator ESS loading sequencer. Starting will occur at step two of the sequence if the "P" signal is present at that time. If the "P" signal occurs between the completion of step two and step six of

the ESS sequence, then starting will occur at the completion of step six of the loading sequence. If the "P" signal occurs after the completion of step six, starting will take place immediately. Automatic starting of the CSS pumps shall not occur unless the pump control switch on the main control board is in the "AUTO" position (References 6.4.001, 6.4.006, 6.4.007, 6.4.008).

3.15.3 Following automatic starting of the CSS pumps from a "P" signal, depressing the CS Reset pushbutton shall reset the containment spray actuation signal. This will allow the CSS pumps to be secured using the MCB control switch for each pump. (References 6.4.006, 6.4.042, 6.4.043)

3.1.5.4 The MCB control switch for each CSS pump shall have "STOP", "START", and "AUTO" positions (spring return to "AUTO") and status lights (green for stop, red for operating status, amber for fault trips). Additionally, fault trips shall actuate a main control board annunciator for each circuit breaker (References 6.4.006, 6.4.038, 6.4.039).

3.1.5.5 The running status of the CSS pumps shall be displayed on an MCB monitor light box (Reference 6.7.020).

3.1.5.6 For I&C requirements of pump discharge test flow instruments FI-944 and FE-944, pump suction pressure indicators PI-945A and PI-946A, and pump discharge pressure indicators PI-945B and PI-946B, see Section 5.26.

3.1.6 EQ Requirements

The CSS pump motors shall be environmentally qualified as detailed in the Master List of Environmentally Qualified Equipment and EQ Package 43.

Maintenance specified in the EQ package shall be performed to maintain qualified life (References 6.7.009, 6.7.010, 6.7.011, 6.7.012).

3.1.7 Interface Requirements

3.1.7.1 The CSS pumps shall be provided with safety-related power from the ac buses. This is to assure the availability of electrical power to support the system safety functions (Reference 6.7.013). The diesel generator loading calculation assumes the pump motor demand shall not exceed 359 kW. This maximum design basis loading was

2F 4160V BUS **AB - 139'** **D-207005**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0006-A	**2F 4160V BUS**	
DF01	**N2R11A0501-N**	**2A STARTUP TRANSFORMER (NORMAL) <<<**	
DF02	**Q2R15A0505-A**	**2K 4160V BUS >>>**	K-1
DF03	**Q2R11B0004-A**	**2D 4160/600V SST >>> ED02 (NORMAL) >>>**	F-2
DF04	Q2P17M0001C-A	2C CCW PUMP	
DF05	Q2P17M0001B-AB	2B CCW PUMP DISC SWITCH Q2R18A0004A-A >>> 2B CCW PUMP (A TRAIN SUPPLY)	
DF06	Q2E21M0001A-A	2A CHARGING/HHSI PUMP	
DF07	Q2E21M0001B-AB	2B CHG PUMP DISC SWITCH Q2R18A0001A-A >>> 2B CHARGING/HHSI PUMP (A TRAIN SUPPLY)	
DF08	**QSR43A0501-A**	**1-2A DIESEL GENERATOR (EMERG) <<<**	
DF09	Q2E11M0001A-A	2A RHR/LHSI PUMP	
DF10	Q2N23M0001A-A	2A AFW PUMP	
DF11	Q2E13M0001A-A	2A CTMT SPRAY PUMP	
DF12	**Q2R16B0008-AB**	**2F 4160/600V SST DISC SW Q2R18A0003A-A >>> 2F 4160/600V SST >>> 2F LOAD CENTER (A TRAIN SUPPLY) >>>**	F-108
DF13	**Q2R15A0503-A**	**2H 4160V BUS >>>**	H-1
DF14	Q2R15BKRDF14	PT COMPARTMENT	
DF15	**N2R11A0502-N**	**2B STARTUP TRANSFORMER (ALT) <<<**	

2G 4160V BUS AB - 121' D-207006

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0007-B	2G 4160V BUS	
DG01	N2R11A0501-N	2A STARTUP TRANSFORMER <<<	
DG02	Q2R15A0506-B	2L 4160V BUS >>>	L-1
DG03	Q2R11B0005-B	2E 4160/600V SST >>> EE02	G-2
DG04	Q2P17M0001A-B	2A CCW PUMP	
DG05	Q2P17M0001B-AB	2B CCW PUMP DISC SWITCH Q2R18A0004B-B >>> 2B CCW PUMP (B TRAIN SUPPLY)	
DG06	Q2E21M0001C-B	2C CHARGING/HHSI PUMP	
DG07	Q2E21M0001B-AB	2B CHG PUMP DISC SWITCH Q2R18A0001B-B >>> 2B CHARGING/HHSI PUMP (B TRAIN SUPPLY)	
DG08	Q2R43A0505-B	2B DIESEL GENERATOR <<<	
DG09	Q2E11M0001B-B	2B RHR/LHSI PUMP	
DG10	Q2N23M0001B-B	2B AFW PUMP	
DG11	Q2E13M0001B-B	2B CTMT SPRAY PUMP	
DG12	Q2R11B0006-AB	2F 4160/600V SST DISC SW Q2R18A0003B-B >>> 2F 4160/600V SST >>> 2F LOAD CENTER (B TRAIN SUPPLY)>>>	F-108
DG13	Q2R15A0504-B	2J 4160V BUS >>>	J-1
DG14	Q2R15BKRDG14	PT COMPARTMENT	
DG15	N2R11A0502-N	2B STARTUP TRANSFORMER <<<	

Normal Power Supply

Emergency power supply

1. 029A1.02 026/NEW//C/A 3.4/3.4/029A1.02/N///

Unit 2 plant conditions are as follows:

- Containment Main Purge system is running.
- Containment radiation levels are **rising**.

Subsequently, R-24A, CTMT PURGE, loses control power.

Which one of the following completes the statements below?

Radiation levels _(1)_ stop rising in the Main Exhaust Plenum.

CTMT Main Purge supply and exhaust fans _(2)_ trip.

	(1)	(2)
A.✓	WILL	will NOT
B.	will NOT	will NOT
C.	WILL	WILL
D.	will NOT	WILL

<u>SOP-45:</u>
3.5 The radiation monitors fail to a “High Radiation” condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

<u>FH4: CP RE 24A or B HI RAD</u>
<u>PROBABLE CAUSE</u>
1. High Radiation Level in the Containment Purge Exhaust Line.
2. The radiation monitors fail to a “High Radiation” condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

<u>AUTOMATIC ACTION</u>
1. Isolates Containment by closing Purge Supply and Exhaust Valves 2-CP-HV-3196, 2-CP-HV-3197, 2-CP-HV-3198A, B, C, & D, 2-CP-HV-2867C & D and 2-CP-HV-2866C & D.

<u>Distracter analysis</u>

A. Correct. First part is correct. Per SOP-45, a radiation monitor that has lost instrument power will initiate its automatic actions.

Second part is correct. R-24A will NOT automatically secure the main purge supply and exhaust fans.

B. Incorrect. First part is incorrect (See B.1). Plausible if candidate does not recall that a loss of control power will cause the actuation of associated automatic functions.

Second part correct (See B.2).

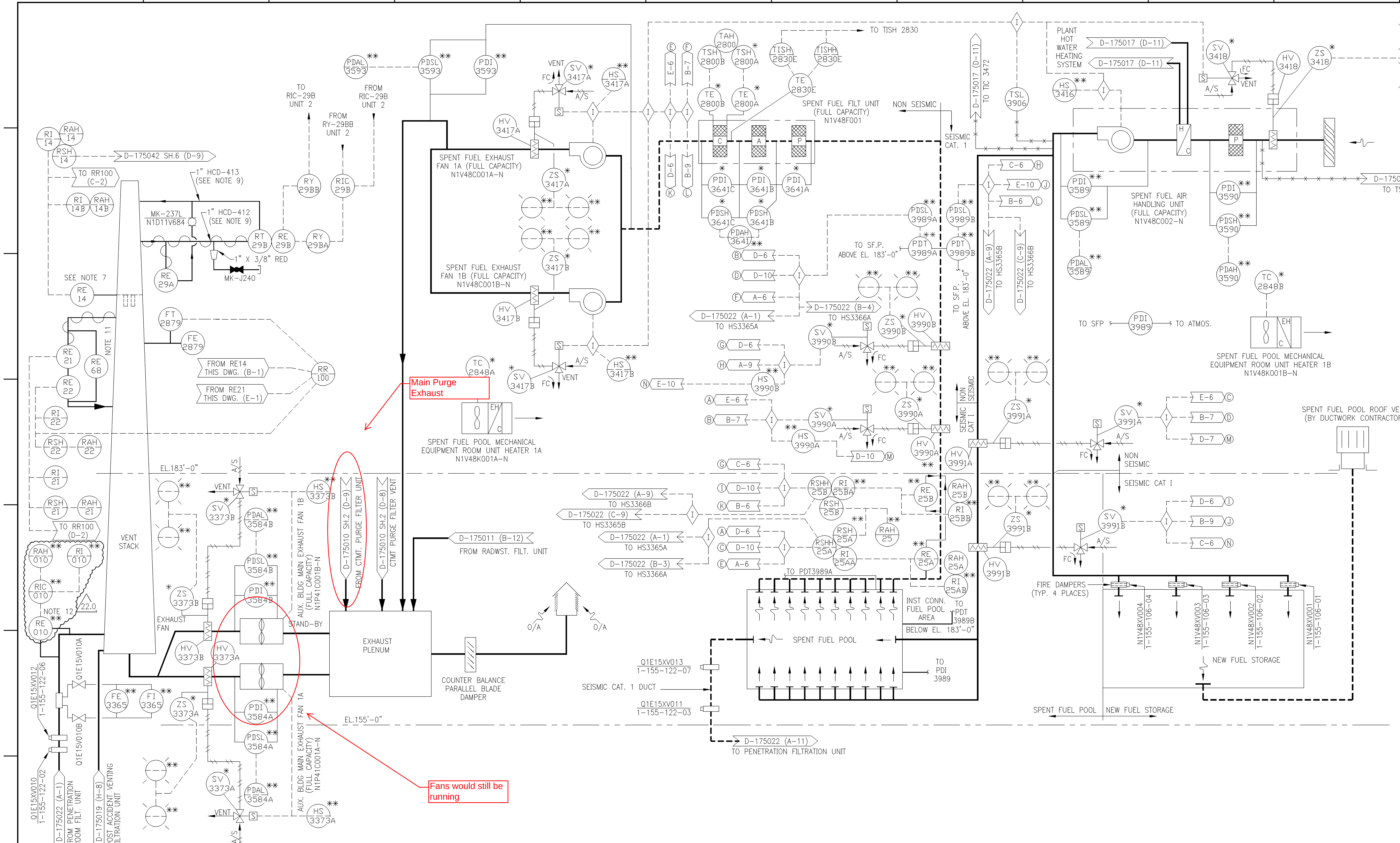
C. Incorrect. First Part is incorrect (See A.1).

Second part is incorrect. (See B.1). Plausible since it could seem logical to the applicant that when the main purge supply and exhaust dampers shut, the fan would also automatically secure.

D. Incorrect. First part is correct (See B.1)

Second part is incorrect (See C.2). Plausible if the applicant does not recall what auto functions are actuated by R-24A and believes that R-24A will trip the fans. Additionally, with the Aux Building main exhaust fan running, the applicant could reason that there is still a negative pressure on the CTMT purge outlet causing CTMT radiation release to the plant vent stack to continue.

K/A: **029A1.02**	Containment Purge System (CPS) - Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Radiation levels
Importance Rating:	3.4 3.4
Technical Reference:	FNP-2-ARP-1.6, FH4, CP RE 24A or B HI RAD, Ver 59 FNP-2-SOP-45.0, Radiation Monitoring System, Ver 38.1
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Containment Ventilation and Purge System, to include those items in Table 6-Component Locations (OPS-40304A02).
Question History:	NEW
K/A match:	Requires the applicant to **predict, based on a loss of control power to R-24A, when the earliest time the radiation release is terminated thereby preventing the off site radiation exposure limit from potentially being exceeded.**
SRO justification:	N/A



NOTES:

1. FOR DUCT MATERIAL & CONSTRUCTION SEE (SPEC. SS-1102-54)
2. FOR DAMPER INFORMATION SEE (SPEC. SS-1102-54).
3. FOR EQUIPMENT DESCRIPTION SEE (SPEC. SS-1102-15, SS-1102-45, SS-1102-80).
4. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX D-175803.
5. FOR SEISMIC CLASSIFICATION OF DUCTWORK SEE (SPEC. SS-1102-54).
6. ** ASSOCIATED INSTRUMENTATION TO BE PROVIDED UNDER INSTRUMENTATION & CONTROL CONTRACT.

   * ASSOCIATED INSTRUMENTATION TO BE PROVIDED BY EQUIP. MFGR.

   ALL OTHER INSTRUMENTATION SHALL BE PROVIDED BY DUCTWORK SUBCONTRACTOR.
7. RADIATION INSTRUMENTS (RE 14) SHALL BE FURNISHED BY WESTINGHOUSE.
8. SEE DWG. D-175015 FOR PROCESS FLOW.
9. SMLS OR WELDED SCH 40 PIPE MAY BE USED FOR 1" HCD-412 AND 1" HCD-413.
10. B-508501 – UNIT NO. 1 & SHARED FIRE DAMPER REPORT.
11. FOR DETAILS SEE B-175983.
12. RADIATION INSTRUMENT (RE 0010) SHALL BE FURNISHED BY GENERAL ATOMICS-ESI.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205045 SH.1

CAD D1750101
ACAD2K JLO – 03

Southern Company Services, Inc. for
ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
SUBJECT HVAC – P&ID
SPENT FUEL POOL VENT SYSTEM
SCALE NONE
SHEET 1 OF – SHEETS
D-175045

FOR SIGNATURES SEE MICROFILM COPY
JLO GDM JMR
VER. NO. 22.0 DATE 6-29-10
REVISED PER ABN 1071862301J019, VER. 1.0

DRAWING CATEGORY: CRITICAL

LOCATION FH4

H4
CP RE24 A OR B HI RAD

SETPOINT: Variable, as per FNP-2-RCP-252

ORIGIN: Radiation Monitor Cabinet Channels R-24A or R-24B Containment Purge

PROBABLE CAUSE

1. High Radiation Level in the Containment Purge Exhaust Line.
2. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

AUTOMATIC ACTION

1. Isolates Containment by closing Purge Supply and Exhaust Valves 2-CP-HV-3196, 2-CP-HV-3197, 2-CP-HV-3198A, B, C, & D, 2-CP-HV-2867C & D and 2-CP-HV-2866C & D.

OPERATOR ACTION

1. Determine which radiation monitor indicates high activity.
2. Verify that any required automatic actions have occurred and if required, secure any running containment purge or mini-purge fans per FNP-2-SOP-12.2, CONTAINMENT PURGE AND PRE-ACCESS FILTRATION SYSTEM.
3. Notify HP personnel of alarm.
4. Implement NMP-EP-110 EMERGENCY CLASSIFICATION DETERMINATION AND INITIAL ACTION.
5. Determine the validity of the high activity indication as follows:
   - 5.1 Verify that the instrument is aligned for normal operation and is functioning properly.
   - 5.2 Sample or survey the affected system or area as required.
6. Determine the source or cause of the high activity and correct or isolate as required.
7. DO NOT allow personnel to enter the affected area without the approval of the Health Physics Foreman.
8. IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.6.
9. IF high activity indication of RCS leakage is present AND accompanied by either;
   - 9.1 Decreasing pressurizer level
   - 9.2 Decreasing VCT level

10 THEN go to FNP-2-AOP-1.0, RCS LEAKAGE.

11. WHEN activity levels have decreased below the alarm setpoint, THEN reset HI alarm on the RAD monitor by depressing the FAIL/RESET pushbutton.

References: A-207100, Sh. 309; U-213901; D-204658; D-204671; D-207199; D-207204; FSAR, Section 11.4; D-205010, Sh. 2

1.0 PURPOSE

1.1 This procedure provides the Initial Conditions, Precautions, Limitations, and Instructions for operating the Radiation Monitoring System.

2.0 INITIAL CONDITIONS

2.1 The electrical distribution system is energized and aligned for normal operation per FNP-2-SOP-36.0, PLANT ELECTRICAL DISTRIBUTION LINE-UP, with exceptions noted.

2.2 120V AC electrical distribution system is energized and aligned for operation per FNP-2-SOP-36.4, 120V A.C. DISTRIBUTION SYSTEMS.

2.3 Power fuses are installed in all radiation monitoring system instrument drawers.

2.4 The radiation monitoring system electrical distribution system is aligned per System Check List FNP-2-SOP-45.0A.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Due to slow filter paper speed of the APD a five hour time period is required for the detector indication to reach equilibrium value after changing filter paper or filter paper speed.

3.2 Alarms on Radiation Monitoring System Panel must be acknowledged to provide main control board annunciator reflash capability.

3.3 A common annunciator on the main control board is actuated on high radiation from any channel. Individual drawers shall be checked to determine the alarming channel(s).

3.4 A common annunciator on the main control board is actuated when any channel is in the test mode.

3.5 The radiation monitors fail to a “High Radiation” condition on loss of instrument and/or control power that will result in actuation of associated automatic functions. Prior to removing a channel from service, **ensure** any automatic functions are either disabled or acceptable with respect to the affect (including reportability consideration) on associated system operation. (**Refer** to Tables A, B and C for monitors with automatic functions.)

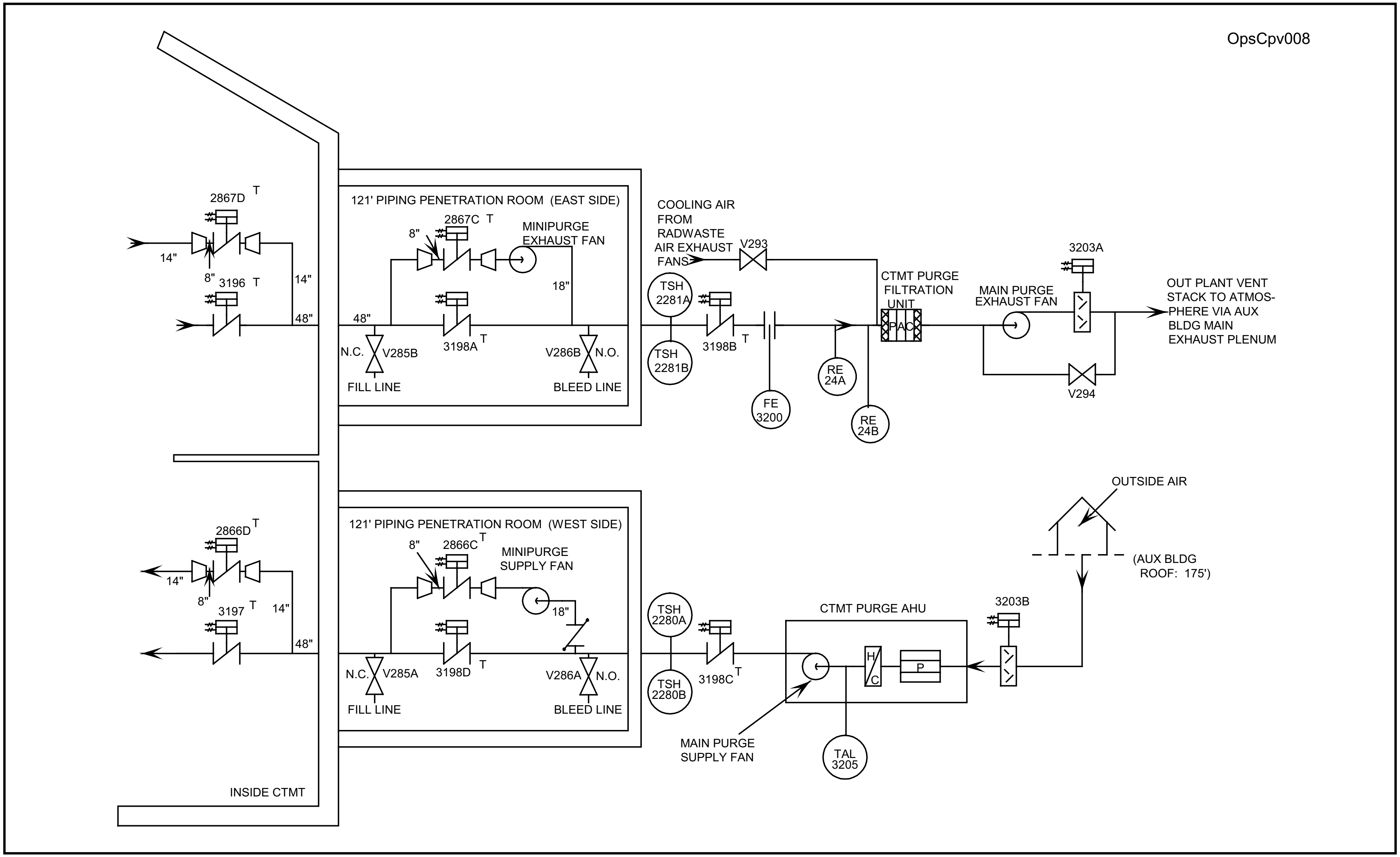


FIGURE 2 - Containment Purge System

1. 033K4.05 027/MOD/NORTH ANNA 08/C/A 3.1/3.3/033K4.05/N///TELL NRC

Unit 1 is operating at 100% power and the following conditions exist:

- A blended make-up to the Spent Fuel Pool (SFP) is occurring.
- A calibration error results in FT-168, PRI WATER MKUP FLOW, providing a flow input to the Reactor Makeup System that is **less** than the actual flowrate.

Which one of the following completes the statements below?

The blended flow makeup resulted in a __(1)__ of the SFP.

Per Tech Spec 3.7.14, Fuel Storage Pool Boron Concentration, the

MINIMUM required SFP boron concentration is __(2)__ ppm.

	(1)	(2)
A.	boration	2000
B.	boration	2200
C.✓	dilution	2000
D.	dilution	2200

Tech Specs 3.7.14
The fuel storage pool boron concentration shall be **≥ 2000 ppm.**
APPLICABILITY: When fuel assemblies are stored in the fuel storage pool.

FSD-A181009

5.30.5.1 **Flow measurement (FT-168) shall be provided downstream of the blender to indicate total makeup flow** to the charging header and as input to the Reactor Makeup Control System.

5.30.5.2 Upstream **boric acid flow measurement (FT-113) shall be provided to serve as input to the Reactor Makeup Control System.**

5.78.1.1 This differential pressure transmitter (and associated orifice flow element) shall provide measurement of the total makeup flow from the boric acid blender. **It shall also provide input to the Reactor Makeup Control System for regulation of RMW flow** and shall alert the operator of a deviation from the selected flow setpoint.

Distracter analysis

A. Incorrect. First part is incorrect (See C.1). Plausible if the candidate thinks that the lower signal results in less RMW added and therefore more acid for a given volume which would result in a boration of the SFP.

Second part is correct (See C.2).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See C.2). Plausible because the applicant could confuse the SFP minimum boron concentration with the accumulators minimum boron concentration which is 2200 ppm.

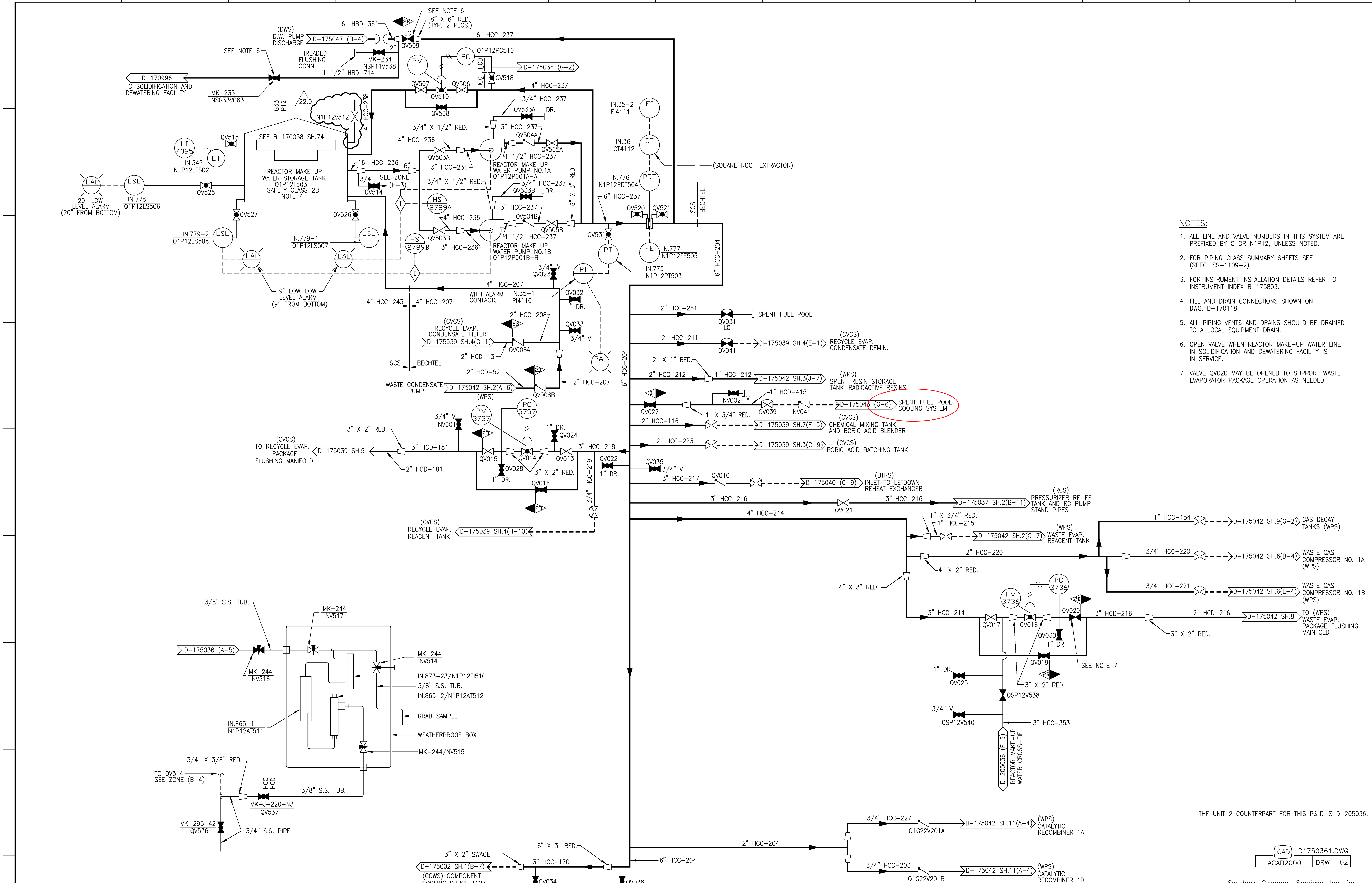
C. Correct. First part is correct. FT-168 will regulate total makeup flow to 120 gpm. This is a combination of acid flow and Reactor Makeup Water (RMW) flow. The amount of acid flow will be determined by the properly functioning FT-113. This means that the malfunctioning LOWER signal sent to FK-168 by FT-168 will cause the system to raise the flow of RMW to achieve a "sensed" total flow of 120 gpm resulting in more RMW than expected therefore a lower boron concentration in the makeup water supplied to the SFP. This will result in a dilution of the SFP.

Second part is correct. Tech Spec 3.7.14 requires the SFP boron concentration to be ≥ 2000 ppm.

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See B.2).

K/A: **033K4.05**	Spent Fuel Pool Cooling System (SFPCS) - Knowledge of design feature(s) and/or interlock(s) which provide for the following: Adequate SDM (boron concentration)
Importance Rating:	3.1 3.3
Technical Reference:	Unit 1 Technical Specifications, Ver 190 D-175043, SH1, Spent Fuel Pool Cooling, ver 27 D-175036, SH 1, Reactor Makeup Water, Ver 22
References provided:	None
Learning Objective:	RECALL AND APPLY the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Spent Fuel Pool Cooling and Purification and Refueling Water Storage Tank Purification Systems components and attendant equipment alignment, to include the following (OPS-52108L01): […] • 3.7.14, Fuel Storage Pool Boron Concentration RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Reactor Makeup Control and Chemical Addition System, to include the following (OPS-40301G02): […] • Inter connections with other systems • Primary Water Makeup Flow Controller, FK-168
Question History:	MOD NORTH ANNA 08
K/A match:	Requires the applicant to evaluate **knowledge of TS requirement for minimum boron concentration, which provides for adequate SDM.** Also evaluates candidates ability to **predict effect of an equipment malfunction which could adversely affect ability to maintain desired boron concentration.**
SRO justification:	N/A



NOTES:

1. ALL LINE AND VALVE NUMBERS IN THIS SYSTEM ARE PREFIXED BY Q OR N1P12, UNLESS NOTED.
2. FOR PIPING CLASS SUMMARY SHEETS SEE (SPEC. SS-1109-2).
3. FOR INSTRUMENT INSTALLATION DETAILS REFER TO INSTRUMENT INDEX B-175803.
4. FILL AND DRAIN CONNECTIONS SHOWN ON DWG. D-170118.
5. ALL PIPING VENTS AND DRAINS SHOULD BE DRAINED TO A LOCAL EQUIPMENT DRAIN.
6. OPEN VALVE WHEN REACTOR MAKE-UP WATER LINE IN SOLIDIFICATION AND DEWATERING FACILITY IS IN SERVICE.
7. VALVE QV020 MAY BE OPENED TO SUPPORT WASTE EVAPORATOR PACKAGE OPERATION AS NEEDED.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205036.

CAD D1750361.DWG
ACAD2000 DRW- 02

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT: J. M. FARLEY NUCLEAR PLANT - UNIT 1

SUBJECT: P&ID-REACTOR MAKE UP WATER SYSTEM

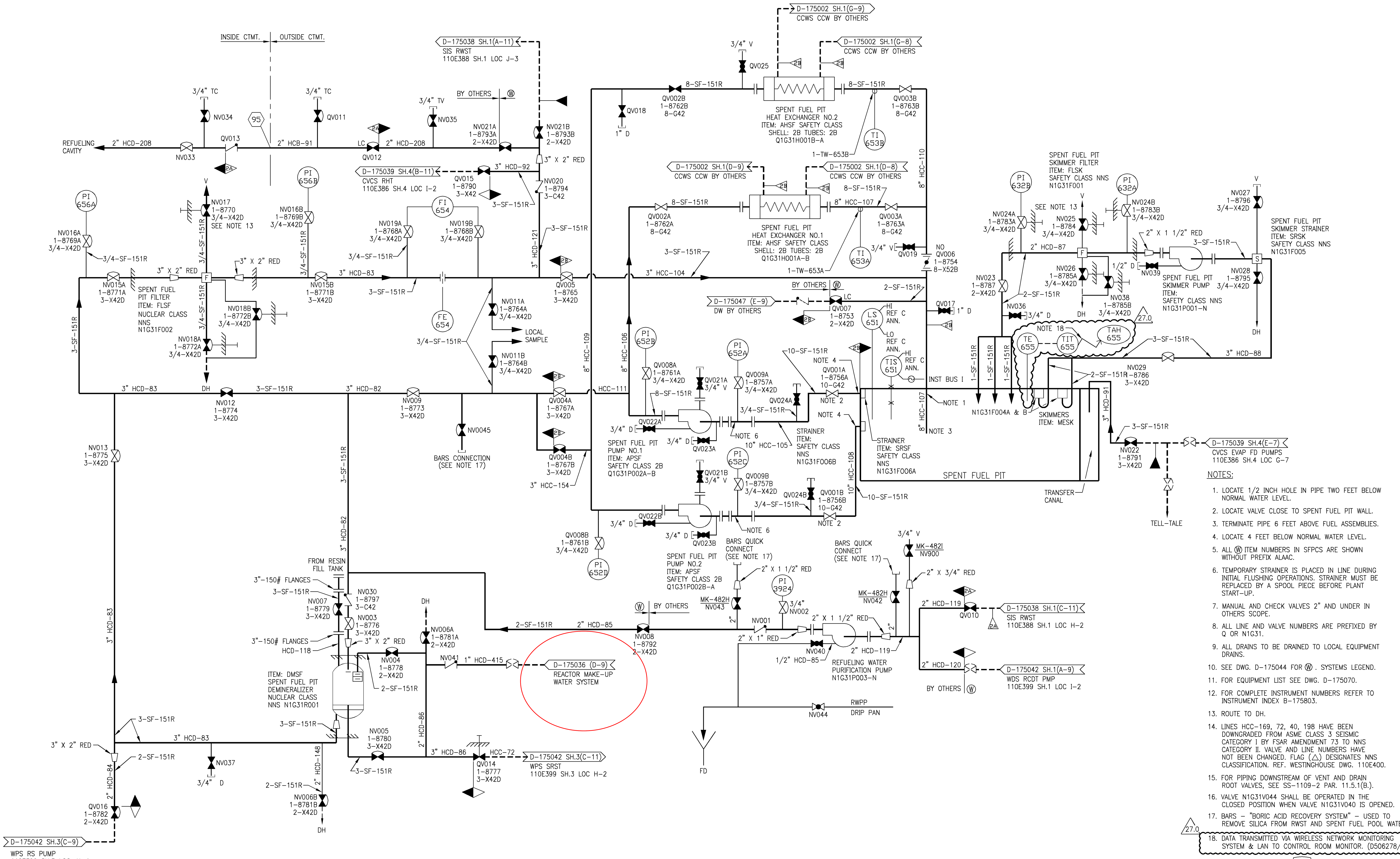
FOR SIGNATURES
SEE MICROFILM
COPY

DRW LCF AAN
VER. NO. 22.0 DATE 07/25/11
REVISED PER ABN-F02327, VER. 1.0

SCALE NONE B/W N/A
SHEET 1 OF — SHEETS

D-175036

DRAWING CATEGORY: CRITICAL



NOTES:

1. LOCATE 1/2 INCH HOLE IN PIPE TWO FEET BELOW NORMAL WATER LEVEL.
2. LOCATE VALVE CLOSE TO SPENT FUEL PIT WALL.
3. TERMINATE PIPE 6 FEET ABOVE SPENT FUEL ASSEMBLIES.
4. LOCATE 4 FEET BELOW NORMAL WATER LEVEL.
5. ALL Ⓦ ITEM NUMBERS IN SFPCS ARE SHOWN WITHOUT PREFIX ALAAC.
6. TEMPORARY STRAINER IS PLACED IN LINE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REPLACED BY A SPOOL PIECE BEFORE PLANT START-UP.
7. MANUAL AND CHECK VALVES 2" AND UNDER IN OTHERS SCOPE.
8. ALL LINE AND VALVE NUMBERS ARE PREFIXED BY Q OR N1G31.
9. ALL DRAINS TO BE DRAINED TO LOCAL EQUIPMENT DRAINS.
10. SEE DWG. D-175044 FOR Ⓦ. SYSTEMS LEGEND.
11. FOR EQUIPMENT LIST SEE DWG. D-175070.
12. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175803.
13. ROUTE TO DH.
14. LINES HCC-169, 72, 40, 198 HAVE BEEN DOWNGRADED FROM ASME CLASS 3 SEISMIC CATEGORY I BY FSAR AMENDMENT 73 TO NNS CATEGORY II. VALVE AND LINE NUMBERS HAVE NOT BEEN CHANGED. FLAG (△) DESIGNATES NNS CLASSIFICATION. REF. WESTINGHOUSE DWG. 110E400.
15. FOR PIPING DOWNSTREAM OF VENT AND DRAIN ROOT VALVES, SEE SS-1109-2 PAR. 11.5.1(B.).
16. VALVE N1G31V044 SHALL BE OPERATED IN THE CLOSED POSITION WHEN VALVE N1G31V040 IS OPENED.
17. BARS - "BORIC ACID RECOVERY SYSTEM" - USED TO REMOVE SILICA FROM RWST AND SPENT FUEL POOL WATER.
18. DATA TRANSMITTED VIA WIRELESS NETWORK MONITORING SYSTEM & LAN TO CONTROL ROOM MONITOR. (D506278/2) 27.0

CAD D1750431
ACAD2K DCP- 03

Southern Company Services, Inc. for
ALABAMA POWER COMPANY
PROJECT: J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT: P&ID - SPENT FUEL POOL COOLING SYSTEM

DCP GDM JMR
VER. NO. 27.0 DATE 03/26/12
REVISED PER ABN 1092885401J008, VER.1.0

DRAWN — CHECKED — DESIGNED
APPROVED FOR SIGNATURES DATE
APPROVED SEE MICROFILM DATE
APPROVED COPY DATE

SCALE NONE
SHEET 1 OF — SHEETS
SUPERSEDES
D-175043

DRAWING CATEGORY: CRITICAL

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

--NOTE---
In MODE 3, with RCS pressure > 1000 psig, the accumulators may be inoperable for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7555 gallons (31.4%) and ≤ 7780 gallons (58.4%).	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 601 psig and ≤ 649 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2200 ppm and ≤ 2500 ppm.	In accordance with the Surveillance Frequency Control Program AND -------NOTE------- Only required to be performed for affected accumulators ---------------------- Once within 6 hours after each solution volume increase of ≥ 12% level, indicated, that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ 2000 psig.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Fuel Storage Pool Boron Concentration

LCO 3.7.14 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	-------------------NOTE-------------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the fuel storage pool boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

1. 033 K4.05 031/NEW//HIGHER//RO/NORTH ANNA/6/2008/NO

- Operators are performing a blended make-up to the Spent Fuel Pool (SFP) following completion of cask loading operations.
- A calibration error results in 1-CH-FT-1114, PG Water to blender flow transmitter indicating less than the actual flowrate.

Which ONE of the following predicts the effect on boron concentration of the blended flow based on these plant conditions, and identifies the Technical Specification minimum required boron concentration for the SFP?

A. Boron concentration of the blended flow will be less than the calculated value; 2500 ppm.

B. Boron concentration of the blended flow will be greater than the calculated value; 2500 ppm.

C.✓ Boron concentration of the blended flow will be less than the calculated value; 2600 ppm.

D. Boron concentration of the blended flow will be greater than the calculated value; 2600 ppm.

1. 035A2.01 028/MOD/VOGTLE 12/C/A 4.5/4.6/035A2.01/N///

Unit 1 is operating at 100% power with the following conditions:

- Reactor power is now 100.5% and slowly rising.
- Tavg is 570.5°F and slowly lowering.
- Pressurizer pressure is 2210 psig and slowly lowering.
- Turbine load is 890 MWe and lowering.
- SG pressures are 720 psig and slowly lowering.
- Containment pressure is 2.1 psig and slowly rising.

Which one of the following completes the statements below?

The event in progress is a (1) line break.

Per AOP-14.0, Secondary System Leakage, the operators are required to (2).

	(1)	(2)
A.	steam	reduce turbine load
B.✓	steam	trip the reactor
C.	feed	reduce turbine load
D.	feed	trip the reactor

AOP-14:

1. [CA] Evaluate plant status for safe operation.
Pressurizer level
GREATER THAN 15%
AND
Pressurizer pressure
GREATER THAN 2000 psig
AND
Steam generator pressure
GREATER THAN 650 psig
AND
Containment pressure
LESS THAN 2 psig
AND
IF main generator on line,
THEN
(check reactor power) - (turbine power + any steam dump power)
mismatch LESS THAN 10%.
AND

1. Perform the following

1.1 **Verify reactor tripped**

1.2 IF reactor tripped,
THEN CLOSE SG
main steam isolation and
bypass valves

IF main generator off line,
THEN check reactor power less than ~ 15%

AOP-14, step 9 and note above step 9 says:
The intent of step 9 is to reduce reactor power to within the capacity of the AFW system if possible and step 9 has the crew reduce power per UOP-3.1 and UOP-2.1 if the above trip criteria is not met.

Distracter analysis

A. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2). Plausible since UOP-3.1, Power Operation, requires a reduction in turbine load if 100% power is exceeded. This would be the correct thing to do if containment pressure did not meet the reactor trip criteria. Also Step 7 and 9 and note above step 9 addresses ramping the unit down to mode 2 if the trip criteria is not exceeded in the previous steps.

B. Correct. First part is correct. All the conditions in the stem - Tavg lowering, RCS Pressure lowering and MWe lowering **are indicative of a steam break** where the steam is exiting the piping before reaching the turbine. The containment parameters show that the break is in containment.

Second part is correct. With containment pressure greater than 2 psig, reactor trip criteria is met.

C. Incorrect. First part is incorrect On a feedline break, Rx power would be stable, turbine MWe would be stable, and RCS pressure would be stable. SG pressure would not lower and Tavg would be rising. Plausible if the applicant misdiagnoses the event. AOP-14 addresses a steam or feed break and they have similar characteristics.

Second part is incorrect (See A.1).

D. Incorrect. First part is incorrect (See C.1).

Second part is correct (See B.2).

K/A: **035A2.01**	Steam Generator System (S/GS) - Ability to (a) predict the impacts of **Faulted or ruptured S/Gs** on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations
Importance Rating:	4.5 4.6
Technical Reference:	FNP-1-AOP-14.0, Secondary System Leakage. Ver 10.0
References provided:	None
Learning Objective:	STATE AND EXPLAIN the operational implications for all Cautions, Notes, and Actions associated with AOP-14, Secondary System Leakage. (OPS-52521O03)
Question History:	MOD VOTGLE 12
K/A match:	Applicant must **predict that a steam break has occurred based on the impact to plant parameters (which is a backward logic way to meet the first part of the KA). The parameters include but are not limited to SGs since the trip criteria in AOP-14 is due to ctmt pressure.** Applicant must **determine the proper procedural response to mitigate a faulted SG inside ctmt.**
SRO justification:	N/A

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-14.0

S
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D

SECONDARY SYSTEM LEAKAGE

PROCEDURE USAGE REQUIREMENTS PER FNP-0-AP-6	SECTIONS
Continuous Use	ALL
Reference Use	
Information Use	

Approved:

David L Reed (for)
Operations Manager

Date Issued: April 22, 2011

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PROCEDURE CONTAINS NUMBER OF PAGES

A. **Purpose**

This procedure provides actions for response to a secondary system leak.

This procedure is applicable in Modes 1, 2 and 3.

B. **Symptoms or Entry Conditions**

This procedure is entered when excessive secondary leakage is indicated by any of the following:

a. Unexplained reduction in Tavg with Tavg below Tref.

b. Unexplained reduction in RCS/PRZR pressure and/or level.

c. Unexplained rise in steam flow.

d. Unexplained rise in feedwater flow.

e. Unexplained reduction in steam generator level and/or pressure.

f. Report of excessive secondary/steam leakage.

g. Unexplained rise in containment temperature, humidity, and/or pressure without RMS indication.

h. Steam generator atmospheric relief valve and/or main steam safety valve stuck open.

i. Steam dump demand less than expected for existing reactor power.

Step **Action/Expected Response** **Response Not Obtained**

NOTE:
- If the secondary leakage is upstream of the MSIVs, closing the MSIVs in step 1 may result in a safety injection.
- Step 1 assumes attempts have been made to stabilize plant status in accordance with normal operating procedure(s) in effect.
- 1% steam dump demand is approximately equal to 0.5% reactor power This reflects the fact that the steam dumps will handle greater than design flow.
- **[CA]** is a continuing action step.

__ 1 **[CA] Evaluate plant status for safe operation.**

- Pressurizer level GREATER THAN 15%

AND

- Pressurizer pressure GREATER THAN 2000 psig

AND

- Steam generator pressure GREATER THAN 650 psig

AND

- Containment pressure LESS THAN 2 psig

AND

- IF main generator on line, THEN (check reactor power) - (turbine power + any steam dump power) mismatch LESS THAN 10%.

AND

- IF main generator off line, THEN check reactor power less than ~ 15%.

1 Perform the following.

1.1 Verify reactor tripped.

1.2 IF reactor tripped,
THEN CLOSE SG main steam isolation and bypass valves.

SG	1A	1B	1C
1A(1B,1C) SG MSIV - TRIP Q1N11HV	[]3369A []3370A	[]3369B []3370B	[]3369C []3370C
1A(1B,1C) SG MSIV - BYPASS Q1N11HV	[]3368A []3976A	[]3368B []3976B	[]3368C []3976C

1.3 IF SG main steam isolation and bypass valves did not close, ,
THEN place associated test switch to TEST position.

Affected SG	1A	1B	1C
1A(1B,1C) SG MSIV - TEST Q1N11HV	[]3369A/ 70A	[]3369B/ 70B	[]3369C/ 70C

° Step 1 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained

NOTE: The following step is a continuing action step.

__ 7 **[CA] Evaluate continued plant operations.**

- Pressurizer level stable at normal programmed value.
- Tavg maintained within ± 1.5°F of expected Tavg for plant conditions.
- RCS pressure 2220-2250 psig.
- SG level stable at normal value of 65%.
- CST level stable.
- IF primary to secondary leakage exists, THEN, secondary release posing NO radiological threat.
- Secondary leakage posing NO hazard to vital equipment or areas.
- Secondary leakage posing NO hazard to electrical equipment.
- Secondary leakage posing NO hazard to personnel.

7 Shutdown the reactor, proceed to step 9.

__ 8 **Check plant shutdown required, (Consult with Operations Manager - conditions permitting)**

8 IF continued plant operation required, THEN return to step 7.

°

__Page Completed

Step | **Action/Expected Response** | **Response Not Obtained**

NOTE: The intent of step 9 is to reduce reactor power to within the capacity of the AFW system if possible.

__ **9** **Reduce Unit load to point of transfer to auxiliary feedwater using FNP-1-UOP-3.1, POWER OPERATION, and FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.**

__ **10** **Check reactor power below capability of auxiliary feedwater (~ 5%).**

10 IF secondary leak upstream of MSIV's OR downstream of main feed stop valves identified,
THEN perform the following,
OTHERWISE proceed to step 11 observe caution prior to step 11.

10.1 Trip the reactor.

10.2 IF reactor tripped,
THEN CLOSE SG main steam isolation and bypass valves.

SG	1A	1B	1C
1A(1B,1C) SG MSIV - TRIP Q1N11HV	[]3369A []3370A	[]3369B []3370B	[]3369C []3370C
1A(1B,1C) SG MSIV - BYPASS Q1N11HV	[]3368A []3976A	[]3368B []3976B	[]3368C []3976C

° Step 10 continued on next page

__Page Completed

NOTES

During steady state power operations the reactor core power level is limited to 2775 MW thermal. (NRC Regulatory Issue Summary 2007-21)

Minor fluctuations of turbine/reactor power are expected due to grid fluctuations and normal tolerances in control systems. Unplanned increases in reactor power due to positive reactivity addition shall be promptly terminated with turbine load reduction, control rod insertion and/or boration. Reactor power shall be returned to at or below the previous steady state power level.

Per the DEH Valve Management Program; a DEH Demand of 947.1 MW will result in all Governor Valves 100% open and Flow Demand equal to 100%.

The return to full power operation should be approached conservatively. Consider using very small ramp rate from 99% to 100% power. (e.g. 0.1 MW/minute) (CR 2010107950)

5.23 **Raise** power to ≤ 100%. ☐

5.23.1 **Ensure** TAVG does NOT exceed 577.2°F.

5.23.2 **Take** prompt action to compensate for excessive positive reactivity additions. ☐

5.23.3 WHEN available, the 15 minute average core thermal power (QC4621M15) should be monitored and maintained below 2775 MWth. ☐

5.23.4 If the 15 minute average value is exceeded, the OATC must reduce power to ensure the 1 hour average power remains below the limit of 2775 MWth. ☐

5.23.5 IF the average power level over any eight hour shift exceeds the full, steady-state licensed power level, THEN **initiate** a condition report to document the event and prompt LER determination. ☐

5.23.6 In no case should 102% power be exceeded. ☐

5.24 **Check** the following feedwater heater bypass valves for proper seating:

- 1A & 2A FW HTRS MANUAL BYPASS, N1N21V519A ☐
- 1B & 2B FW HTR MANUAL BYPASS, N1N21V519B ☐
- 3A & 4A FW HTRS MANUAL BYPASS, N1N21V522A ☐
- 3B& 4B FW HTRS MANUAL BYPASS, N1N21V522B ☐
- 5A FW HTR CNDS BYPASS, N1N21V529A ☐
- 5B FW HTR CNDS BYPASS, N1N21V529B ☐
- 6A & 6B FW HTRS BYPASS, N1N21V510 ☐

1. 040AA1.20 001/1/1/SLB-CNMT PRESS/TEMP/H-4.1/4.2/NEW/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

- Unit 1 is at 100% power.

Current conditions:

- Reactor power is 100.4% and slowly rising.
- RCS pressure is 2212 psig and slowly lowering.
- Turbine load is 1200 MWe and lowering.
- Containment pressure is 1.4 psig and rising.
- Containment temperature is 117.5ºF and rising.

Which one of the following correctly completes the following statement?

A ___(1)___ break is the event in progress

and

per 18008-C, "Secondary Coolant Leakage", the FIRST action the operators will perform is to ___(2)___ .

	__(1)__	__(2)__
A.✔	steamline	reduce turbine load
B.	steamline	manually insert rods
C.	feedline	reduce turbine load
D.	feedline	manually insert rods

QUESTIONS REPORT
for ILT 36 RO NRC Exam Version 6

1. 036AG2.1.7 029/MOD/FNP 05/C/A 4.4/4.7/APE036G2.1.7/N///

Unit 1 is operating at 100%. A fuel shuffle is being performed in the Unit 1 SFP.

At 1000:

- The following alarm is received:
  – EH2, SFP LVL HI-LO

At 1015:
The SRO in charge of refueling reports that a fuel assembly has been dropped.

- The following alarm is received:
  – FH5, SFP AREA RE-25 A OR B HI RAD
- The Unit Operator reports the following conditions:
  – R-25A & B, SPENT FUEL BLDG EXH, reads off scale high.

Which one of the following completes the statement below?

The operating crew is required to enter __(1)__ .

The crew is required to dispatch personnel to __(2)__ per the applicable AOP.

Procedures:
- AOP-30.0, Refueling Accident
- AOP-49.3, Spent Fuel Pool Emergency

	(1)	(2)
A.	AOP-49.3	make up to the SFP using the RWST
B.	AOP-30.0	make up to the SFP using the RWST
C.	AOP-49.3	ensure all SFP hatches and doors are closed
D✓	AOP-30.0	ensure all SFP hatches and doors are closed

AOP-30 Symptoms or entry conditions
1. This procedure is entered when a fuel handling accident causes damage to a fuel assembly in conjunction with a high radiation indication on any of the following:
[] R-2 CTMT 155 ft
[] R-5 SFP ROOM
[] R-24A(B) CTMT PURGE
[]R-25A(B) SPENT FUEL BLDG EXH

Step 1.6. Dispatch personnel to close all spent fuel area fuel handling hatches.

Step 5. Verify all access doors to accident area - CLOSED

Step 20 has the operator makeup to the refueling cavity from the RHR system if the cavity is low.

AOP-49.3 B. Symptoms or entry conditions
1. A report of damage to and/or leakage from the SPENT FUEL POOL caused by an external threat is received.

2. Any condition outside the design basis of the plant that will result in a long term loss of Spent Fuel Pool cooling.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible based on the name of the procedures. The applicant may believe that since there is no refueling occurring (Unit 1 at 100%) , AOP-30 does not apply and AOP-49.3 applies since there is "an emergency" in the SFP.

Second part is incorrect (See D.2). Plausible since this is an action of AOP-49.3 to keep all assemblies covered with water and would seem logical since there was a SFP HI-LO level alarm and a damaged fuel assembly lying on the racks. The applicant may think that keeping the damaged assembly covered with water is a required action.

B. Incorrect. First part is correct (See D.1)

Second part is incorrect (See D.2). Plausible since AOP-30 directs filling the refueling cavity. The applicant could easily confuse this action with filling the SFP. It could seem logical since there was a SFP HI-LO level alarm and a damaged fuel assembly lying on the racks. The applicant may think that keeping the damaged assembly covered with water is a required action.

C. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2). This is a logical connection to AOP-49.3 since during a SFP Emergency, the applicant could assume the affected area would be isolated as radiation levels are high.

D. Correct. First part is correct. This scenario meets the entry requirements of AOP-30.0.

Second part is correct. This action is taken per step 1.6 and 5.

K/A: **036AG2.1.7**	Fuel Handling Incidents - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Importance Rating:	4.4 4.7
Technical Reference:	FNP-1-AOP-30.0, Refueling Accident, Ver 19
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if entry into AOP-30.0, Refueling Accident is required. (OPS-52521H02)
Question History:	MOD FNP 05
K/A match:	Requires the applicant to **interpret plant instrumentation (rad monitors and SFP alarm) and reports from the field and determine the applicable procedure to enter and the appropriate action to take (operational judgment).**
SRO justification:	N/A

A. Purpose

This procedure provides actions for response to fuel handling accident or a loss of refueling cavity water level.

This procedure is applicable at all times.

B. Symptoms or Entry Conditions

1. This procedure is entered when a fuel handling accident causes damage to a fuel assembly in conjunction with a high radiation indication on any of the following:

[ ] R-2 CTMT 155 ft
[ ] R-5 SFP ROOM
[ ] R-24A(B) CTMT PURGE
[ ] R-25A(B) SPENT FUEL BLDG EXH

2. This procedure is entered when a dry storage activity causes damage to a fuel assembly in conjunction with a high radiation indication on radiation monitor R-5(SFP ROOM).
3. This procedure is entered when rapidly falling refueling cavity level is observed.
4. This procedure may be entered at the discretion of the Shift Supervisor when any abnormal fuel handling incident occurs.

Step	Action/Expected Response	Response Not Obtained
1.6	Dispatch personnel to close all spent fuel area fuel handling hatches.	
1.7	IF required, THEN have Maintenance personnel close the containment equipment hatch.	
__ **2**	**[CA] While continuing with the remainder of this procedure, perform** ATTACHMENT 1, CONTROL ROOM ISOLATION WITH CREFS IN SERVICE **to ensure control room isolation with CREFS placed in service.**	

****

CAUTION: Dangerously high dose rates will result from uncovering reactor internals.

****

Step	Action/Expected Response	Response Not Obtained
__ **3**	**IF reactor internals are being moved, THEN place them in a safe location.**	

__Page Completed

Step	Action/Expected Response	Response Not Obtained

****

CAUTION: Rapidly falling refueling cavity level will affect the containment and the spent fuel room.

****

__ **4** **Evacuate all non-essential personnel from accident area.**

__ **5** **Verify all access doors to accident area - CLOSED.**

__ **6** **Evaluate actuating the plant emergency alarm.**

__ **7** **[CA] IF fuel damage has resulted in elevated containment radiation levels, THEN perform the following.**

7.1 Verify containment ventilation isolation.

7.1.1 Verify containment purge dampers - CLOSED

[ ] 3197
[ ] 3198D
[ ] 3198C
[ ] 3196
[ ] 3198A
[ ] 3198B

° Step 7 continued on next page

__Page Completed

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-49.3

SPENT FUEL POOL EMERGENCY

S
A
F
E
T
Y

R
E
L
A
T
E
D

PROCEDURE USAGE REQUIREMENTS-per NMP-AP-003	SECTIONS
Continuous Use	
Reference Use	**ALL**
Information Use	

Approved:

____
Operations Manager

Date Issued:______________

TABLE OF CONTENTS

A. <u>Purpose</u>

Purpose of the procedure is to provide methods for stopping a spent fuel pool leak and for emergency make up to the spent fuel pool due to damage caused by an external threat gaining access to the spent fuel pool. It assumes the external threat has been eliminated and access to the areas outside the spent fuel pool are accessible. Additionally, sections of this procedure can be used for emergency make up due to prolong LOSP conditions or other condition that take the unit outside its design basis for spent fuel pool cooling or make up. Condition beyond the design basis, are not governed by the requirements of 10CFR50.59. Therefore, 10CFR50.54(x) must be invoked to allow performance of some sections of this guideline. <u>These steps are not required to be completed exactly as written and may be deviated from with appropriate management or supervisory approval based on exact conditions at the time of the event</u>. It is intended that this procedure be used in conjunction with other abnormal or emergency operating procedures.

Any revisions to this procedure concerning B.5.b strategies that are to be implemented require validation. Validation means measures have been implemented that verify the strategies can be executed. These measures include drills, exercises, walk throughs, table top exercises, or classroom discussions; so that stategies remain feasible.

An engineering review has been performed and it is not credible event to drain the spent fuel pool below the top of the fuel assemblies due to the construction of the pool which is bounded by earth on three sides and the new fuel pit on the other side.

B. <u>Symptoms or Entry Conditions</u>

1 **A report of damage to and/or leakage from the SPENT FUEL POOL caused by an external threat is received.**

2 **Any condition outside the design basis of the plant that will result in a long term loss of Spent Fuel Pool cooling.**

C. <u>Procedure Steps</u>

NOTE:
- The following steps provide guidance for temporarily stopping a spent fuel pool leak. Other methods or ideas may be substituted if they are more expedient or provide more reliability.
- Continuous Health Physics coverage will be required due to the possibility of excess contamination and higher than normal radiation levels.

1 TEMPORARY REPAIRS TO SPENT FUEL POOL

1.1 USE OF THE STAINLESS STEEL FME BARRIER

1.1.1 **Obtain** the section of the stainless steel barrier that has been staged in the SFP Room for a potential leak barrier.

NOTE:
- Air operated vice grips are located in the Engineering Support gang box located outside the Unit 1 spent fuel pool.
- Use of the air operated vice grips will require a Nitrogen cylinder.

1.1.2 **Attach** air operated vice grips to the stainless steel FME barrier.

1.1.3 **Lower** the stainless steel sheet into the spent fuel pool to cover the breach.

1.2 USE OF SHEET METAL <u>OR</u> OTHER SIMILAR MATERIAL

1.2.1 **Have** Maintenance deliver sheet metal or other sheet type material to the spent fuel pool.

1.2.2 **Lower** the material in place as in step 1.1.

1.3 BREACH OF THE TRANSFER CANAL.

1.3.1 <u>IF</u> the weir gate is not already in place,
<u>THEN</u> **replace** the weir gate.

NOTE:

- The following steps provide guidance for various methods of make up to the the Spent Fuel Pool. It will be up to the Shift Manager to determine the most expedient and available method to utilize based on current plant conditions.
- Lock valves are designated with the appropriate key numbers for the install locks, however master keys may be used in place of the designated keys.

2 MAKE UP TO THE SPENT FUEL POOL

TABLE 1	
Step 3.0	MAKE UP FROM RWST
Step 4.0	FILLING THE SFP TRANSFER CANAL FROM THE RHTS
Step 5.0	MAKE UP FROM DEMINERALIZED WATER SYSTEM
Step 6.0	MAKE UP TO SFP THROUGH TEMPORARY CONNECTIONS
Step 7.0	MAKE UP FROM REACTOR MAKE UP WATER SYSTEM
Step 8.0	ALT MAKE UP FROM DEMINERALIZED WATER SYSTEM
Step 9.0	ALT MAKE UP FROM FIRE PROTECTION SYSTEM
Step 10.0	ALT FIREMAIN SUPPLY FROM UNIT 1 CIRC WATER CANAL
Step 11.0	ALT FIREMAIN SUPPLY FROM UNIT 2 CIRC WATER CANAL

2.1 Use the methods listed in the Table 1 above when installed plant piping is available.

QUESTIONS REPORT
for 036AG2.1.7 FNP 05

1. AOP-30.0-52521H04 006/HLT//C/A (LEVEL 2/3) PROC/APE036AK1.01////

Both units are in MODE 1 and Unit 1 is conducting a fuel shuffle in the SFP when the following occurs:

- EH2, SFP LVL HI-LO, annunciator comes into alarm.
- R-5, SFP Area Radiation Monitor, comes into alarm and reads 10 R/HR.
- FH5, SFP AREA RE-25 A OR B HI RAD, annunciator comes into alarm.
- R-25A & B, SFP Ventilation Radiation Monitor, reads off scale high.

All automatic actions have taken place as expected. AOP-30, REFUELING ACCIDENT, is entered. Which one of the following is the correct action for the conditions stated?

A. Immediately commence filling the SFP and monitor the SFP level.

B. Have HP verify the R-5 alarm is valid and the extent of damage to the fuel.

C. Place a second control room air conditioning unit on service and place the control room exhaust fans in service.

D✓ Dispatch personnel to close ALL SFP hatches and doors and evacuate the SFP area.

1. 037AK1.01 030/MOD/CATAWBA 09/C/A 2.9/3.3/APE037AK1.01/Y///

Unit 1 is performing the actions of AOP-2.0, Steam Generator Tube Leakage, due to a tube leak on the 1A SG. The following conditions exist:

- RCS pressure is currently being reduced to minimize break flow.

The following parameters are observed:

- SG pressures are:

1A SG	1B SG	1C SG
885 psig	965 psig	960 psig

  – RCS pressure is 947 psig.
  – The highest reading non-upperhead CETC is 518°F.
  – PRZR level is 43%.

- BOTH Subcooled Margin Monitors are malfunctioning.

Which one of the following completes the statements below?

The current value of subcooling is approximately _(1)_.

The RCS pressure reduction _(2)_ required to be stopped.

Reference Provided

	(1)	(2)
A.	14°F	IS
B.	14°F	is NOT
C.	22°F	IS
D.✓	22°F	is NOT

AOP-2:
Step 34.2
[CA] WHEN one of the following conditions occur, THEN stop the RCS pressure reduction.
[] RCS pressure is less than affected SG pressure, AND pressurizer level greater than 15%.
OR
[] Pressurizer level greater than 63%.
OR
[]SUBCOOLED MARGIN MONITOR indication less than 16°F subcooled in CETC mode.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since the applicant may determine subcooling based on ruptured SG pressure vs. RCS pressure. This would be a common misconception since in a SGTR procedure the ruptured SG is the focus for determining SG pressure less than RCS pressure and is the pressure referred to when determining the required CETC temperature to cooldown to.

885 psig + 15 = 900 psia which is 532°F
532°F - 518°F = 14°F subcooling.

Second part is incorrect (See D.2). Plausible since subcooling in the first part is below the 16°F value if calculated incorrectly. This would make this a correct answer if subcooling was 14°F.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2). Plausible since even with the error in choosing the SG to calculate subcooling, if the applicant improperly believed that ALL 3 requirements must be met to stop the cooldown then this would be a correct answer.

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See D.1). Plausible if the applicant compared RCS pressure to the **intact** SG pressure they would improperly arrive at this as an answer.

D. Correct. First part is correct.
947 psig + 15 = 962 psia which is 540°F
540°F - 518°F = 22°F subcooling.

Second part is correct. Subcooling is greater than 16°F and the other pressure reduction termination criteria are not met.

K/A: **037AK1.01**	Steam Generator (S/G) Tube Leak - Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Use of steam tables
Importance Rating:	2.9* 3.3
Technical Reference:	FNP-1-AOP-2.0, Steam Generator Tube Leakage, Ver 35 Properties of saturated and superheated steam, 1967
References provided:	Steam tables and AOP-2.0 step 34.2 ver 35.0
Learning Objective:	ANALYZE plant conditions and DETERMINE the successful completion of any step in AOP-2.0, SG Tube Leakage. (OPS-52520B07)
Question History:	MOD CATAWBA 09
K/A match:	Applicant is required to **use the steam tables** to determine current value of subcooling and the **whether or not the RCS depressurization is required to be stopped during a SG tube leak scenario.**
SRO justification:	N/A

Step	Action/Expected Response	Response Not Obtained

THIS IS THE REFERENCE PROVIDED TO THE APPLICANTS

34.2 [CA] WHEN one of the following conditions occur,
THEN stop the RCS pressure reduction.

[ ] RCS pressure is less than affected SG pressure, AND pressurizer level greater than 15%.

OR

[ ] Pressurizer level greater than 63%.

OR

[ ] SUBCOOLED MARGIN MONITOR indication less than 16°F subcooled in CETC mode.

Step	Action/Expected Response	Response Not Obtained
34.2	[CA] WHEN one of the following conditions occur, THEN stop the RCS pressure reduction. [ ] RCS pressure is less than affected SG pressure, AND pressurizer level greater than 15%. OR [ ] Pressurizer level greater than 63%. OR [ ] SUBCOOLED MARGIN MONITOR indication less than 16°F subcooled in CETC mode.	
34.3	Verify both normal pressurizer spray valves - CLOSED. 1A(1B) LOOP [ ] PK 444C [ ] PK 444D	34.3 Perform the following. 34.3.1 Deenergize rod control system. [ ] Open both RX TRIP BKRs. OR [ ] Open 1A and 1B MG SET SUPP BKRs. 34.3.2 Stop associated 1A & 1B RCPs. 34.3.3 IF any rod bottom light not lit, THEN emergency borate RCS using FNP-1-AOP-27.0, EMERGENCY BORATION.
34.4	Verify auxiliary spray valve - CLOSED. RCS PRZR AUX SPRAY [ ] Q1E21HV8145	34.4 Isolate auxiliary spray line. CHG PMPS TO REGENERATIVE HX [ ] Q1E21MOV8107 closed [ ] Q1E21MOV8108 closed
34.5	Verify both PRZR PORVs - CLOSED.	34.5 Close PRZR PORV ISO for any open PRZR PORV.

Step 38 continued on next page

Question: 21
(1 point)

Unit 1 was operating at 50% power. Given the following:

- A steam generator tube leak occurred on 1A S/G
- The crew depressurized the NC system per AP/1/A/5500/010 (Reactor Coolant Leak), Case I (Steam Generator Tube Leak)
- Both ICC monitors are unavailable
- Current NC pressure is 665 psig
- Core exit thermocouple temperatures are 488°F
- T-Colds are 487.7°F

1. Based on current conditions, what subcooling value would the ICC monitors be reading if they were available?
2. Based on current conditions, what is steam header pressure?

<u>Reference provided</u>

A. 1. -8 ° F
   2. 608 psig

B. 1. -8 °F
   2. 593 psig

C. 1. +12° F
   2. 608 psig

D. 1. +12° F
   2. 593 psig

---

1. 038EK3.02 031/NEW//C/A 4.4/4.5/EPE038EK3.02/N///

Unit 1 has experienced a tube rupture on the 1C SG.

The operating crew is at the step in EEP-3.0, Steam Generator Tube Rupture, to "Check SI termination criteria."

- The following plant conditions are observed:
  - RCS Subcooling is 22°F and slowly rising.
  - RCS pressure is 950 psig and slowly rising.
  - Pressurizer level is 45% and slowly rising.
  - AFW flow is 450 gpm.
  - 1A SG NR level is 29% and slowly rising.
  - 1B SG NR level is 26% and slowly rising.
  - 1C SG NR level is 95% and rising rapidly.

Which one of the following completes the statements below?

SI termination criteria __(1)__ been met.

Per EEP-3.0, SI termination is necessary to prevent overfilling the __(2)__.

	(1)	(2)
A.	has NOT	Steam Generator
B.	has NOT	Pressurizer
C.✓	HAS	Steam Generator
D.	HAS	Pressurizer

EEP-3
20 [CA] Check SI termination criteria.

20.1 Check SUBCOOLED MARGIN MONITOR indication - GREATER THAN 16°F{45°F} SUBCOOLED IN CETC MODE.

20.2 Check secondary heat sink available.

Total feed flow to SGs - GREATER THAN 395 gpm AVAILABLE.

Narrow range level in at least one intact SG - GREATER THAN 31%{48%}.

20.3 Check RCS pressure - STABLE OR RISING.

20.4 Check pressurizer level - GREATER THAN 13%{43%}.

EEB-3

Step 20 Basis: [...] **If SI flow is not terminated, leakage into the secondary will eventually fill the steam generator with water and lift the atmospheric relief valves.** This could damage the relief valve and main steamline which would complicate subsequent recovery and aggravate the radiological consequences. Hence, SI must be terminated when the criteria in subsequent steps are satisfied to prevent steam generator overfill

Distracter analysis

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant believes that due to the SI, adverse numbers are applicable. If they were applicable, then this would be a correct answer due to subcooling.

Second part is correct (See C.1)

B. Incorrect. First part is incorrect. (See A.1).

Second part is incorrect (See A.2). Plausible since this is the reason to terminate SI in EEP-0 and ESP-1.1 for a spurious SI. The applicant could confuse the basis for these procedures with the basis for the step in EEP-3.

C. Correct. First part in correct. SI termination criteria has been met.

Second part is correct. EEP-3 background document - **If SI flow is not terminated, leakage into the secondary will eventually fill the steam generator with water and lift the atmospheric relief valves.** This could damage the relief valve and main steamline which would complicate subsequent recovery and aggravate the radiological consequences**.** Hence, SI must be terminated when the criteria in subsequent steps are satisfied to prevent steam generator overfill

D. Incorrect. First part in correct (See C.2).

Second part is incorrect (See B.2)

This question was written with these values for the following reasons:
RCS subcooling is low but above the 16°F{45°F}. If adverse numbers were used it makes plausibility greater for this parameter.
Przr level is about where you would expect it after cooldown and depress and still above both parameters. 13%{43%}. and to meet plausibility for KA.

SG NR is below value of **Narrow range level in at least one intact SG - GREATER THAN 31%{48%}. but AFW flow is > 395 gpm. one does not meet si termination and one does, ans SGWL for 1C SG is so high to meet plausibility for KA.**

K/A: **038EK3.02**	Steam Generator Tube Rupture (SGTR) - Knowledge of the reasons for the following responses as the apply to the SGTR: Prevention of secondary PORV cycling
Importance Rating:	4.4 4.5
Technical Reference:	FNP-1-EEP-3.0, Steam Generator Tube Rupture, Ver 27 FNP-0-EEB-3.0, Specific Background Document For FNP-1/2-EEP-3.0, Ver 2
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with EEP-3, Steam Generator Tube Rupture. (OPS-52530D03)
Question History:	NEW
K/A match:	Requires the applicant to know that **preventing the SG PORVs from lifting due to filling the SGs solid will prevent a radiological release from the atmospherics.**
SRO justification:	N/A

Section: **Procedure**

Unit 1 ERP Step: **Unit 2 ERP Step:** ***ERG Step No:*** *5*

ERP StepText: Step moved to Attachment 2.

ERG StepText: *Verify FW Isolation*

Purpose:

Basis:

Knowledge:

References: DW-96-038

Justification of Differences:

1 Attachment 2 has been provided to expedite the performance of EEP-0 verification steps to reduce the chance of over-filling the pressurizer due to a spurious SI. Providing this information in a separate attachment improves procedure efficiency by assigning dedicated personnel for the task while minimizing the impact on the flowpath of the procedure. Reference DW-96-038.

Section: **Procedure**

Unit 1 ERP Step: 20 CAUTION-1 **Unit 2 ERP Step:** 20 CAUTION-1 ***ERG Step No:*** *19 CAUTION-1*

ERP StepText: To prevent overfilling ruptured SG(s), SI must be terminated as soon as possible after SI termination criteria are satisfied.

ERG StepText: SI MUST BE TERMINATED when termination criteria are satisfied to prevent overfilling the ruptured SG(s).

Purpose: To alert the operator that primary-to-secondary leakage will continue until SI flow is terminated.

Basis: As previously demonstrated (see Step 16), SI termination is necessary to control reactor coolant inventory and stop primary-to-secondary leakage. If SI flow is not terminated, leakage into the secondary will eventually fill the steam generator with water and lift the atmospheric relief valves. This could damage the relief valve and main steamline which would complicate subsequent recovery and aggravate the radiological consequences. Hence, SI must be terminated when the criteria in subsequent steps are satisfied to prevent steam generator overfill.

Knowledge: Voiding in the upper head region should not preclude SI termination. Such a void is expected to be confined to the upper head region and will not expand below the top of the hot legs if RCS subcooling is maintained at the core exit. Although an upper head void may hinder pressurizer pressure and level control, it is not a sufficient safety concern to prevent SI termination if the specified criteria are met.

References:

Justification of Differences:

1. Wording enhanced without changing intent.

Section: **Procedure**

Unit 1 ERP Step: 2 **Unit 2 ERP Step:** 2 ***ERG Step No:*** *4*

ERP StepText: Stop all but one CHG PUMP.

ERG StepText: *Stop All But One Charging/SI Pump And Place In Standby*

Purpose: To reduce flow into the RCS from the charging/SI pumps

Basis: Satisfaction of conditions for SI flow reduction implies that control can be maintained by the operator without all of the charging/SI pumps running. In this step, all but one charging/SI pump are stopped and placed in standby for future use. It should be noted that sequences to stop SI pumps other than the one outlined in this guideline are permissible as long as control of the plant can be maintained and there are no additional problems associated with the sequence. For example, it is permissible to align the flow path through the charging line to control flow with two charging/SI pumps running only if 1) the charging line can handle full flow from both pumps, 2) there are no runout problems, and 3) the flow lost by the realignment of both pumps to the charging line is not excessively larger than stopping one pump with flow through the BIT.

Knowledge: N/A

References: DW-96-038

Justification of Differences:

1 Changed to make plant specific. FNP design uses dual purpose Charging/HHSI pumps.

2 Step sequence and actions were modified to enhance ability to stop SI flow when termination conditions are met in order to minimize potential for a sold pressurizer. Some actions regarding this event are now accomplished in EEP-0. Relocated phase A and instrument air steps to later in the procedure since these are not as time sensitive (reference DW-96-038).

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: To prevent overfilling ruptured SG(s), SI must be terminated as soon as possible after SI termination criteria are satisfied.

****

___20	**[CA] Check SI termination criteria.**	
20.1	Check SUBCOOLED MARGIN MONITOR indication - GREATER THAN 16°F{45°F} SUBCOOLED IN CETC MODE.	20.1 Go to FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.
20.2	Check secondary heat sink available. • Total feed flow to SGs - GREATER THAN 395 gpm AVAILABLE. OR • Narrow range level in at least one intact SG - GREATER THAN 31%{48%}.	20.2 Go to FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.
20.3	Check RCS pressure - STABLE OR RISING.	20.3 Go to FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.
20.4	Check pressurizer level - GREATER THAN 13%{43%}.	20.4 Return to Step 4. OBSERVE CAUTION PRIOR TO STEP 4.
___21	**Stop all but one CHG PUMP.**	

1. 039K1.09 032/MOD/ROBINSON 04/MEM 2.7/2.7/039K1.09/N///

Concerning R-70A/B/C, 1A/1B/1C SG TUBE LEAK DET, on Unit 1:

Which one of the following completes the statements below?

The R-70s are located __(1)__ of the MSIVs.
A minimum reactor power level that the R-70s can accurately estimate a SG leak rate is __(2)__ .

	(1)	(2)
A.✔	upstream	25%
B.	downstream	25%
C.	upstream	1%
D.	downstream	1%

FSD-A181015
3.2.8 These detectors are located to monitor the main steam lines upstream of the safety relief valves for the presence of Nitrogen-16 activity in the steam lines and alert the operator when setpoints are exceeded.

SOP-69
Step 3.1 - **The system receives a reactor power input from power range channel N-43. IF N-43 fails OR is in Test OR is less than 20% power, THEN the system cannot accurately estimate a leak rate in the AV mode, and the indicators will display "PN <20%".** If desired, the Counting Room can configure the N-16 system in the ME counts per second (C/S) mode using FNP-0-CCP-31, LEAK RATE DETERMINATION. While not able to provide a leak rate determination, this mode can be used to indicate if leakage is increasing based on the indication trending up. The AV mode is the preferred mode of operation above 20% reactor power. The ME mode should only be utilized below 20% reactor power.

Distracter analysis

A. Correct. First part is correct. R-70s are located to monitor the main steam lines upstream of the safety relief valves.

Second part is correct. R-70s are accurate at reactor power >20%.

B. Incorrect. First part is incorrect (See A.1). Plausible if the applicant does not recall the location of these monitors.

Second part is correct (See A.2).

C. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2). Plausible since N-16 gammas would be produced once the reactor is critical and the applicant may believe that the R-70s are accurate to estimate leak rate once the reactor is critical at the POAH.

D. Incorrect. First part is incorrect (See B.1).

Second part is incorrect (See C.2).

K/A: 039K1.09	Main and Reheat Steam System (MRSS) - Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: RMS
Importance rating:	2.7 2.7
Technical Reference:	FSD-A181015, Radiation Monitoring System, Ver 14 FNP-1-SOP-69, N-16 Primary to Secondary Leak Detection System, Ver 5 D-175033, SH1, Main and Aux Steam, ver 38
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Radiation Monitoring System to include those items in Table 4- Remote and Local Indications and Controls (OPS-40305A02). RECALL AND DESCRIBE the physical in-plant location of those components associated with the Radiation Monitoring System, to include those items in Table 4- Remote and Local Indications and Controls (OPS-40305A03).
Question History:	MOD ROBINSON 04
K/A match:	The applicant is required to know the **physical location /connection** of the R-70s in relation to the main steam system and the **cause-effect** of the Rad monitors going into alarm, which is no effect in this case.
SRO justification:	N/A

3.2.8 Main Steam Line Nitrogen 16 (N-16) Monitor

TPNS No.

ND11RE 0070A, B, C **3.2.8.1** Basic Function

These detectors are located to monitor the main steam lines upstream of the safety relief valves for the presence of Nitrogen-16 activity in the steam lines and alert the operator when setpoints are exceeded.

In July of 1987, North Anna Unit 1 suffered a steam generator tube rupture accident that was subsequently attributed to high cyclic fatigue of some of the steam generator tubes. As a result of the event, the NRC issued NRC Bulletin 88-02 requesting several items from all licensees, including an enhanced primary to secondary leak rate monitoring system. The RE-0070 monitors were added to the plant design to provide this feature (References 6.7.038, 6.7.039, and 6.7.040,).

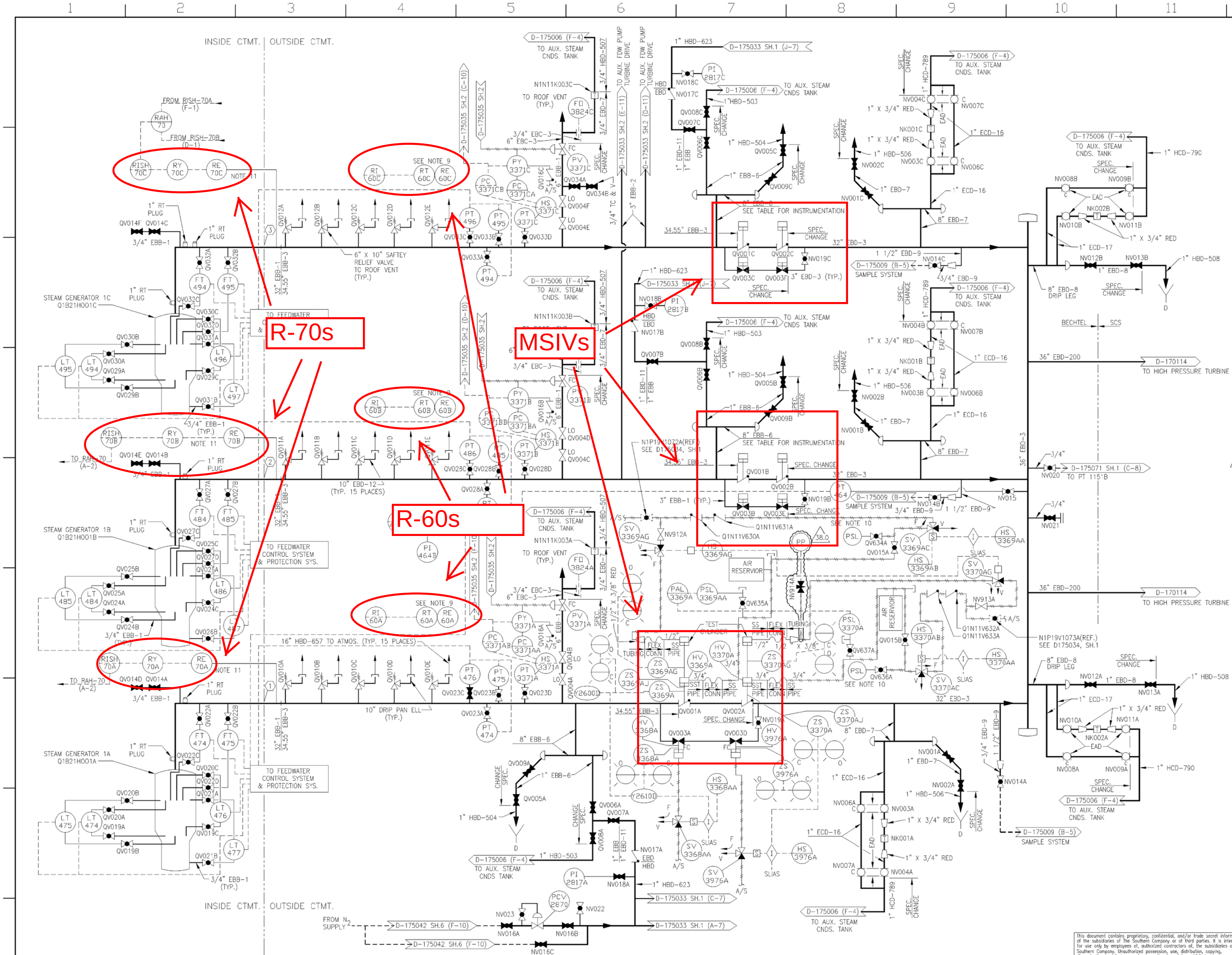
Radiolysis of water in the reactor core causes dissociation of the primary system water into free hydrogen and oxygen. Nitrogen-16 is produced from neutron activation of Oxygen-16 and subsequent emission of a proton. The nitrogen then decays, with a half-life of approximately 7.2 seconds, by beta and gamma emission. Since this reaction should be limited to the primary system, the presence of Nitrogen-16 activity in the main steam lines is indicative of primary to secondary side steam generator tube leakage. By utilizing a discriminating detector, the gamma energy peaks particular to the Nitrogen-16 activity can be identified and quantified.

3.2.8.2 Functional Requirements

A detector shall be mounted in the vicinity of each main steam line to read direct radiation levels through the steam header (Reference 6.4.320). This is possible because the high energy peaks of concern (6.13 and 7.1 MeV) are significantly high and background radiation levels are low (References 6.7.041 and 6.7.080).

3.2.8.3 I&C Requirements

3.2.8.3.1 A scintillation detector shall be provided to monitor a window of gamma energies from 0 keV to 7.65 MeV. The monitor shall provide a flat (±1 percent) response for gamma energies between 500 keV and 7 MeV (Reference 6.4.320).



MAIN STEAM STOP VALVES INSTRUMENTATION		
QV001A&2A QV003A QV003D HS3368AA HV3368A SV3368AA ZS3368A	QV001B&2B QV003B QV003E HS3368BA HV3368B SV3368BA ZS3368B	QV001C&2C QV003C QV003F HS3368CA HV3368C SV3368CA ZS3368C
HS3369AA HS3369AB HS3369AG HV3369A PAL3369A PSL3369AA QV630A QV631A QV634A QV635A N1P19V1072A SV3369AC SV3369AG ZS3369A ZS3369AG ZS3369AJ	HS3369BA HS3369BB HS3369BG HV3369B PAL3369B PSL3369BA QV630B QV631B QV634B QV635B N1P19V1072B SV3369BC SV3369BG ZS3369B ZS3369BG ZS3369BJ	HS3369CA HS3369CB HS3369CG HV3369C PAL3369C PSL3369CA QV630C QV631C QV634C QV635C N1P19V1072C SV3369CC SV3369CG ZS3369C ZS3369CG ZS3369CJ
HS3370AA HS3370AB QV632A QV633A QV636A QV637A N1P19V1073A HV3370A PSL3370AA SV3370AC SV3370AG ZS3370A ZS3370AG ZS3370AJ HS3976A HV3976A SV3976A ZS3976A	HS3370BA HS3370BB QV632B QV633B QV636B QV637B N1P19V1073B HV3370B PSL3370BA SV3370BC SV3370BG ZS3370B ZS3370BG ZS3370BJ HS3976B HV3976B SV3976B ZS3976B	HS3370CA HS3370CB QV632C QV633C QV636C QV637C N1P19V1073C HV3370C PSL3370CA SV3370CC SV3370CG ZS3370C ZS3370CG ZS3370CJ HS3976C HV3976C SV3976C ZS3976C
NV914A	NV914B	NV914C

38.0

NOTES:

1. FOR PIPING CLASS SUMMARY SHEETS SEE (SPEC SS-1109-2).
2. FOR VALVE INFORMATION SEE MASTER VALVE LIST. DWGS. B-175500, B-175504, B-175509.
3. FOR EQUIPMENT LIST SEE DWG. D-175070.
4. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INST. INDEX. B-175803.
5. ALL VALVE, EQUIP AND LINE NOS. ON THIS SHEET ARE PREFIXED BY Q OR N, 1N11 UNLESS OTHERWISE NOTED.
6. WORK THIS P&I DIAGRAM WITH DWG D-175300 D-175301, & D-175302
7. DRAINS TO BE DRAINED TO THE AUXILIARY STEAM CONDENSATE TANK OR TO THE MAIN STEAM ROOM DRAIN RETURN UNIT.
8. FOR COMPLETE INSTRUMENT NUMBERS AND ASSOCIATED CONTROL VALVE NUMBERS REFER TO INSTRUMENT INDEX B-175803.
9. RE-60 A,B,&C MONITOR THE STEAM DISCHARGE FROM SAFTEY VALVES QV010, QV011 AND QV012, AND ATMOSPHERIC RELIEF VALVES PV-3371 A, B & C.
10. PSL FOR TEST PURPOSES ONLY.
11. DETECTORS RE-70A,B,C ARE N16 MONITORS TO DETECT PRIMARY TO SECONDARY STEAM GENERATORS TUBE LEAKAGE, DETECTORS ARE INDEPENDENTLY STANCHION MOUNTED AND ARE NOT CONNECTED TO THE PIPE.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205033

CAD D1750331
ACAD2K | JLO - 09

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1

SUBJECT P&ID – MAIN STEAM AND AUXILIARY STEAM SYSTEM

DRAWN ___ CHECKED ___ DESIGNED ___

APPROVED FOR SIGNATURES DATE ___

APPROVED SEE MICROFILM DATE ___

APPROVED COPY DATE ___

SCALE NONE B/M ___

SHEET 1 OF — SHEETS

SUPERSEDES

D-175033

JLO MWD JMR 38.0 DATE 5-13-09

REVISED PER ABN 1072447901M012, VER. 1.0

DRAWING CATEGORY: CRITICAL

FARLEY NUCLEAR PLANT
UNIT 1
SYSTEM OPERATING PROCEDURE SOP-69.0

N-16 PRIMARY TO SECONDARY
LEAK DETECTION SYSTEM

1.0 Purpose

To provide guidance for operation of the Primary to Secondary Leak Detection System.

2.0 Initial Conditions

2.1 120V Regulated Instrumentation Panel 1B is energized per FNP-1-SOP-36.4, 120V A.C. DISTRIBUTION SYSTEMS.

3.0 Precautions and Limitations

3.1 The system receives a reactor power input from power range channel N-43. IF N-43 fails OR is in Test OR is less than 20% power, THEN the system cannot accurately estimate a leak rate in the AV mode, and the indicators will display "PN <20%". If desired, the Counting Room can configure the N-16 system in the ME counts per second (C/S) mode using FNP-0-CCP-31, LEAK RATE DETERMINATION. While not able to provide a leak rate determination, this mode can be used to indicate if leakage is increasing based on the indication trending up. The AV mode is the preferred mode of operation above 20% reactor power. The ME mode should only be utilized below 20% reactor power.

3.2 The N-16 Leak Detection System cannot determine the location of a leak within a specific Steam Generator. The system can however provide a more accurate leak rate determination if the location of the leak is known to be in one of the following locations:

Cold Leg - CB, Hot Leg - HB or U-Bend region - BE

WHEN a leak location is selected (CB, HB or BE), THEN the processor displays a leak rate that assumes the leak is at the location you have selected. The AV mode is essentially the average of the three leak rates at the specific locations.

3.3 The N-16 system is limited to an upward range of 1,000 gallons per day.

NOTE:
- **Changes between the AV and ME modes of operation is performed by the counting room using the guidance of FNP-0-CCP-31, LEAK RATE DETERMINATION, APPENDIX B.**
- **AV / gallons per day mode is the preferred mode of operation above 20% reactor power. ME / counts per second (C/S) mode does not indicate a leak rate, but is useful for trending changes in leak rate below 20% power. It is possible to indicate ME gallons per day (GPD), however this is misleading. While GPD units may be indicated, the value is actually representing counts per second. If indicating 'ME ---- GPD', the counting room should be requested to change the display to C/S.**

4.1.4 Check that all three indicating modules initialize and provide a display as follows:

- IF reactor power is ≥20%, THEN the display should indicate **'AV ---- GPD '**

OR

- IF reactor power is <20%, THEN the display should indicate **'AV PN <20% '** OR , **'ME ---- C/S '**

4.2 System Shutdown

4.2.1 Depress the ON/OFF pushbuttons on N1D11RISH0070A, B & C to deenergize the indicating modules.

4.2.2 Open the three circuit breakers located in panels (A)-CR191, (B)-CR191 and (C)-CR191. These panels are located in analyzer cabinet N1H21NFSGL2631-N (121' Rod Control Room) [key C415A].

4.2.3 Open breaker 9 in 120V Regulated A.C. Instrumentation Panel 1B, N1R22L001B-N (121' 1B Battery Charger Room).

38. 039 K1.09 001

Which ONE (1) of the following describes the location and use of Main Steam Line Radiation Monitors R-31 A, B, and C?

A. Mounted upstream of MSIVs. Reliable under non-accident conditions ONLY.

B. Mounted downstream of MSIVs. Reliable under non-accident conditions ONLY.

C.✓ Mounted upstream of MSIVs. Designated as ACCIDENT monitors.

D. Mounted downstream of the MSIVs. Designated as ACCIDENT monitors.

A-Incorrect. Designated as accident monitors
B-Incorrect. Mounted upstream of MSIVs, between the containment and the SG PORVs
C-Correct
D-Incorrect. Mounted upstream
Question 016
Tier 2 / Group 1
K/A Importance Rating - RO 2.7 SRO 2.7

Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: RMS.

Reference(s) - RMS SD-019
Proposed References to be provided to applicants during examination - None
Learning Objective - RMS004
Question Source - New
Question History -
Question Cognitive Level - Memory
10 CFR Part 55 Content - 41.7,11
Comments -

1. 045K1.19 033/MOD/CALLOWAY AUG 05/MEM 3.4*/3.6/045K1.19/N///

Which one of the following coincidences will cause an anticipated transient without trip (ATWT) mitigation system actuation circuitry (AMSAC) Main Turbine Trip?

_(1)_ Turbine impulse pressure channels > 40%

AND

_(2)_ SG NR levels < 10% for > 25 seconds.

	(1)	(2)
A.	1 of 2	2 of 3
B.✓	2 of 2	2 of 3
C.	1 of 2	1 of 3
D.	2 of 2	1 of 3

FSD- A181007 pg 2-37
C-20 Interlock. Control interlock C-20 is used to enable the Anticipated Transient Without Trip (ATWT) Mitigation System Actuation Circuitry (AMSAC) . When turbine load is > 40% on 2/2 turbine impulse channel detectors and steam generator narrow range water level decreases to <10% on 2/3 steam generators for 25 seconds, the AMSAC system will trip the main turbine and provide an auto start signal to all AFW pumps. There is a time delay drop out associated with the impulse pressure portion of the signal such that for 260 sec after impulse pressure decreases below 40%, AMSAC is still enabled.

A. Incorrect. First part is incorrect (See B.1). Plausible since various control and permissive interlocks use a 1 of 2 logic to enable or disable functions. The applicant could confuse AMSAC (C-20) with any of these.

Second part is correct (See B.2).

B. Correct. First part is correct. 2 of 2 turbine impulse channels > 40% enables AMSAC.

Second part is correct. 2 of 3 SG NR levels < 10% for > 25% actuates AMSAC.

C. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See B.2). Plausible because the Low Low SGWL is 1 of 3 SGWL less that 28% NR. The applicant could improperly believe that AMSAC is 1 of 3 as is the Low Low SGWL logic.

D. Incorrect. First part is correct (See B.1).

Second part is incorrect (See C.2).

K/A: 045K1.19	Main Turbine Generator (MT/G) System - Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: ESFAS
Importance Rating:	3.4* 3.6
Technical Reference:	FSD-A181007, Reactor Protection System, Ver 18
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Main Turbine and Auxiliaries System components and equipment, to include the following (OPS-40202A07): […] • Turbine Trips Actions needed to mitigate the consequence of the abnormality.
Question History:	MOD CALLOWAY AUG 05
K/A match:	AMSAC is listed as a back up to the reactor trip system and ESFAS in the FSAR. This question requires the applicant to know the **cause and effect** of relationship between AMSAC and the Main Turbine. Conditions which cause AMSAC to be enabled and produce a turbine trip.
SRO justification:	N/A

4. Pressurizer water level

5. Reactor coolant flow

6. Reactor coolant pump operation status (feeder voltage and bus frequency)

7. Steam generator steam flow

8. Steam generator water level

9. Turbine-generator operational status (auto stop oil pressure and stop valve position)

(References 6.1.002, 6.1.008, 6.7.001, 6.7.002, 6.7.003, 6.7.013, 6.7.050, 6.7.051, 6.7.067, 6.7.068)

2.4 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM

The Engineered Safety Features Actuation System (ESFAS) acts to limit the consequences of Condition III events (infrequent faults such a primary coolant spillage from a small rupture which exceeds normal charging system makeup and requires action of the safety injection system) and shall mitigate Condition IV events (limiting faults which include the potential for significant release of radioactive material).

The specific functions which rely on the Engineered Safety Feature Actuation System for automatic initiation are:

1. A reactor trip and main turbine trip, provided one has not already been generated by the Reactor Trip System.

2. Cold leg injection isolation valves which are opened for injection of borated water by the centrifugal charging pumps into the cold legs of the Reactor Coolant System.

3. Charging pumps, residual heat removal pumps and associated valving which provide emergency makeup water to the cold leg of the Reactor Coolant System following a loss of coolant accident.

4. Containment air recirculation fans and coolers, which serve to cool the containment and limit the potential for release of fission products from the containment by reducing the pressure following an accident.

5. Component cooling pumps and valves.

condenser. To activate C-9, both condenser pressures shall be < 8 inches Hg vacuum, and 1/2 circulation water pump motor breakers must be shut. (References 6.4.007, 6.4.017)

C-11 Interlock. Control interlock C-11 prevents misalignment of rod position counters and errors in bank overlap and insertion limits by blocking automatic withdrawal of control bank D when control bank D position is at 220 steps. (References 6.4.007, 6.4.016)

C-20 Interlock. Control interlock C-20 is used to enable the Anticipated Transient Without Trip (ATWT) Mitigation System Actuation Circuitry (AMSAC) . When turbine load is > 40% on 2/2 turbine impulse channel detectors and steam generator narrow range water level decreases to <10% on 2/3 steam generators for 25 seconds, the AMSAC system will trip the main turbine and provide an auto start signal to all AFW pumps. There is a time delay drop out associated with the impulse pressure portion of the signal such that for 260 sec after impulse pressure decreases below 40%, AMSAC is still enabled. (References 6.2.012, 6.4.007, 6.4.014, 6.4.022,)

2.9 CONTROL SYSTEM INTERFACE AND CONTROL ROOM INDICATIONS

The reactor control system shall be composed of those controllers, indicators, alarms and associated hardware whose primary function is to maintain the reactor within an allowable deviation of the steady state operation. In addition, the control system causes the NSSS power to follow the turbine demand in a controlled fashion during load transients. Indicators are provided for monitoring system operation. Alarms and annunciators are provided to alert the plant operator of control system malfunctions or abnormal operating conditions.

The reactor control system is designed to be independent of the reactor protection system. In certain applications, the control signals and other nonprotective functions are derived from individual protective channels through isolation amplifiers. The isolation amplifiers are classified as protection. The functional isolation of the signals shall meet the requirements of IEEE-279-1971. The signals obtained through the isolation amplifier shall never be returned to the protection rack.

The RPS shall be designed to provide the operator with accurate, complete, and timely information pertinent to its own status and to generating station safety. Moreover, protective actions shall be indicated and identified in the control room.

The control room shall include indicators, recorders, alarms and status information that are necessary to monitor the performance of the protection system during normal and accident conditions as described below:

A. Indicators and Recorders

All the transmitted signals which actuate reactor trips, rod stops, or permissive

TABLE T-4 - ENGINEERED SAFEGUARDS ACTUATION SIGNALS

TURBINE DRIVEN AUXILIARY FEEDWATER PUMP START (CONTINUED)

ACTUATION SIGNAL	SETPOINT	COINCIDENCE	INTERLOCKS & BLOCKS	PROTECTION PROVIDED FOR	MODES OF OPERATION	FSD SECTION
AMSAC Loop 1 LB474 Loop 2 LB485 Loop 3 LB496	10% Steam generator Narrow Range Span	1/1 AMSAC level channels $\leq$ setpoint on 2/3 Steam generators	-Actuation is auto blocked below C-20 -Actuation is auto unblocked above C-20 -Auto start of TDAFW pump with 25 second time delay	Loss of main feedwater	1, 2	2.7.1 Fig. F-2 Sht. 14 15
Manual	N/A	N/A	Local control overrides all signals	Operator discretion		2.7.1 Fig. F-2 Sht. 14

TABLE T-4 - ENGINEERED SAFEGUARDS ACTUATION SIGNALS

MOTOR DRIVEN AUXILIARY FEEDWATER PUMP START (CONTINUED)

ACTUATION SIGNAL	SETPOINT	COINCIDENCE	INTERLOCKS & BLOCKS	PROTECTION PROVIDED FOR	MODES OF OPERATION	FSD SECTION
AMSAC Loop 1 LB474 Loop 2 LB485 Loop 3 LB496	10% Steam generator NR span with a 25 second time delay	1/1 $\leq$ setpoint on 2/3 steam generators	-Actuation is auto blocked below C-20 -Actuation is auto unblocked above C-20	Loss of main feedwater	1, 2, 3	2.7.1 Fig. F-2 Sht. 14, 15
Manual	N/A	Individual switch for each pump	Local control overrides all other signals	Operator discretion		2.7.1 Fig. F-2 Sht. 14

Appendix G
OPERATIONAL PERMISSIVES AND CONTROL INTERLOCKS

PERMISSIVES

Permissive	Source	Setpoint	Coincidence & Light Status	Function
1. P-4 Reactor Trip Interlock	Reactor Trip and Bypass Breakers	Breakers Open	RTA & BYA Open or RTB & BYB Open No Light	Prevents a rapid cooldown of primary system after a reactor trip. 1. Trips Turbine 2. Trips F.W. Reg Valves on Low T_{avg} 3. Seals in F.W. Reg Valve Trips from S.I. and S/G Hi Hi Level 4. Allows S.I. signal to be blocked after S.I. initiation 5. Resets Hi Stm Flow Setpoint 6. Arms steam dump system, enables plant trip controller and disables loss of load controller.
2. P-6 I/R Power Escalation Permissive	NIS 35 and 36	10^{-10} amps	1/2 > Setpoint Lit > Setpoint Permission to Block Source Range	Allows power escalation into the IR by turning Both Train A & B Source Range Block switches to Block. Above setpoint 1. Blocks SR Hi ∅ Reactor Trip 2. Turns off Hi volt to SR Instr. Below setpoint Auto reinstates Hi volt to SR Instr.

PERMISSIVES

	Permissive	Source	Setpoint	Coincidence & Light Status	Function
3.	P-7 Low Power Rx Trip Block Permissive	P-10 P-13	10% Rx Power 10% Turbine Power	Lit < setpoint permission to trip RCPs, etc. (2/4 > setpoint) or (1/2 > setpoint)	Prevents unnecessary reactor trips at low power by auto blocking following Rx trips below setpoint. 1. Low Rx Coolant Flow 2. RCP U.V. 3. RCP Bus U.F. 4. PZR Low Pressure 5. PZR High Level Auto unblocks above Rx Trips above setpoint.
4.	P-8 Single Loop Loss of Flow Permissive	NIS 41, 42, 43, and 44	30% Rx Power	2/4 > setpoint Lit < setpoint Permission to stop 1 pump	Prevents a Rx trip from loss of flow OR RCP undervoltage in a single loop. Auto reinstates above Rx trips.
5.	P-9 Turbine Trip Permissive	NIS 41, 42, 43, and 44	35% Rx Power	2/4 > setpoint 3/4 < setpoint Lit < setpoint Permission to trip turbine	Auto reinstates a Rx trip from turbine trip. Prevents a Rx trip from a turbine trip.

PERMISSIVES

Permissive	Source	Setpoint	Coincidence & Light Status	Function
6. P-10 PR Power Escalation Permissive	NIS 41, 42, 43, and 44	10% Rx Power	2/4 > setpoint Lit > setpoint permission to block trips	Allows power escalation into the PR up to 100% power by 1. Turning Both Train A & B IR Blocks switches to BLOCK. a. Blocks IR Hi ∅ Rx trip b. Blocks IR Hi ∅ rod stop 2. Turning Both Train A & B PR Block switches to BLOCK blocking PR Hi ∅ (Low setpoint) Rx trip. 3. Interlocks the SR Hi volt by inhibiting manual reset Auto unblocks IR Hi ∅ Rx trip IR Hi ∅ rod stop PR Hi ∅ (low setpoint) Rx trip Allows manual reinstatement of SR Hi Volt & SR Hi ∅ Rx trip by turning both Train A & B source range block switches to reset.

PERMISSIVES

Permissive	Source	Setpoint	Coincidence & Light Status	Function
7. P-11 Pzr Low Press. Permissive Interlock	Pzr Press Instr. 455, 456, and 457	2000 psig	2/3 > setpoint no light	Below Setpoint 1. Allows manual block of SI from pzr lo press by turning Both Train A & B Pzr Press SI block switches to block. 2. Auto interlocks pzr power relief valves shut. Above Setpoint 1. Auto reinstates pzr lo press. SI. 2. Allows pzr power relief valves to be opened. 3. Opens SI accumulator isolation valves, if shut and handswitch is in AUTO.
8. P-12 Lo Lo T_{avg} Interlock	Primary Loop Temp. Instr. 412, 422, and 432	543°F	2/3 < setpoint Lit < setpoint permission to block SI	Below Setpoint Prevents rapid cooldown of primary sys. from a stm line rupture or blowdown through the stm dumps. 1. Provides temp. sig. portion of Hi stm flow with Lo Lo T_{avg} stm line isolation. 2. Interlocks stm dump valves shut. Allows manual block of low stm line press safety injection by turning both Train A & B steam line SI block switches to BLOCK. Above Setpoint 1. Auto unblocks low stm line pressure safety injection. 2. Allows stm dump valves to open.

PERMISSIVES

	Permissive	Source	Setpoint	Coincidence & Light Status	Function
9.	P-13 Turb Low Power Trip Block Permissive	Turb Impulse Press Inst. 446 and 447	10% Turb. Power	1/2 > setpoint Lit < setpoint permission to trip pumps, etc.	Provides input to P-7
10.	P-14 S/G Hi Hi Level Permissive Interlock	S/G Level Instr loop 1- 474, 475, and 476 loop 2- 484, 485, and 486 loop 3- 494, 495, and 496	82.0% of Level Span	2/3 Instr. 1/3 SG > setpoint no light	Above Setpoint Prevents carryover of water to main turbine 1. Trips all F.W. pumps 2. Trips main turbine 3. Interlocks all Main FW reg. valves and bypass valves closed Below Setpoint Allows main FW reg. valves and bypass valves to be opened.

CONTROL INTERLOCKS

Interlock	Source	Setpoint	Coincidence & Light Status	Function
1. C-1 IR Hi ∅ Rod Stop	NIS 35, and 36	Current Eq. to 20% Rx Power	1/2 > setpoint no light	1. Blocks auto and manual rod withdrawal. 2. May be manually blocked by turning Both IR Block 3. May be bypassed by level trip at NIS rack
2. C-2 PR Hi ∅ Rod Stop	NIS 41, 42, 43, and 44	103% Rx Power	1/4 > setpoint no light	1. Blocks auto and manual rod withdrawal. 2. Each PR inst. input may be manually bypassed at the NIS racks misc drawer
3. C-3 OTΔT Rod Stop	OTΔT Instr. 412, 422, and 432	Variable 3% Below OTΔT Rx Trip	2/3 > setpoint lit > setpoint	1. Blocks auto and manual rod withdrawal. 2. Cannot be blocked or bypassed.
4. C-4 OPΔT Rod Stop	OPΔT Instr. 412, 422, and 432	Variable 3% Below OPΔT Rx Trip	2/3 > setpoint lit > setpoint	1. Block auto and manual rod withdrawal. 2. Cannot be blocked or bypassed.
5. C-5 Low Turb. Power Rod Stop	Turb. Impulse Instr. 446 447 Sel. Sw. on MCB	15% Turb. Power	1/1 > setpoint lit > setpoint	Above setpoint allows auto rod control. Below setpoint blocks rod withdrawal in auto.

CONTROL INTERLOCKS

Interlock	Source	Setpoint	Coincidence & Light Status	Function
6. C-7 Sudden Loss of Load	Turb. Impulse Press Instr. 447 rate ckt.	15% Turb. Power Reduction 120 sec Time Const.	1/1 > setpoint lit > setpoint	Arms steam dump valves. Manually reset by placing the control mode selector switch to reset momentarily.
7. C-9 Condenser Available	Cond. Press. Switch and Circ Water Pump Bkrs.	8" Hg Vac. and Closed	2/2 > setpoint and 1/2 > setpoint lit > setpoint	Allows steam dump valves to be armed.
8. C-11 Bank D Stop	P/A Converter Bank D Position	220 Steps	1/1 = setpoint lit after being at setpoint for 3 min.	Stops outward rod motion in auto.
9. C-20 AMSAC Enabled	Turb. Impulse Pressure Inst. 2446 and 2447	> 40%	2/2 > setpoint; also sealed in for 260 seconds after Pimp lowers to < 40%; lit < setpoint	Allows AMSAC to be armed.

Reactor Trip	Instrumentation (TSLB)	Setpoint	Coincidence
10. Pressurizer High Water Level	LI-459A,460,461 (TSLB2 18-1,18-2,18-3	92%	2/3 (RX Pwr >10%)
11. Low Reactor Coolant Flow	FI-414,415,416 FI-424,425,426 FI-434,435,436 (TSLB2 4-1,4-2,4-3, 5-1,5-2,5-3, 6-1,6-2,6-3)	90%	2/3 per loop on 1/3 loops (Rx Pwr > 30%) 2/3 per loop on 2/3 loops (30% > Rx Pwr > 10%)
12. RCP Undervoltage	RCP Undervoltage Relays (TSLB2 1-1,1-2,1-3)	2680 V (0.6 sec time delay)	1/2 detectors on 2/3 RCPs (Rx Pwr > 10%)
13. RCP Bus Underfrequency	BUS Underfrequency Relays (TSLB2 2-1,2-2,2-3)	57 Hz (0.3 sec . time delay)	1/2 detectors on 2/3 Busses (Rx Pwr > 10%)
14. Low Low SG Water Level	LI-474,475,476 LI-484,485,486 LI-494,495,496 (TSLB4 4-1,4-2,4-3, 5-1,5-2,5-3, 6-1,6-2,6-3)	28%	2/3 Detectors on 1/3 SGs
15. Turbine Trip	DEHC (TSLB2 13-1,13-2,13-3, 14-1,14-2,14-3, 14-4)	Low Auto Stop Oil at 45 psig or Throttle Valves Closed	2/3 for Auto Stop Oil or 4/4 Throttle Valves Closed (Rx Pwr > 35%)
16. SI	N/A	Any SI Signal	1/2 Trains
17. General Warning	N/A	N/A	2/2 Trains
18. Manual	N/A	N/A	1/2

II. The following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit.

b. Rapid decrease in neutron level indicated by nuclear instrumentation.

c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

7.8 ATWS MITIGATION SYSTEM ACTUATION CIRCUITRY (AMSAC)

7.8.1 DESCRIPTION

7.8.1.1 System Description

The ATWS (anticipated transient without scram) mitigation system actuation circuitry (AMSAC) provides a backup to the reactor trip system (RTS) and ESF actuation system (ESFAS) for initiating turbine trip and auxiliary feedwater flow in the event of an anticipated transient (e.g., in the complete loss of main feedwater). The AMSAC is independent of and diverse from the RTS and ESFAS, with the exception of the final actuation devices. The AMSAC equipment, with the exception of the output isolation relays, is classified as control-grade equipment. It is a highly-reliable, microprocessor-based, single-train system powered by a non-Class 1E source.

The AMSAC continuously monitors level in the steam generators, which is an anticipatory indication of a loss of heat sink, and initiates certain functions when the level drops below a predetermined setpoint for at least a preselected time and for two of the three steam generator levels. These initiated functions are the tripping of the turbine, the initiation of auxiliary feedwater, and isolation of the steam generator blowdown and sample lines.

The AMSAC is designed to be highly reliable, resistant to inadvertent actuation, and easily maintained. Reliability is assured through the use of internal redundancy and continual self testing by the system. Inadvertent actuations are minimized through the use of internal redundancy and majority voting at the output stage of the system. The time delay on low steam generator level and the coincidence logic used also minimize inadvertent actuations.

The AMSAC automatically performs its actuations when above a preselected power level (determined using turbine impulse chamber pressure) and remains armed sufficiently long after that pressure drops below the setpoint to ensure that its function will be performed in the event of a turbine trip.

7.8.1.2 Equipment Description

The AMSAC consists of a single train of equipment located primarily in a seismically qualified cabinet. The output isolation relays, however, are located in two separate qualified wall-mounted cabinets.

It should be noted that following a reactor trip, feedwater will be isolated prior to reaching the no load T_{avg} set point of 547°F. This helps to ensure that T_{avg} can be stopped at 547°F.

AMSAC System

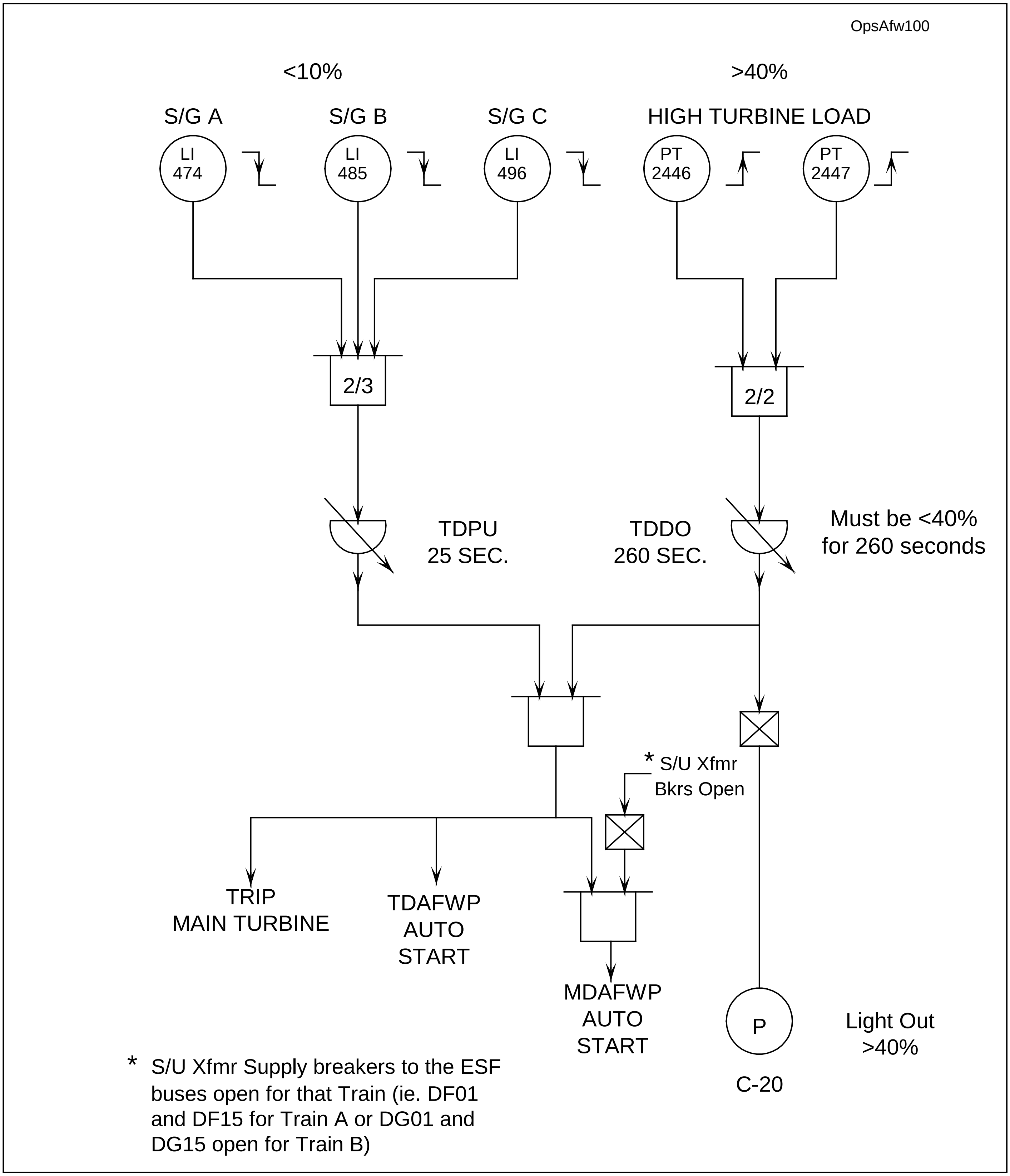
The AMSAC system is designed to mitigate the consequences of the most limiting ATWT conditions for Westinghouse PWRs. The most limiting ATWT conditions cause the largest primary system pressure rise and therefore challenge the integrity of the reactor coolant system. Analysis has shown that either of two conditions, loss of all normal main feedwater flow or the loss of electrical load, without the protection system initiated functions of reactor trip, turbine trip, and AFW starting are the most limiting conditions. Analysis has also shown that, if the turbine is tripped within 30 seconds and AFW is started within 60 seconds of the initiating event, that the resulting pressure transient will be within acceptable limits.

Figure 8 shows the logic circuitry for the AMSAC system. AMSAC is enabled when impulse pressure indicates turbine load is $\geq$ 40% on 2 out of 2 channels and remains enabled for 260 seconds after both channels are less than 40%. The time delay after going below 40% ensures that on a loss of electrical load, the system remains enabled long enough to detect steam generator levels shrinking below 10% level. A light on the MCB bypass and permissive panel (low turbine impulse pressure AMSAC trip blocked) is lit when the AMSAC system is blocked from performing its function. This block/enabling function is control interlock C-20. The impulse pressure channels used are two new channels, PT-2447 and PT-2446, that receive power from the AMSAC uninterruptible power supply (UPS). The AMSAC UPS is located on the 155' elevation and can be accessed via ladder from the 139' elevation, across the hall from the cable spreading room.

When enabled, AMSAC will trip the main turbine and provide an auto start signal to all AFW pumps when narrow range level in 2/3 steam generators drops below 10% for $\geq$ 25 seconds. The time delay prior to actuation is provided to permit recovery from a partial loss of feedwater. The AFW auto start will provide the normal functions of starting the pump, opening of flow control valves, and isolating blowdown. The instruments used for SG level are LT-474, -485, and -496 for steam generators A, B, and C, respectively.

TABLE 4
PERMISSIVES AND CONTROL INTERLOCKS

Permissive/Interlock	Source	Set Point	Coincidence & Light Status	Function
P-4 Reactor Trip Interlock	Reactor Trip and Bypass Breakers	Breakers Open	RTA & BYA Open or RTB & BYB Open No Light	Prevents a rapid cooldown of primary system after a reactor trip. 1. Trips Turbine 2. Trips FW Reg Valves on Low T_{avg} 3. Seals in FW Reg Valve Trips from SI and S/G Hi Hi Level 4. Allows SI signal to be blocked after SI initiation 5. Resets Hi Stm Flow Set Point 6. Arms the steam dumps
P-14 S/G Hi Hi Level Permissive Interlock	S/G Level Instr loop 1-474, 475, and 476 loop 2-484, 485, and 486 loop 3-494, 495, and 496	82% NR Level	2/3 Instruments on 1/3 SG > set point	Above Set Point, prevent carryover of water to main turbine. 1. Trips all FW pumps 2. Trips main turbine 3. Interlocks all main feed reg and bypass valves closed Below Set Point, allows main FW reg. and bypass valves to be opened, if P-4 not present.
C-20 AMSAC	Turb Impulse Pressure	40%	2/2 Turbine impulse pressure > setpoint will enable the AMSAC signal. 1 or both < setpoint for >260 seconds blocks AMSAC signal. Permissive status light not lit > setpoint	Allows AMSAC to arm.
AMSAC Actuation	SG Level Control Channels	≤10%	1/1 on 2/3 S/G with 25 second time delay	Trip main turbine. Start MDAFW. Start TDAFW when armed by C-20.



AMSAC Logic Diagram

Figure 8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045 K1.19	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: ESFAS

Proposed Question: Common 36

Which ONE (1) of the following is the minimum required for AMSAC to be "ARMED"?

A. 1 of 2 turbine impulse channels greater than 33% power.

B. 2 of 2 of the turbine impulse channels greater than 33% power.

C. 2 of 4 steam generator levels below 12% for 25 seconds.

D. 3 of 4 steam generator levels below 12% for 25 seconds.

Proposed Answer: B

Explanation (Optional):
A. Incorrect. Both channels required.
B. Correct. Both channels arm AMSAC.
C. Incorrect. This would partially satisfy actuation logic (2 of 4 low is reactor trip).
D. Incorrect. This is actuation logic.

Technical Reference(s): AMSAC (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: AMSAC B. (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

1. 051AK3.01 034/BANK/FNP EXAM BANK/C/A 2.8*/3.1/APE051AK3.01/N///

Unit 1 is operating at 40% power when Condenser vacuum rapidly rises to 6 psia.

Subsequently, vacuum stabilizes at 12 psia.

Which one of the following completes the statements below?

The Steam Dump __(1)__ controller is enabled.

The Steam Dumps are __(2)__ .

	(1)	(2)
A.✓	Plant Trip	CLOSED
B.	Plant Trip	OPEN
C.	Loss of Load	CLOSED
D.	Loss of Load	OPEN

FSD-A181007 Pg 2-36/37
C-9 Interlock. C-9 is the condenser-available interlock. This interlock allows the steam dump valves to be armed if the condenser is available. It also prevents an overpressure condition which could damage the condenser. To activate C-9, both condenser pressures shall be < 8 inches Hg vacuum, and 1/2 circulation water pump motor breakers must be shut.

8 inches of Hg vacuum is 10.8 psia.

See references Figure 2, Sheet 10 of FSD-A181007.

Distracter analysis

A. Correct. First part is correct. A turbine trip results which causes a reactor trip, thus enabling the plant trip controller.

Second part is correct. C-9 is NOT enabled at 12 psia therefore the steam dumps do not operate and are closed.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2) Plausible if the applicant cannot recall that the vacuum setpoint for the C-9 interlock is <10.8 psia and believes that adequate condenser vacuum exists for steam dump operation.

C. Incorrect. First part is incorrect (See A.1). Plausible if the applicant fails to recognize that the turbine trip causes a reactor trip at this power. If rx power were less than 35% then a rx trip would not occur and the turbine trip would cause the LOL controller to be the controlling controller.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2)

K/A: 051AK3.01	Loss of Condenser Vacuum - Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum
Importance Rating:	2.8* 3.1
Technical Reference:	FSD-A181007, Reactor Protection System, Ver 18
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Steam Dump System to include the components found on Figure 5, Steam-Dump Control (OPS-52201G02).
Question History:	FNP EXAM BANK
K/A match:	Requires the applicant to know on a **loss of vacuum** which controller the steam dumps will operate on and **the reason** the steam dumps will not operate (loss of capability). On a loss of vacuum the reason is **because** the C-9 interlock (vacuum) is not met. This is not stated in the stem but is inherent to the question.
SRO justification:	N/A

range channels exceeds a current equivalent to 20 percent reactor power. The rod stop may be manually blocked when above the P-10 setpoint, but is automatically reinstated below P-10 (3/4). This can be manually bypassed at NIS racks. (References 6.4.007, 6.4.011, 6.4.016)

C-2 Interlock. The C-2 overpower rod stop blocks automatic and manual control rod withdrawal. The block action occurs when 1/4 power range channels exceeds 103 percent reactor power. Each power range channel may be manually bypassed at the NIS racks. All power range channels cannot be bypassed at the same time. Only two power range channels may be blocked at one time using two switches located on the NIS Miscellaneous Control and Indication Drawer. (References 6.4.007, 6.4.011, 6.4.016)

C-3 Interlock. The C-3 control interlock is generated by the OTDT circuitry. The setpoint is 3 percent below the variable OTDT reactor trip setpoint. C-3 generates a block of automatic and manual rod withdrawal, when 2/3 loop delta Ts exceed their setpoint. The function of the rod block is to eliminate the cause of the impending trip, thereby preventing it. Since relatively slow transients are typical of those requiring OTDT protection, there is sufficient time for a load reduction to correct the situation. (References 6.2.003, 6.4.007, 6.4.012, 6.4.016)

C-4 Interlock. The C-4 control interlock is generated by the OPDT circuitry. The setpoint is 3 percent below the variable OPDT reactor trip setpoint. C-4 generates a block of automatic and manual rod withdrawal, when 2/3 loop delta Ts exceed their setpoint. The function of the rod block is to eliminate the cause of the impending trip, thereby preventing it. (References 6.2.003, 6.4.007, 6.4.012, 6.4.016)

C-5 Interlock. The C-5 interlock ensures that automatic rod withdrawal system is prevented when less than 15 percent power. It also prevents automatic rod withdrawal when power falls below 15 percent. The setpoint is 15 percent power as detected by turbine first stage impulse pressure. (References 6.4.007, 6.4.016, 6.4.022)

C-7 Interlock. This control interlock arms the steam dumps upon a load rejection (when in coincidence with C-9). The steam dump demand interlock (C-7) is actuated when turbine load is reduced by greater than 15 percent with a 120 second time constant. Rate differentiation of the first stage turbine impulse chamber pressure signal provides the equivalent turbine load signal. It must be manually reset. (Only PT-447 provides input to C-7.) (References 6.4.007, 6.4.017)

C-9 Interlock. C-9 is the condenser-available interlock. This interlock allows the steam dump valves to be armed if the condenser is available. It also prevents an overpressure condition which could damage the

condenser. To activate C-9, both condenser pressures shall be < 8 inches Hg vacuum, and 1/2 circulation water pump motor breakers must be shut. (References 6.4.007, 6.4.017)

C-11 Interlock. Control interlock C-11 prevents misalignment of rod position counters and errors in bank overlap and insertion limits by blocking automatic withdrawal of control bank D when control bank D position is at 220 steps. (References 6.4.007, 6.4.016)

C-20 Interlock. Control interlock C-20 is used to enable the Anticipated Transient Without Trip (ATWT) Mitigation System Actuation Circuitry (AMSAC) . When turbine load is > 40% on 2/2 turbine impulse channel detectors and steam generator narrow range water level decreases to <10% on 2/3 steam generators for 25 seconds, the AMSAC system will trip the main turbine and provide an auto start signal to all AFW pumps. There is a time delay drop out associated with the impulse pressure portion of the signal such that for 260 sec after impulse pressure decreases below 40%, AMSAC is still enabled. (References 6.2.012, 6.4.007, 6.4.014, 6.4.022,)

2.9 CONTROL SYSTEM INTERFACE AND CONTROL ROOM INDICATIONS

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The control room shall include indicators, recorders, alarms and status information that are necessary to monitor the performance of the protection system during normal and accident conditions as described below:

A. Indicators and Recorders

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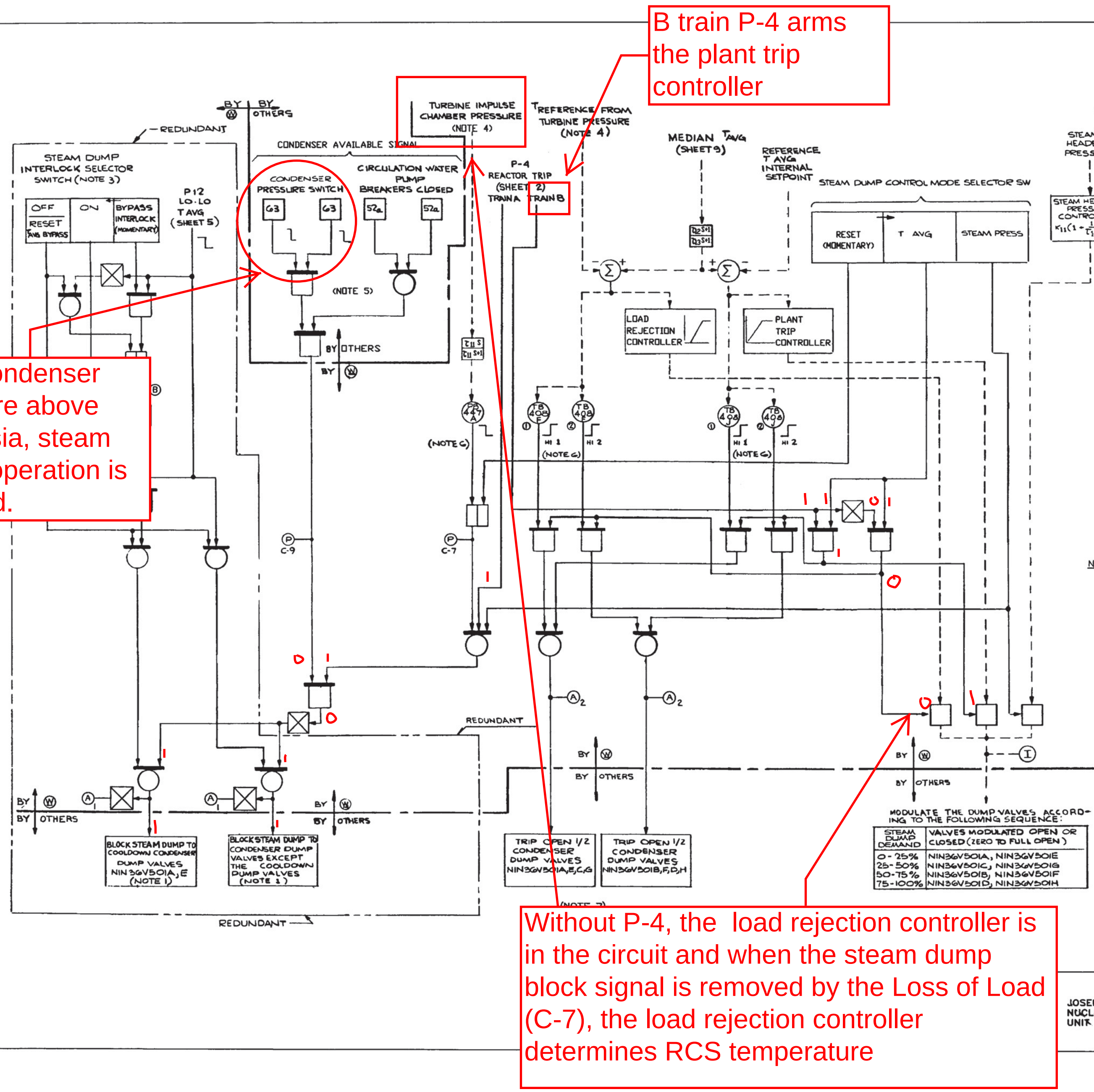
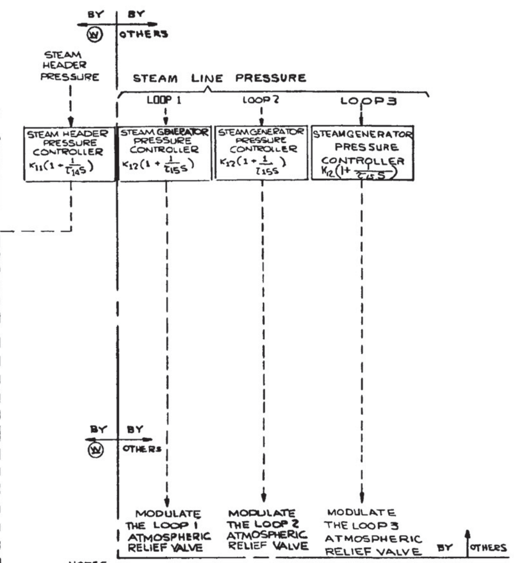


STEAM DUMP DEMAND	VALVES MODULATED OPEN OR CLOSED (ZERO TO FULL OPEN)
0 - 25%	NIN3GV501A, NIN3GV501E
25 - 50%	NIN3GV501C, NIN3GV501G
50 - 75%	NIN3GV501B, NIN3GV501F
75 - 100%	NIN3GV501D, NIN3GV501H



NOTES

1. STEAM DUMP IS BLOCKED BY BLOCKING AIR TO THE DUMP VALVES AND VENTING THE DIAPHRAGM. THE REDUNDANT LOGIC OUTPUT OPERATES 2 SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND ITS ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZED TO VENT CAUSING THE MAIN DUMP VALVE TO CLOSE IN FIVE SECONDS.
2. CIRCUITRY ON THIS SHEET IS NOT REDUNDANT EXCEPT WHERE INDICATED REDUNDANT.
3. SELECTOR SWITCH WITH THE FOLLOWING 3 POSITIONS -
   ON- STEAM DUMP IS PERMITTED.
   BYPASS- T AVG INTERLOCK IS BYPASSED FOR LOW T AVG. SPRING RETURN TO ON POSITION.
   OFF- STEAM DUMP IS NOT PERMITTED AND RESET T AVG BYPASS.

   THE REDUNDANT INTERLOCK SEKECTOR SWITCH CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
4. THE TWO ANALOG SIGNAL INPUTS COMING FROM TURBINE PRESSURE MUST COME FROM DIFFERENT PRESSURE TAPS TO MEET THE SINGLE FAILURE CRITERION.
5. THE CONDENSER AVAILABLE SIGNAL LOGIC IS TYPICAL. ACTUAL IMPLEMENTATION MAY BE DIFFERENT.
6. ALL TEMPERATURE BISTABLES ON THIS SHEET AND TURBINE IMPULSE CHAMBER PRESSURE BISTABLE #PB-447A ARE "ENERGIZE TO ACTUATE".
7. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH DUMP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.

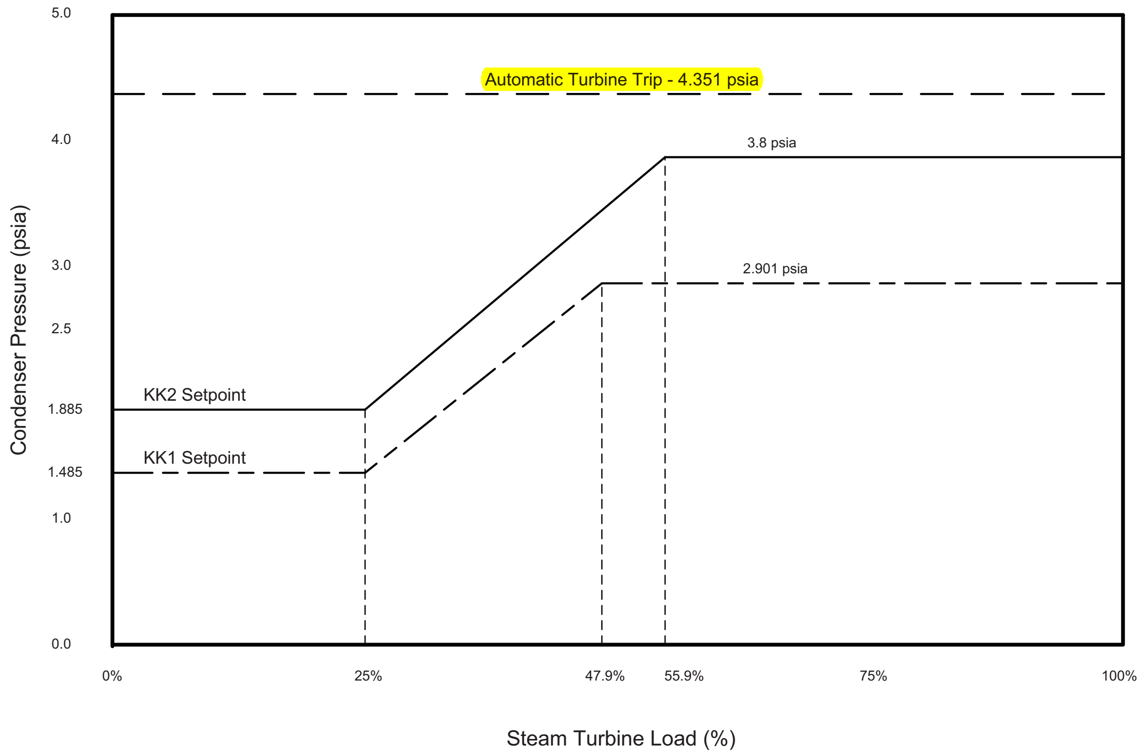
REV. 4 7/98

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM DUMP CONTROL
	FIGURE F-2, SHEET 10

PERMISSIVES

	Permissive	Source	Setpoint	Coincidence & Light Status	Function
3.	P-7 Low Power Rx Trip Block Permissive	P-10 P-13	10% Rx Power 10% Turbine Power	Lit < setpoint permission to trip RCPs, etc. (2/4 > setpoint) or (1/2 > setpoint)	Prevents unnecessary reactor trips at low power by auto blocking following Rx trips below setpoint. 1. Low Rx Coolant Flow 2. RCP U.V. 3. RCP Bus U.F. 4. PZR Low Pressure 5. PZR High Level Auto unblocks above Rx Trips above setpoint.
4.	P-8 Single Loop Loss of Flow Permissive	NIS 41, 42, 43, and 44	30% Rx Power	2/4 > setpoint Lit < setpoint Permission to stop 1 pump	Prevents a Rx trip from loss of flow <u>OR</u> RCP undervoltage in a single loop. Auto reinstates above Rx trips.
5.	P-9 Turbine Trip Permissive	NIS 41, 42, 43, and 44	35% Rx Power	2/4 > setpoint 3/4 < setpoint Lit < setpoint Permission to trip turbine	Auto reinstates a Rx trip from turbine trip. Prevents a Rx trip from a turbine trip.

LOCATION KK1



Steam Turbine Load	Setpoint	Steam Turbine Load	Setpoint
25%	1.485 PSIA	37%	2.223 PSIA
26%	1.546 PSIA	38%	2.284 PSIA
27%	1.607 PSIA	39%	2.346 PSIA
28%	1.669 PSIA	40%	2.407 PSIA
29%	1.730 PSIA	41%	2.469 PSIA
30%	1.792 PSIA	42%	2.531 PSIA
31%	1.853 PSIA	43%	2.592 PSIA
32%	1.915 PSIA	44%	2.654 PSIA
33%	1.976 PSIA	45%	2.715 PSIA
34%	2.038 PSIA	46%	2.777 PSIA
35%	2.100 PSIA	47%	2.838 PSIA
36%	2.161 PSIA	47.9%	2.901 PSIA

References: A-177100, Sh. 491; D-172803; D-170812, Sh. 2; U-162213, Tab 5; Westinghouse Customer Advisory Letter 86-02; DCP P-95-1-8943; DCP 1090247701

1. 054AA2.05 035/NEW//C/A 3.5/3.7/APE054AA2.05/N///

Unit 1 is operating at 4% power. The following conditions exist:

- 1A SGFP is running.
- All SG NR levels are in the programmed band.
- FCV-479/489/499,1A/1B/1C SG FW BYP FLOW, controllers are in MANUAL and 35% open.

Subsequently, the 1A SGFP trips.

Which one of the following completes the statements below?

MOV-3232A/B/C, MAIN FW TO 1A/1B/C SG, will __(1)__.

FCV-479/489/499,1A/1B/1C SG FEED FLOW BYPASS FCVs, will __(2)__.

	(1)	(2)
A.	remain OPEN	remain OPEN
B.	remain OPEN	CLOSE
C.✓	CLOSE	remain OPEN
D.	CLOSE	CLOSE

For this event the applicant has to analyze the situation. For a SGFP trip, aop-13 is required to be entered and a Rx trip is initiated >5% power. If the applicant thought the Rx was tripped, then the dumps would be controlling at 547°F and a FWI signal would be generated. This would directly affect the bypass valves. Since the bypass valves are rarely used, an applicant may not realize the link and open/close signals.
Since we are <5% power, the RTBs are not opened and AFW will auto start to raise SGWL due to both SGFPs tripped. This will keep level high. MOV-3232A/B/C close when both SGFPs are tripped. This has to be analyzed and known for these two particular valves.

<u>Distracter analysis</u>

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant thinks that this valve only automatically shuts on a feedwater isolation (FWI). A FWI has NOT occurred at this time.

Second part is correct (See C.1).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect. (See C.1). Plausible if the applicant thinks that a FWI has occurred.

C. Correct. First part is correct. D175073, Sheet 1 shows that these valves close on a SGFP trip.

Second part is correct. The bypass valves are in manual and therefore remain open since there is NO feedwater isolation (FWI). A FWI occurs with a P-4 signal (Rx Trip) coincident with a low Tavg, Safety Injection and a Hi-Hi SGWL (P-14).

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See B.2).

K/A: **054AA2.05**	Loss of Main Feedwater (MFW) - Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Status of MFW pumps, regulating and stop valves
Importance Rating:	3.5 3.7
Technical Reference:	D-175073, SH 1, Main Feedwater System, Ver 18 FSD-181007, Reactor Protection System, Ver 18
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if entry into AOP-13, Loss of Main Feedwater is required. (OPS-52520M02)
Question History:	NEW
K/A match:	Requires the applicant to **determine the status of feed system STOP valves and bypass FCVs upon a loss of Main Feedwater.**
SRO justification:	N/A

for the required engineered safety features lines. Phase B isolation is initiated by containment pressure High-3 (27 psig) or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

The Containment Ventilation Isolation isolates the containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on the completion of the SI logic, high radioactivity levels in the purge exhaust, or by manual initiation of either Phase A Containment Isolation or Phase B Isolation/Containment Spray Actuation. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.080)

3. Main Steam Line Isolation

Isolation of the Main Steam lines limits the effects of an uncontrolled release of steam either inside or outside the containment. For a break upstream of the isolation valves (MSIV) in the steamlines, valves closure will limit the release to the blowdown of the one affected steam generator. A break downstream of the valves is limited to the depressurization of the pipe volume downstream of the valves. This results in a rapid termination of the event and significantly reduces the mass lost from the secondary.

The Main Steam Line Isolation is initiated by the following:

a. High steam line flow with low-low T_{avg} , 1/2 steam flow channels above setpoint (40% of full steam flow between 0-20% load and increasing linearly to 110% at full load) on 2/3 steam lines with $T_{avg} \leq$ P-12

b. Low steam pressure; $\leq$ 585 psig on 2/3 S.G.

c. High-2 containment pressure; $\geq$ 16.2 psig on 2/3

d. Manual. By closing each MSIV by operating individual hand switches. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main Feed line Isolation:

a. Safety injection

b. High-high steam generator water level (P-14) set at ≥ 82% of narrow range steam generator span on 2/3 coincidence

c. Low T_{avg}; ≤ 554°F in coincidence with reactor trip P-4 manual reset to clear.

(References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.21)

5. Containment Spray

Containment Spray reduces the containment building pressure and temperature following a LOCA or high-energy line rupture and reduces airborn fission products in the containment atmosphere following a LOCA. Actuation of Containment Spray starts the containment spray pumps, aligns the discharge of the pumps to the containment spray nozzle header in the upper levels of the containment and opens the spray additive tank outlet valves. Containment spray is initiated by:

a. High-3 containment pressure (≥ 27 psig on 2/4).

b. It can also be initiated manually by actuating two MCB mounted push button handswitches simultaneously (2/4 switches in two groups)

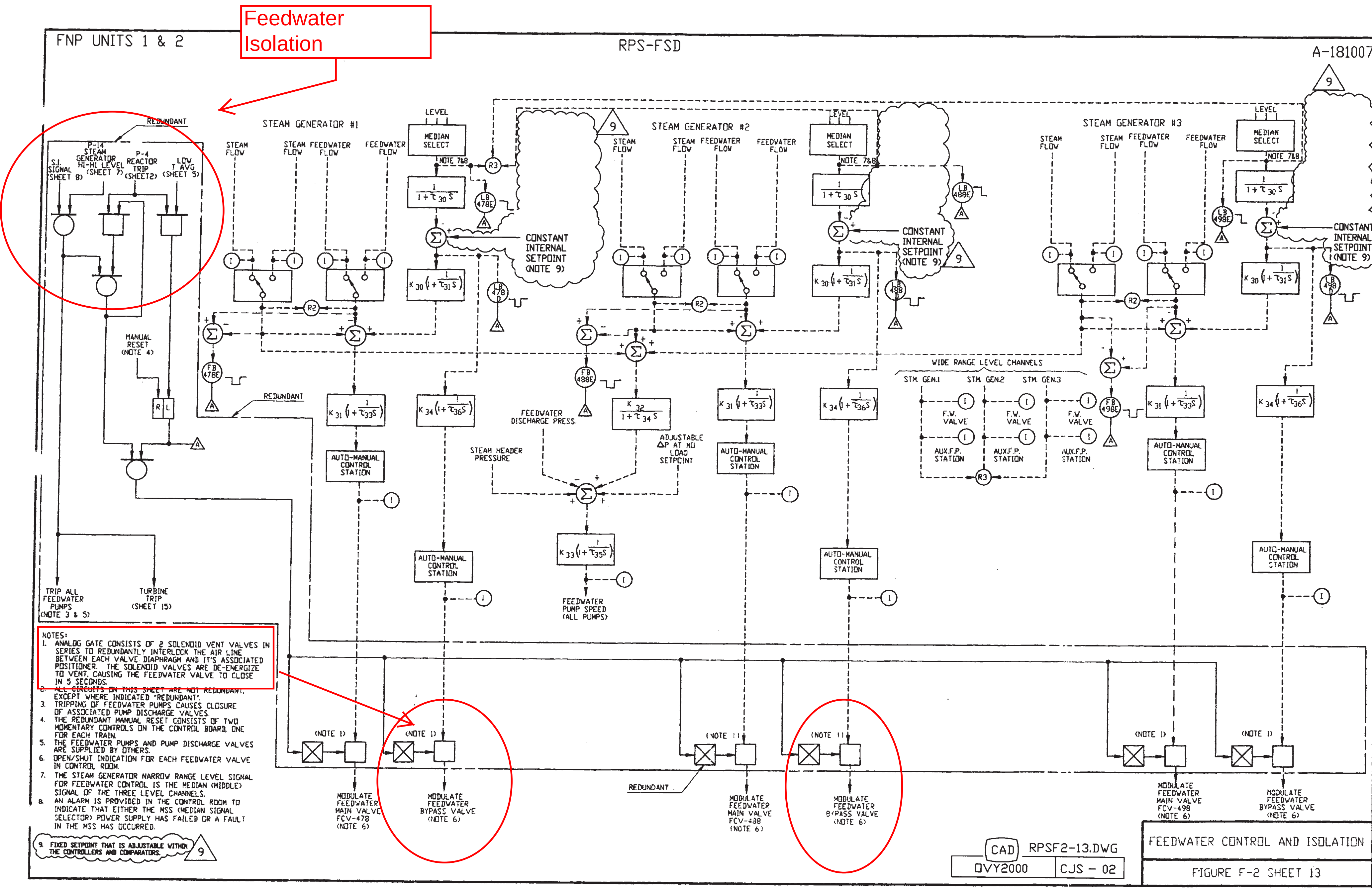
(References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

6. Auxiliary Feedwater

The Auxiliary Feedwater System (AFW) provides water to the steam generators (SG), assuring the availability of a heat sink for the primary side, when the Main Feedwater System is unavailable. The AFW typically supplies feedwater for normal plant operations when the plant is heating up, cooling down, or at a power level below the Main Feedwater System's capability e.g., ≤ 7% RTP.

Assurance that adequate feedwater is available to mitigate a postulated feedline break is provided by the auxiliary feedwater system, which includes two motor-driven pumps and one turbine-driven pump. The motor-driven pumps are initiated automatically by one of the following signals:

a. Low-Low steam generator (SG) water level in 1/3 steam generators ≤ 28% of narrow range span (NR)



NOTES:

1. ANALOG GATE CONSISTS OF 2 SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND IT'S ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZE TO VENT, CAUSING THE FEEDWATER VALVE TO CLOSE IN 5 SECONDS.
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT, EXCEPT WHERE INDICATED 'REDUNDANT'.
3. TRIPPING OF FEEDWATER PUMPS CAUSES CLOSURE OF ASSOCIATED PUMP DISCHARGE VALVES.
4. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
5. THE FEEDWATER PUMPS AND PUMP DISCHARGE VALVES ARE SUPPLIED BY OTHERS.
6. OPEN/SHUT INDICATION FOR EACH FEEDWATER VALVE IN CONTROL ROOM.
7. THE STEAM GENERATOR NARROW RANGE LEVEL SIGNAL FOR FEEDWATER CONTROL IS THE MEDIAN (MIDDLE) SIGNAL OF THE THREE LEVEL CHANNELS.
8. AN ALARM IS PROVIDED IN THE CONTROL ROOM TO INDICATE THAT EITHER THE MSS (MEDIAN SIGNAL SELECTOR) POWER SUPPLY HAS FAILED OR A FAULT IN THE MSS HAS OCCURRED.
9. FIXED SETPOINT THAT IS ADJUSTABLE WITHIN THE CONTROLLERS AND COMPARATORS.

03:25:10 PM 11/16/01 CLHAYES

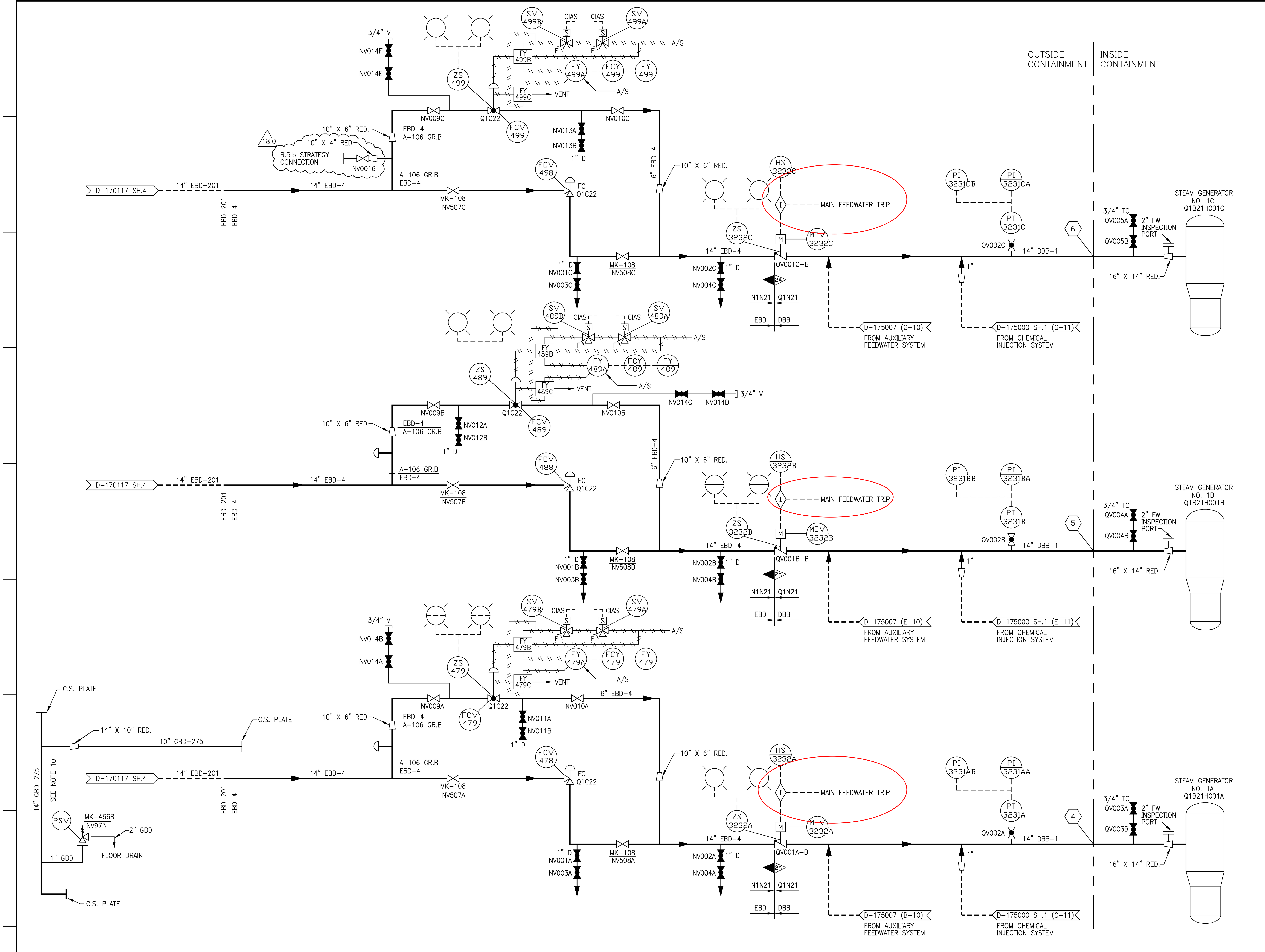


TABLE-A

Q1N21LSH2828A-A Q1N21LSH2828B-A Q1N21LSH2828C-A	PUMP NO.1A, 1B
Q1N21LSH2829A-B Q1N21LSH2829B-B Q1N21LSH2829C-B	PUMP NO.1A, 1B

NOTES:

1. PIPING CLASS SUMMARY SHEETS SEE SPEC. SS-1109-1, OR SS-1109-2.
2. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX B-175803.
3. FOR P&ID LEGEND SEE DRAWING D-175016.
4. DRAINS ARE TO BE PUMPED OR DRAINED TO THE AUXILIARY STEAM CONDENSATE TANK, TO THE CONDENSATE STORAGE TANK, OR TO THE MAIN STEAM ROOM DRAIN RETURN UNIT.
5. UNLESS OTHERWISE NOTED, ALL VALVE, EQUIPMENT, AND LINE NUMBERS ON THIS SHEET ARE PREFIXED BY Q OR N1N21.
6. FOR COMPLETE INSTRUMENT NUMBERS AND ASSOCIATED CONTROL VALVE NUMBERS REFER TO INSTRUMENT INDEX B-175803.
7. FOR INFORMATION RELATING TO THE FEEDWATER CONTROL SYSTEM REFER TO DRAWINGS U-169614, U-175939D, U-183472, U-167941G.
8. MAIN STEAM ROOM FLOODING SENSORS LISTED IN TABLE "A" ARE SET TO TRIP THE MAIN FEEDWATER PUMPS IN ACCORDANCE WITH LOGIC B-175810 SH.126.
9. FOR INFORMATION RELATING TO THE THERMAL LINER ASSEMBLY SEE DRAWINGS U-265126, U-265127, U-265128, AND U-265129.
10. THIS PORTION OF PIPE IS ABANDONED IN PLACE. REFER TO DCP 1061089801.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205073 SH.1.

CAD D1750731.DWG
ACAD2K | Vep - 02

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT P&ID - MAIN FEEDWATER SYSTEM

DRAWN ____ CHECKED ____ DESIGNED ____
APPROVED FOR SIGNATURES DATE ____
APPROVED SEE MICROFILM DATE ____
APPROVED COPY DATE ____

VER. NO. 18.0 DATE 4-23-08
REVISED PER ABN 1072193901M004, VER. 1.0

SCALE NONE B/M ____
SHEET 1 OF - SHEETS
SUPERSEDES

D-175073

DRAWING CATEGORY: CRITICAL

NOTES:

1. ANALOG GATE CONSISTS OF 2 SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND IT'S ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZE TO VENT, CAUSING THE FEEDWATER VALVE TO CLOSE IN 5 SECONDS.
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6. OPEN/SHUT INDICATION FOR EACH FEEDWATER VALVE IN CONTROL ROOM.
7. THE STEAM GENERATOR NARROW RANGE LEVEL SIGNAL FOR FEEDWATER CONTROL IS THE MEDIAN (MIDDLE) SIGNAL OF THE THREE LEVEL CHANNELS.
8. AN ALARM IS PROVIDED IN THE CONTROL ROOM TO INDICATE THAT EITHER THE MSS (MEDIAN SIGNAL SELECTOR) POWER SUPPLY HAS FAILED OR A FAULT IN THE MSS HAS OCCURRED.
9. FIXED SETPOINT THAT IS ADJUSTABLE WITHIN THE CONTROLLERS AND COMPARATORS. A1

CAD U166243.DWG
R14OVY2 JOK-02

FARLEY U-166243

7597-03-E46.3-29

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U.S.A.
TITLE ALABAMA POWER COMPANY
JOSEPH M. FARLEY UNIT 1 A1
FUNCTIONAL DIAGRAMS
FEEDWATER CONTROL & ISOLATION
SCALE ~
DO NOT SCALE
565SD37
SHEET 13
SUB 4 A0 A1

REV. NO. A0 DATE: 7-22-94 SCS REV. PER PCN B-93-1-8538, REV. 0-4
REV. NO. A1 DATE: 8-24-00 REVISED BY SCS PER DCP 98-1-9357, 001-007

ALA/APR-300 1 / ALA/APR-300 4 / CHANGE
SO 90624V/90G242 / ECN-3122O / REDRAWN FROM SUB 3. D. PETROCELLI 6/20/78

VENDOR: WESTINGHOUSE
P.O.: FNP-002
DIST. CODE: 214
STATUS CODE: 9
FUNCTIONAL DIAGRAMS MAIN FEEDWATER CONTROL & ISOLATION
DWG. NO. U 166243 H
PLANT FARLEY
UNIT 1

FSAR PROJECT DRAWING

1. 055A3.03 036/MOD/FNP 11/C/A 2.5*/2.7*/055A3.03/N///

Unit 1 is at 70% power with the following conditions:

- **R-15A, SJAE EXH**, is in alarm.
- **R-15B, TURB BLDG VNTL**, is in alarm.
- AOP-2.0, Steam Generator Tube Leakage, is in progress.
- The Turbine Building SO has placed the SJAE Filtration System in service.

Which one of the following completes the statement below?

After the SJAE Filtration system is placed in service, the reading on **R-15B** will _(1)_ and the SJAE Filtration system will _(2)_ .

	(1)	(2)
A.	decrease	be aligned in a recirc alignment
B.	remain the same	be aligned in a recirc alignment
C.✓	decrease	discharge to the Turbine Bldg roof
D.	remain the same	discharge to the Turbine Bldg roof

D170064/D-175027:
These drawings show that R-15A is upstream of the normally off service SJAE filtration system. R-15B is downstream of the SJAE filtration system. When the SJAE filtration sysytem is placed on service, R-15B reading will decrease. The SJAE filtration system discharges directly to the turbine building roof and cannot be diverted elsewhere.

Distracter analysis

A. Incorrect. First part is correct (See C.1).

Second part is incorrect (See C.2). Plausible since the turbine building ventilation system is capable of bypassing the SJAE filter system which is similar to a recirc alignment. Recirc would seem reasonable to minimize radioactive release to the outside atmosphere. The SJAE filtration system discharges directly to the turbine building roof and cannot be diverted elsewhere. The Penetration Room Filtration system on the rad side does have recirc alignment MOVs and a student could confuse the two systems or apply the concepts from one system to the other.

B. Incorrect. First part is incorrect (See C.1). Plausible if the applicant cannot recall the location of R-15B and believes it is upstream of the SJAE filtration system. R-15A is located before the SJAE filtration system and remain the same would be the correct answer.

Second part is incorrect (See A.2).

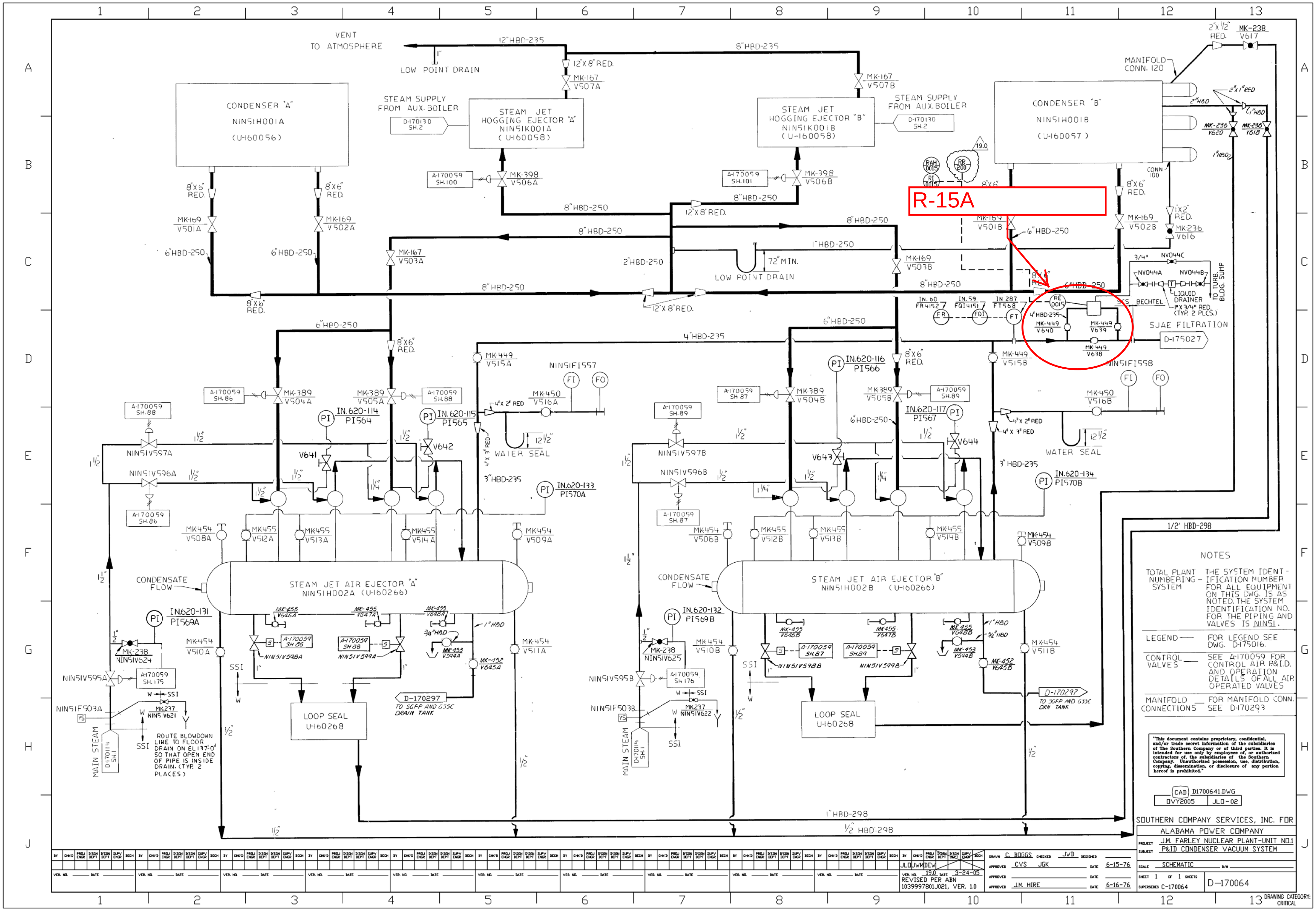
C. Correct. First part is correct. R-15B is downstream of the SJAE filtration system. When the SJAE filtration system is placed on service, R-15B reading will decrease.

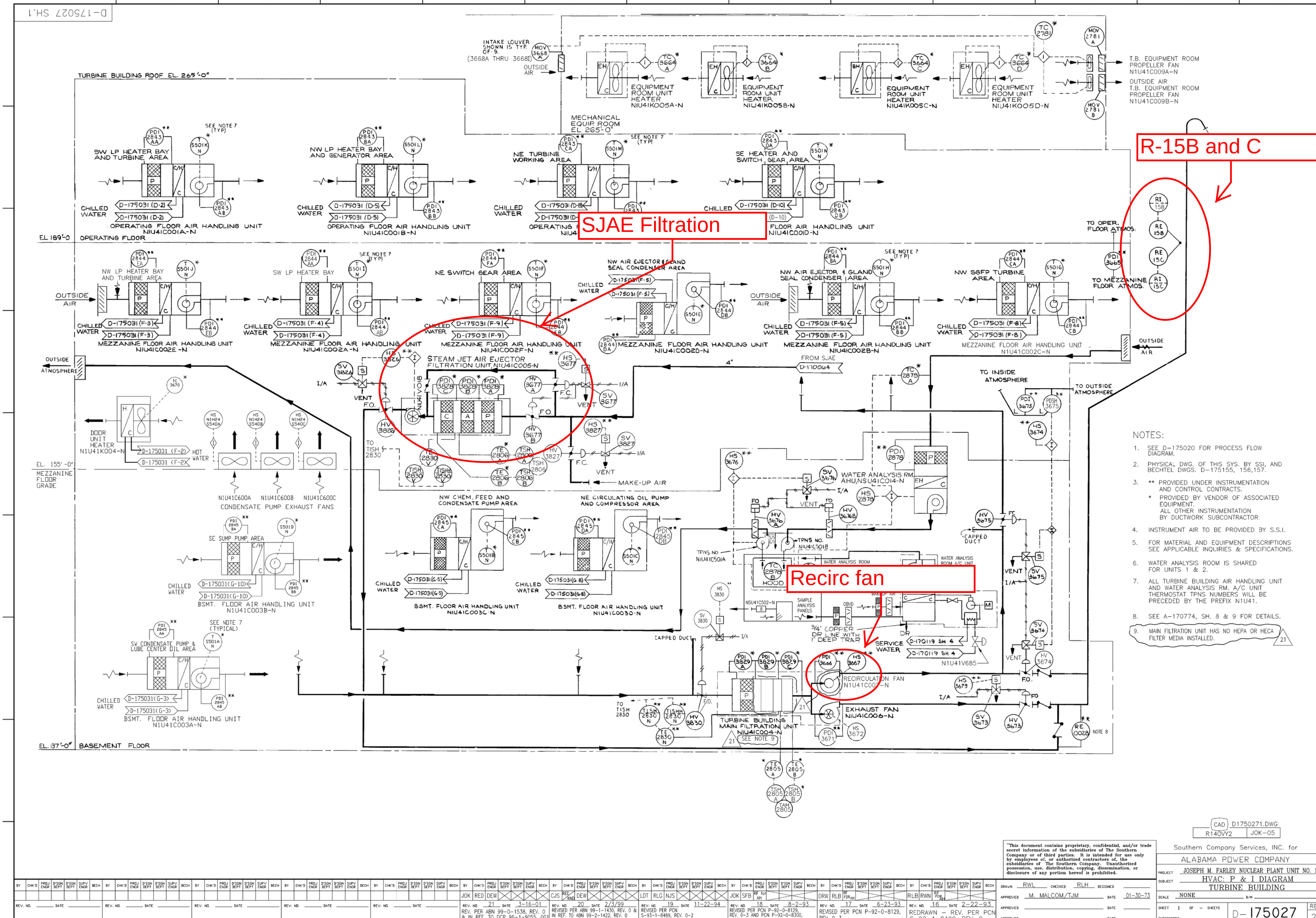
Second part is correct. The SJAE filtration system discharges directly to the turbine building roof and cannot be diverted elsewhere.

D. Incorrect. First part is incorrect (See B.1).

Second part is correct (See C.2).

K/A: **055A3.03**	Condenser Air Removal System (CARS) - Ability to monitor automatic operation of the CARS, including: Automatic diversion of CARS exhaust
Importance Rating:	2.5* 2.7*
Technical Reference:	D-170064, SH1, Condenser Vacuum System, Ver 19 D-175027, SH 1, HVAC: TUBINE BLDG, Ver 21
References provided:	None
Learning Objective:	LABEL, DRAW AND ILLUSTRATE the Condensate and Feedwater System flow paths, to include the components on the following figures (OPS-40201B05, Part A): • Figure 3, Condenser Air Removal System
Question History:	MOD FNP 11
K/A match:	Requires the applicant to **monitor the R-15B reading and determine its response when the SJAE filtration system is placed on service.** FNP has no automatic diversion of the SJAE exhaust system. (10/24/12) Chief Examiner said using manual diversion based on our plant design is acceptable.
SRO justification:	N/A


R-15A
CONDENSER "A"
NIN51H001A
(U-160056)
CONDENSER "B"
NIN51H001B
(U-160057)
STEAM JET HOGGING EJECTOR "A"
NIN51K001A
(U-160058)
STEAM JET HOGGING EJECTOR "B"
NIN51K001B
(U-160058)
STEAM JET AIR EJECTOR "A"
NIN51H002A (U-160266)
STEAM JET AIR EJECTOR "B"
NIN51H002B (U-160266)
LOOP SEAL
U-160268
VENT TO ATMOSPHERE
LOW POINT DRAIN
STEAM SUPPLY FROM AUX. BOILER
WATER SEAL
CONDENSATE FLOW
MAIN STEAM
SJAE FILTRATION
D-175027
LIQUID DRAINER
TO TURB. BLDG. SUMP
ROUTE BLOWDOWN LINE TO FLOOR DRAIN ON EL 137'-0" SO THAT OPEN END OF PIPE IS INSIDE DRAIN. (TYP. 2 PLACES)
NOTES
TOTAL PLANT NUMBERING SYSTEM — THE SYSTEM IDENTIFICATION NUMBER FOR ALL EQUIPMENT ON THIS DWG. IS AS NOTED. THE SYSTEM IDENTIFICATION NO. FOR THE PIPING AND VALVES IS N1N51.
LEGEND — FOR LEGEND SEE DWG. D175016.
CONTROL VALVES — SEE A170059 FOR CONTROL AIR P&I.D. AND OPERATION DETAILS OF ALL AIR OPERATED VALVES
MANIFOLD CONNECTIONS — FOR MANIFOLD CONN. SEE D170293
SOUTHERN COMPANY SERVICES, INC. FOR
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT-UNIT NO.1
P&ID CONDENSER VACUUM SYSTEM
SCHEMATIC
D-170064
CAD D170064I.DWG
DVY2005 JLD-02
REVISED PER ABN
DRAWING CATEGORY: CRITICAL


R-15B and C
SJAE Filtration
Recirc fan
TURBINE BUILDING ROOF EL. 265'-0"
OPERATING FLOOR
EL 189'-0
EL. 155'-0" MEZZANINE FLOOR GRADE
EL 137'-0" BASEMENT FLOOR
INTAKE LOUVER SHOWN IS TYP. OF 9. (3668A THRU 3668I)
MECHANICAL EQUIP ROOM EL 265'-0"
EQUIPMENT ROOM UNIT HEATER N1U41K005A-N
EQUIPMENT ROOM UNIT HEATER N1U41K005B-N
EQUIPMENT ROOM UNIT HEATER N1U41K005C-N
EQUIPMENT ROOM UNIT HEATER N1U41K005D-N
T.B. EQUIPMENT ROOM PROPELLER FAN N1U41C009A-N
OUTSIDE AIR T.B. EQUIPMENT ROOM PROPELLER FAN N1U41C009B-N
SW LP HEATER BAY AND TURBINE AREA
OPERATING FLOOR AIR HANDLING UNIT N1U41C001A-N
NW LP HEATER BAY AND GENERATOR AREA
OPERATING FLOOR AIR HANDLING UNIT N1U41C001B-N
NE TURBINE WORKING AREA
SE HEATER AND SWITCH GEAR AREA
FLOOR AIR HANDLING UNIT N1U41C001D-N
TO OPER. FLOOR ATMOS.
TO MEZZANINE FLOOR ATMOS.
NW LP HEATER BAY AND TURBINE AREA
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002E-N
SW LP HEATER BAY
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002A-N
NE SWITCH GEAR AREA
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002F-N
NW AIR EJECTOR & GLAND SEAL CONDENSER AREA
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002D-N
NW AIR EJECTOR & GLAND SEAL CONDENSER AREA
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002B-N
NW SGFP TURBINE AREA
MEZZANINE FLOOR AIR HANDLING UNIT N1U41C002C-N
STEAM JET AIR EJECTOR FILTRATION UNIT N1U41C005-N
DOOR UNIT HEATER N1U41K004-N
N1U41C600A N1U41C600B N1U41C600C
CONDENSATE PUMP EXHAUST FANS
SE SUMP PUMP AREA
BSMT. FLOOR AIR HANDLING UNIT N1U41C003B-N
NW CHEM. FEED AND CONDENSATE PUMP AREA
BSMT. FLOOR AIR HANDLING UNIT N1U41C003C-N
NE CIRCULATING OIL PUMP AND COMPRESSOR AREA
BSMT. FLOOR AIR HANDLING UNIT N1U41C003D-N
SW CONDENSATE PUMP & LUBE CENTER OIL AREA
BSMT. FLOOR AIR HANDLING UNIT N1U41C003A-N
WATER ANALYSIS RM AHU, N1U41C014-N
TURBINE BUILDING MAIN FILTRATION UNIT N1U41C004-N
RECIRCULATION FAN N1U41C007-N
EXHAUST FAN N1U41C006-N
TO INSIDE ATMOSPHERE
TO OUTSIDE ATMOSPHERE
MAKE-UP AIR
CAPPED DUCT
SERVICE WATER
3/4" COPPER DR LINE WITH 1' DEEP TRAP
NOTES:
1. SEE D-175020 FOR PROCESS FLOW DIAGRAM.
2. PHYSICAL DWG. OF THIS SYS. BY SSI, AND BECHTEL DWGS. D-175155, 156,157.
3. ** PROVIDED UNDER INSTRUMENTATION AND CONTROL CONTRACTS.
* PROVIDED BY VENDOR OF ASSOCIATED EQUIPMENT. ALL OTHER INSTRUMENTATION BY DUCTWORK SUBCONTRACTOR.
4. INSTRUMENT AIR TO BE PROVIDED BY S.S.I.
5. FOR MATERIAL AND EQUIPMENT DESCRIPTIONS SEE APPLICABLE INQUIRIES & SPECIFICATIONS.
6. WATER ANALYSIS ROOM IS SHARED FOR UNITS 1 & 2.
7. ALL TURBINE BUILDING AIR HANDLING UNIT AND WATER ANALYSIS RM. A/C UNIT THERMOSTAT TPNS NUMBERS WILL BE PRECEDED BY THE PREFIX N1U41.
8. SEE A-170774, SH. 8 & 9 FOR DETAILS.
9. MAIN FILTRATION UNIT HAS NO HEPA OR HECA FILTER MEDIA INSTALLED.
CAD D1750271.DWG R14OVY2 JOK-05
Southern Company Services, INC. for
ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT UNIT NO. 1
HVAC: P & I DIAGRAM TURBINE BUILDING
D-175027 REV. 21
OPERATIONS CRITICAL
D-175027 SH.1

4.3 STEAM JET AIR EJECTOR FILTRATION UNIT, N1U41C005-N Operation

NOTE

The SJAE discharge line drain trap drains to the Turbine Building Sump. IF the filtration unit is being placed in service due to a tube leak, THEN consideration should be given to EITHER isolating the drain trap OR routing the drainage to a poly bottle with a filtered vent. Use administrative controls for drain trap configuration should be considered. ☐

4.3.1 To place SJAE FILTRATION UNIT in FILTER operation, **perform** the following:

4.3.1.1 At LCS SJAE FILTRATION, N1U41G529-N **place** local control handswitch for SJAE filtration unit valves in FILTER. ☐

4.3.1.2 **Verify** open SJAE FILTER SUCT DMPR, N1U41HV3677B. ☐

4.3.1.3 **Verify** closed SJAE FILTER BYP DMPR, N1U41HV3677A. ☐

4.3.1.4 **Close** SJAE FILTER BYP MAN ISO, N1U41V018. ☐

NOTE

The SJAE discharge line drain trap drains to the Turbine Building Sump. ☐

4.3.1.5 If the filtration unit is being **placed** in service due to a tube leak, then consideration should be given to EITHER:

- **Isolating** the drain trap. ☐

OR

- **Routing** the drainage to a poly bottle with a filtered vent. ☐

4.3.2 To bypass SJAE filter, **perform** the following:

4.3.2.1 **Verify** open SJAE FILTER BYP MAN ISO, N1U41V018. ☐

4.3.2.2 At LCS SJAE FILTRATION, N1U41G529-N **place** local control handswitch for SJAE filtration unit valves in BYPASS. ☐

4.3.2.3 **Verify** closed SJAE FILTER SUCT DMPR, N1U41HV3677B. ☐

4.3.2.4 **Verify** open SJAE FILTER BYP DMPR, N1U41HV3677A. ☐

1. 055K1.06 035/NEW/RO/MEM 2.6/2.6/055K1.06/N///D

Unit 1 is at 70% power with the following conditions:

- R-15A, SJAE EXH, is in alarm.
- AOP-2.0, Steam Generator Tube Leakage, is in progress.
- The Turbine Building SO has placed the SJAE Filtration System in service.

Which one of the following completes the statement below?

After the SJAE Filtration system is placed in service, the reading on R-15A will _(1)_ and the SJAE Filtration system will _(2)_ .

	(1)	(2)
A.	decrease	be aligned in a recirc alignment
B.	remain the same	be aligned in a recirc alignment
C.	decrease	discharge to the Turbine Bldg roof
D.✓	remain the same	discharge to the Turbine Bldg roof

1. 056AK3.02 037/NEW//C/A 4.4/4.7/APE056AK3.02/N///

Unit 1 is operating at 100% when a LOSP occurred. The following conditions exist:

- The Emergency Diesel Generators failed to energize the ESF busses.
- The operating crew is conducting a secondary depressurization per ECP-0.0, Loss Of All AC Power.
- SG pressures are as follows:
  - 1A SG: 245 psig and lowering
  - 1B SG: 247 psig and lowering
  - 1C SG: 244 psig and lowering

Which one of the following completes the statements below?

Per ECP-0.0, this secondary pressure reduction is required to _(1)_.

The reason the secondary pressure reduction is required to be stopped at the SG pressure specified in ECP-0.0 is to prevent _(2)_.

	(1)	(2)
A.✔	be STOPPED	injection of accumulator nitrogen into the RCS
B.	CONTINUE	injection of accumulator nitrogen into the RCS
C.	be STOPPED	a challenge to the Integrity Critical Safety Function
D.	CONTINUE	a challenge to the Integrity Critical Safety Function

ECP-0.0:
17. Reduce intact SGs pressure to 260 psig.

ECB-0.0:
The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi).

Distracter analysis

A. Correct. First part is correct. Per ECP- 0.0, Reduce intact SGs pressure to 260 psig.

Second part is correct: Per ECB-0.0, [...] Should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS.

B. Incorrect. First part is incorrect (See A.1). Plausible since the limit in the background document is 160 psig. The limit in the procedure adds a 100 psig for margin of controllability and the applicant could confuse these two numbers and believe that the depressurization must continue.

Second part is correct (See A.2).

C. Incorrect. First part is correct (See A.1)

Second part is incorrect (See A.2). Plausible since this is the reason for the Tcold temperature limit of 280°F during the pressure reduction but NOT the reason for stopping at 260 psig.

D. Incorrect. First part is incorrect (See B.1)

Second part is incorrect (See C.2)

K/A: **056AK3.02**	Loss of Offsite Power - Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power
Importance Rating:	4.4 4.7
Technical Reference:	FNP-1-ECP-0.0, Loss Of All AC Power, Ver 26 FNP-0-ECB-0.0, Specific Background Document for FNP-1/2-ECP-0.0, Ver 3.1
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with (1) ECP-0.0, Loss of All AC Power; [...] (OPS-52532A03)
Question History:	NEW
K/A match:	This question presents a scenario where a **Loss of Offsite Power** occurs and the Emergency DGs fail to energize the ESF busses. The Applicant is required to **know the reason that the secondary depressurization is stopped at 260 psig (reasons for the actions contained in the EOP).**
SRO justification:	N/A

Section: Procedure

Unit 1 ERP Step: 17 **Unit 2 ERP Step:** 17 **ERG Step No:** 16

ERP StepText: Reduce intact SGs pressure to 260 psig.

ERG StepText: Depressurize Intact SGs To (O.08) PSIG

Purpose: To depressurize the intact steam generators

Basis: This step depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss. The advantages to performing this action, as well as restrictions that apply during the action, are detailed in Subsection 2.3.

During SG depressurization, SG level must be maintained above the top of the SG U-tubes in at least one SG. Maintaining the U-tubes covered in at least one SG will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. Substep a requires that SG level be in the narrow range in at least one SG before SG depressurization is initiated in substep b. If level is not in the narrow range in at least one SG, RNO a instructs the operator to maintain maximum AFW flow until narrow range level is established in one SG. When narrow range level is established, SG depressurization can be started or continued via substep b.

Substep b instructs the operator to reduce SG pressures by depressurizing the intact SGs. Depressurization should be accomplished by opening the PORVs on the intact SGs to establish a cooldown rate of less than 100°F/hr in the RCS cold legs. By maintaining RCS cooldown rate less than 100°F/hr, the RCP seal temperatures are reduced in a controlled manner to prevent thermal shock. The step is structured assuming that the operator can open and control SG PORVs from the control room. This structure assumes that the PORVs are air-operated and have dc control power and pneumatic power (i.e., either air reservoirs or nitrogen bottles) available. Some plants may not have the capability to open the SG PORVs from the control room. These plants should evaluate their capability to accomplish this step locally via PORV handwheels. Such an evaluation should consider accessibility and communications necessary to accomplish local PORV operation.

Once depressurization is initiated, the depressurization rate should be controlled to maintain RCS cooldown rate near 100°F/hr. The depressurization rate should be sufficiently fast to expeditiously reduce SG pressures, but not so fast that SG pressures and RCS cooldown cannot be controlled. It is important that the depressurization not reduce SG pressures in an uncontrolled manner that undershoots the pressure limit, thus permitting potential introduction of nitrogen from the accumulators into the RCS.

During SG depressurization, AFW flow may have to be increased to maintain the required SG narrow range level. Control of AFW flow will have to be performed from the control room or locally depending on plant specific design. Full AFW flow should be established to any SG in which level drops out of the narrow range.

Section: Procedure

RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function. This check is included in substep c since guideline ECA_0.0 has priority over the Function Restoration Guidelines and the operator is instructed to not implement a Function Restoration Guideline even if a Critical Safety Function challenge is detected by the Critical Safety Function Status Trees. Consequently, Step 16c implicitly protects the Integrity Critical Safety Function. The SG depressurization should not result in a challenge to the Integrity Critical Safety Function since the resultant RCS cold leg temperatures should not approach the temperature limit (i.e., T2 temperature) at which a challenge will exist.

Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored.

The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi). To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tanks pressure (P1), initial nitrogen gas volume (V1), and final nitrogen gas volume (V2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

The RCS pressure at empty tank conditions (P2) is determined from:

$$P_1V_1^{\gamma} = P_2V_2^{\gamma}$$

where $\gamma = 1.0$ for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta p from P2 and adding the margin to controllability. The RCS to SG delta p should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the RCS.

In addition to the accumulator nitrogen limitation on SG depressurization, Subsection 2.3 also discusses the core criticality concern that exists when the RCS is cooled down. This concern was evaluated for the reference plant for various fuel burnups, assuming equilibrium xenon, all rods inserted and no addition of borated water (which will occur when RCS pressure is decreased below approximately 650 psig). The evaluation showed that only at the end of core life did the criticality concern (418°F for the reference plant) become more limiting than the accumulator nitrogen concern (410°F saturation for the reference plant). For the assumed conditions, the accumulator nitrogen concern dominated the criticality concern. Consequently, this step is structured to explicitly address the accumulator nitrogen concern. For the majority of circumstances, this will cover the criticality concern. The next step explicitly addresses the criticality concern by terminating the SG depressurization if core shutdown is lost.

Knowledge:

1. PTS concerns o RCP seal integrity concerns o Relationship of RCP seal leakage to RCS pressure o Basis for SG pressure limit on SG depressurization.
2. Basis for maintaining SG level above U-tubes.

References: DW-06-014
DW-06-005

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Accumulator nitrogen injection into the RCS may result from reduction of SG pressure to less than 160 psig.

****

NOTE:
- The SGs should be depressurized at a rate sufficient to maintain a cooldown rate in the RCS cold legs near 100°F/hr. This will minimize RCS inventory loss while cooling the RCP seals in a controlled manner.
- PRZR level may be lost and reactor vessel upper head voiding may occur due to depressurization of SGs. Depressurization should not be stopped to prevent these occurrences.

___17 **Reduce intact SGs pressure to 260 psig.**

17.1 Check at least one intact SG narrow range level - GREATER THAN 31%{48%}.

17.1 Perform the following:

17.1.1 Maintain maximum AFW flow to intact SGs until narrow range SG level greater than 31%{48%} in at least one SG.

TDAFWP
SPEED CONT
[] SIC 3405 adjusted to 100%

17.1.2 WHEN narrow range level in at least one intact SG is greater than 31%{48%}, THEN perform steps 17.2 through 17.7.

17.1.3 Proceed to step 18.

Step 17 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

17.2 WHEN P-12 light lit,
THEN perform the following.

17.2.1 Block low steam line pressure SI.

STM LINE PRESS SI
BLOCK - RESET
[] A TRN to BLOCK
[] B TRN to BLOCK

17.2.2 Verify blocked indication.

BYP & PERMISSIVE
STM LINE ISOL.
SAFETY INJ.
[] TRAIN A BLOCKED light lit
[] TRAIN B BLOCKED light lit

17.3 WHEN pressurizer pressure less than 2000 psig,
THEN perform the following.

17.3.1 Block low pressurizer pressure SI.

PRZR PRESS SI
BLOCK - RESET
[] A TRN to BLOCK
[] B TRN to BLOCK

17.3.2 Verify blocked indication.

BYP & PERMISSIVE
PRZR. SAFETY
INJECTION
[] TRAIN A BLOCKED light lit
[] TRAIN B BLOCKED light lit

Step 17 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

17.4 Dump steam from intact SGs to maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR

17.4.1 Locally control intact SG atmospheric relief valves with handwheel. (127 ft, AUX BLDG main steam valve room)

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

17.5 Maintain at least one intact SG narrow range level - GREATER THAN 31%{48%}.

17.5.1 Control TDAFWP speed.

TDAFWP
SPEED CONT
[] SIC 3405 adjusted

17.5.2 Control TDAFWP flow control valve with manual handwheels.

SG	1A	1B	1C
TDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3228A	[]3228B	[]3228C

Response NOT Obtained:

17.5 IF all intact SG narrow range levels less than 31%{48%}, THEN perform the following.

a) Locally close intact SG atmospheric relief valves with handwheels.

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

b) WHEN at least one intact SG narrow range level greater than 31%{48%}, THEN continue dumping steam from intact SGs.

17.6 Check RCS cold leg temperatures - GREATER THAN 280°F{280°F}.

RCS COLD LEG TEMP
[] TR 410

Response NOT Obtained:

17.6 Perform the following.

17.6.1 Locally control intact SG atmospheric relief valves with handwheels to stop SG pressure reduction.

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

17.6.2 Proceed to step 18.

Step 17 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

17.7 Check intact SG pressures - LESS THAN 260 psig.

17.7 Perform the following.

17.7.1 WHEN intact SG pressures less than 260 psig, THEN perform step 17.8.

17.7.2 Proceed to step 18.

17.8 Locally control intact SG atmospheric relief valves with handwheels to maintain intact SG pressure at 260 psig.

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

___18 **Check startup rate - LESS THAN OR EQUAL TO ZERO.**

SR1(2)
S/U RATE
[] NI 31D
[] NI 32D

IR1(2)
S/U RATE
[] NI 35D
[] NI 36D

18 Establish subcriticality.

18.1 Locally control intact SG atmospheric relief valves with handwheels to raise SG pressure.

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

18.2 WHEN startup rate less than or equal to zero, THEN locally control intact SG atmospheric relief valves with handwheels to maintain stable pressure in all intact SGs.

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. 059A4.01 038/BANK/DIABLO CANYON 12-07/MEM 3.1*/3.1*/059A4.01/N///

Unit 1 is operating at 100% power when the 1B SGFP trips.

Which one of the following completes the statements below for the 1B SGFP?

The HIGH PRESS. GOV. VALVE CLOSED light is __(1)__ .

The LOW PRESS. GOV. VALVE CLOSED light is __(2)__ .

	(1)	(2)
A.✓	LIT	LIT
B.	LIT	NOT lit
C.	NOT lit	LIT
D.	NOT lit	NOT lit

U-161792 - Tab 3, Section 5, Page 1

When a trip condition occurs, signals from the electronic controller close the steam valves.

Ran on desktop simulator. Inserted SGFP trip from 100% power and both governor valves went closed.

From OPS-52104C Ver 2 pg 15 -

Initially, as the feed pump turbine accelerates from operation on the turning gear to operating speed, both the LP and HP stop valves are open. The first governor valve to open on an increase speed signal from the control system is the LP governor. Since reheat steam is not available, the turbine speed does not increase. Once the LP governor valves begin to reach their fully open position, the HP governor valve begins to open. The turbine now accelerates to the demanded speed using the main steam supply via the HP governor valve.

As main turbine load is increased, reheat steam pressure in the shell side of the MSRs also increases. At approximately 25 percent main turbine power, the reheat steam pressure is high enough to cause the feed pump turbine speed to increase. In an effort to maintain the desired feed pump turbine speed, the control system begins to shut the HP governor valve. Once the HP governor valve approaches the fully shut position, the control system starts closing the LP governor valves. During 100 percent power operation, the governor valve alignment is as follows:

1. The HP governor valve is fully shut.
2. The LP governor valve is throttled partially shut and consequently controls feed pump turbine speed.

** Some validators selected the correct answer but stated that they struggled with*

determining the response of governor valves when the SGFP tripped.

Distracter analysis

A. Correct. First part is correct. The Servo controller shuts the HP governor valve when the SGFP turbine trips.

Second part is correct. The Servo controller shuts the LP governor valve when the SGFP turbine trips.

B. Incorrect. First part is correct (See A.1). Logical connection to the second part because this is the normal position of the HP governor valve at 100% power.

Second part is incorrect (See A.2). Plausible if the applicant thinks that the HP and LP Governor Valves remain in their pre-trip positions on a SGFP trip because the HP and LP STOP valves go shut.

C. Incorrect. First part in incorrect (See A.1). Plausible if the applicant thinks that the HP and LP Governor Valves remain in their pre-trip positions on a SGFP trip because the HP and LP STOP valves go shut. If the applicant thought the HP governor valve controlled speed at high power then it would make this a plausible correct answer coupled with the second part.

Second part is correct (See A.2). Logical connection to the first part if the applicant thinks that the HP governor valve controls speed at high power.

D. Incorrect. First part is incorrect (See A.1) Plausible if the applicant thinks that the HP and LP Governor Valves remain in their pre-trip positions on a SGFP trip because the HP and LP STOP valves go shut. Logical connection to the second part if the applicant thinks that both the HP and LP governor valves are open at 100% power.

Second part is incorrect (See A.2) Plausible if the applicant thinks that the HP and LP Governor Valves remain in their pre-trip positions on a SGFP trip because the HP and LP STOP valves go shut. Logical connection to the first part if the applicant thinks that both the HP and LP governor valves are open at 100% power.

K/A: **059A4.01**	Main Feedwater (MFW) System - Ability to manually operate and monitor in the control room: MFW turbine trip indication
Importance Rating:	3.1* 3.1*
Technical Reference:	U-161792, SGFP Drive Turbine and Accessories, Ver 12
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if entry into AOP-13, Loss of Main Feedwater is required. (OPS-52520M02)
Question History:	DIABLO CANYON 12-07
K/A match:	Requires the applicant to monitor MCB indications and determine the proper SGFP GOV valve positions on a SGFP trip.
SRO justification:	N/A

5.81.4 **Continue** opening main feed regulating valves as power increases while keeping main feed regulating bypass valves in AUTO and at a reliable controlling position. This will continue until approximately 35% power. Control will then be shifted to the main feed regulating valves per FNP-1-UOP-3.1, Power Operation. ☐

NOTE

The preferred supply for 1A, 1B and 1C 4160V busses is from the unit auxiliary transformers. At the Shift Supervisor's direction, the busses may remain aligned to the startup transformers. ☐

5.82 IF desired, **transfer** 4160V buses 1A, 1B and 1C to the unit auxiliary transformers per FNP-1-SOP-36.2, 4160V AC Electrical Distribution System. ______

______ / ______

NOTE

IF the heater breakers are closed per FNP-1-SOP-26.0A, CIRCULATING WATER SYSTEM, the heaters will automatically de-energize and energize as the cooling tower fans are started and stopped. ☐

5.83 * At the 1A cooling tower local control station/panel, **perform** the following per FNP-1-SOP-26.0, CIRCULATING WATER SYSTEM:

5.83.1 **Start** the non-running fans on 600V LC 1U. ☐

5.83.2 **Start** the non-running fans on 600V LC 1V. ☐

5.83.3 **Start** the non-running fans on LC # 11. ☐

5.84 **Notify** ACC that the generator is ready for additional load. ☐

5.85 **Go To** FNP-1-UOP-3.1, Power Operation, for increasing reactor power greater than 25%. ☐

5.86 *Shift Support Supervisor complete Data Sheet 1. ______

______ / ______

____

Shift Supervisor Review

5.7.9 IF a main power transformer (MPT) has been replaced and it is desired to obtain third harmonic voltage readings during the initial power ascension following MPT replacement, THEN **perform** the following:

- **Hold** power ascension at 380 MWs ☐
- **Adjust** MVARs as directed by the applicable work order. ☐
- **Contact** the ES Manager to gather the required data. ☐

NOTE

In the following step, the valves will stay open on the non-running SGFP if both stop valves are closed. The valves should close on the running SGFP.

5.7.10 **Perform** the following as turbine load exceeds 40%.

5.7.10.1 **Verify** that air has been restored to 1A SGFP casing and steam line drain bypass valves: (Ref. D-170111)

- IA TO N1N39V627A 1A SGFP HP STM LINE DRN ORIF BYP, N1P19V930A (East of 1A HDT) ☐
- IA TO N1N39V628A 1A SGFP HP STOP BEFORE SEAT DRN ORIF BYP, N1P19V931A (East of 1A HDT) ☐
- IA TO N1N39V629A 1A SGFP LP STOP BEFORE SEAT DRN ORIF BYP, N1P19V932A (East of 1B HDT) ☐

5.7.10.2 **Verify** that air has been restored to 1B SGFP casing and steam line drain bypass valves: (Ref. D-170111)

- IA TO N1N39V627B 1B SGFP HP STM LINE DRN ORIF BYP, N1P19V930B (East of 1A HDT) ☐
- IA TO N1N39V628B 1B SGFP HP STOP BEFORE SEAT DRN ORIF BYP, N1P19V931B (East of 1A HDT) ☐
- IA TO N1N39V629B 1B SGFP LP STOP BEFORE SEAT DRN ORIF BYP, N1P19V932B (East of 1B HDT) ☐

CAUTION

During the startup of a second SGFP or any other activity such as hotwell blowdown SGFP suction pressure may be adversely affected.

5.7.11 **Start** the second SGFP per FNP-1-SOP-21.0, Condensate and Feedwater System. ☐

5.7.11.1 Once the second SGFP is running, **place** it in service supplying main feed to reduce secondary contaminants. ☐

NOTES

The heater drain pumps should be started at approximately 45% power or when HDT level is controlling on the dump. (AI 2004201198)

If starting the HDTP's early, refer to guidance for Heater Drain Tank Pump operations per FNP-1-SOP-21.0, section titled "Heater Drain Tank Pump Operations".

Due to the initial decrease when the drain pump is started, heater drain tank level should be above normal level prior to starting the pump

5.7.12 **Start** 1A and 1B heater drain pumps.

- **Place** handswitch for 1A HDP, N1N26P001A in START, AND **verify** handswitch red light illuminated. ☐
- **Place** handswitch for 1B HDP, N1N26P001B to START AND **verify** handswitch red light illuminated. ☐

5.7.13 **De-energize** 1A and 1B heater drain pump motor heaters.

- **Have** handswitch, START HTR DRN PUMP 1A MOTOR HTR, N1R58G542-N **placed** to OFF. (TB 137 West wall) ☐
- **Have** handswitch START HTR DRN PUMP 1B MOTOR HTR, N1R58G543-N **placed** to OFF. (TB 137 West wall) ☐

5.8 Prior to exceeding 48% reactor power, verify steam generator chemistry in spec within limits required by FNP-0-CCP-202, CHEMICAL SPECIFICATIONS.

Steam Generator Feed Pump Turbine Steam Supply

Main steam and reheat steam from moisture separator reheaters (MSRs) 1A and 2A are supplied to the feed pump turbine through separate headers, stop valves, and governor valves. Main steam is used to initially roll the turbine and supply the driving force until reheat steam is available in sufficient quantity. Main steam supplies the steam requirements of the feed pump turbine to approximately 25 percent power when reheat steam, controlled by the LP governor valves, takes over.

Bumpless transfer between the steam supplies is accomplished by the control system.

Initially, as the feed pump turbine accelerates from operation on the turning gear to operating speed, both the LP and HP stop valves are open. The first governor valve to open on an increase speed signal from the control system is the LP governor. Since reheat steam is not available, the turbine speed does not increase. Once the LP governor valves begin to reach their fully open position, the HP governor valve begins to open. The turbine now accelerates to the demanded speed using the main steam supply via the HP governor valve.

As main turbine load is increased, reheat steam pressure in the shell side of the MSRs also increases. At approximately 25 percent main turbine power, the reheat steam pressure is high enough to cause the feed pump turbine speed to increase. In an effort to maintain the desired feed pump turbine speed, the control system begins to shut the HP governor valve. Once the HP governor valve approaches the fully shut position, the control system starts closing the LP governor valves. During 100 percent power operation, the governor valve alignment is as follows:

1. The HP governor valve is fully shut.
2. The LP governor valve is throttled partially shut and consequently controls feed pump turbine speed.

SGFP Low Pressure Oil System and EH Fluid Systems

(Figure 6, Table 6)

The relationships of the various elements that comprise the LP oil system and the high pressure fluid system are shown diagrammatically in Figure 6. The two systems are separate and distinct from one another, serving different functions and each employing a different hydraulic medium. The LP oil system uses refined mineral oil, while the HP system requires a synthetic fluid. The two systems are linked together by the interface trip valve.

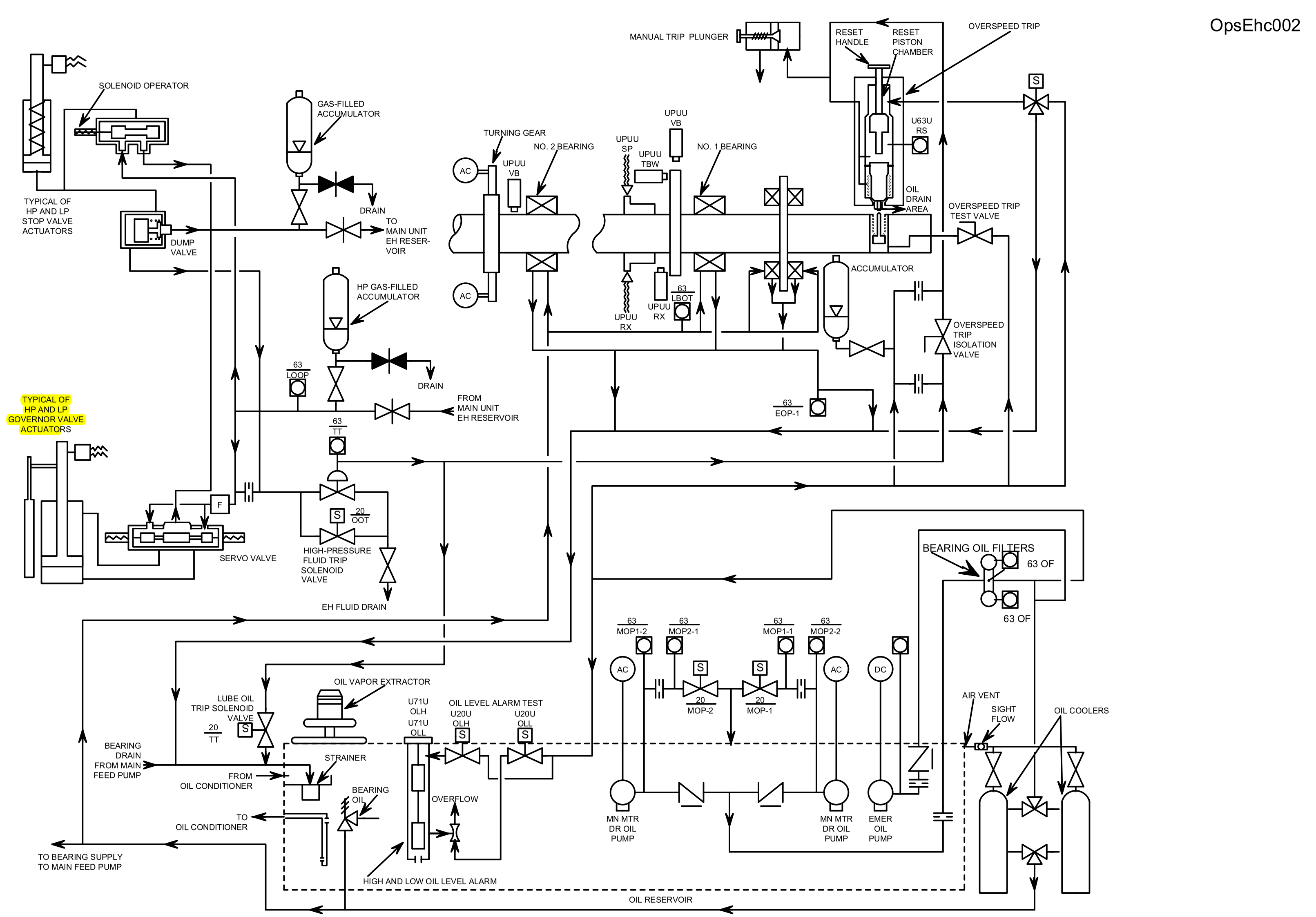


FIGURE 6 - SGFP Low-Pressure Oil & EH Fluid Systems

with the reset pressure switch (63/RS). The pushbutton and indicating light remain illuminated until the turbine is again tripped.

The overspeed trip mechanism can be tested during normal operation without actually overspeeding the turbine. The test arrangement includes a hand operated swing valve for isolating the trip mechanism and a second hand operated swing valve for testing. These test valves are located on the side of the turbine console.

To perform the test, first close the spring-loaded isolation valve to prevent the turbine from tripping. This valve must be maintained closed throughout the test and afterward until the trip mechanism has been reset. Next turn the test valve handle to open it to admit high pressure oil through passages to the chamber beneath the overspeed trip weight within the rotor shaft. The oil pressure will cause the weight to move outward to its trip position. Sufficient movement of the weight to have normally tripped the turbine will be evident by the audible click of the weight striking the trip trigger. Further evidence that the trip mechanism has functioned will be the turbine "Latch" pushbutton light and indicating light on the instrument console going off.

After test, return the test valve handle to the closed position while still maintaining the isolation valve closed. Reset the overspeed trip and verify by observing the turbine "Latch" pushbutton or indicating light on the instrument console illuminating. After assurance the trip is reset, release the isolation valve handle enabling the valve to spring return to its normally open position. The turbine overspeed protection is again restored.

A WORD OF CAUTION: Failure to maintain the isolation valve closed throughout the test or releasing the isolation valve before resetting the trip mechanism will cause the turbine to be tripped. Opening the test valve before closing the isolation valve will likewise trip the turbine.

HIGH PRESSURE FLUID SYSTEM

The function of the "EH" (Electro-Hydraulic) high pressure fluid system is to provide the motive force which positions the turbine steam valves in response to electrical commands from the electronic controller, acting through hydraulic servo-actuators. The high pressure fluid requirements are supplied from the main turbine unit fluid system. Detailed description of the fluid supply unit, its components and operation will be found in the main turbine instruction book.

The major elements of the "EH" fluid system are the H.P. accumulator, the drain accumulator and the steam valve servo-actuators.

The H.P. and drain accumulators are mounted on a rack at the inlet end of the turbine bedplate, the assembly of which is shown in Figure 2. Both accumulators are the bag type with a 5 gallon capacity and charged with nitrogen gas admitted through charging valves located on top of each.

The function of the drain accumulator is to absorb excess drain pressure from the servo-actuators which may develop during a trip, and discharge it gradually back to the fluid supply unit. The accumulator thus prevents drain pressure build-up which would impede servo-actuator function. The H.P. accumulator serves as a pressure storage tank in the fluid supply line. As such, this accumulator prevents drastic pressure drops during normal transient conditions. Accumulator gas pressure should be checked periodically and recharged if necessary. Checking gas pressure may be done while the unit is in service by first closing the fluid supply shut-off valve, then opening the smaller drain valve. The gas pressure may then be read on the gauge mounted on top of the accumulator. Since the temperature surrounding the gas will affect this pressure, pressure checks should be made after surrounding temperature stabilizes. The accumulator rack houses the shut-off and drain valves for testing purposes.

The accumulator rack also serves as the dispersing station between the high pressure fluid supply unit and the individual steam valve servo-actuators. A manifold with internal passages directs the flow of high pressure fluid, emergency trip fluid as well as drain fluid.

Contained within the manifold is an orifice which supplies metered high pressure fluid to the emergency trip header. Loss of emergency trip

SERVO-ACTUATOR

GENERAL

The servo-actuator shown in Figure 1 is a mechanism designed to produce rapid and precise steam valve movement by electrical control of high pressure fluid. It consists essentially of a stainless steel block (3), drilled to facilitate fluid flow, to which is assembled all of the components necessary to translate electrical signals from the electronic controller into steam valve opening or closure. The servo-actuator, by design, is the double acting type in that fluid power is utilized to move the steam valve in either the open or closed direction. The modular design of the servo-actuator lends itself to a wide range of applications with a minimum of external piping connections.

OPERATION

High pressure fluid is admitted initially through the filter (14); then, through communicating passages, to the servo-valve (1). Within the servo-valve, the fluid enters a chamber containing a shifting spool whose movement uncovers ports which direct fluid flow to and from the hydraulic cylinder (5).

The servo-valve is an electrically-controlled valve which responds to voltage signals from the electronic controller. A change in the voltage balance between two coils causes the servo-valve spool to shift, with the resulting movement of the steam valve being dependent on the direction of spool shift. Should the voltage signal from the electronic controller create an imbalance to initiate valve opening, the servo-valve spool shift connects high pressure fluid to the chamber above the piston in the hydraulic cylinder and connects the chamber beneath piston to drain, resulting in the piston rod moving the valve in the open direction. Conversely, when voltage imbalance seeks to close valve the spool shift connects high pressure fluid to the chamber beneath the piston in the hydraulic cylinder and connects chamber above piston to drain, resulting in piston rod moving valve in the closed direction.

The Linear Variable Differential Transformer (L.V.D.T.) (7) provides the feed-back signal to the electronic controller to confirm valve movement. The L.V.D.T. body is fastened to the control block (3), while the movable slide is rigidly attached to the hydraulic cylinder piston rod and duplicates rod movement. Movement of the L.V.D.T. slide generates a feed-back signal to the electronic controller in opposition to the control input signal to the servo-preamplifier. When the L.V.D.T. signal balances the control input signal, the servo-valve spool returns to its null (balanced voltage) position and the flow of high pressure fluid stops. The position of the valve is then maintained until another control input signal is received by the servo-valve.

In the event of loss of signal to the servo-actuator, the servo-valve is mechanically biased to move the hydraulic cylinder piston upward, thus closing the steam valve.

When a trip condition occurs, signals from the electronic controller close the steam valves.

LIST OF PARTS

The following list has been compiled to facilitate ordering Spare or Renewal Parts. In ordering parts it is of utmost importance to give the Turbine Serial Number, Instruction Leaflet 1150-675, Item Number and Name of Part.

Item	Name
1	Servo-Valve
2	Terminal Block
3	Control Block
4	Bushing
5	Hydraulic Cylinder
6	Lever (L.V.D.T.)
7	Linear Variable Differential Transformer (L.V.D.T.)
8	Cover (L.V.D.T.)
9	Cover (Actuator)
10	Mounting Block (L.V.D.T.)

1. 059AK2.01 039/BANK/FNP 06/MEM 2.7/2.8/APE059AK2.01/N///

Unit 2 is operating at 100% power when a SG tube leak occurs.

Which one of the following completes the statement below?

When R-23A, SGBD HX OUTLET, alarms, it will cause _____ , to automatically close.

A. HV-7614A/B/C, 1A/B/C SGBD ISO

B.✓ FCV-1152, SGBD HX OUTLET FCV

C. RCV-023B, SGBD DISCH TO ENVIRONMENT

D. HV-7697A/B, 7698A/B and 7699A/B, STM GEN 1A/B/C BLDN ISO

FH1

Automatic Actions:

R23A :(Steam Generator Blowdown Processing) closes 1-BD-FCV-1152 S/G Blowdown Heat Exchanger Discharge Valve.

R-23A is in the SGBD line after FCV-1152 but before the SGBD Surge tank. R-23B is after the surge tank and is the last rad monitor and isolation signal before an accidental release would make it to the environment. R-23 A and B are often confused on the functions and locations. Two other sets of valves isolate SGBD due to other conditions, such as High Penetration room pressure and AFW autostart. All of these valves complete the same function but for different reasons.

Distracter analysis

A. Incorrect. See B. Plausible since these valves will isolate SG Blowdown (SGBD) and automatically close on an AFW autostart. The applicant could believe they also close on a high radiation signal.

B. Correct. R-23A automatically closes FCV-1152. FCV-1152 also closes on SGBD high ST level, High pressure in the SGBD system and High flow.

C. Incorrect. See B. Plausible since R-23B automatically closes RCV-23B and the applicant could confuse which radiation monitor closes which valve. RCV-023B will isolate SGBD to the environment and is downstream of FCV-1152.

D. Incorrect. See B. Plausible since these valves are two series isolation valves located inside the containment on each line from the steam generator. The air-operated isolation valves (7697A/B, 7698A/B, 7699A/B) automatically close when high pressure (0.28-0.33 psig) is sensed in any room outside the containment where the blowdown piping, upstream of the heat exchanger, is located. Since these valves isolate on High pressure in the PPRs, they could be confused with closing signals for FCV-1152.

K/A: 059AK2.01	Accidental Liquid Radwaste Release - Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-liquid monitors
Importance Rating:	2.7 2.8
Technical Reference:	FNP-2-ARP-1.6, FH1 - RMS HI-RAD, Ver 70
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Radiation Monitoring System to include those items in Table 4-Remote and Local Indications and Controls (OPS-40305A02).
Question History:	FNP 06
K/A match:	In this scenario, **a SG tube leak results in an accidental liquid radwaste release. The applicant is required to know the interrelations between R-23A and the SGBD system that will terminate the accidental liquid radwaste release.**
SRO justification:	N/A

NOTES (CONT'D)

20. PROCESS SHOWN IS NORMAL REACTOR POWER OPERATION WITH THE DEMINERALIZERS BYPASSED.

NOTES (CONT'D)

13. THE FOLLOWING VALVE AND INSTRUMENT NUMBERS ARE APPLIED TO THE SIMILAR LOCATIONS ON LINES CBB-6 & CBB-7.

AIR OPERATED GLOBE VALVES

LINE	BECHTEL NO.	WESTINGHOUSE NO.
CBB-6	QV006B	7698B
CBB-6	QV005B	7698A
CBB-7	QV006C	7699B
CBB-7	QV005C	7699A

MANUALLY OPERATED GLOBE VALVES

LINE	BECHTEL NO.
CBB-6	QV002B
CBB-6	QV004B
CBB-7	QV002C
CBB-7	QV004C
CBB-9	QV001B
CBB-10	QV001C
CBB-6	QV009B
CBB-6	QV010B
CBB-7	QV009C
CBB-7	QV010C
CBB-6	QV211B
CBB-6	QV212B
CBB-7	QV211C
CBB-7	QV212C

14. FOR COMPLETE INSTRUMENT NUMBERS, REFER TO INSTRUMENT INDEX B-205803.
15. SAMPLE POINT LOCATED AS CLOSE AS POSSIBLE TO THE STEAM GENERATOR.
16. MULTI LOOP POWER SUPPLY CONTROL BOARD MTD. (NOT SHOWN).
17. SENSOR LOCATIONS ARE SHOWN ON D-205146.
18. CONTENTS DELETED
19. VALVES NV205A, B, & C ARE PROVIDED BY BECHTEL.

NOTES:

1. PLANT LAYOUT SHOULD ENSURE SUFFICIENT NPSH FOR DRAIN PUMP.
2. NEEDLE VALVE INSERTED TO PRESET RATE OF SET POINT AFTER HI/LO RANGE TRANSFER. NEEDLE VALVE AND 3-WAY SOLENOID VALVE SUPPLIED WITH TC-1150.
3. ALL Ⓦ ITEM NO. ON THIS DWG. ARE SHOWN WITHOUT PREFIX ALASB.
4. VALVE CLOSES ON AN AUTOMATIC AUXILIARY FEEDWATER SYSTEM START SIGNAL, OR ON MANUAL ISOLATION SIGNAL (REF. STEAM SYSTEMS DESIGN MANUAL, SECTION V-8 WCAP7451).
5. SBP-STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM PANEL.
6. STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM PANEL DRAWING Ⓦ 271C880.
7. LOW PRESSURE PORTION OF SAMPLING SYSTEM MUST BE 150 LB. DESIGN.
8. CONTAINMENT ISOLATION VALVES ARE LOCATED IMMEDIATELY OUTSIDE THE CONTAINMENT BOUNDARY.
9. CONTENTS DELETED.
10. FOR P&ID LEGEND SEE DWG. D-175044.
11. FOR EQUIPMENT LIST SEE DWG. D-205070.
12. ALL LINE AND VALVE NUMBERS ON THIS DRAWING ARE PREFIXED BY Q2G24 OR N2G24 UNLESS NOTED OTHERWISE.

REFERENCES:

FOR ALPHA REFERENCES SEE FLOW DIAGRAM LEGEND Ⓦ 1210E88.

TABLE A
ROOM PRESSURE SENSING INSTRUMENTATION

INSTRUMENT	TRAIN
PSH2853A	A
B	B
C	A
D	B
E	A
F	B
G	A
H	B
I	A
PSH2853J	B

THIS DRAWING WAS ORIGINALLY REPRODUCED FROM WESTINGHOUSE DWG. 1210E96.

THE UNIT 1 COUNTERPART FOR THIS P&ID IS D-175071 SH.1.

CAD D2050711.DWG
ACAD2000 | DFV - 02

Southern Company Services, INC. for
ALABAMA POWER COMPANY
PROJECT: J. M. FARLEY NUCLEAR PLANT - UNIT 2
SUBJECT: P&ID - STM. GEN. BLWDN. PROC. SYSTEM

FOR SIGNATURES
SEE MICROFILM
COPY

SCALE NONE B/M N/A
SHEET 1 OF — SHEETS
D-205071

REV. NO. 17 DATE 10-24-03 REVISED PER ABN 03-0-2602, REV. 1 (DFV AJC DLL)
REV. NO. 16 DATE 7-30-02 REDRAWN PER ABN 02-2-2405, REV. 0 (JAB LCF DEW)

OPERATIONS CRITICAL

LOCATION ED3

D3
AUX BLDG RM PRESS HI

SETPOINT: 0.28 + 1.0 PSIG - 0.0

ORIGIN: Any of the following pressure switches:

Q2N12PSH2850A (TD Aux Feedwater Pump Room)
2850B (100' El Equipment Room)
2850C (MD Aux Feedwater Pump Room)
2850D (TD Aux Feedwater Pump Room)
2850E (MD Aux Feedwater Pump Room)
2850F (100' El Equipment Room)

Q2G12PSH2851A (Recycle Holdup Tank Room)
2851B (BTRS Valve Compartment)
2851C (BTRS Valve Compartment)
2851D (BTRS Valve Compartment)
2851E (BTRS Valve Compartment)
2851F (Recycle Holdup Tank Room)

Q2G21PSH2852A (100' El Piping Penetration Room)
2852B (Letdown Heat Exchanger Room)
2852C (Letdown Heat Exchanger Room)
2852D (Letdown Heat Exchanger Room)
2852E (Letdown Heat Exchanger Room)
2852F (Recycle Holdup Tank Room)
2852G (Recycle Holdup Tank Room)
2852H (100' El Piping Penetration Room)

Q2G24PSH2853A (121' El East Corridor)
2853B (121' El East Corridor)
2853C (Blowdown Heat Exchanger Room)
2853D (Blowdown Heat Exchanger Room)
2853E (121' El East Corridor)
2853F (121' El East Corridor)
2853G (Recycle Evap Package Room)
2853H (Recycle Evap Package Room)
2853I (Recycle Evap Package Room)
2853J (Sluice Filter Room)

Either letdown line isolation valve Q2E21HV8175A OR Q2E21HV8175B handswitch on the PRIP taken to the CLOSE position denergizes the “MR” relay This will cause this alarm while the handswitch is held in CLOSE.

PROBABLE CAUSE

1. A High Energy Pipe break in any of the listed locations.
2. Loss of DC in the Multiplying Relay Cabinets.
3. Loss of DC to letdown isolation valves HV8175A or B.
4. Closing letdown isolation valves HV8175A or B.

AUTOMATIC ACTION

1. IF the alarm is due to a PSH in the 2851A, B, D (C, E, F) or 2852A, B, C, F (D, E, G, H) series, THEN LTDN LINE PENE RM ISO Q2E21HV8175A(B) in the Letdown line closes.
2. IF the alarm is due to a PSH in the 2853A, C, E, G, I (B, D, F, H, J) series, THEN 2A, 2B, and 2C SGBD PENE RM ISO Q2G24HV7697A(B), 7698A(B) and 7699A(B) close.

OPERATOR ACTIONS

1. IF letdown line isolation has occurred, THEN **Go To** FNP-1-AOP-16.0, CVCS MALFUNCTION.
2. **Dispatch** Operations and Health Physics personnel to survey the rooms listed to identify the cause of the high pressure.
3. **Isolate** any indicated piping breaks.
4. **Refer To** Technical Specification TR 13.3.2 for any required actions.

NOTE:
- Steps 5 thru 13 reflect manual actions specified in the FSAR, Appendix 3K, to mitigate auxiliary steam system breaks, or plant hot water heating system breaks. (ref. ABN 93-0-0042 and IN 90-53)
- Steam releases resulting from system piping breaks can affect electrical equipment in much the same way as a fire (i.e., result in short circuits or grounded circuits). Therefore, FNP-0-AOP-29.0, PLANT FIRE, may provide additional guidance depending on the location and severity of the break.
- Closing Aux Stm to Aux Bldg N2P20V525 will result in a loss of Gland Seal Steam to Main Turbine and SGFPs if Gland Seal is being supplied by Aux Steam.

5. IF high pressure due to Aux Stm & Condensate Recovery system pipe break, THEN **isolate** break by closing N2P20V525 from MCB OR by closing both N2P20V024 on the Aux Bldg roof AND N2P20V026 in the 100 ft. Plant Heating Equipment Room.
6. IF TDAFWP starts due to spurious signals and is not needed, THEN **secure** by closing the Trip Throttle Valve Q2N12MOV3406.

LOCATION FH1

H1
RMS HI-RAD

SETPOINT: 1. Variable, as per FNP-2-RCP-252

ORIGIN: Any of the below listed Area, Process or Gaseous and Particulate Monitors: R01B, R02, R04, R05, R06, R07, R08, R09, R10, R11, R12, R13, R14, R15, R17A, R17B, R18, R19, R20A, R20B, R21, R22, R23A or R23B

PROBABLE CAUSE

1. High Radiation Level in the System, Area or at the Component monitored.
2. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions.

AUTOMATIC ACTION

1. The following actions will occur if a High Radiation Alarm is actuated on the associated Radiation Monitor.
   a) R14: (Plant Vent Gas) closes Waste Gas Release Valve 2-GWD-HV-014.
   b) R16: (Boron Recycle System) diverts 2-CVC-RCV-016 Recycle Evaporator discharge from Reactor Makeup Water System to the Recycle Evaporator Demineralizer.
   c) R17A or B: (Component Cooling Water) closes 2-CCW-RCV-3028 CCW SRG TANK VENT.
   d) R18: (Liquid Waste Processing) closes Liquid Waste Release Valve 2-LWP-RCV-018.
   e) R19: (Steam Generator Blowdown) isolates Steam Generator Blowdown Sample Lines.
   f) R23A: (Steam Generator Blowdown Processing) closes 2-BD-FCV-1152 S/G Blowdown Heat Exchanger Discharge Valve.
   g) R23B: (Steam Generator Blowdown Processing) closes 2-BD-RCV-023B Dilution Discharge Valve.

STEAM GENERATOR BLOWDOWN

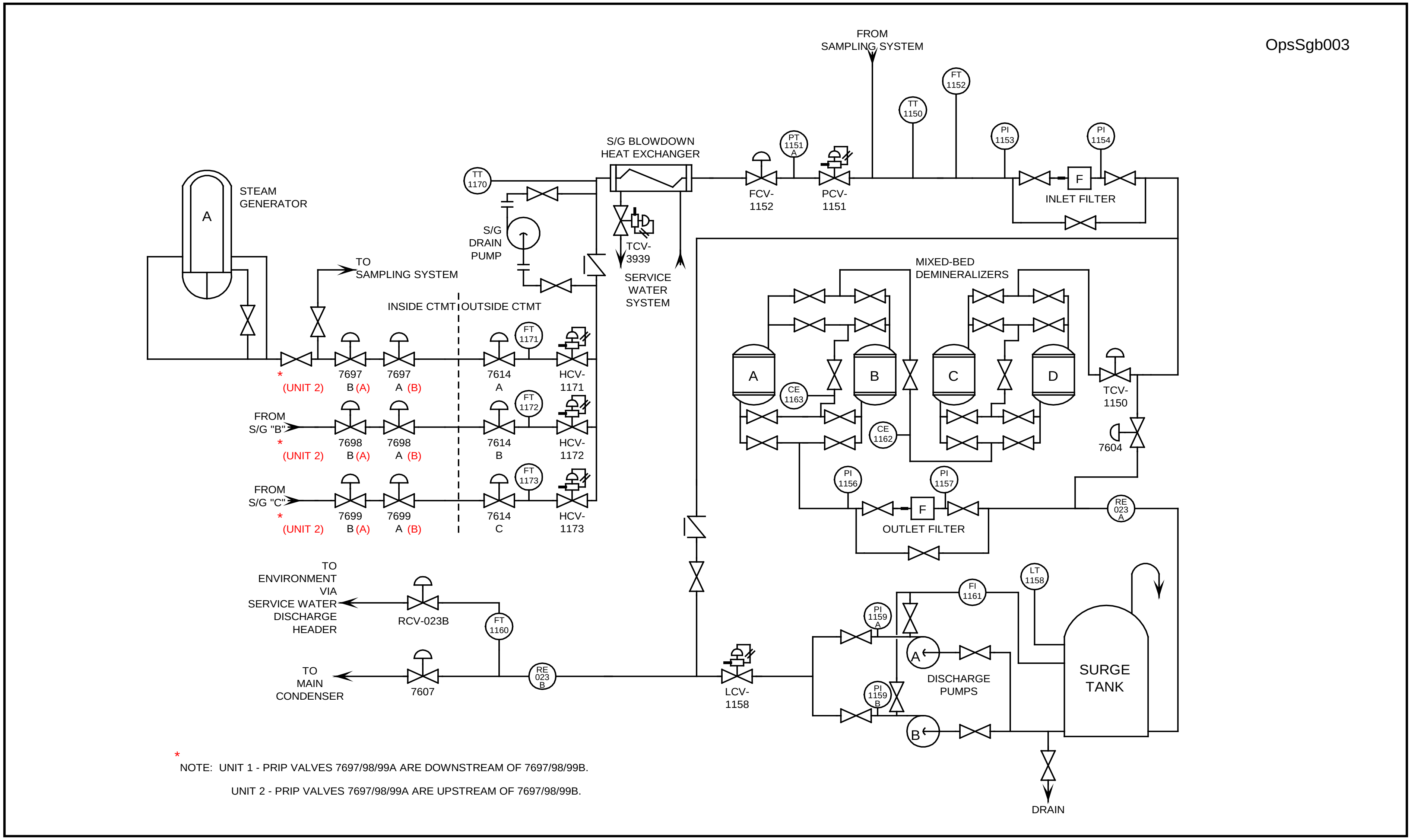


FIGURE 2 - S/G Blowdown Processing System

1. 059G2.2.44 040/NEW//C/A 4.2/4.4/059G2.2.44/N///

Unit 1 is operating at 33% power and the following conditions exist:

- 1A and 1B Condensate pumps are running.
- 1C Condensate pump is in OFF with a CAUTION TAG that says, "EMERGENCY USE ONLY".
- 1A SGFP is running.

Subsequently, the 1B Condensate pump trips and the following conditions are observed:

KB4, SGFP SUCTION PRESS LOW, comes into alarm and the operating crew observes the following on PR4039, SGFP SUCT PRESS:

Time				
0 sec	10 sec	20 sec	30 sec	40 sec
300 psig	275 psig	265 psig	270 psig	285 psig

At time 20 seconds, the 1C condensate pump was started.

Which one of the following completes the statements below?

At time 30 seconds, the 1A SGFP __(1)__ be tripped.

The operating crew is required to __(2)__.

A. 1) will NOT

2) rapidly reduce Turbine load using AOP-17.1, Rapid Turbine Power Reduction

B.✓ 1) will NOT

2) check SGFP suction pressure stabilizes

C. 1) WILL

2) trip the Reactor and enter EEP-0.0, Reactor Trip or Safety Injection.

D. 1) WILL

2) trip the Main Turbine and enter AOP-3.0, Turbine Trip Below P-9 Setpoint.

Not a true 2+2 question to improve the plausibility of the distracters.

KB4 comes into alarm at 300 psig.
At 275 psig decreasing on 2/3 pressure switches (PS625, PS626, PS627),
1. The standby condensate pump will start after 10 sec delay. (63IP relay)
2. The SGFP(s) will trip after 30 sec. delay (63IPX relay).

AOP-13
6.1 Check SGFP suction pressure stabilizes above 275 psig.

6.1.2 RNO:
IF suction pressure still falling, THEN reduce turbine load rapidly using FNP-1-AOP-17.1, RAPID TURBINE POWER REDUCTION.

<u>**Distracter analysis**</u>

A. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2). Plausible since this is the action to take if the SGFP suction does NOT stabilize (6.1.2 RNO).

B. Correct. First part is correct. The SGFP's will trip 30 seconds after suction pressure falls below 275 psig which would be at 40 seconds in this scenario.

Second part is correct. This is the correct action per AOP-13 step 6.1 since suction pressure is rising and within the band to keep the SGFP from tripping at time 40 sec

C. Incorrect. First part is incorrect (See B.1) Plausible if the applicant confuses the condensate pump autostart setpoint with the SGFP trip. The standby condensate pump, if in AUTO, would start 10 seconds after SGFP suction pressure falls below 275 psig.
OR plausible if the applicant thought that when the low pressure alarm comes in the SGFP would trip 30 sec later.

Second part is incorrect (See B.2) Plausible since this is the correct response if the SGFP tripped.

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2) Plausible since power is less than 35% (P-9) and tripping the turbine would stop most of the steam flow from the SG. This was the correct actions to take until 2 years ago when the station decided the most conservative action would be to trip the reactor if power is >5% power

K/A: **059G2.2.44**	Main Feedwater System - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
Importance Rating:	4.2 4.4
Technical Reference:	FNP-1-AOP-13, Condensate and Feedwater Malfunction, Ver 33
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing AOP-13, Loss of Main Feedwater. (OPS-52520M06).
Question History:	NEW
K/A match:	This question requires the applicant to **interpret the SGFP suction pressure to verify the status of the MFW system (SGFP is not tripped). Also, understand how operator actions, starting the 1C cond pump, and directives (AOP-13.0) affect the plant status which is to check that the suction pressure is rising and not reduce power or trip the reactor.**
SRO justification:	N/A

Step	Action/Expected Response	Response Not Obtained
1.11	Check parameters within limits for continued at power operation. • Pressurizer level greater than 15% • Pressurizer pressure greater than 2100 psig • SG narrow range levels 35%-75% • TAVG 541°F - 580°F • Control rod bank position Lo-Lo Annunciator FE2 Clear • Delta I within limits specified in the COLR	1.11 IF the Team is NOT confident that a parameter is being restored, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
__ 2	**Check Both SGFPs - TRIPPED**	2 Proceed to step 3 OBSERVE CAUTION prior to step 3.
2.1	Check Reactor Power - LESS THAN 5%.	2.1 Trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
2.2	Reduce reactor power to approximately 2%.	
2.2.1	Verify rod control in MANUAL.	
2.2.2	Stabilize TAVG by adjusting rod position and/or boron concentration.	
2.2.3	Check for proper operation of steam dumps.	

Step 2 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
	5.4.3 Refer to FNP-1-SOP-20.0, FEED WATER HEATER EXTRACTION, VENT, AND DRAIN SYSTEM for restoration of automatic dump and drain valve operations	
	5.5 Restore feedwater heater(s) to service using FNP-1-SOP-21.0, CONDENSATE AND FEEDWATER SYSTEM.	
__ 6	**Check SGFP suction pressure - ABNORMAL.**	6 Proceed to step 7.
	6.1 Check SGFP suction pressure stabilizes above 275 psig.	6.1 Perform the following

****

CAUTION: IF all condensate pumps are tripped and a standby condensate pump is not immediately started, THEN a condensate pump should NOT be started due to water hammer concerns.

****

Step	Action/Expected Response	Response Not Obtained
		6.1.1 Verify standby condensate pump started.
		6.1.2 IF suction pressure still falling, THEN reduce turbine load rapidly using FNP-1-AOP-17.1, RAPID TURBINE POWER REDUCTION.
		6.1.3 IF SGFP(s) trip occurs, THEN return to step 1.
		6.1.4 If SGFP speed is elevated return to step 3 OBSERVE CAUTION prior to step 3.

Step 6 continued on next page

__Page Completed

LOCATION KB4

B4
SGFP SUCT PRESS LO

SETPOINT: 300 PSIG Decreasing

ORIGIN: PS4X Relay from Pressure Switch N1N21PS690, with a 2 second time delay

PROBABLE CAUSE

1. 1A or 1B heater drain pump tripped.
2. A malfunction of SGWLC causing SGFP speed to rise.
3. A malfunction of one or more of the following valves, due to a failure of L & N power supply, instrument air problem, or mechanical problem:
   a) CNDS MINIMUM FLOW FCV - NlN2lV908
   b) 1A SGFP RECIRC FCV - NlN2lV909A
   c) 1B SGFP RECIRC FCV - NlN2lV909B
   d) SJAE BYP - NlN2lV90l
   e) GS COND BYP FCV - NlN2lV902

NOTE: In the following AUTOMATIC ACTION discussion, all times are concurrent.

AUTOMATIC ACTION

At 275 psig decreasing on 2/3 pressure switches (PS625, PS626, PS627),

1. The standby condensate pump will start after 10 sec delay. (63IP relay)
2. The SGFP(s) will trip after 30 sec. delay (63IPX relay).

If suction pressure is restored to greater than the actuation setpoint 275 psig, no automatic actions will take place.

3.2.17 Operation with only two condensate pumps available

- In the event of a condensate pump trip above 65% power with no backup pump available, it is management's expectation that the reactor will be tripped and FNP-1-EEP-0.0 Reactor Trip Or Safety Injection entered. In the event of a condensate pump trip from 65% power level or below, the crew is expected to enter AOP-13 at step 6 for SGFP low suction pressure..

3.3 FEEDWATER SYSTEM

3.3.1 The SGFP turbine lube oil temperature must be above 70°F prior to turbine startup.

3.3.2 SGFP lube oil circulation shall be maintained a minimum of four hours after turbine shutdown.

3.3.3 Maintain SGFP turbine lube oil temperature leaving the oil cooler at 110°F-120°F.

3.3.4 SGFP Rotor Vibration Limitations

3.3.4.1 When SGFP speed is increased above 3000 rpm and is under low flow conditions, the vibration reading will be erratic in nature until the pump is loaded, i.e. flow increased above 3500 KBH.

3.3.4.2 OE has shown that SGFP rotor vibrations are minimized when feed flow is increased (Above ~3500 KBH) and the use of the miniflow is minimized.

3.3.4.3 Sustained is defined as greater than one minute. (CR# 20101172710)

3.3.4.4 Limit With flow greater than 3500 KBH

- Greater than 3 mils: Alert range requires investigation of SGFP.
- Greater than or equal to 7 mils: Excessive vibration (sustained) requires pump to be removed from service.

3.3.4.5 Limit With flow less than 3500 KBH

- Less than 10 mils Sustained

3.3.5 SGFP Turbine Thrust Bearing Wear Limits

- Alarm Setpoints: -22 mils (Reverse) and +7 mils (Forward)
- Automatic Trip Setpoints: -25 mils (Reverse) and +10 mils(Forward)

QUESTIONS REPORT
for ILT 36 RO NRC Exam Version 6

1. 061A3.01 041/NEW//C/A 4.2/4.2/061A3.01/N///

Unit 1 is in Mode 3 with the following conditions:

- 1A MDAFW pump was started per UOP-1.2, Startup of Unit From Cold Shutdown to Hot Standby.
- There are no other AFW pumps running.
- All SG NR levels are 65%.

Subsequently, power is lost to the 1A Startup Transformer.

Which one of the following completes the statements below?

The TDAFW Pump __(1)__ be running.

Total design AFW flow rate will be approximately __(2)__ gpm.

	(1)	(2)
A.✔	will NOT	350
B.	will NOT	700
C.	WILL	700
D.	WILL	1050

Not a true 2+2 question to improve distracter plausibility.

The applicant has to evaluate how the loss of the 1A Startup transformer (SUT) affects the TDAFW and MDAFW pumps auto starts. Since the 1A SUT powers the 1A bus and the 1B SUT powers the 1B and 1C busses, only the 1A bus is lost. The opposite is true on Unit 2 so the applicant has to recall how each unit is configured.

Secondly, the applicant has to recall how the loss of power affects the MDAFW pumps. The 1B MDAFWP is unaffected since the 1G bus did not lose power as it is powered from 1B SUT. The 1A MDAFW pump did lose power and will be sequenced on the bus when the DG starts and the LOSP sequencer runs.

Thirdly, once the applicant determines which AFW pumps are running, then they will have to recall design flow rates for each (350 gpm for the MDAFW and 700 gpm for the TDAFW pump) to determine total approximate flow.

<u>**Distracter analysis**</u>

A. Correct. First part is correct. FSD-A181010 - 3.9.2.3 - The TDAFW pump shall start by opening the steam supply valves to the turbine drive **on a loss of power signal**, low-low water level signals from two out of three level transmitters of any two out of three steam

generators, or an AMSAC signal. The loss of power signal comes from the loss of power to 2 of 3 RCP busses (1A, 1B and 1C). **Since ONLY the 1A bus loses power, the TDAFW pump does not start.)**

Second part is correct. The 1A MDAFWP pump will autostart and the FCV's will open fully providing ~350 gpm design flow.

.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2). Plausible if the applicant incorrectly thinks an LOSP has occurred which would start 2 MDAFWP's and provide approx 700 gpm flow.

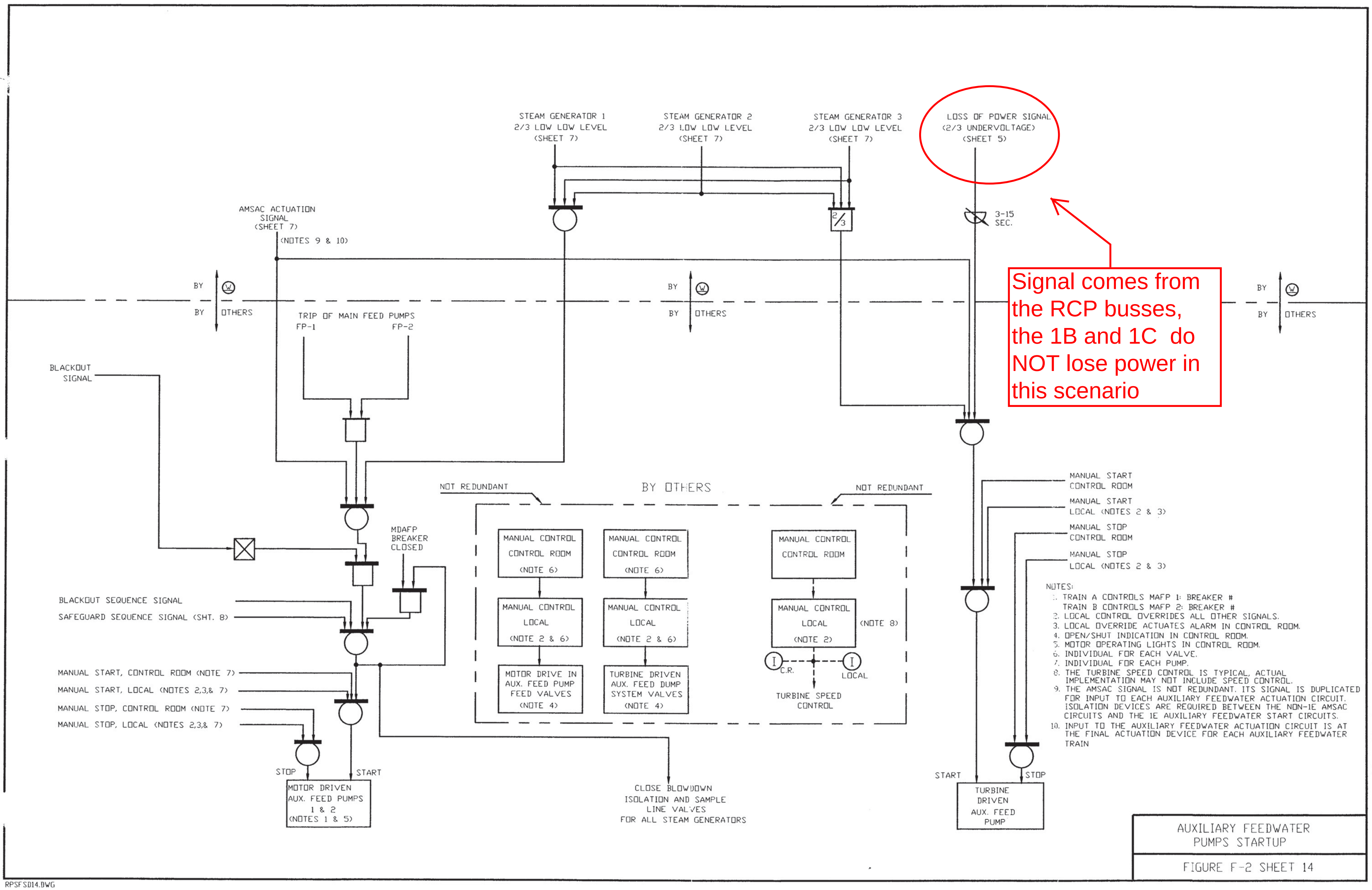
C. Incorrect. First part is incorrect (See A.1). Plausible if the applicant confuses the Startup transformer alignment with Unit 2. The TDAFW pump on Unit 2 would start under these conditions.

Second part is incorrect (See A.2). Plausible if the applicant doesn't recall that the MDAFWP receives and auto start signal during an LOSP. This would make this a logical connection to the first part and a correct answer if the applicant thought that only the TDAFW pump started.

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See A.2).Plausible since this is the design flow for one MDAFW pump and the TDAFW pump and a logical connection to the first part if the applicant thought that the TDAFW pump started.

K/A: **061A3.01**	Auxiliary / Emergency Feedwater (AFW) System - Ability to monitor automatic operation of the AFW, including: AFW startup and flows
Importance Rating:	4.2 4.2
Technical Reference:	FSD-A181010, Auxiliary Feedwater System, Ver 25 FSD-A181007, Reactor Protection, Ver 18 U166235, Primary Coolant Trip Signals, Ver 2 A506250, U1 Load List, Ver 74
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the AFW System to include the components found on Figure 2, Auxiliary Feedwater System, Figure 3, TDAFWP Steam Supply, and Figure 4, Air Supply to TDAFWP Steam Admission Valves (OPS-40201D02). NAME AND IDENTIFY the Bus power supplies (Off-site sources and Emergency source-to- Load), for those electrical components associated with the AFW System to include those items in Table 3- Power Supplies (OPS-40201D04). SELECT AND ASSESS the AFW System instrument/equipment response expected when performing auxiliary feedwater evolutions including (OPS-52102H05): [...] The Failed Condition [...] Associated Trip Setpoint(s) [...]
Question History:	NEW
K/A match:	The applicant is required to **evaluate the loss of power and determine which AFW pump auto starts (monitor startup) and the resultant flow (monitor flow).**
SRO justification:	N/A


STEAM GENERATOR 1
2/3 LOW LOW LEVEL
(SHEET 7)
STEAM GENERATOR 2
2/3 LOW LOW LEVEL
(SHEET 7)
STEAM GENERATOR 3
2/3 LOW LOW LEVEL
(SHEET 7)
LOSS OF POWER SIGNAL
(2/3 UNDERVOLTAGE)
(SHEET 5)
3-15 SEC.
Signal comes from the RCP busses, the 1B and 1C do NOT lose power in this scenario
AMSAC ACTUATION
SIGNAL
(SHEET 7)
(NOTES 9 & 10)
BY
BY OTHERS
TRIP OF MAIN FEED PUMPS
FP-1
FP-2
BLACKOUT
SIGNAL
NOT REDUNDANT
BY OTHERS
MDAFP
BREAKER
CLOSED
MANUAL CONTROL
CONTROL ROOM
(NOTE 6)
MANUAL CONTROL
LOCAL
(NOTE 2 & 6)
MOTOR DRIVE IN
AUX. FEED PUMP
FEED VALVES
(NOTE 4)
TURBINE DRIVEN
AUX. FEED DUMP
SYSTEM VALVES
(NOTE 4)
MANUAL CONTROL
CONTROL ROOM
MANUAL CONTROL
LOCAL
(NOTE 2)
(NOTE 8)
C.R.
LOCAL
TURBINE SPEED
CONTROL
BLACKOUT SEQUENCE SIGNAL
SAFEGUARD SEQUENCE SIGNAL (SHT. 8)
MANUAL START, CONTROL ROOM (NOTE 7)
MANUAL START, LOCAL (NOTES 2,3,& 7)
MANUAL STOP, CONTROL ROOM (NOTE 7)
MANUAL STOP, LOCAL (NOTES 2,3,& 7)
STOP
START
MOTOR DRIVEN
AUX. FEED PUMPS
1 & 2
(NOTES 1 & 5)
CLOSE BLOWDOWN
ISOLATION AND SAMPLE
LINE VALVES
FOR ALL STEAM GENERATORS
MANUAL START
CONTROL ROOM
MANUAL START
LOCAL (NOTES 2 & 3)
MANUAL STOP
CONTROL ROOM
MANUAL STOP
LOCAL (NOTES 2 & 3)
START
STOP
TURBINE
DRIVEN
AUX. FEED
PUMP
NOTES:
1. TRAIN A CONTROLS MAFP 1: BREAKER #
TRAIN B CONTROLS MAFP 2: BREAKER #
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS.
3. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
4. OPEN/SHUT INDICATION IN CONTROL ROOM.
5. MOTOR OPERATING LIGHTS IN CONTROL ROOM.
6. INDIVIDUAL FOR EACH VALVE.
7. INDIVIDUAL FOR EACH PUMP.
8. THE TURBINE SPEED CONTROL IS TYPICAL. ACTUAL IMPLEMENTATION MAY NOT INCLUDE SPEED CONTROL.
9. THE AMSAC SIGNAL IS NOT REDUNDANT. ITS SIGNAL IS DUPLICATED FOR INPUT TO EACH AUXILIARY FEEDWATER ACTUATION CIRCUIT. ISOLATION DEVICES ARE REQUIRED BETWEEN THE NON-1E AMSAC CIRCUITS AND THE 1E AUXILIARY FEEDWATER START CIRCUITS.
10. INPUT TO THE AUXILIARY FEEDWATER ACTUATION CIRCUIT IS AT THE FINAL ACTUATION DEVICE FOR EACH AUXILIARY FEEDWATER TRAIN
AUXILIARY FEEDWATER
PUMPS STARTUP
FIGURE F-2 SHEET 14
RPSFSD14.DWG

3.1.2.2 Evaluation of the Net Positive Suction Head (NPSH) margin shows that sufficient margin exists over the complete range of pump flows from shut-off to run-out when a single MDAFW pump is in operation. When dual MDAFW pumps are in operation, the NPSH margin is depleted when each individual pump is operating at approximately 580 gpm (Reference 6.3.003). The NPSH calculation (Reference 6.3.003) assumes an empty Condensate Storage Tank (water elevation at 156'-5") and uses a composite NPSH required curve which combines the worst case portions of the Unit 1 and 2 NPSH curves for conservatism.

The NPSH required for the MDAFW pumps at the design flow rate of 350 gpm is 13 feet for Unit 1 and 17 feet for Unit 2 (References 6.4.076, 6.4.077, 6.4.078, 6.4.079).

The acceptability of the NPSH for the various normal and accident flow conditions is addressed in the Unit 1 and 2 Verification of Auxiliary Feedwater Bases calculations (References 6.3.010, 6.3.011).

3.1.2.3 The MDAFW pumps shall be of a horizontal centrifugal design, driven by a continuous duty electric motor to meet the system operating requirements during normal and emergency conditions (Reference 6.5.003).

3.1.2.4 The MDAFW pumps shall be able to be vented and drained for maintenance (Reference 6.5.003).

3.1.2.5 The MDAFW pump bearings shall be air cooled and equipped with constant level oilers (Reference 6.5.003).

3.1.2.6 The AFW pumps shall operate in an ambient air temperature environment of up to 104°F with no degradation of motor winding insulation.

A calculation has established that the post-accident temperature in the Auxiliary Feedwater Rooms (191/2191 - 192/2192) will reach 116 F at the end of a post-accident period. The electrical equipment in the pump rooms has been evaluated and found acceptable for operation during the temperatures which will exist during the post-accident period.

Operation of the pump motors at ambient temperatures above 104 F will result in reduced motor life and will require further evaluation of their qualified life on a case-by case basis. (References 6.5.003, 6.7.082)

3.9.1.3 In the event of a Station Blackout (SBO), the TDAFW pump shall deliver the required flow to the steam generators at conditions described in Section 2.2.8.

3.9.2 Functional Requirements

3.9.2.1 The TDAFW pumps' design head shall be sufficient to deliver the required flow to the steam generators at conditions described in Sections 2.1 and 2.2 (References 6.2.001, 6.3.010, 6.3.011). The TDAFW pump has been sized to provide 700 gpm at 2835 feet (References 6.3.002, 6.5.003). See Table T-4 for minimum/maximum TDAFW pump performance curves which meet the analyzed design conditions (References 6.3.010, 6.3.011, 6.7.087). The TDAFW pump curve was degraded and enhanced by decreasing and increasing the pump head a selected percentage at each point along the curve.

3.9.2.2 Evaluation of the Net Positive Suction Head (NPSH) margin for the TDAFW pump shows that sufficient margin exists over the complete range of pump flows from shut-off to run-out (Reference 6.3.003). The NPSH calculation (Reference 6.3.003) assumes an empty Condensate Storage Tank (water elevation at 156'-5") for conservatism.

The NPSH required for the TDAFW pump at the design flow rate of 700 gpm and 3960 rpm is 21 feet for Unit 1 and 19 feet for Unit 2 (References 6.4.081, 6.4.082).

The acceptability of the NPSH for the various normal and accident flow conditions is addressed in the Unit 1 and 2 Verification of Auxiliary Feedwater Bases calculations (References 6.3.010, 6.3.011).

3.9.2.3 The TDAFW pump shall start by opening the steam supply valves to the turbine drive on a loss of power signal, low-low water level signals from two out of three level transmitters of any two out of three steam generators, or an AMSAC signal (Reference 6.2.001).

3.9.2.4 The TDAFW pump is designed to be automatically initiated and controlled for at least 2 hours independent of any ac power (Reference 6.7.042).

The operation of the TDAFWP and its supporting components are not adversely affected by loss of the ventilation system in the TDAFWP room (References 6.3.012, 6.7.080).

3.10 TDAFW PUMP TURBINE DRIVE

TPNS No. QN23P003

3.10.1 Basic Function

The turbine drive for the TDAFW pump shall take steam from two of the three main steam lines and allow the steam to expand in the turbine, thereby providing power to the TDAFW pump (References 6.7.071, 6.4.003, 6.4.088). Two sources of steam are required for turbine drive operation in the event of a failure of one main steam line or a faulted steam generator.

3.10.2 Functional Requirements

3.10.2.1 The turbine shall start by a remote manual or automatic signal and shall load the driven pump to its rated capacity and head within 1 minute from starting at rest (Reference 6.5.003).

3.10.2.2 The turbine drive is designed to operate for at least 2 hours following a loss of all ac power (Reference 6.7.016).

3.10.2.3 Emergency trip speed of the turbine shall not exceed 115 percent of the rated speed of 3960 rpm for the turbine (4554 rpm). This overspeed trip setpoint prevents overpressurization of the AFW system design pressure rating. At 115% turbine overspeed, the shut-off pressure developed by the TDAFW pump is 2026.8 psig. Based on that overpressurization is an infrequent event and can be categorized as an emergency condition, the TDAFW pump, discharge flange, piping, and associated fittings are adequate for a pressure of 2026.8 psig (References 6.7.043, 6.7.044, 6.7.045, and 6.7.084).

3.10.2.4 The steam turbine shall be a horizontal, noncondensing, single-stage, mechanical drive unit (Reference 6.5.003).

3.10.2.5 The required horsepower output of the turbine shall be 685 hp (Reference 6.5.003). The rated horsepower of the turbine is 687 hp at 3960 rpm and a steam rate of 38.0 lb/hp/hr (Reference 6.4.043).

3.10.2.6 The design pressures and temperatures of the inlet and exhaust steam of the turbine drive shall be 1250 psig and 574 °F, and 10

1A 4160V BUS **AB - 139'** **D177002**

BKR	TPNS	DESCRIPTION	SEE PAGE
	N1R15A0001-N	**1A 4160V BUS**	
DA01	**N1R12A0502-N**	**1B UNIT AUX TRANSFORMER (NORMAL) <<<**	
DA02	N1R15BKRDA02	PT COMPARTMENT	
DA03	N1N21M0001A-N	1A CONDENSATE PUMP	
DA04	N1B41M0001A-N	1A REACTOR COOLANT PUMP	
DA05	N1P26M0001A-N	1A CIRC WATER PUMP	
DA06	**N1R11B0007-N**	**1I 4160/600V SST >>> EI02**	**A-2**
DA07	**N1R11A0501-N**	**1A STARTUP TRANSFORMER (ALTERNATE) <<<**	

Correct alignment in Mode 3

1B 4160V BUS **AB - 139'** **C177003**

BKR	TPNS	DESCRIPTION
	N1R15A0002-N	**1B 4160V BUS**
DB01	**N1R12A0502-N**	**1B UNIT AUX TRANSFORMER (NORMAL)** <<<
DB02	N1R15BKRDB02	PT COMPARTMENT
DB03	N1B41M0001B-N	1B REACTOR COOLANT PUMP
DB04	N1P26M0001B-N	1B CIRC WATER PUMP
DB05	**N1R11A0502-N**	**1B START-UP TRANSFORMER (ALTERNATE)** <<<

Correct alignment
in Mode 3

1C 4160V BUS AB - 139' C177004

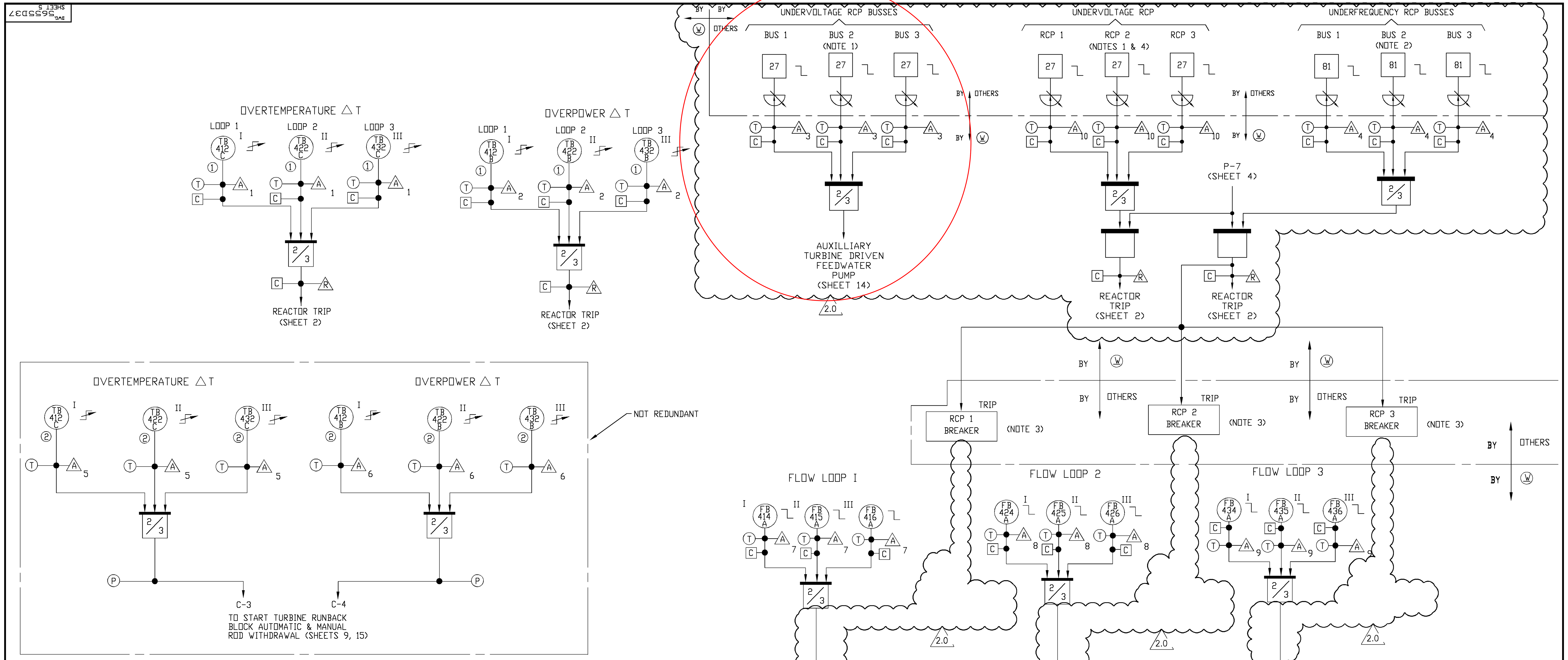
BKR	TPNS	DESCRIPTION	SEE PAGE
	N1R15A0003-N	**1C 4160V BUS**	
DC01	**N1R11A0502-N**	**1B START-UP TRANSFORMER (NORMAL)** <<<	
DC02	N1R15BKRDC02	PT COMPARTMENT	
DC03	N1B41M0001C-N	1C REACTOR COOLANT PUMP	
DC04	**N1R12A0502-N**	**1B UNIT AUX TRANSFORMER (ALTERNATE)** <<<	
DC05	**NSR31G0501-N**	**10 4160/600V SST** >>> **E002**	**C-2**
	N1R11G0510-N	**1T 4160/600V SST** >>> **ET02**	**C-50**

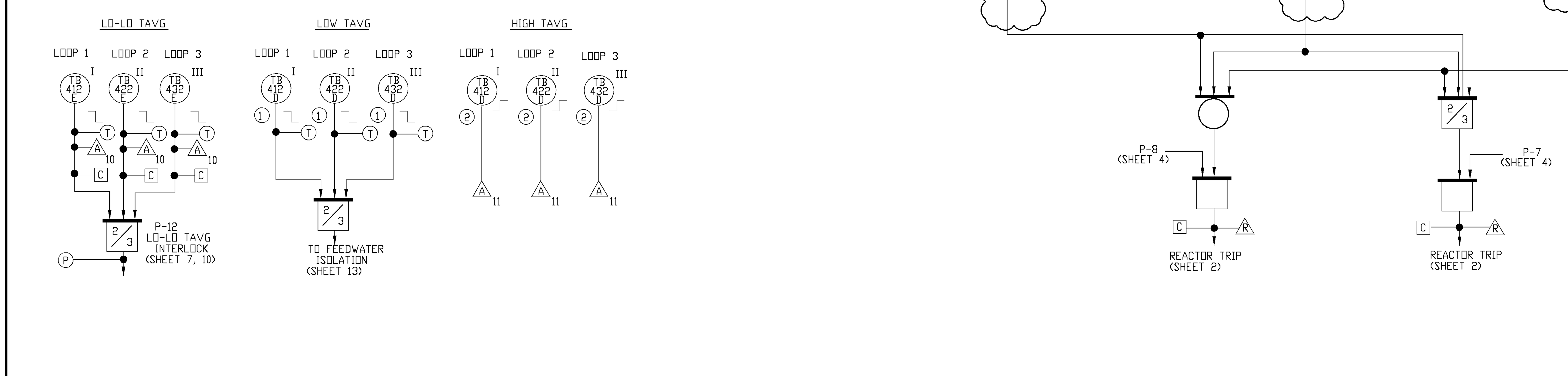
ATTACHMENT 10

ESS AND LOSP LOAD SEQUENCE

SEQUENCERS	B1F		B1G	
COMPONENTS	ESS STEP	LOSP STEP	ESS STEP	LOSP STEP
1A CHG PUMP	1	1		
1B CHG PUMP	1	1	1	1
1C CHG PUMP			1	1
1A RHR PUMP	2			
1B RHR PUMP			2	
1A CS PUMP	2			
1B CS PUMP			2	
1A SW PUMP	3	2		
1B SW PUMP	3	3		
1D SW PUMP			3	2
1E SW PUMP			3	3
1A CCW PUMP			4	4
1B CCW PUMP	4	4	4	4
1C CCW PUMP	4	4		
A TRN CTMT CLR FAN	4	4		
B TRN CTMT CLR FAN			4	4
1A MDAFWP	5	5		
1B MDAFWP			5	5
1A BATT CHARGER	6	6		
1B BATT CHARGER			6	6
1A CRDM CLG FAN				2
1B CRDM CLG FAN		2		
1A RX CAV CLG FAN		1		
1B RX CAV CLG FAN				1
1A RX CAV H2 DILUTION FAN	5			
1B RX CAV H2 DILUTION FAN			5	
1A LC EMERG SUPPLY ED08/ EA09 / 1C AIR COMP. EA15	6	6		

-END-





NOTES:

1. SETPOINT FOR UNDERVOLTAGE RELAYS SHOULD BE APPROXIMATELY 70% NOMINAL VOLTAGE. THE MAXIMUM ALLOWABLE DETECTOR TIME DELAY BETWEEN REACHING THE SETPOINT VALUE AND PASSING ON THE SIGNAL TO THE REACTOR TRIP AND TURBINE DRIVEN AUX. FW PUMP ACTUATION CIRCUITRY SHOULD NOT EXCEED 0.6 SECONDS. THIS DOES NOT INCLUDE THE MOTOR EMF DECAY TIME OF 0.6 SECONDS. 2.0
2. SETPOINT FOR UNDERFREQUENCY RELAYS SHOULD BE ADJUSTABLE BETWEEN 54 CPS AND 59 CPS. THE MAXIMUM ALLOWABLE DETECTOR TIME DELAY, INCLUDING THE ADJUSTABLE DELAY, BETWEEN REACHING THE SETPOINT VALUE AND PASSING ON THE SIGNAL TO THE TRIP CIRCUITS SHOULD NOT EXCEED 0.3 SECONDS. THIS DOES NOT INCLUDE THE FREQUENCY DECAY TIME AT A RATE OF 5 HZ/SECOND.
3. THE MAXIMUM ALLOWABLE TIME DELAY BETWEEN THE TIME THE RCP BREAKERS RECEIVE A TRIP SIGNAL AND THE TIME THE BREAKERS HAVE TRIPPED AND PASSED ON AN OPEN SIGNAL TO THE REACTOR TRIP LOGIC SHOULD NOT EXCEED 0.1 SECONDS. 2.0
4. THE UNDERVOLTAGE SENSORS FOR THE RCP BUS UNDERVOLTAGE REACTOR TRIP MUST BE LOCATED ON THE MOTOR SIDE OF THE RCP CIRCUIT BREAKER.

DFTM.	M. HABERCHAK 2-2-71
CHKR.	
DES. ENG.	G. G. Patton 4-20-71
MFG. ENG.	
MTLS. ENG.	
APP.	
APP.	
APP.	
DFTG. SUPV.	

Westinghouse Electric Corporation
ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.

TITLE: ALABAMA POWER COMPANY JOSEPH M. FARLEY UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRIMARY COOLANT SYSTEM TRIP SIGNALS

DIMENSIONS IN INCHES
DO NOT SCALE
5655D37
SHEET 5
SUB. ~~1~~ ~~2~~ ~~3~~ 4

CAD U166235
OVY2000 | KB

THIS DWG. REFERENCED IN VENDOR MANUAL N/A	
TAB/SECT.	N/A
PAGE	N/A
FIGURE	N/A

VERSION 2.0 DATE 03/23/11
REVISED BY SNC PER ABN 1082083901J003, VER. 1.0
SEE MICROFILM FOR PREVIOUS VER. SIGNATURES.

BY	CHK'D	APPR.1	APPR.2
KB	GDM	JMR	X

Southern Nuclear Operating Company, Inc.
FOR
JOSEPH M. FARLEY NUCLEAR PLANT UNIT NO. 1

TITLE: FUNCTIONAL DIAGRAMS PRIMARY COOLANT SYSTEM TRIP SIGNALS

VENDOR: WESTINGHOUSE P.O.#: FNP-2
DRAWING NO. U-166235

1. 061K5.01 042/BANK/INDIAN POINT 07/C/A 3.6/3.9/061K5.01/N///

Unit 1 was operating at 100% power when a Reactor Trip occurred and the following conditions exist:

- ESP-0.1, Reactor Trip Response, has just been entered.
- Pressurizer level is 12% and slowly lowering.
- SG NR levels are 40% and slowly rising.
- Tavg is 543°F and slowly lowering.
- RCS pressure is 2050 psig and slowly lowering.

Which one of the following actions will be performed FIRST as required by ESP-0.1 to address the cooldown?

A.✓ Minimize total AFW flow.

B. Emergency borate the RCS.

C. Close all MSIVs and MSIV Bypass Valves.

D. Manually initiate SI and return to EEP-0.0, Reactor Trip or Safety Injection.

ESP-0.1 -
Step 1.1 RNO:
IF RCS temperature less than 547°F and falling, THEN perform the following. IF NOT, THEN proceed to RNO Step 1.2.

Step 1.1.4 RNO:
IF cooldown continues, THEN minimize total AFW flow.

Distracter analysis

A. Correct — Step 1.1.4 RNO of ESP-0.1 has the operator minimize AFW to stop the cooldown.

B. Incorrect. — See A.
Plausible since this is an action in ESP-0.1 if Tavg falls below 525°F. This action is at step 4 and would not be required since Tavg is >525°F. The applicant could confuse this temperature limit to emergency borate with P-12, 543°F Lo-Lo Tavg.

C. Incorrect. — See A. This is done AFTER AFW flow is reduced at step 1.1.5 . Plausible since this would address the cooldown. Also there are a number of steps completed before the AFW flow is addressed that equates to steam in the TB reduced, and stm dumps checked.

D. Incorrect. — See A. Plausible because the Pzr level meets the SI reinitiation criteria (13%) for a number of other Emergency procedures (such as ESP-1.1) and the applicant could confuse it with the correct Pzr level SI initiation criteria of ESP-0.1 foldout page of 4%.

K/A: **061K5.01**	Auxiliary / Emergency Feedwater (AFW) System - Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer
Importance Rating:	3.6 3.9
Technical Reference:	ESP-0.1, Reactor trip Response, Ver 32.
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing ESP-0.1, Reactor Trip Response. (OPS-52531B06)
Question History:	INDIAN POINT 07
K/A match:	The applicant is required to know that in order **to stop the excessive RCS cooldown (operational implication), they must know that reducing AFW flow will reduce the heat transfer rate of the RCS.**
SRO justification:	N/A

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: [CA] To ensure proper plant response, FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION, must be entered upon any SI actuation.

****

NOTE:
- Foldout page should be monitored continuously.
- [CA] Verification that RCS temperature is being maintained stable at or approaching 547°F is a continuing action until directed otherwise.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**[CA] Check RCS temperature.**	1 Perform the following.
	• IF any RCP running, THEN check RCS average temperature - STABLE AT OR APPROACHING 547°F. TAVG 1A(1B,1C) RCS LOOP [] TI 412D [] TI 422D [] TI 432D OR • IF no RCP running, THEN check RCS cold leg temperature - STABLE AT OR APPROACHING 547°F. RCS COLD LEG TEMP RECORDER [] TR 410	1.1 IF RCS temperature less than 547°F and falling, THEN perform the following. IF NOT, THEN proceed to RNO Step 1.2. 1.1.1 Verify steam dumps closed. STM DUMP INTERLOCK [] A TRN in OFF RESET [] B TRN in OFF RESET 1.1.2 Verify atmospheric reliefs closed. 1A(1B,1C) MS ATMOS REL VLV [] PC 3371A [] PC 3371B [] PC 3371C

Step 1 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
		1.1.3 IF MSIVs are open, THEN isolate turbine building steam loads while continuing with RNO step 1.1.4. 1.1.3.1 Isolate main steam drain pots. MS LINE DRN POT A(B,C,D) TO COND ISO (155 ft, TURB BLDG) [] N1N11V905A closed [] N1N11V905B closed [] N1N11V905C closed [] N1N11V905D closed 1.1.3.2 Isolate MSRs. [] MSRs reset MSR 1A(2A,1B,2B) 2ND STG STM SUPP ISO (189 ft, TURB BLDG) [] N1N11V503A closed [] N1N11V503B closed [] N1N11V503C closed [] N1N11V503D closed MSR 1A(2A,1B,2B) 2ND STG RAMP VLV ISO (189 ft, TURB BLDG) [] N1N11V619A closed [] N1N11V619B closed [] N1N11V619C closed [] N1N11V619D closed 1.1.3.3 IF gland sealing steam required, THEN align to Unit 2 steam supply using ATTACHMENT 1.

Step 1 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

1.1.3.4 IF two SJAEs in service, THEN secure one SJAE.

1A SJAE
[] A SECTION ISO closed
[] B SECTION ISO closed

1B SJAE
[] A SECTION ISO closed
[] B SECTION ISO closed

1.1.3.5 Verify SG blowdown - ISOLATED.

1A(1B,1C) SGBD ISO
[] Q1G24HV7614A closed
[] Q1G24HV7614B closed
[] Q1G24HV7614C closed

Step 1 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: To provide adequate heat sink total AFW flow must remain greater than 395 gpm until at least one SG narrow range level is greater than 31%.

****

1.1.4 IF cooldown continues, THEN minimize total AFW flow.

AFW FLOW TO
1A(1B,1C) SG
[] FI 3229A
[] FI 3229B
[] FI 3229C

AFW
TOTAL FLOW
[] FI 3229

- Control MDAFWP flow.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3227A in MOD	[]3227B in MOD	[]3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

Step 1 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

- Control TDAFWP flow.

  TDAFWP FCV 3228
  [] RESET reset

  TDAFWP
  SPEED CONT
  [] SIC 3405 adjusted

SG	1A	1B	1C
TDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3228A in MOD	[]3228B in MOD	[]3228C in MOD
TDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3228AA adjusted	[]3228BA adjusted	[]3228CA adjusted

1.1.5 IF cooldown continues, THEN close main steam isolation and bypass valves.

1A(1B,1C) SG
MSIV - TRIP
[] Q1N11HV3369A
[] Q1N11HV3369B
[] Q1N11HV3369C
[] Q1N11HV3370A
[] Q1N11HV3370B
[] Q1N11HV3370C

1A(1B,1C) SG
MSIV - BYPASS
[] Q1N11HV3368A
[] Q1N11HV3368B
[] Q1N11HV3368C
[] Q1N11HV3976A
[] Q1N11HV3976B
[] Q1N11HV3976C

Step 1 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
___ 2	**[CA] WHEN RCS average temperature less than 554°F, THEN verify feedwater status.**	
	2.1 Verify main feedwater flow control and bypass valves - CLOSED. 1A(1B,1C) SG FW FLOW [] FCV 478 [] FCV 488 [] FCV 498 1A(1B,1C) SG FW BYP FLOW [] FCV 479 [] FCV 489 [] FCV 499	2.1 Close main feedwater stop valves. MAIN FW TO 1A(1B,1C) SG STOP VLV [] Q1N21MOV3232A [] Q1N21MOV3232B [] Q1N21MOV3232C
	2.2 Defeat MDAFWP auto start on SGFP trip. MDAFWP AUTO/DEFEAT [] 1A in DEFEAT [] 1B in DEFEAT	
	2.3 Verify both steam generator feed pumps - TRIPPED.	2.3 Stop both EH fluid pumps. EH PUMP [] 1A [] 1B
	2.4 Verify total feed flow to SGs - GREATER THAN 395 gpm. AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229	

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Verify all reactor trip breakers and reactor trip bypass breakers - OPEN.** [] Reactor trip breaker A [] Reactor trip breaker B [] Reactor trip bypass breaker A [] Reactor trip bypass breaker B	3 Dispatch an operator to locally trip the reactor trip and bypass breakers.
___ 4	**[CA] Check emergency boration not required.** • [CA] Check all control rods - FULLY INSERTED. • [CA] Check RCS TAVG - GREATER THAN 525°F.	4 Emergency borate RCS. 4.1 Start a boric acid transfer pump. BATP [] 1A [] 1B 4.2 Establish emergency boration flow path. 4.2.1 Align normal emergency boration flow path. EMERG BORATE TO CHG PUMP SUCT [] Q1E21MOV8104 open 4.2.2 IF normal emergency boration flow path NOT available, THEN align manual emergency boration flow path. MAN EMERG BORATION [] Q1E21V185 open (100 ft, AUX BLDG chemical mixing tank area) BORIC ACID TO BLENDER [] Q1E21FCV113A open

Step 4 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
1	**Monitor SI criteria.**	
1.1	Greater than 16°F subcooled in CETC mode and PRZR level above 4%.	1.1 Verify SI actuated, check for CSF red or orange paths, go to FNP-1-EEP-0.
2	**Monitor switchover criteria.**	
2.1	CST level greater than 5.3 ft.	2.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0.
3	**Monitor spent fuel pool conditions:** • Verify annunciator EH1, SFP TEMP HI CLEAR. • Verify annunciator EH2, SFP LEVEL HI-LO CLEAR. • IF applicable, THEN check time to restore power is LESS THAN the time TO REACH 200°F in the spent fuel pool; using Unit 1 Core Physics Curve Book, PCB-1-VOL1-CRV79.	3 Perform ATTACHMENT 5.

Step	Action/Expected Response	Response NOT Obtained
1	**Monitor SI reinitiation criteria following HHSI isolation.**	
1.1	Greater than 16°F{45°F} subcooled in CETC mode and PRZR level above 13%{43%}.	1.1 Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 5, RE-ESTABLISHING HHSI FLOW.
2	**Monitor FNP-1-EEP-2 and FNP-1-EEP-3 branch criteria.**	
2.1	No SG pressure falling in an uncontrolled manner or less than 50 psig.	2.1 IF affected SG NOT previously isolated, THEN go to FNP-1-EEP-2.
2.2	No high secondary radiation or SG level rising uncontrolled.	2.2 Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 5, RE-ESTABLISHING HHSI FLOW THEN go to FNP-1-EEP-3.
3	**Monitor switchover criteria.**	
3.1	RWST level greater than 12.5 ft.	3.1 Go to FNP-1-ESP-1.3.
3.2	CST level greater than 5.3 ft.	3.2 Align AFW pumps suction to SW using FNP-1-SOP-22.0.
4	**Monitor charging miniflow criteria (during SI).**	
4.1	RCS pressure less than 1900 psig.	4.1 Verify miniflow valves open.
4.2	RCS pressure greater than 1300 psig.	4.2 Verify miniflow valves closed.
5	**Monitor adverse containment criteria.**	
5.1	CTMT pressure less than 4 psig and radiation less than 10^5 R/hr.	5.1 Utilize bracketed adverse CTMT condition numbers.

Step	Action/Expected Response	Response NOT Obtained
1	**Following SI termination monitor SI reinitiation criteria.**	
1.1	Greater than 16°F{45°F} subcooled in CETC mode and PRZR level above 13%{43%}.	1.1 Establish HHSI flow using ATTACHMENT 3, RE-ESTABLISHING HHSI FLOW and go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
2	**Monitor FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION branch criteria.**	
2.1	No SG pressure falling in an uncontrolled manner or less than 50 psig.	2.1 IF affected SG NOT previously isolated, THEN go to FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION.
3	**Monitor switchover criterion.**	
3.1	CST level greater than 5.3 ft.	3.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0.
3.2	RWST level greater than 12.5 ft.	3.2 Go to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION.
4	**Monitor charging miniflow criteria (during SI).**	
4.1	RCS pressure less than 1900 psig.	4.1 Verify miniflow valves open.
4.2	RCS pressure greater than 1300 psig.	4.2 Verify miniflow valves closed.
5	**Monitor adverse containment criteria.**	
5.1	CTMT pressure less than 4 psig and radiation less than 10^5 R/hr.	5.1 Utilize bracketed adverse CTMT condition numbers.

1. 062AA1.07 043/BANK/FNP 12/C/A 2.9/3.0/APE062AA1.07/N///

Unit 2 is operating at 100% power. The following conditions exist:

- SGBD is on service.
- #1 WMT release is in progress.
- The service water pond level has dropped to 179 feet, 10 inches.

Which one of the following combinations predicts the plant response to the low pond level?

A.✓ 1) SW Dilution Flow on FR-4107, SW DILUTION FLOW, will lower;

2) RCV-023B, SGBD Dilution Discharge Valve, will automatically close.

B. 1) SW Dilution Flow on FR-4107, SW DILUTION FLOW, will lower;

2) RCV-018, Liquid Waste Discharge Valve, will automatically close.

C. 1) SW Pressure on PI-3001A & B, SW TO CCW HX HDR PRESS, will lower;

2) PCV-562 and 563, Dilution Bypass Valves, will fully open.

D. 1) SW Pressure on PI-3001A & B, SW TO CCW HX HDR PRESS, will lower;

2) MOV-538 and 539, Master Recirculation Isolation Valves, will fully open.

Not a true 2+2 to improve distracter plausibility.

SOP-16.1
4.5 Defeating the Low SW Dilution Flow Trip of N2G24RCV023B
NOTES
• At low dilution flow below 14,500 GPM SGBD will isolate [...]

AOP-31
2. At a pond level of 180 ft 0 in the following sequence of events occurs

• SW A(B) HDR EMERG RECIRC TO POND valves on both units will open.
• SW HDR NORMAL DISCH ISO A(B) TRN valves on both units will close.
• SW TO WET PIT EAST(WEST) HDR ISO valves will open.
• SW TO POND EAST(WEST) HDR ISO will partially close to divert approximately 50% of the SW recirculation flow to the wet pit.

Ran on desk top simulator and Discharge pressure ROSE ~1.5 psig

__**Distracter analysis**__

A. Correct — First part is correct. When the pond level drops to 180 ft 0 in, the SW the SW system changes valve alignments such that the

emergency recircs to the pond open and the discharges from each train closes which lowers the dilution flow as seen on FR-4107.

Second part is correct. The dilution line flow drops to less than 14,500 gpm (goes to 0 gpm), which in turn causes the auto-closure of RCV-023B, terminating this Release path. S ee D200013 for line up.

B. Incorrect. First part is correct (See A.1)

Second part is incorrect (See A.2). RCV-018 does not have a low dilution line auto closure. Plausible since RCV-023B is also a radioactive release point isolation and will close on both High Radiation and Low Flow so the applicant could think it also closed on low flow.

C. Incorrect. First part is incorrect (See A.1). SW discharge pressure is virtually unchanged due to the lineup. When run on desk top simulator, pressure ROSE ~1.5 psig. Plausible since the SW header will operate on RECIRC back to the POND, the applicant may believe this would cause an lowered backpressure on SW header which would translate into a higher flow but at a lower pressure (Centrifugal pump curves).

Second part is incorrect (See A.2). Since SW discharge pressure is virtually unaffected then these valves will NOT fully open because discharge pressure is less than 110 psig. Plausible if the applicant believes that the system "DILUTION BYPASS" valves open to ensure a minimum dilution flow is maintained for Radioactive releases.

.

D. Incorrect. First part is incorrect (See C.1).

Second part is correct. These valves open on a low level in the SW pond but will not lower pressure.

K/A: **062AA1.07**	Loss of Nuclear Service Water - Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Flow rates to the components and systems that are serviced by the SWS; interactions among the components
Importance Rating:	2.9 3.0
Technical Reference:	FNP-2-SOP-16.1, SG Blowdown Processing System, Ver 43.2 FNP-0-AOP-31, Loss of Service water Pond, Ver 12 FNP-2-SOP-24, Service Water System, Ver 73 FSD-A181001, Service Water System, Ver 61 D-200013, Sh 8, Service Water System, Ver 36
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Service Water System components and equipment, to include the following (OPS-40101B07): [...] • Automatic actuation including setpoint (example SI, Phase A, LOSP) • Protective isolations such as high flow, low pressure, low level including setpoint • Protective interlocks
—	[...]
Question History:	FNP 12. The bank was checked and this question is the only one that meets this K/A. We have spent hours developing this question and have encountered difficulty in the final product due to our system design.
K/A match:	Requires the applicant to know what they are expecting to see **(monitor) on the MCB (PI-3001 and FR4107) and what will occur due to the flow to other system components (interactions among the components).** The candidate will have to know what happens to the SW system on low pond level (loss of SW) and then the effects of the new valve line up on system pressure and flow to other system components (ie. RCV-18 and 23B and PCV-562 and MOV-538).
SRO justification:	N/A

"B" SW return line. Train "A" power shall be provided to the other two valves in each train of the SW return line. (Reference 6.4.151 - 6.4.158 and Engineering Judgement)

3.46.7 Failure Modes and Effects Analysis

These motor operated valves are normally open and fail "as-is" on loss of power. These valves can be manually isolated if necessary by the handwheel located on the valve operator.

Since each train of service water has two isolation valves, Turbine Building isolation is possible even with a single valve failure. Additionally, each isolation valve on a single train is powered by different trains, one by electrical train A and one by electrical train B. Turbine Building isolation is therefore possible with only one electrical train operating.

A spurious action closing one of these valves would prevent SW return flow to that train. Turbine Building flow would be reduced and the return flow would be forced into the other train. Such a condition would be indicated to the plant operators through the valve position indicating lights in the control room.

3.47 SERVICE WATER DILUTION BYPASS CONTROL VALVES

TPNS Nos	Unit 1	Unit 2
Train A	Q1P16V563	Q2P16V563
Train B	Q1P16V562	Q2P16V562

3.47.1 Basic Functions

3.47.1.1 These valves provide over pressurization protection for each train of service water by bypassing a portion of the service water supply flow when the service water supply pressure exceeds 110 psig. (References 6.4.040, 6.4.073, 6.4.078, 6.4.103, 6.4.196, 6.4.239, 6.7.141 and 6.7.142)

3.47.1.2 These valves provide an alternate flow path during outage conditions to direct service water supply flows to the dilution line during periods of low dilution line flow. (Reference 6.1.001)

3.47.1.3 These valve actuators have been modified to limit valve disc travel to 30 degrees in the open direction. This modification allows the valve to adequately relieve any over pressure

condition while limiting the impact on the service water system if the valve failed open. (Reference 6.7.141 and 6.7.142)

3.47.2 Functional Requirements

3.47.2.1 The valves must be designed for pressure and temperature conditions of 150 psig and 200°F. (Reference 6.5.015)

3.47.2.2 The valve shall have a maximum design flow of 15,000 gpm with a corresponding maximum pressure drop of 85 psig. The valve travel has been limited to 30 degrees in the open direction. (Reference 6.5.015, 6.7.141 and 6.7.142)

3.47.2.3 The valve operator and associated pressure controller shall be designed to modulate the valve from full closed to full open with an air supply of 80 to 100 psig. The valve travel has been limited to 30 degrees in the open direction in order to minimize the inventory loss from the Service Water System. In addition, the actuator shall be furnished with needle valves to enable to adjust the response time (0 sec. -30 sec., adjustable) in order to prevent slamming of the disc. (Reference 6.5.015, 6.7.141, 6.7.142, 6.7.201 and 6.7.202)

3.47.2.4 Each valve shall be provided with a spring-return type actuator to close the valve in the event of a loss of air supply. (References 6.5.015, 6.7.141, 6.7.142, 6.7.201 and 6.7.202).

3.47.2.5 A handwheel shall be provided on each valve operator to manually stroke the valve. (References 6.4.103, 6.4.196, 6.4.239, 6.4.240, 6.4.261, 6.4.262, 6.4.264, and 6.4.270)

3.47.2.6 These valves shall fail closed upon a loss of air supply. (Reference 6.5.015)

3.47.3 Code Requirement

3.47.3.1 These valves shall be designed and fabricated in accordance to ASME B&PV Code, Section III, Class 3, 1971 Edition including addenda through summer of 1972 along with additional codes and standards are listed in sections 6.6 and 6.7 of Specification SS-1102-036. (Reference 6.5.015)

TURBINE BUILDING
AUXILIARY BUILDING
VALVE BOX 2VB-1
VALVE BOX 2VB-2
VALVE BOX 2VB-3
VALVE BOX 2VB-4
VALVE BOX 2VB-5
DIESEL GENERATOR BUILDING
DILUTION BY-PASS
TO STORAGE POND FLUME
TO RIVER
VENT TO ATMOSPHERE
CIRCULATING WATER CANAL
COOLING TOWER BLOWDOWN
WASTE PROCESSING SYS.
TURB. BLDG. SUMP DISCH.
CHEMICAL INJECTION SYSTEM SUPPLIED BY CONTRACTOR
STANDPIPE SURGE TANK
PS CONTACTS SET TO CLOSE AT 60 PSIG DECREASING
PDS CONTACTS SET TO CLOSE AT 11 PSID
FROM UNIT 1 "A" TRAIN RETURN (V539)
FROM UNIT 1 "B" TRAIN RETURN (V538)
New flow path
Close on low pond level
Open on low pond level
FR-4107
Normal flow
FOR UNIT 1 COUNTERPART SEE D-170119 SH.2
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Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT – UNIT NO.2
P&ID – RIVER WATER, SERVICE WATER AND CIRCULATING WATER SYSTEM
VER. NO. 36.0 DATE 11-29-12
REVISED PER ABN F02805, VER. 1.0
SCALE NONE
SHEET 8
D-200013
CAD D2000138 ACAD2K JJB – 10

A. Purpose

This procedure provides actions for water pond.

This procedure is applicable at all

This procedure is Revision 12. Some pages have revision 11 in the header.

B. Symptoms or Entry Conditions

I. This procedure is entered when a loss of the service water pond is indicated by any of the following.

a. Actuation of Unit 1 SW POND LVL A TRN LO annunciator AG3 or SW POND LVL B TRN LO annunciator AH3 (184 ft 4 in)

b. Actuation of Unit 1 SW WET PIT LVL A TRN LO annunciator AG4 or SW WET PIT LVL B TRN LO annunciator AH4 (170 ft 0 in)

C. Operational Concerns

1 RW TO POND A(B) TRN valves will close and RW EMERG SUPP TO SW WET PIT valves will open at 184 ft 0 in.

2 At a pond level of 180 ft 0 in the following sequence of events occurs

[] SW A(B) HDR EMERG RECIRC TO POND valves on both units will open.
[] SW HDR NORMAL DISCH ISO A(B) TRN valves on both units will close.
[] SW TO WET PIT EAST(WEST) HDR ISO valves will open.
[] SW TO POND EAST(WEST) HDR ISO will partially close to divert approximately 50% of the SW recirculation flow to the wet pit.

SOP-24 addresses these valves

3 SW pond level will fall to 181 ft 10 in within approximately 4 days following loss of makeup.

4 Technical Specification 3.7.9 details SW pond level LCO requirements.

5 [CA] Indicates a continuing action step.

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Step 2.3 will isolate CW canal makeup and the water treatment plant clarifier.
- Step 2.3 will isolate flow to the SW dilution. No liquid waste releases are permitted.

2.3 Close SW normal discharge valves.

SW HDR NORMAL
DISCH ISO A(B) TRN
[] Q1P16V546
[] Q1P16V545

SW HDR NORMAL
DISCH ISO A(B) TRN
[] Q2P16V546
[] Q2P16V545

2.4 Close Cooling Tower Blowdown Isolation Valves.

UNIT 1 CTBD ISO
[] N1P16V586
UNIT 2 CTBD ISO
[] N2P16V586

___ 3 **[CA]<u>IF</u> SW pond level less than 184 ft 0 in, <u>THEN</u> place both units in Mode 5 in accordance with appropriate plant procedures as needed.**

[] Trip units per FNP-1(2)-EEP-0
[] RAPID TURBINE POWER REDUCTION per FNP-1(2)-AOP-17.1
<u>OR</u>
[] Normal Ramp per UOP path to cold shutdown.

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Step 2.3 will isolate CW canal makeup and the water treatment plant clarifier.
- Step 2.3 will isolate flow to the SW dilution. No liquid waste releases are permitted.

2.3 Close SW normal discharge valves.

SW HDR NORMAL
DISCH ISO A(B) TRN
[] Q1P16V546
[] Q1P16V545

SW HDR NORMAL
DISCH ISO A(B) TRN
[] Q2P16V546
[] Q2P16V545

2.4 Close Cooling Tower Blowdown Isolation Valves.

UNIT 1 CTBD ISO
[] N1P16V586
UNIT 2 CTBD ISO
[] N2P16V586

___ 3 **[CA]<u>IF</u> SW pond level less than 184 ft 0 in,**
<u>THEN</u> place both units in Mode 5 in accordance with appropriate plant procedures as needed.

[] Trip units per FNP-1(2)-EEP-0
[] RAPID TURBINE POWER REDUCTION per FNP-1(2)-AOP-17.1
<u>OR</u>
[] Normal Ramp per UOP path to cold shutdown.

4.4.6 **Determine** SGBD total flow as follows:

(Final level - Initial level) × 14.45 gal/% = ____________ gal/min
{Step 4.4.3} {Step 4.4.1} 4 min

4.5 Defeating the Low SW Dilution Flow Trip of N2G24RCV023B

NOTES

- At low dilution flow below 14,500 GPM SGBD will isolate and defeating the SW Dilution Flow Trip will be required to maintain SGBD flow.
- For SGBD continuous releases dilution flow rate should be greater tahn or equal to 10,000 GPM unless approval is obtained from Chemistry Supervision (10CFR20 and dose acceptance criteria are met) to use a lower flow rate. (ref. FNP-ODCM sec 2.0) ☐

4.5.1 IF SGBD discharge is required AND the following conditions can not be met:

- Dilution flow above 14,500 GPM, ☐
- N2P16FR4107, S.W. DILUTION FLOW RECORDER, is operable, ☐

THEN, the dilution line low flow trip of N2G24RCV023B (N2G24V138, SGBD DISCH TO ENVIRONMENT) may be defeated using:

- Appendix 1 Defeating the Low SW Dilution Flow Trip of N2G24RCV023B (N2G24V138) ☐
- FNP-2-STP-212.1, Service Water Discharge Dilution Line Flow N2P16FT0571 Loop Calibration and Functional Test. ☐

3.13 The service water emergency recirc to pond MOVs operate as follows:

3.13.1 The A Train MOV Q2P16V539-A will open on:

- Lo-Lo pond level 180'0"
- Hi-Hi surge tank level 191'1"
- Hi dilution line press 40 psig
- Lo dilution line press 5 psig

3.13.2 The associated SW to dilution line MOV Q2P16V546-A will close once the emergency recirc valve is open with the exception of Hi dilution line press above. In this instance, valve Q2P16V546-A will remain open.

3.13.3 The B Train MOV Q2P16V538-B will open on:

- Lo-Lo pond level 180'0"
- Hi surge tank level 189'7"
- Hi dilution line press 40 psig
- Lo dilution line press 5 psig

3.13.4 The associated SW to dilution line MOV Q2P16V545-B will close once the emergency recirc valve is open with the exception of Hi surge tank level and Hi dilution line press above. In these instances, valve Q2P16V545-B will remain open.

3.14 The power must be removed from 2C SW pump discharge valves Q2P16V506 and Q2P16V507, EXCEPT when shifting trains, to meet appendix R requirements.

3.15 Power is removed from SW TO 2B CCW HX valve Q2P16MOV3130B to meet Appendix R requirements when 2B CCW HX is in service on A Train.

3.16 The motor heaters should be turned on in any pump that is not running to prevent moisture buildup in the windings. The motor heaters should be off for any pump which is running.

3.17 If SW TO DILUTION LINE Q2P16V549 must be closed, then B and A HDR EMERG RECIRC TO POND Q2P16V538-B and Q2P16V539-A must be opened prior to closing Q2P16V549 and power must be removed from Q2P16V538-B and Q2P16V539-A to prevent spurious valve closure.

3.18 Shifting the Service Water discharge from pond recirculation to river discharge places a large transient on the system. To minimize the transient, the discharge should be shifted one train at a time on only one unit at a time. At least five minutes should be allowed between each train shifted (Ref. letter NT-82-0692).

3.19 Service Water supply to the standby CCW HX must be closed, except when shifting the standby heat exchanger.

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. 062K3.01 044/BANK/FNP EXAM BANK/MEM 3.5/3.9/062K3.01/N///

The following conditions exist on Unit 2:

- DG02-2, 2G 4160 V bus tie to 2L 4160 V bus, has tripped opened.

Which one of the following completes the statement below?

The ____ has lost Service Water cooling.

A. 2C Instrument Air Compressor

B.✓ 2C Reactor Coolant Pump Motor Air Cooler

C. 2C Component Cooling Water Heat Exchanger

D. Steam Generator Blowdown Heat Exchanger

AOP-10:
Step 15. Minimize SW loads in affected train.
15.2 For **'A' train** affected minimize 'A' TRAIN SW LOADS as required.
15.2.1 **Secure SGBD** using FNP-2-SOP-16.1, STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM.
15.2.2 Close SW to blowdown and BTRS heat exchangers valve.
SW TO BLDN HX & BTRS CHLRS
[] Q2P16MOV3149 - closed

15.3 For **'B' train** affected minimize B TRAIN SW LOADS, as required.
15.3.1 **Close SW to RCP motor air coolers.**
SW TO RCP
MTR AIR CLRS
[] Q2P16MOV3135 - closed

U2 Load List:
2L 4160V bus is the power supply to the B Train SW pumps. When that power supply is lost, All B Train SW pumps will be lost and cooling to B Train components are affected.

Distracter analysis

A. Incorrect.	See B. Plausible since the 2C designation could make the applicant believe this is a 'B' train component. All instrument air compressors are normally supplied from a common SW header, which is fed from both trains of SW.
B. Correct.	ALL RCP motor air coolers are supplied from "B" Train SW.
C. Incorrect.	See B. 2C CCW Heat Exchanger is supplied from "A" Train SW. Plausible since 2C is an A Train component and 2A is B Train component and this is a common mistake made for these components.
D. Incorrect.	See B. The SGBD Heat Exchanger is supplied only from "A" Train SW. Plausible since this and the RCPs each are supplied from different trains and a common mistake made by students as to which train supplies which components.

K/A: **062K3.01**	A.C. Electrical Distribution - Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: Major system loads
Importance Rating:	3.5 3.9
Technical Reference:	FNP-2-AOP-10, Loss of Service Water, Ver 18 A-351199, Unit 2 Load List, Ver 61.
References provided:	None
Learning Objective:	RELATE AND DESCRIBE the effect(s) on the Service Water System for a loss of an AC or DC bus, or a malfunction of the Instrument Air System (OPS-40101B06).
Question History:	FNP EXAM BANK
K/A match:	**The 2L 4160V bus has been lost due to a malfunction** and the **effect is the loss of cooling to various major systems loads. The applicant will have to know which SW pumps have lost power and then equate that to which major system load has lost cooling.**
SRO justification:	N/A

2L 4160V BUS

SECTION L
TABLE OF CONTENTS
SORT BY PAGE # and 4KV BREAKER

2L 4160V BUS SW - 188' DWG D-207044

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q2R15A0506-B	**2L 4160V BUS <<< DG02**	
DL01	Q2R15BKRDL01	PT COMPARTMENT	
DL02	**Q2R11B0505-B**	**2L 4160/600V SST >>> EL08**	L-2
DL03	Q2P16M0001D-B	2D SW PUMP	
DL04	Q2P16M0001E-B	2E SW PUMP	
DL05	Q2P16M0001C-AB	2C SW PUMP (B TRAIN SUPPLY)	

TURBINE BUILDING
AUXILIARY BUILDING
VALVE BOX VB-1
VALVE BOX VB-2
VALVE BOX VB-3
VALVE BOX VB-4
COOLING TOWER BLOWDOWN
DIESEL GENERATOR BUILDING
Supply to air compressor
D-170119 SH.5
D-170119 SH.4
24" HBD-255
24" HBD-254
4" HBD-346
ABANDONED
D-170071
TRAIN "B"
TRAIN "A"
D-175003 SH.1
30" X 24" RED
24" HBC-26
24" HBC-46
30" HBC-211
24" HBC-210
24" HBC-209
12" HBC-201
DILUTION BY-PASS
LOCA SIGNAL
LOSP SIGNAL
STORAGE POND FLUME
SERVICE WATER WETPIT LEVEL
CIRCULATING WATER CANAL
SURFACE MOUNTED ULTRASONIC TRANSDUCERS
RADWASTE DISCHARGE
STEAM GEN. BLOWDOWN
TURB BLDG. SUMP DISCH.
SAMPLE LINE AT CIRCULATING WATER PUMP STRUCTURE
CHEMICAL INJECTION NOZZLE
CHEMICAL INJECTION SYS. SUPPLIED BY CONTRACTOR
STANDPIPE SURGE TNK
MANHOLE
TO RIVER
FROM UNIT 2
60" KBD-200
SERVICE WATER TRAIN "A"
SERVICE WATER TRAIN "B"
HEADER
DETAIL "A"
FOR GENERAL NOTES SEE SH.1
CAD D1701192.DWG
ACAD2010 Vep - 04
Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
P&ID - SERVICE WATER SYSTEM
SCALE NONE
SHEET 2
D-170119
REVISED PER ABN SNC461051M002, VER. 1.0
47.0 2-19-13
FOR SIGNATURES
SEE MICROFILM
COPY
DRAWING CATEGORY: CRITICAL
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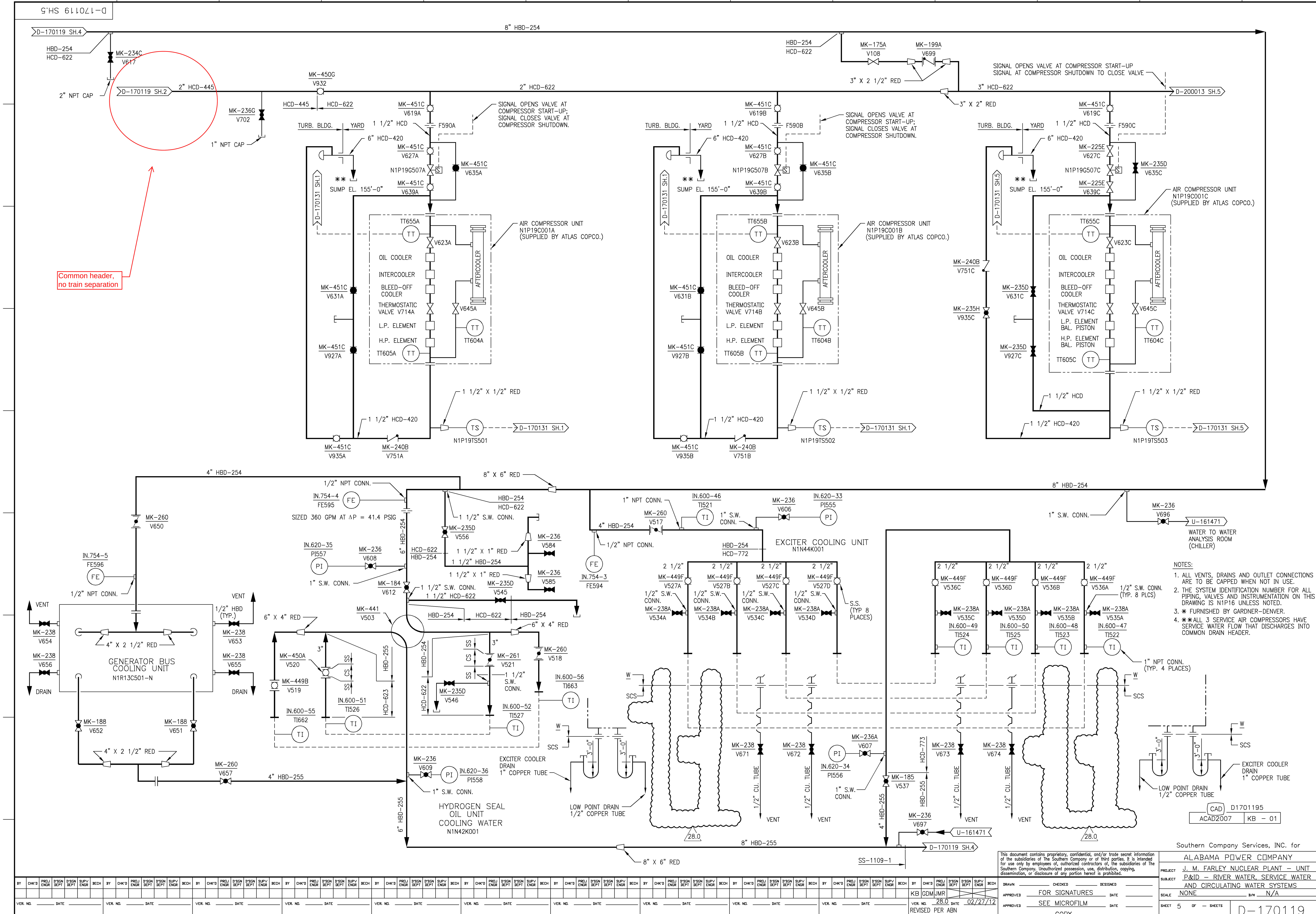
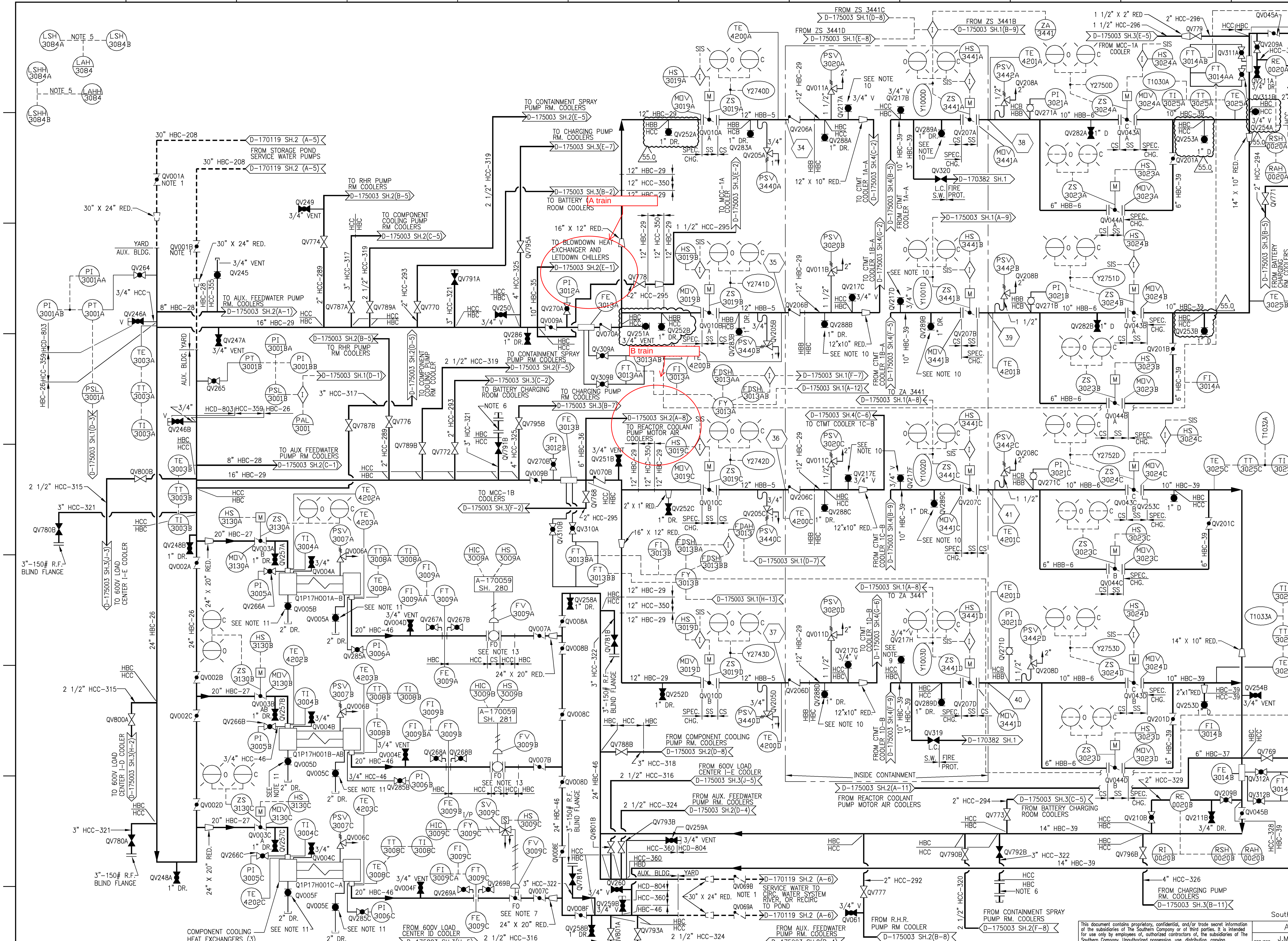

D-170119 SH.5
8" HBD-254
2" HCD-622
3" HCD-622
AIR COMPRESSOR UNIT N1P19C001A (SUPPLIED BY ATLAS COPCO.)
AIR COMPRESSOR UNIT N1P19C001B (SUPPLIED BY ATLAS COPCO.)
AIR COMPRESSOR UNIT N1P19C001C (SUPPLIED BY ATLAS COPCO.)
SIGNAL OPENS VALVE AT COMPRESSOR START-UP; SIGNAL CLOSES VALVE AT COMPRESSOR SHUTDOWN.
SIGNAL OPENS VALVE AT COMPRESSOR START-UP SIGNAL AT COMPRESSOR SHUTDOWN TO CLOSE VALVE
OIL COOLER
INTERCOOLER
BLEED-OFF COOLER
THERMOSTATIC VALVE V714A
L.P. ELEMENT
H.P. ELEMENT
AFTERCOOLER
TURB. BLDG.
YARD
SUMP EL. 155'-0"
Common header, no train separation
GENERATOR BUS COOLING UNIT N1R13C501-N
HYDROGEN SEAL OIL UNIT COOLING WATER N1N42K001
EXCITER COOLING UNIT N1N44K001
SIZED 360 GPM AT ΔP = 41.4 PSIG
WATER TO WATER ANALYSIS ROOM (CHILLER)
EXCITER COOLER DRAIN 1" COPPER TUBE
LOW POINT DRAIN 1/2" COPPER TUBE
8" HBD-255
NOTES:
1. ALL VENTS, DRAINS AND OUTLET CONNECTIONS ARE TO BE CAPPED WHEN NOT IN USE.
2. THE SYSTEM IDENTIFICATION NUMBER FOR ALL PIPING, VALVES AND INSTRUMENTATION ON THIS DRAWING IS N1P16 UNLESS NOTED.
3. ✳ FURNISHED BY GARDNER-DENVER.
4. ✳✳ALL 3 SERVICE AIR COMPRESSORS HAVE SERVICE WATER FLOW THAT DISCHARGES INTO COMMON DRAIN HEADER.
CAD D1701195 ACAD2007 KB - 01
Southern Company Services, INC. for
ALABAMA POWER COMPANY
PROJECT J. M. FARLEY NUCLEAR PLANT - UNIT 1
SUBJECT P&ID - RIVER WATER, SERVICE WATER AND CIRCULATING WATER SYSTEMS
SCALE NONE
SHEET 5
D-170119
REVISED PER ABN 1060838201M004, VER. 1.0
28.0 DATE 02/27/12
FOR SIGNATURES SEE MICROFILM COPY
DRAWING CATEGORY: CRITICAL



NOTES:

1. VALVE IS LOCATED OUTSIDE AUXILIARY BUILDING
2. RELIEF VALVE MUST BE LOCATED AT LEAST 6" ABOVE THE TOP OF THE LOOP SEAL. BOTTOM OF THE LOOP MUST EXTEND AT LEAST 6" BELOW THE CONNECTION TO THE MAIN PIPE.
3. FOR COMPLETE INSTRUMENT NUMBERS, REFER TO INSTRUMENT INDEX B-175803.
4. LINE NUMBERS FOR VENT, DRAIN, AND INSTRUMENT LINES ON EQUIPMENT, UP TO AND INCLUDING THE ROOT VALVES:
   FOR HEAT EXCHANGERS AND HVAC COOLERS SAME AS SERVICE WATER OUTLET LINE NUMBERS.
   FOR PIPING DOWNSTREAM OF VENT AND DRAIN ROOT VALVES, SEE SPEC. SS-1109-2, PARA. 11.5.1(b).
   RELIEF VALVE DISCHARGE LINES SHALL BE IN ACCORDANCE WITH SPEC. SS-1109-2, PARA, 11.5.1(e).
5. COMPONENT COOLING WATER PUMP AREA FLOOD DETECTORS.
6. REMOVABLE FLANGED SPOOL PIECE. REQUIRED FOR SEISMIC 1 INTEGRITY OF PIPING STUB.
7. CCW HX FLOW CONTROL VALVE FV3009C IS LIMITED TO A NOMINAL 35° OPEN POSITION.
8. ALL VALVE AND LINE NUMBERS BEGIN WITH THE PREFIX OF Q1P16 OR N1P16.
9. VALVE Q1P16V217H IS PIPING CLASS HCC PER MDC 1040935001. THE PIPING NIPPLES WHICH CONNECT VALVE Q1P16V217H TO THE PROCESS ARE SCHEDULE 80 STAINLESS STEEL PER MDC 1040935001.
10. Q1P16V217A,D,E,&F; Q1P16V288B&D; Q1P16V289A,B,&C; AND Q1P16V216A ARE PIPING CLASS HCC PER MDC 1051657401. THE PIPING NIPPLES WHICH CONNECT THESE VALVES TO THE PROCESS ARE SCHEDULE 80 SS PER MDC 1051657401.
11. Q1P16V005A, B, C, D, E & F, ARE PIPING CLASS HCC PER MDC 1051657401. THE PIPING NIPPLES WHICH CONNECT THESE VALVES TO THE PROCESS ARE SCHEDULE 80 SS PER MDC 1051657401.
12. PIPE FLANGE ARE 300# FLANGES TO MATCH COOLER FLANGES SUPPLIED BY VENDOR.
13. CCW HX FLOW CONTROL VALVES FV3009A AND B ARE LIMITED TO A NOMINAL 45° OPEN POSITION.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205003, SH. 1

CAD D1750031
ACAD2K | KB - 06

KB WLM JMR
55.0 06/25/12
REVISED PER ABN 1092714601M002, VER. 1.0.

DRAWN ______ CHECKED ______ DESIGNED ______
APPROVED FOR SIGNATURES DATE ______
APPROVED SEE MICROFILM DATE ______
APPROVED COPY DATE ______

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1

SUBJECT P&ID – SERVICE WATER SYSTEM

SCALE NONE B/M

SHEET 1 OF — SHEETS

SUPERSEDES

D-175003

DRAWING CATEGORY: CRITICAL

Step	Action/Expected Response	Response NOT Obtained

___15 **Minimize SW loads in affected train.**

15.1 If required, secure BTRS chillers using FNP-2-SOP-3.0, BORON THERMAL REGENERATION SYSTEM.

15.2 For 'A' train affected minimize 'A' TRAIN SW LOADS as required.

15.2.1 Secure SGBD using FNP-2-SOP-16.1, STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM.

15.2.2 Close SW to blowdown and BTRS heat exchangers valve.

SW TO BLDN HX
& BTRS CHLRS
[] Q2P16MOV3149 - closed

15.2.3 Stop 'A' train containment cooler fans.

2A CTMT CLR FAN
SLOW SPEED
[] Q2E12H001A - stopped

2A CTMT CLR FAN
FAST SPEED
[] Q2E12H001A - stopped

2B CTMT CLR FAN
SLOW SPEED
[] Q2E12H001B - stopped

2B CTMT CLR FAN
FAST SPEED
[] Q2E12H001B - stopped

Step 15 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
15.3	For 'B' train affected minimize B TRAIN SW LOADS, as required.	
15.3.1	Close SW to RCP motor air coolers. SW TO RCP MTR AIR CLRS [] Q2P16MOV3135 - closed	
15.3.2	Stop 'B' train containment cooler fans. 2C CTMT CLR FAN SLOW SPEED [] Q2E12H001C - stopped 2C CTMT CLR FAN FAST SPEED [] Q2E12H001C - stopped 2D CTMT CLR FAN SLOW SPEED [] Q2E12H001D - stopped 2D CTMT CLR FAN FAST SPEED [] Q2E12H001D - stopped	

Step 15 continued on next page.

1. 063A1.01 045/MOD/FNP 11/12/MEM 2.5/3.3/063A1.01/N///

Unit 1 has experienced a Reactor trip with the following conditions:

- A Loss of All AC has occurred.
- ECP-0.0, Loss of All AC Power, is in progress.

Which one of the following completes the statements below?

The 1B Aux Building DC bus voltage will __(1)__ .

Per ECP-0.0, there may not be enough DC capacity to start a DG and sequence needed loads if power is not restored to the 125V DC battery chargers on each train within a MINIMUM of __(2)__.

A.✓ 1) drop slowly at first; then later drop rapidly as the battery nears exhaustion

2) 30 min

B. 1) drop slowly at first; then later drop rapidly as the battery nears exhaustion

2) 90 min

C. 1) drop at a constant, linear rate the entire time the battery discharges

2) 30 min

D. 1) drop at a constant, linear rate the entire time the battery discharges

2) 90 min

DOE Fundamentals Handbook Vol 2 of 4, Jun 1992 (This is a reference for lesson plan OPS-30501D, Batteries.) - During Battery discharge, voltage will slowly drop until the battery approaches exhaustion. As the battery approaches exhaustion, voltage will decrease exponentially until exhaustion.

ECP-0.0 Caution prior to Step 5:
IF power is not restored to the 125 V DC battery chargers on each train within 30 minutes, THEN there may not be enough DC capacity to start a DG and sequence needed loads.

Distracter analysis:

A. Correct. First part is correct. The battery voltage will drop slowly then at an exponential rate towards the end of discharge per the graph in the references.

Second part is correct. Per the Note, 30 minutes is the minimum time in which the battery charger must be restored to ensure the DG can start and sequences loads.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.1). Plausible since the design capacity of the Aux building battery is 2 hours. 90 minutes would give a 30 minute buffer so the applicant could confuse the 30 minutes in the note with "30 minutes left" of the 2 hour design battery capacity.

C. Incorrect. First part is incorrect (See A.1). Plausible if the applicant is not familiar with battery discharge characteristics.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2).

K/A: **063A1.01**	D.C. Electrical Distribution - Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate
Importance Rating:	2.5 3.3
Technical Reference:	FNP-1-ECP-0.0, Loss of All AC Power, Ver 26. DOE Fundamentals Handbook Vol 2 of 4, Jun 1992
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with (1) ECP-0.0, Loss of All AC Power; [...] P-0.2, Loss of All AC Power Recovery, With SI Required. (OPS-52532A03)
Question History:	MOD FNP11/12 NRC exam
K/A match:	Requires the applicant to **predict the change in battery voltage (parameters) as the battery is discharged (capacity versus discharge rate)** during a Loss of All AC event. In addition, knowledge of the time expected to restore the battery charger (30 mins) to prevent the LOSS of DC POWER, which equates to the time limit (how long can we operate this way) that the battery capacity is affected.
SRO justification:	N/A

E3.0 CRITICAL COMPONENT FUNCTIONAL DESIGN REQUIREMENTS

This section contains functional design requirements for all major components contained within the auxiliary building 125 V dc Class 1E electrical distribution system that are considered to be critical to safety-related system functions (i.e., failure of the component could lead to loss or impairment of safety-related system functions). The functions of each component, in addition to performance and interface requirements, are discussed in the following sections.

E3.1 125 V DC BATTERIES

TPNS Nos. QR42E002A-A
QR42E002B-B

E3.1.1 Basic Functions

The auxiliary building Class 1E 125 V station batteries shall provide power to the required dc and vital ac loads in case of loss of auxiliary system power or in the event of failure of a battery charger (Reference E6.1.001).

E3.1.2 Functional Requirements

E3.1.2.1 The 125 V station batteries shall be stationary type, consisting of 60 cells connected in series to establish a nominal 125 V station dc power supply (Reference E6.7.020). Under both normal and accident conditions the batteries shall be capable of providing the voltage required for operation of the non-safety-related and safety-related components considering an aging factor of 25% and electrolyte temperature within the range of 60°F to 110°F (Reference E6.7.025).

E3.1.2.1.1 During the normal plant operation, the batteries shall be capable of carrying the loads necessary to support plant operation for two hours. The two hour duration is based on the time required for the operators to connect the spare battery charger to the system if the connected battery charger fails on either train. During this two hour period, the redundant train of the dc system with operable battery charger is available for accident mitigation, if required. The normal load on the batteries during the two hour period will not exceed 250 amps for batteries 1A, 2A and 2B, and 300 amps for battery 1B (Reference E6.7.025).

The decrease in specific gravity on discharge is proportional to the ampere-hours discharged. While charging a lead-acid battery, the rise in specific gravity is not uniform, or proportional, to the amount of ampere-hours charged (Figure 6).

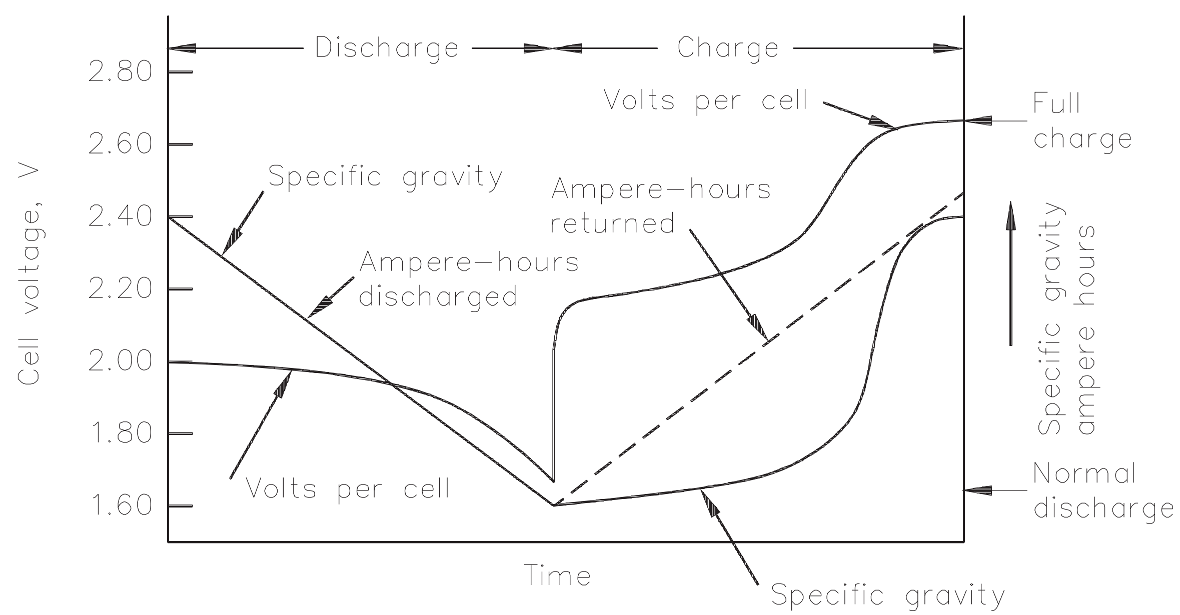


Figure 6 Voltage and Specific Gravity During Charge and Discharge

The electrolyte in a lead-acid battery plays a direct role in the chemical reaction. The specific gravity decreases as the battery discharges and increases to its normal, original value as it is charged. Since specific gravity of a lead-acid battery decreases proportionally during discharge, the value of specific gravity at any given time is an approximate indication of the battery's state of charge. To determine the state of charge, compare the specific gravity, as read using a hydrometer, with the full charge value and the manufacturer's published specific gravity drop, which is the decrease from full to nominal charge value.

Example: A lead-acid battery reads 1.175 specific gravity. Its average full charge specific gravity is 1.260 and has a normal gravity drop of 120 points (or.120) at an 8 hour discharge rate.

Solution:

Fully charged - 1.260
Present charge - 1.175

The battery is 85 points below its fully charged state. It is therefore about 85/120, or 71%, discharged.

Step	Action/Expected Response	Response NOT Obtained

4.3 Verify AFW flow path to each SG.

TDAFWP FCV 3228
[] RESET reset

TDAFWP
TO 1A(1B,1C) SG
[] Q1N23HV3228A in MOD
[] Q1N23HV3228B in MOD
[] Q1N23HV3228C in MOD

TDAFWP TO 1A(1B,1C) SG
FLOW CONT
[] HIC 3228AA open
[] HIC 3228BA open
[] HIC 3228CA open

****

CAUTION: IF power is not restored to the 125 V DC battery chargers on each train within 30 minutes, THEN there may not be enough DC capacity to start a DG and sequence needed loads.

****

___ 5 **[CA] Restore power to any emergency bus.**

5.1 Verify supply breakers for major loads on emergency 4160 V busses - OPEN.

[] BKR DF01 (1A S/U XFMR TO 1F 4160 V BUS)
[] BKR DF15 (1B S/U XFMR TO 1F 4160 V BUS)
[] BKR DF-13-1 (1F 4160 V BUS TIE TO 1H 4160 V BUS)
[] BKR DG01 (1A S/U XFMR TO 1G 4160 V BUS)
[] BKR DG15 (1B S/U XFMR TO 1G 4160 V BUS

[] 1C CCW PUMP BKR DF-04-1
[] 1B CCW PUMP BKR DF-05-1
[] 1B CCW PUMP BKR DG-05-1
[] 1A CCW PUMP BKR DG-04-1

Step 5 continued on next page.

1. 063A1.01 046/MOD/RO/M 2.5/3.3/063A1.01/N///A

Unit 1 has experienced a Reactor trip with the following conditions:

- A Loss of All AC has occurred.
- ECP-0.0, Loss of All AC Power, is in progress.

Which one of the following completes the statements below?

The 1B Aux Building DC bus voltage will __(1)__ .

DC loads are minimized in ECP-0.0 to __(2)__ .

A.✔ 1) drop slowly at first; then later drop rapidly as the battery nears exhaustion

2) prolong battery life ONLY

B. 1) drop slowly at first; then later drop rapidly as the battery nears exhaustion

2) prevent damage to the DC components AND prolong battery life

C. 1) drop at a constant, linear rate the entire time the battery discharges

2) prolong battery life ONLY

D. 1) drop at a constant, linear rate the entire time the battery discharges

2) prevent damage to the DC components AND prolong battery life

QUESTIONS REPORT

for 063A1.01 FNP 12

1. 058AK1.01 043/MOD/FNP-NO NRC/MEM 2.8/3.1/APE058AK1.01/N///

Which one of the following completes the statement below in accordance with the CAUTION listed in FNP-1-ECP-0.0, Loss of All AC Power?

CAUTION: If power is not restored to the 125V DC battery chargers on each train within ____, THEN _____ .

	TIME	CONSEQUENCE
A.	30 minutes	the TDAFW Pump will become unreliable
B.✓	30 minutes	there may not be enough DC capacity to start a DG and sequence needed loads
C.	90 minutes	the TDAFW Pump will become unreliable
D.	90 minutes	there may not be enough DC capacity to start a DG and sequence needed loads

1. 064K3.02 046/BANK/FNP 07/C/A 4.2/4.4/064K3.02/N///

A loss of all AC power has occurred on Unit 1 and the following conditions exist:

- VA2, 1B DG GEN FAULT TRIP, has come into alarm.
- The crew has completed the step in ECP-0.0, Loss Of All AC Power, to verify breakers for major loads OPEN.
- A Safety Injection occurs on Unit 1 at this time.

Which one of the following completes the statements below?

The 2C DG will be started from the EPB in _(1)_ using the START pushbutton.

All ESF loads will _(2)_ .

A. 1) Mode 2

2) automatically start

B. 1) Mode 2

2) have to be manually aligned

C. 1) Mode 1

2) automatically start

D.✓ 1) Mode 1

2) have to be manually aligned

ECP-0.0
5.2.1 RNO Perform 2C DG SBO start as follows.

5.2.1.1 RNO Verify 2C DG MODE SELECTOR switch in MODE 1.

Note before Step 5.2.1.5 -
NOTE: The LOSP sequencer should run when output breaker closes, if no SI signal is present. If an SI signal is present, neither sequencer will run and SI loads must be started manually.

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since all other DGs would be started in Mode 2 in ECP-0.0.

Second part is incorrect (See D.1). Plausible since the ESF sequencer would run if it were the 1-2A or 1B DG that was started. The operation of the 2C DG in this scenario is complicated and easily confused.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2)

C. Incorrect. First part is correct (See D.1)

Second part incorrect (See A.2)

D. Correct. First part is correct. Step 5.2.1.1 RNO of ECP-0.0 starts the 2C DG in Mode 1.

Second part is correct. The note before step 5.2.1.5 RNO of ECP-0.0 states that under the conditions in the stem, the SI sequencer will NOT run and ESF loads must be manually aligned.

K/A: **064K3.02**	Emergency Diesel Generators (ED/G) - Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following ESFAS controlled or actuated systems.
Importance Rating:	4.2 4.4
Technical Reference:	FNP-1-ECP-0.0, Loss of All AC Power, Ver 26
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing (1) ECP-0.0, Loss of All AC Power; [...] (OPS-52532A06) ANALYZE plant conditions and DETERMINE if actuation or reset of any Engineered Safety Features Actuation Signal (ESFAS) is necessary. (OPS-52532A05)
Question History:	FNP 07
K/A match:	This requires the applicant to **know what effect a 1B DG malfunction has on the ESFAS system in that ESF loads must be manually aligned.**
SRO justification:	N/A

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: A running diesel generator will overheat if adequate SW flow is not provided within 3 minutes. Steps 5.3 through 5.7 must be performed immediately to verify adequate SW flow once a diesel generator has been started.

****

NOTE: Load shed of bus loads must be completed prior to closing the output breaker for any diesel generator.

5.2 Check 1-2A, 1C or 1B diesel generator running for Unit 1.

- Check DIESEL SPEED indication - GREATER THAN 0 rpm.

OR

- Check FREQUENCY METER indication - GREATER THAN 58 Hz.

OR

- Check DIESEL AT SYN SPEED light - LIT.

5.2 Perform the following:

5.2.1 Perform 2C DG SBO start as follows.

5.2.1.1 Verify 2C DG MODE SELECTOR switch in MODE 1.

5.2.1.2 Place 2C DG UNIT SELECTOR switch in UNIT 1.

5.2.1.3 WHEN load shed verified, THEN depress 2C DG DIESEL START pushbutton.

5.2.1.4 Verify 2C DG starts.

Step 5 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

NOTE: The LOSP sequencer should run when output breaker closes, if no SI signal is present. If an SI signal is present, neither sequencer will run and SI loads must be started manually.

Step	Action/Expected Response	Response NOT Obtained
		5.2.1.5 Check Unit 1 2C DG output breaker DJ06 closes.
		5.2.1.6 Verify breaker DG13 closed. (1G 4160 V bus tie to 1J 4160 V BUS)
		5.2.1.7 Verify breaker DG02 closed. (1G 4160 V bus tie to 1L 4160 V bus)
		5.2.1.8 IF 1G 4160V bus energized, THEN proceed to step 5.7.

Step 5 continued on next page.

Step | **Action/Expected Response** | **Response NOT Obtained**

NOTE:
- Starting 1-2A or 1B diesel generator should be attempted first.
- Any attempted start of a diesel generator which has automatically tripped should be performed locally in Mode 4.

5.2.2 IF unable to start 2C diesel generator AND energize 1G 4160V bus, THEN start any diesel generator.

5.2.2.1 Start diesel generator from EPB in Mode 2 using START pushbutton.

OR

5.2.2.2 Direct diesel operator to start diesel generator in MODE 4 using FNP-0-SOP-38.1, EMERGENCY STARTING OF A DIESEL GENERATOR.

OR

5.2.2.3 Direct diesel operator to perform a manual emergency start of 1-2A or 1B diesel generator using FNP-0-SOP-38.1, EMERGENCY STARTING OF A DIESEL GENERATOR.

OR

5.2.2.4 IF attempts to start a diesel generator fail, THEN proceed to step 5.9.

Step 5 continued on next page.

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. 064K6.07 047/NEW//MEM 2.7/2.9/064K6.07/N///

Unit 1 is operating at 100% power with the following conditions:

- A problem with 1B DG starting air system has occurred.
- The B Air receiver has been tagged out.

Which one of the following completes the statement below?

A MINIMUM of __(1)__ psig must be available in the remaining air receiver to ensure five (5) start attempts are available.

1B DG's required minimum time to reach rated speed and voltage is __(2)__ seconds after receiving an emergency start signal.

	(1)	(2)
A.	200	7
B.	200	12
C.	350	7
D.✓	350	12

FSD - A181005:
2.1.2 - The DGS shall be capable of achieving > 3952 V and > 57 Hz within 12 seconds after receipt of an engine start signal

Tech Specs Bases: 3.8.3 - With both starting air receiver pressures on a DG < 350 psig for the 4075 kW DGs or < 200 psig for DG 1C, sufficient capacity for five successive DG start attempts does not exist.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. First part is incorrect (See D.1). Plausible if candidate thinks that the 1B DG is a "little DG" (Fairbanks Morse) instead of a "Big DG" (Colt Peilstick) which would make this a correct answer.

Second part is incorrect (See D.2). Plausible if the candidate confuses the required time to reach 115 RPM for the Fail to Start DG trip with the time to achieve rated voltage and speed. 7 seconds in not a subset of 12 seconds. 12 seconds is the requirement which implies a maximum and any time > 7 seconds but ≤ 12 seconds would be acceptable also.

B. Incorrect. First part is incorrect (A.1).

Second part is correct (See D.2).

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

D. Correct. First part is correct. The 1B DG is required to have one air receiver >350 psig to have 5 start attempts available.

Second part is correct. The DGS shall be capable of achieving > 3952 V and > 57 Hz within 12 seconds after receipt of an engine start signal

K/A: **064K6.07**	Emergency Diesel Generators (ED/G) - Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers
Importance Rating:	2.7 2.9
Technical Reference:	FSD-A181005, Diesel Generator, Ver 44 FNP Tech Specs Bases, Amendment No. 58
References provided:	None
Learning Objective:	RECALL AND APPLY the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Diesel Generator and Auxiliaries System components and attendant equipment alignment, to include the following (OPS-52102I01): [...] 3.8.3, Diesel Fuel Oil, Lube Oil, Starting Air
Question History:	NEW
K/A match:	Requires the applicant to know how a **malfunction of the DG air start receivers affect the operation of the ED/G system.**
SRO justification:	N/A

(Reference 6.5.018 and 6.7.080)

Diesel generators 1-2A, 1C or 1B shall be operable with Unit 1 in modes 5 and 6. (Reference 6.1.017)

Diesel generator 1-2A, 1C or 2B shall be operable with Unit 2 in modes 5 and 6. (Reference 6.1.017)

2.1.1 Nonaccident Condition

The design basis generator shall be aligned for automatically starting the diesel engines and capable of supplying required emergency loads in the event of an emergency start signal (Reference 6.4.173, 6.4.177, 6.4.181, 6.4.228 and 6.7.080).

2.1.2 Accident Condition

The DGS shall have sufficient capacity to supply the emergency loads identified in Appendix B (Reference 6.7.040, 6.1.005 and 6.7.028).

The DGS shall be capable of achieving $\geq$ 3952 V and $\geq$ 57 Hz within 12 seconds after receipt of an engine start signal (Reference 6.1.005, 6.5.009, 6.7.007, 6.7.080).

Upon a safety injection actuation signal (SIAS) the DGS shall start the appropriate diesel engines and be capable of loading the emergency loads for the associated train should a LOSP occur (Reference 6.4.173, 6.4.177, 6.4.181, 6.4.228 and 6.7.080).

Upon a LOSP the DGS shall start the appropriate diesel engines and load the affected emergency bus (Reference 6.4.173, 6.4.177, 6.4.181 6.4.228 and 6.7.080).

Upon operator assessment of an SBO condition, diesel gnerator 2C will be manually started and automatically aligned to and loaded with train B LOSP loads for the affected unit (Reference 6.7.080).

2.2 INTERFACE REQUIREMENTS

The electrical distribution system (EDS) shall provide power from independent, redundant, safety-related buses to the diesel generator system components. The diesel generator equipment shall be compatible with the voltage and frequency ranges provided by the EDS (Reference 6.4.108).

internals. The rest of the components operate within support systems and are described in Sections 3.2 through 3.10 as outlined above.

3.1.2 Functional Requirements

Each diesel engine, including associated equipment, is independent of the other diesel engines as well as from any outside power source for startup and operation, except for the 125 Vdc power supply from the station battery for control and instrumentation (References 6.5.005, 6.1.004, and 6.1.002).

Diesel engines 1-2A, 1B and 2B shall achieve 115 rpm within 7 sec ± 1 sec and accelerate to at least 514 rpm within 12 sec. If 115 rpm is not reached within 7 sec the air start timers (T2A and T2B) will remove starting air (References 6.1.005, 6.4.519, 6.7.007, 6.4.173, 6.4.177, 6.4.228 and 6.5.009).

Diesel engines 1C and 2C shall achieve 250 rpm within 7 sec and accelerate to at least 900 rpm within 12 sec. If 250 rpm is not reached within 7 sec ± 1 sec the air start timers (T2A and T2B) will remove starting air (References 6.1.005, 6.4.181, 6.4.185, 6.4.508, 6.5.009 and 6.7.007).

The diesel engines are designed to run at 110% of rated speed without damage. Rated speed is 514 rpm for diesel engines 1-2A, 1B and 2B and 900 rpm for diesel engines 1C and 2C (References 6.5.005, 6.4.509 and 6.4.521).

The diesel engines shall be capable of operating up to 3 minutes with no cooling water flow since service water will not be available until the service water pumps are loaded by the sequencers onto the diesel generator system. This 3 minutes is based upon the jacket water and lubricating oil temperature being at the keep warm conditions of 110°F water and 130°F oil (References 6.5.005, 6.1.005 and 6.7.035).

Diesel engine overspeed shutdown shall be independent of the speed control governor. Engine overspeed shutdown is accomplished by a mechanical overspeed governor. The overspeed governor consists of a single weight and spring. The overspeed governor is adjusted to trip at a speed (rpm) slightly higher than the speed control governor setpoint.

Upon overspeed the governor lever rod trips the overspeed switch which, through the governor control linkage, shuts off the fuel and air supplies. The overspeed governor must be manually reset prior to restarting the engine (References 6.4.509, 6.4.510, 6.4.519 and 6.4.537).

BASES

ACTIONS (continued)

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With both starting air receiver pressures on a DG < 350 psig for the 4075 kW DGs or < 200 psig for DG 1C, sufficient capacity for five successive DG start attempts does not exist. However, as long as at least one receiver pressure per DG is > 150 psig for the 4075 kW DGs or 90 psig for DG 1C, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air

(continued)

BASES

ACTIONS

E.1 (continued)

receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of useable fuel oil in the shared storage tanks (25,000 gallons each) to support the operation of the required DG(s) for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The inventory may consist of a combination of lube oil in storage and the useable sump volume above the manufacturer recommended minimum sump level or a total volume of lube oil in storage that is in addition to the lube oil normally maintained in each DG sump. The 238 gal requirement for the 4075 kW DGs and the 167 gal requirement for DG 1C are based on the DG manufacturer consumption values for 7 days of operation at full rated load. Implicit in this SR is the requirement to verify the capability

(continued)

1. 065AG2.4.11 048/BANK/FNP EXAM BANK/C/A 4.0/4.2/APE065AG2.4.11/N///

-Unit 1 was operating at 100% power when the following occurred:

- A complete loss of instrument air caused an automatic Reactor Trip.

The following conditions exist:

- All AFW pumps are running.
- All SG NR Levels are 25% and rising.
- The Shift Supervisor has directed AFW flow to be reduced.

Per AOP-6.0, Loss of Instrument Air, which one of the following methods below will be successful in reducing AFW flow?

Valve nomenclature:
- HV-3228A / B / C, TDAFWP TO 1A/1B/1C SG
- MOV-3764A / D / F, MDAFWP TO 1A/1B/1C SG ISO
- MOV-3350A / B / C, AFW TO 1A/1B/1C SG STOP VLV

A. Stop BOTH MDAFW pumps on the MCB.

B. Throttle HV-3228A / B / C on the MCB.

C.✔ Close MOV-3764A / D / F on the BOP.

D. Close MOV-3350A / B / C on the MCB.

AOP-6
Step 8. Maintain SG narrow range levels between 35-69%.

8.1 RNO WHEN required to limit SG level rise,
THEN perform the following:.

a) Alternately cycle closed and open one
MDAFWP isolation valve to each SG.

[ ] MDAFWP TO 1A SG ISO,
Q1N23MOV3764A(E)
[ ] MDAFWP TO 1B SG ISO,
Q1N23MOV3764B(D)
[ ] MDAFWP TO 1C SG ISO,
Q1N23MOV3764C(F)

b) STOP/START MDAFWPs as required.
[ ] 1A MDAFWP
[ ] 1B MDAFWP

Distracter analysis

A. Incorrect. See C. Plausible since this is a method per AOP-6 step 8.1 RNO but since SG NR Level is less than 28%, the MDAFW pumps cannot be stopped due to the auto-start signal. The applicant may not recall the MDAFW pump start logic and believe the pump can be stopped.

B. Incorrect. See C. Plausible if the applicant believes that the air receiver that keeps the TDAFW pump steam admission valves open is also used to control the TDAFW pump FCVs.

C. Correct. Of the available choices, this is the only method to control AFW flow per AOP-6. Step 8 of AOP-6 also directs the use of the MDAFW and TDAWF FCV's locally but these are not an available choice due to the loss of air

D. Incorrect. See C. Plausible since these valves are not addressed in AOP-6.0 and do not have power supplied during full power ops. Plausible since these valves are used in AOP-4.0 on loss of RCP flow to stop AFW flow and could be used to stop flow to all SGs if power was supplied to the MOV.

K/A: **065AG2.4.11**	Loss of Instrument Air - Knowledge of abnormal condition procedures.
Importance Rating:	4.0 4.2
Technical Reference:	FNP-1-AOP-6.0, Loss Of Instrument Air, Ver 40
References provided:	None
Learning Objective:	ANALYZE plant conditions and DETERMINE the successful completion of any step in AOP-6.0, Loss of Instrument Air. (OPS-52520F07)
Question History:	FNP EXAM BANK
K/A match:	Applicant must know what equipment is directed to be used by AOP-6 to control the cooldown rate.
SRO justification:	N/A

relays connected to any two of the buses (time delayed to prevent spurious trips caused by short term frequency perturbations) will trip the reactor if the power level is above P-7 (10% RTP) and the frequency drops below setpoint (57 hz). Since no control is derived from this protection function, IEEE 279 is satisfied by 1/2 sensors on 2/3 buses on either of the two trains. In addition to tripping the reactor, the trip signal causes the opening of each of the three pump motor power supply breakers, separating their windings from the underfrequency problem, to preclude any coastdown reduction. (References 6.1.003, 6.4.007, 6.7.012, 6.7.047)

2.6.5 Steam Generator Trips

PARAGRAPH DELETED (References 6.7.051, 6.7.058)

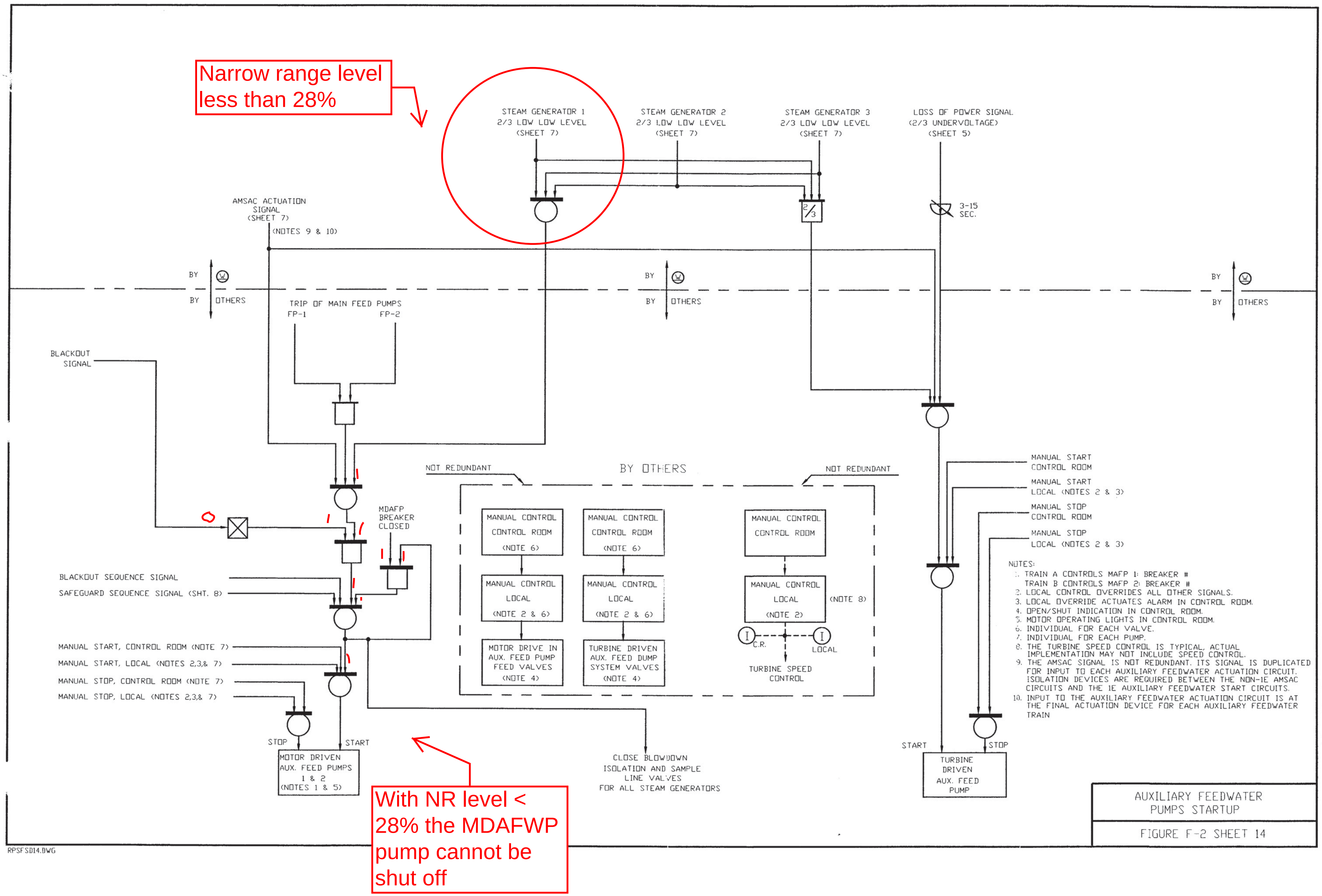
1. Low-Low Steam Generator Water Level Trip

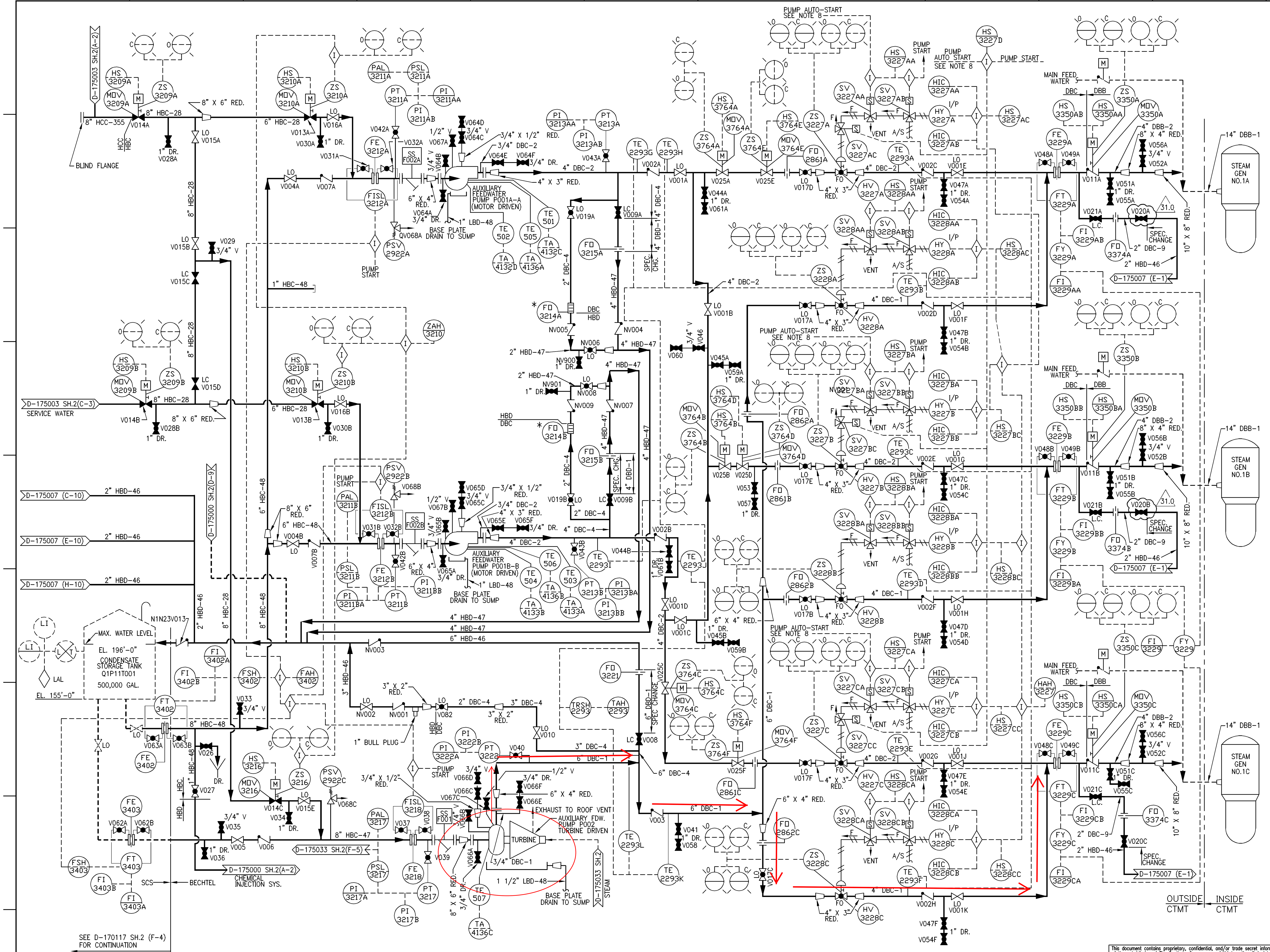
This is a primary trip which protects the reactor from loss of heat sink. This trip is actuated on two out of three low-low water level signals occurring in any steam generator, with a trip setpoint initiated at 28% of narrow range instrument span. The low-low level setpoint is due to the recent implementation of steam generator Level Tap Relocation program which increased the span of the narrow range protection system instrumentation. The increased operating margin reduces challenges to the safety systems by providing more forgiving steam generator water level control responses and provides additional transient capabilities to further improve plant reliability and availability.

Each low-low steam generator level trip signal is also part of the auxiliary feedwater auto start logic, which starts the auxiliary feedwater pumps to minimize the thermal transient on the reactor coolant system and steam generators.

Steam generator level provides input to both the control and protection systems. For Farley, with 2/3 actuation logic, IEEE 279 is satisfied by reliance on a median signal select circuit.

A Median Signal Selector card is installed in each of the 7300 Control System cabinets for use with the Steam Generator Water level Control System. Westinghouse has evaluated and demonstrated that the control and protection system interaction criteria of IEEE 279 is satisfied with the implementation of the Median Signal Selector in the Steam Generator Water Level Control System. Isolated protection grade steam generator water level signals are used as input to the control system median signal


Narrow range level less than 28%
STEAM GENERATOR 1 2/3 LOW LOW LEVEL (SHEET 7)
STEAM GENERATOR 2 2/3 LOW LOW LEVEL (SHEET 7)
STEAM GENERATOR 3 2/3 LOW LOW LEVEL (SHEET 7)
LOSS OF POWER SIGNAL (2/3 UNDERVOLTAGE) (SHEET 5)
3-15 SEC.
AMSAC ACTUATION SIGNAL (SHEET 7) (NOTES 9 & 10)
BY
BY OTHERS
TRIP OF MAIN FEED PUMPS FP-1 FP-2
BLACKOUT SIGNAL
MDAFP BREAKER CLOSED
NOT REDUNDANT
BY OTHERS
MANUAL CONTROL CONTROL ROOM (NOTE 6)
MANUAL CONTROL LOCAL (NOTE 2 & 6)
MOTOR DRIVE IN AUX. FEED PUMP FEED VALVES (NOTE 4)
TURBINE DRIVEN AUX. FEED DUMP SYSTEM VALVES (NOTE 4)
MANUAL CONTROL CONTROL ROOM
MANUAL CONTROL LOCAL (NOTE 2)
(NOTE 8)
C.R.
LOCAL
TURBINE SPEED CONTROL
BLACKOUT SEQUENCE SIGNAL
SAFEGUARD SEQUENCE SIGNAL (SHT. 8)
MANUAL START, CONTROL ROOM (NOTE 7)
MANUAL START, LOCAL (NOTES 2,3,& 7)
MANUAL STOP, CONTROL ROOM (NOTE 7)
MANUAL STOP, LOCAL (NOTES 2,3,& 7)
STOP
START
MOTOR DRIVEN AUX. FEED PUMPS 1 & 2 (NOTES 1 & 5)
CLOSE BLOWDOWN ISOLATION AND SAMPLE LINE VALVES FOR ALL STEAM GENERATORS
With NR level < 28% the MDAFWP pump cannot be shut off
MANUAL START CONTROL ROOM
MANUAL START LOCAL (NOTES 2 & 3)
MANUAL STOP CONTROL ROOM
MANUAL STOP LOCAL (NOTES 2 & 3)
TURBINE DRIVEN AUX. FEED PUMP
NOTES:
1. TRAIN A CONTROLS MAFP 1: BREAKER #
TRAIN B CONTROLS MAFP 2: BREAKER #
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS.
3. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
4. OPEN/SHUT INDICATION IN CONTROL ROOM.
5. MOTOR OPERATING LIGHTS IN CONTROL ROOM.
6. INDIVIDUAL FOR EACH VALVE.
7. INDIVIDUAL FOR EACH PUMP.
8. THE TURBINE SPEED CONTROL IS TYPICAL, ACTUAL IMPLEMENTATION MAY NOT INCLUDE SPEED CONTROL.
9. THE AMSAC SIGNAL IS NOT REDUNDANT. ITS SIGNAL IS DUPLICATED FOR INPUT TO EACH AUXILIARY FEEDWATER ACTUATION CIRCUIT. ISOLATION DEVICES ARE REQUIRED BETWEEN THE NON-1E AMSAC CIRCUITS AND THE 1E AUXILIARY FEEDWATER START CIRCUITS.
10. INPUT TO THE AUXILIARY FEEDWATER ACTUATION CIRCUIT IS AT THE FINAL ACTUATION DEVICE FOR EACH AUXILIARY FEEDWATER TRAIN
AUXILIARY FEEDWATER PUMPS STARTUP
FIGURE F-2 SHEET 14
RPSFSD14.DWG



NOTES:

1. THE SYSTEM IDENTIFICATION NUMBER FOR ALL NUCLEAR PIPING, VALVES, AND EQUIPMENT ON THIS DRAWING IS Q1N23.
   THE IDENTIFICATION NUMBER FOR NON-NUCLEAR COMPONENTS IS N1N23. NON-NUCLEAR VALVES ARE SHOWN WITH THE LETTER "N" PRECEDING THE VALVE NUMBERS.
2. FOR P&I LEGEND SEE DWG. D-175016 SHEETS 1 & 2.
3. * MINIMUM FLOW ORIFICES ARE FURNISHED BY PUMP MFGR.
4. PORTIONS OF THE SYSTEM CONTAINING CONDENSATE STORAGE TANK GRADE WATER ARE TO BE DRAINED OR PUMPED TO THE AUX STEAM CONDENSATE TANK OR TO THE CONDENSATE STORAGE TANK. PORTIONS OF THE SYSTEM CONTAINING SERVICE WATER ARE TO BE DRAINED VIA THE SERVICE WATER DRAIN PUMP TO THE SERVICE WATER SYSTEM.
5. FOR COMPLETE INSTRUMENT NUMBERS, REFER TO INSTRUMENT INDEX B-175803.
6. DURING PLANT POWER OPERATION, VALVES MOV 3350A, B, & C ARE TO BE IN THE OPEN POSITION WITH THE ELECTRICAL BREAKERS RACKED OUT.
7. (DELETED)
8. SEE U-166244 FOR MOTOR DRIVEN AUX. FEEDWATER PUMP AUTO-START INITIATION SIGNALS.
9. TE's 2293 A-L ARE STRAP ON SURFACE TEMP. MONITORS AND DO NOT PENETRATE THE PIPING PRESSURE BOUNDARY.

CAD D1750071.DWG
ACAD2K | CYN - 08

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT P&ID - AUX. FEEDWATER SYSTEM

DRAWN ____ CHECKED ____ DESIGNED ____
APPROVED FOR SIGNATURES DATE ____
APPROVED SEE MICROFILM DATE ____
APPROVED COPY DATE ____

CYN | LCF | PML
VER. NO. 31.0 DATE 12/08/09
REVISED PER ABN-F01634, VER. 1.0

SCALE NONE B/M N/A
SHEET 1 OF — SHEETS
SUPERSEDES

D-175007

DRAWING CATEGORY: CRITICAL

Step	Action/Expected Response	Response Not Obtained

NOTE: Steps 1 through 2.1 are IMMEDIATE OPERATOR actions.

__ 1 **Check 1A and 1B RCPs - RUNNING.**

1 Manually close pressurizer spray valve for affected RCP.

[ ] 1A RCS loop spray valve PK-444C
[ ] 1B RCS loop spray valve PK-444D

__ 2 **[CA] Maintain SG narrow range level stable at approximately 65% using:**

[ ] Main Feedwater Regulating Valves
[ ] Main Feedwater Bypass Regulating Valves.
[ ] Auxiliary Feedwater Control Valves.

2 IF SG level rise cannot be controlled, THEN perform the following:

2.1 IF main feedwater in service, THEN close the affected SG Main Feedwater Stop Valve(s)

[ ] 1A SG Q1N21MOV3232A
[ ] 1B SG Q1N21MOV3232B
[ ] 1C SG Q1N21MOV3232C

2.2 IF auxiliary feedwater in service, THEN close the affected SG AFW Stop Valve(s)

[ ] 1A SG Q1N23MOV3350A
[ ] 1B SG Q1N23MOV3350B
[ ] 1C SG Q1N23MOV3350C

__ 3 **[CA] Monitor Tavg for all three RCS loops ≥ 541°F. (TS 3.4.2)**

3 Perform the following..

3.1 IF the main generator is ON LINE, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION

3.2 IF the main generator is OFF LINE, THEN raise Tavg ≥ 541°F within 30 minutes

3.2.1 Adjust steam dumps to reduce secondary power demand as necessary

Step 3 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
__ 7	**Maintain PRZR level between 20-50%.**	7 WHEN required to limit level rise, THEN alternately cycle open and closed one of the following MOVs for charging control as required.
7.1	Go to FNP-1-AOP-16.0, CVCS MALFUNCTION while continuing with this procedure.	CHG PUMPS TO REGENERATIVE HX [ ] Q1E21MOV8107 [ ] Q1E21MOV8108
__ 8	**Maintain SG narrow range levels between 35-69%.**	

****

CAUTION: With SG pressure less than 400 psig, AFW pumps at full flow may approach run out.

****

Step	Action/Expected Response	Response Not Obtained
8.1	Locally manually control MDAFWP flow control valves with handwheels. (MSVR) MDAFWP TO 1A(1B, 1C) SG [ ] Q1N23HV3227A [ ] Q1N23HV3227B [ ] Q1N23HV3227C	8.1 WHEN required to limit SG level rise, THEN perform the following:. a) Alternately cycle closed and open one MDAFWP isolation valve to each SG. [ ] MDAFWP TO 1A SG ISO, Q1N23MOV3764A(E) [ ] MDAFWP TO 1B SG ISO, Q1N23MOV3764B(D) [ ] MDAFWP TO 1C SG ISO, Q1N23MOV3764C(F) OR b) STOP/START MDAFWPs as required. [ ] 1A MDAFWP [ ] 1B MDAFWP

Autostart signal is present therefore this will not work

1

Step 8 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
8.2	IF TDAFWP required, THEN perform the following:	8.2 STOP TDAFWP.
8.2.1	Locally manually control TDAFWP flow control valves with handwheels. (MSVR). TDAFWP TO 1A(1B,1C) SG [ ] Q1N23HV3228A [ ] Q1N23HV3228B [ ] Q1N23HV3228C	8.2.1 Control TDAFWP speed. TDAFWP SPEED CONT [ ] SIC 3405 adjusted

****

CAUTION: The TDAFWP steam admission valves will fail closed within two hours if emergency air is not aligned.

****

Step	Action/Expected Response	Response Not Obtained
8.2.2	Align emergency air using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.	8.2.2 Manually operate TDAFWP per FNP-1-SOP-22.0, Appendix I, TDAFWP MANUAL OPERATION.
__ 9	**Verify SW to standby CCW heat exchanger isolated.** [ ] SW TO 1C CCW HX Q1P16MOV3130C [ ] SW TO 1B CCW HX Q1P16MOV3130B [ ] SW TO 1A CCW HX Q1P16MOV3130A	

NOTE: PORV BKUP air supply Q1P19HV2228 fails closed on a loss of 'B' train DC.

Step	Action/Expected Response	Response Not Obtained
__ 10	**Align nitrogen supply to PRZR PORVs using FNP-1-SOP-62.1, BACKUP-UP AIR OR NITROGEN SUPPLY TO THE PRESSURIZER POWER OPERATED RELIEF VALVES.**	

°
°

__Page Completed

1. 068AK2.07 049/NEW//C/A 3.3/3.4/068AK2.07/N///

There is a fire in the Control Room and the following conditions exist for Unit 1:

- FNP-1-AOP-28.2, Fire In The Control Room, has been entered.

Which one of the following completes the statements below?

During the conduct of AOP-28.2, the Diesel Generators are required to be placed in __(1)__ and the output breakers __(2)__ automatically close when the DGs are started after a Loss of Offsite Power.

	(1)	(2)
A.	MODE 3	WILL
B.	MODE 3	will NOT
C.	MODE 4	WILL
D.✓	MODE 4	will NOT

AOP-28.2
Step 6.3.1 - Dispatch personnel to the diesel building to perform ATTACHMENT 18, PLACING DIESEL GENERATORS IN LOCAL CONTROL.

Attachment, 18 Step 1 - PLACE 1B DIESEL IN MODE 4.

Attachment 21, Note prior to Step 1.10 - Diesel generator and diesel generator output breaker must be controlled locally (155' DG BLDG) while diesel generator is in MODE 4.

<u>Distracter analysis</u>

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant assumes that the most reliable condition of the DGs in this scenario would be in Mode 3 as in AOP-49.2, Complete Loss of Service Water.

Second part is incorrect (See C.2). Plausible because this would be the correct answer if the DG was required to be in Mode 1 per AOP-28.2. Once the DG's are no longer in MODE 1 (MODE 2, 3 or 4), applicants can have difficulty recalling how the output breaker responds on a DG start for an LOSP)

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2). This is a logical connection to the first part if the applicant improperly believes that control power to the DG output breakers is removed as part of shifting local control of the Main Control Room operated equipment to the HSDP. The RCP breakers are tripped locally and have control power removed per AOP-28.2.

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See C.2). Plausible if the applicant failed to recall that the DG output breaker will NOT automatically close when started in Mode 4 after a loss of power.

D. Correct. First part is correct. Step 6.3.1 requires the DGs to be placed in MODE 4.

Second part is correct. While in MODE 4, the DG output breaker will NOT automatically close.

K/A: **068AK2.07**	Control Room Evacuation - Knowledge of the interrelations between the Control Room Evacuation and the following: ED/G.
Importance Rating:	3.3 3.4
Technical Reference:	FNP-1-AOP-28.2, Fire In The Control Room, Ver 28
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing AOP-28.1, Fire or Inadvertent Fire Protection System Actuation in the Cable Spreading Room, and AOP-28.2, Fire in the Control Room. (OPS-52521C06)
Question History:	NEW
K/A match:	Requires the applicant to know the **interrelation of the DG mode of operation to a control room evacuation.**
SRO justification:	N/A

___ 1 **Invoke 10CFR50.54(x) to allow performance of this procedure since this condition is beyond the design basis and is not governed by the requirements of 10CFR50.59**

___ 2 **Secure Diesel Generators.**

2.1 IF all Service Water has been lost to the Diesel Generators, THEN perform the following:

2.1.1 Verify Diesel Generators are secured, using Attachment 1 or Placard on EPB.

[] 1C DG
[] 2B DG
[] 2C DG
[] 1-2A DG
[] 1B DG

2.1.2 Verify the Diesel Generator Mode Selector switches in Mode 3.

[] 1C DG Mode Selector Switch
[] 2B DG Mode Selector Switch
[] 2C DG Mode Selector Switch
[] 1-2A DG Mode Selector Switch
[] 1B DG Mode Selector Switch

___ 3 **Notify the NRC within 1 hour of plant status.**

___ 4 **Consider securing TB components affected by loss of SW, i.e. condensate pumps, heater drain pumps, etc.**

___ 5 **Consider securing Letdown (to reduce heat load on CCW, and extend time that CCW will serve as a heat sink without SW cooling.)**

___ 6 **Consider securing RCP's, or at least reducing number of running RCP's (to reduce heat load on CCW, and extend time that CCW will serve as a heat sink without SW cooling.)**

6.3 Perform the following attachments.

6.3.1 Dispatch personnel to the diesel building to perform ATTACHMENT 18, PLACING DIESEL GENERATORS IN LOCAL CONTROL.

6.3.2 Dispatch personnel to place equipment in local control per ATTACHMENT 19, OPERATOR ACTIONS IN THE SWIS BLDG, and ATTACHMENT 20, OPERATOR ACTIONS IN THE NON RAD AUX BLDG.

NOTE: The following step is a continuous action step.

6.4 IF at anytime an LOSP occurs, AND automatic actions do not restore power, THEN perform ATTACHMENT 21, RESPONSE TO LOSS OF OFFSITE POWER CONTINGENCY ACTIONS..

NOTE: 1C air compressor should automatically align when 1A 600 V LC emergency section aligns to 1D 600 V LC during a LOSP or SI/LOSP. To ensure adequate supply voltage to class 1E loads and to meet short circuit analysis constraints, only one air compressor, 1C (preferred) or 1A, should be powered from the diesel generator. One air compressor will consume 0.16 MW of diesel generator load.

__ **7 Verify instrument air available to Penetration Rooms and containment.**

7.1 Check any selected air compressor N1P19C001A, B, or C running.

7.1.1 If required, start any deselected air compressor from the MCB by taking its associated handswitch to the START/RUN position and returned to AUTO.

7.1.2 IF required, dispatch personnel to the Turbine Building to investigate the operation of the air compressor sequencer per FNP-1-SOP-31.0, COMPRESSED AIR SYSTEM.

7.1.3 IF required, dispatch personnel to the Turbine Building to start an air compressor with air compressor panel key switch in the LOCAL position per FNP-1-SOP-31.0, COMPRESSED AIR SYSTEM.

° Step 7 continued on next page

__Page Completed

Step | Action/Expected Response | Response Not Obtained

ATTACHMENT 18

PLACING DIESEL GENERATORS IN LOCAL CONTROL

****

CAUTION: Diesel generators are only capable of operating 3 minutes without cooling water flow.

****

__ 1 **PLACE 1B DIESEL IN MODE 4.**

Action/Expected Response	Response Not Obtained
1.1 Check 1B diesel generator is not running. [ ] At the DLCP place the mode selector switch in mode 4.	1.1 IF the 1B diesel generator is running, THEN perform the following at the DLCP. 1.1.1 At the DLCP place the mode selector switch in mode 4. 1.1.2 Monitor generator frequency. [ ] Adjust frequency to 60 Hz by positioning the SPEED CONTROL switch to RAISE or LOWER as required. 1.1.3 Monitor generator voltage [ ] Adjust voltage to 4160V by positioning the AUTOMATIC VOLTAGE ADJUST to RAISE or LOWER as required. 1.1.4 Monitor diesel generator lube oil temperature temperature. [ ] TI-552 D/G 1B LUBE OIL HEAT EXCHANGER OUTLET TEMPERATURE INDICATOR

° Step 1 continued on next page

ATTACHMENT 21

RESPONSE TO LOSS OF OFFSITE POWER CONTINGENCY ACTIONS

NOTE: Diesel generator and diesel generator output breaker must be controlled locally (155' DG BLDG) while diesel generator is in MODE 4.

1.10 Check 1-2A, 1B, 1C or 2C diesel generator output.

1.10.1 Verify diesel generator frequency at 58-62 Hz.

1.10.2 Verify diesel generator voltage - 4.0-4.3 kV.

1.10 Restore diesel generator field.

a) Locally reset exciter. (DG BLDG at local control panel)

b) IF diesel generator frequency indicates 58-62 Hz AND diesel generator voltage indicates 4.0-4.3 kV,
THEN proceed to step 1.11,
IF NOT, verify diesel generator DC control power breakers closed. (121 ft, AUX BLDG battery charger rooms)

Running DG	1-2A	1C	1B	2C
Battery Supply	[ ] BKR LA05	[ ] BKR LA05	[ ] BKR LB18	[ ] BKR LB18
DC Control Panel Supply	[ ] BKR LA17	[ ] BKR LA06	[ ] BKR LB19	[ ] BKR LB11

c) IF unable to restore excitation to diesel generator,
THEN secure the diesel generator AND proceed to step 1.13.

° Step 1 continued on next page

****

CAUTION: To prevent inadvertent steamline differential pressure SI, steam generator atmospheric relief valves should be adjusted to provide approximately equal SG demands. Consideration must be given to the steam load supplied to the TDAFWP.

****

__ **17 Adjust steam generator atmospheric relief valves to maintain SG pressures at approximately 1005 psig. (A HSDP Rm 254)**

17.1 Monitor 1A (1B, 1C) SG PRESS (A-HSDP)

[ ] Q1N11PI3371A
[ ] Q1N11PI3371B
[ ] Q1N11PI3371C

17.2 Adjust as required 1A (1B, 1C) MS ATMOS REL VLV (A HSDP)

[ ] Q1N11PCV3371A
[ ] Q1N11PCV3371B
[ ] Q1N11PCV3371C

__ **18 Locally trip all reactor coolant supply breakers.**

[ ] 1A RCP BKR DA-04 (139 ft, AUX BLDG Rm 346)
[ ] 1B RCP BKR DB-03 (139 ft, AUX BLDG Rm 343)
[ ] 1C RCP BKR DC-03 (139 ft, AUX BLDG Rm 343)

18.1 Verify breaker open by observing open indicating light illuminated.

18.1.1 Open the door; verify breaker open by observing mechanical indicator window.

18.1.2 Open DC control power breaker.

__ **19 Isolate the Main Steam system.**

19.1 IF inadvertent SI occurs while closing MSIVs,
THEN perform recovery from inadvertent SI using ATTACHMENT 1, RESPONSE TO SPURIOUS OR INADVERTENT SAFETY INJECTION.

19.2 Align MSIVs for local operation at G HSDP.

19.2.1 Place local / remote handswitch for 1A (1B, 1C) SG MSIV in LOCAL

[ ] Q1N11HV3369A to LOCAL
[ ] Q1N11HV3369B to LOCAL
[ ] Q1N11HV3369C to LOCAL

Step 19 continued on next page

__Page Completed

19.2.2 Close 1A (1B, 1C) SG MSIVs (G-HSDP).

[ ] Q1N11HV3369A
[ ] Q1N11HV3369B
[ ] Q1N11HV3369C

__ **20 IF a LOSP has occurred,**
THEN energize emergency sections of 1A and 1C 600 V load centers.

20.1 Verify 1D 600 V LC energized (139 ft, AUX BLDG switchgear Rm 335)

20.2 Locally verify open 1A 600 V LC normal/emergency tie breaker EA-08 (139 ft, AUX BLDG switchgear Rm 335)

20.3 Locally verify closed 1A 600 V LC feeder breakers. (139 ft, AUX BLDG switchgearRm)

[ ] BKR ED-08
[ ] BKR EA-09

20.4 Check 1A 600 V LC energized

20.5 Verify 1E 600 V LC energized (121 ft, AUX BLDG switchgear Rm 229)

20.6 Locally Open 1C 600 V LC normal/emergency tie breaker EC-08. (121 ft, AUX BLDG switchgear Rm 229)

20.7 Locally close 1C 600 V LC feeder breakers.

[ ] BKR EE-07
[ ] BKR EC-10

20.8 Check 1C 600 V LC energized.

__ **21 Align the following A-HSDP components for local operation.**

1A PRZR HTR GROUP

[ ] BACKUP to LOCAL

1A BATP

[ ] Q1E21P005A to LOCAL

LTDN ORIF ISO 45(60) GPM

[ ] Q1E21HV8149A to LOCAL
[ ] Q1E21HV8149B to LOCAL
[ ] Q1E21HV8149C to LOCAL

__ **22 Align the following G-HSDP components for local operation.**

RX VESSEL HEAD VENT OUTER ISO

[ ] Q1B13SV2213A to LOCAL

RX VESSEL HEAD VENT INNER ISO

[ ] Q1B13SV2214A to LOCAL

__ **23 Align the following C-HSDP components for local operation.**

1B BATP

[ ] Q1E21P005B to LOCAL

1B PRZR HTR GROUP

[ ] BACKUP to LOCAL

RX VESSEL HEAD VENT OUTER ISO

[ ] Q1B13SV2213B to LOCAL

RX VESSEL HEAD VENT INNER ISO

[ ] Q1B13SV2214B to LOCAL

CCW TO SECONDARY HXS

[ ] Q1P17MOV3047 to LOCAL

__ **24 Verify open the block valve for each operable PRZR PORV.**

PRZR PORV ISO

[ ] Q1B31MOV8000A (G-HSDP)

[ ] Q1B31MOV8000B (B-HSDP)

__Page Completed

1. 071K3.05 050/NEW//C/A 3.2/3.2/071K3.05/N///

Unit 1 is operating at 100% power with the following conditions:

- The 1A Waste Gas Compressor is running and aligned to #7 WGDT.
- R-13, WGC SUCT, alarms.

Subsequently, the #7 WGDT relief valve lifts and fails to reseat.

Which one of the following completes the statements below?

R-22, VENT STACK GAS, _(1)_ trend up.

#7 WGDT relief valve _(2)_ be manually isolated.

	(1)	(2)
A.✔	WILL	CANNOT
B.	WILL	CAN
C.	will NOT	CANNOT
D.	will NOT	CAN

D-175045 SH 1: Shows R-22 located in the vent stack.

D175042 SH 6, Shows that the #7 WGDT relief discharges to the vent stack and has no manual isolations.

Distracter analysis

A. Correct. First part is correct. #7 WGDT relief valve discharges to the vent stack and would cause R-22 to trend up since the 1A Waste Gas Compressor is aligned to it and the compressor suction has a high rad alarm.

Second part is correct. There is no manual isolations for the #7 WGDT relief valve.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2). Plausible since some systems have isolation valves upstream of their relief valves such as LP Feedwater heaters (See 170116 SH 1 in reference material).

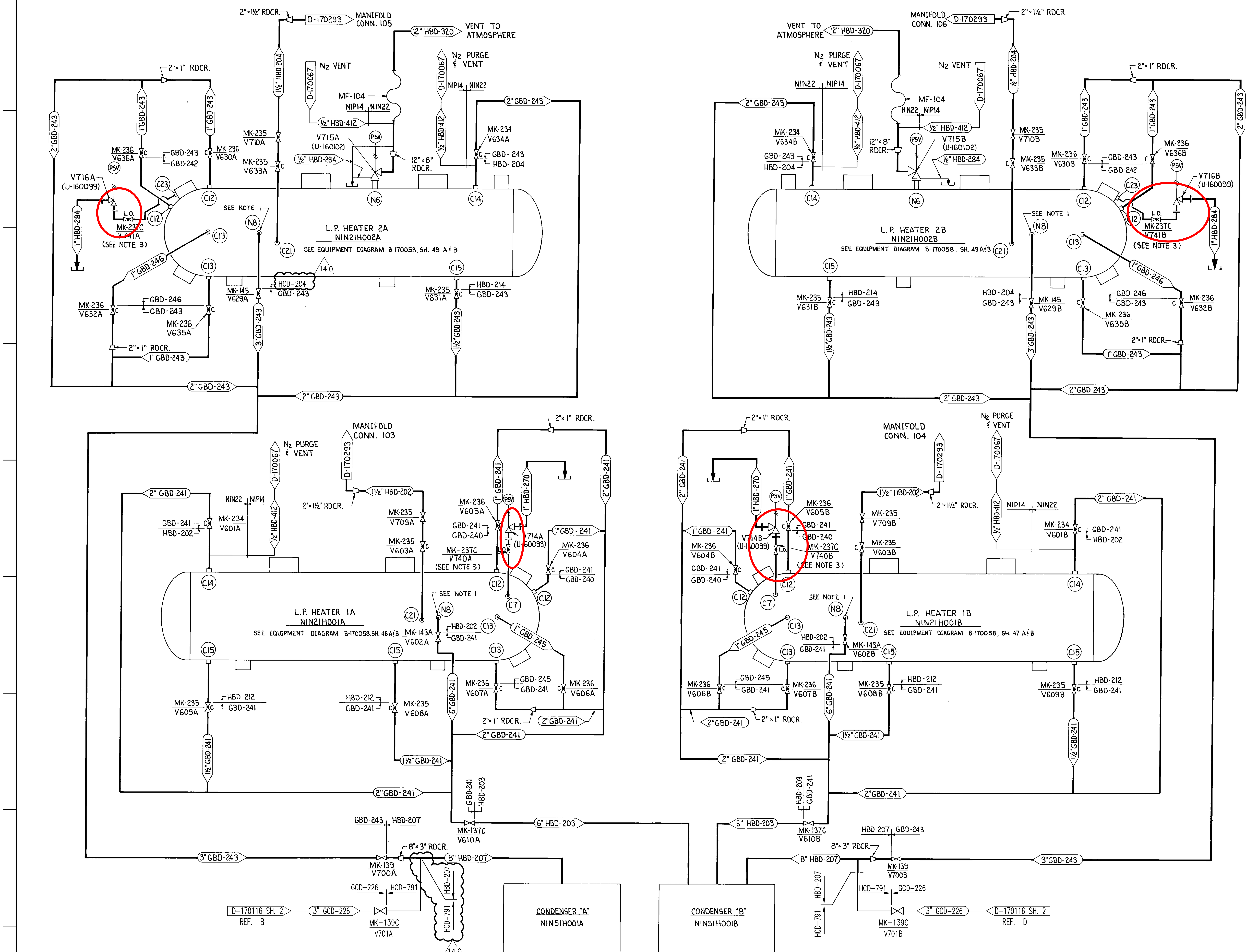
C. Incorrect. First part is incorrect (See B.1). Plausible because WGDT 1 though 6 relieve to #8 WGDT and the applicant could think that #7 also relieved to #8 WGDT.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2).

K/A: **071K3.05**	Waste Gas Disposal System (WGDS) - Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following: ARM and PRM systems
Importance Rating:	3.2 3.2
Technical Reference:	D175045, Unit 1 HVAC - P&ID SFP Vent Sys, Sheet 1, Ver 22.0 D175042, Unit1 Waste Processing System, Sheet 6, Ver 33.0
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Radiation Monitoring System components and equipment, to include the following (OPS-40305A07): [...] • Automatic actuation • Protective isolations • Protective interlocks • Actions needed to mitigate the consequence of the abnormality
Question History:	NEW
K/A match:	Requires the applicant to **know the effect on R-22 (PRMS) when WGDT #7 relieves to the vent stack (malfunction of the WG system resulting in relief lifting).**
SRO justification:	N/A



NOZZLE SCHEDULE

NOZZLE	DESCRIPTION
N6	SHELL SAFETY RELIEF VALVE
C7	CHANNEL SAFETY RELIEF VALVE - HTR. 1A & 1B
N8	OPERATING AIR OUTLET
C12	CHANNEL START-UP TEST VENT
C13	CHANNEL DRAIN
C14	SHELL START-UP TEST VENT
C15	SHELL DRAIN
C21	DRAIN COOLER START-UP VENT
C23	CHANNEL SAFETY RELIEF VALVE - HTR. 2A & 2B

NOTES:

1. CRITICAL FLOW ORIFICE PROVIDED IN HEATER CONNECTION.
2. TOTAL PLANT NUMBERING SYSTEM - THE SYSTEM IDENTIFICATION NUMBER FOR ALL PIPING & VALVES ON THIS DRAWING IS N1N22, UNLESS NOTED. THE NUMBER FOR ALL OTHER EQUIPMENT IS AS NOTED.
3. ISOLATION VALVES V740A, V740B, V741A & V741B ARE TO REMAIN LOCKED OPEN AND BE CLOSED ONLY WITH PROPER AUTHORIZATION. (REF. ASME BOILER AND PRESSURE VESSEL CODE, 1980, SEC. 8, DIV. 1, APPENDIX M, PARA M-5).

REFERENCES:

B-170058	EQUIPMENT DIAGRAMS
D-170067	P&ID - NITROGEN PURGING SYSTEM
D-170116 SH. 2	P&ID - L.P. HEATER 3 & 4 - VENTS & DRAINS
D-170293	PIPING - CONDENSER DRAIN MANIFOLDS
U-160099	CHANNEL SAFETY RELIEF VALVES
U-160102	SHELL SAFETY RELIEF VALVES
U-162454	FLEXIBLE HOSE
U-416349	OUTLINE - L.P. HEATER 2A & 2B
U-416437	OUTLINE - L.P. HEATER 1A
U-416438	OUTLINE - L.P. HEATER 1B
U-160055	OUTLINE - CONDENSER A & B
U-160056	OUTLINE - CONDENSER A
U-160057	OUTLINE - CONDENSER B

CAD D1701161
ACADOVY | RCR - 02

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH
RCR	CFC	JMR				

VER. NO. 14.0 DATE 11-21-07
REVISED PER ABN's
1062801001M002, VERSION 1.0
1062841401M002, VERSION 1.0

DRAWN DOOM CHECKED SYX DESIGNED DEK
APPROVED JWW JSK JRC DATE 1-24-73
APPROVED F.E. EHRENSPERGER DATE 1-30-73
APPROVED JAMES W. WALLACE DATE 2-01-73

Southern Company Services, Inc. for
ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT P.&I.D.
L.P. HEATER 1 & 2 - VENTS AND DRAINS
SCALE NONE B/M
SHEET 1 OF — SHEETS
SUPERSEDES
D-170116
DWG. CATEGORY: CRITICAL

FOR NOTES AND REFERENCES SEE SHEET 5

THIS DRAWING WAS ORIGINALLY REPRODUCED FROM WESTINGHOUSE DRAWING 110E399 SH.5

CAD 17504210.DWG
ACAD2000 | JOK-05

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SOUTHERN COMPANY SERVICES, INC. FOR

ALABAMA POWER COMPANY

PROJECT: JOSEPH M. FARLEY NUCLEAR PLANT NO.1

SUBJECT: P & I DIAGRAM

WASTE PROCESSING SYSTEM

SCALE: NO SCALE

SHEET 10 OF 4 SHEETS

D-175042

REV. NO.	DATE	DESCRIPTION	BY	CHK'D
4	5-2-02	REVISED PER ABN 02-1-2320, REV. 0	JOK	RKF DEW
3	3-29-90	REV'D. PER PCN S-89-1-6959, REV.0.		KGL DML
2	12-12-89	REV'D. PER PCN S-88-1-5473, REV.0.		
1	6-16-89	REVISED PER PCN S-88-1-5152, REV.0	BST	RLB DML JRC
0	5-19-89	APPROVED-ISSUED PER PCN S-88-1-5152, REV.0 ALL VENT/DRAIN VALVES CHECKED REDRAWN, WAS PART OF SHEET 2		

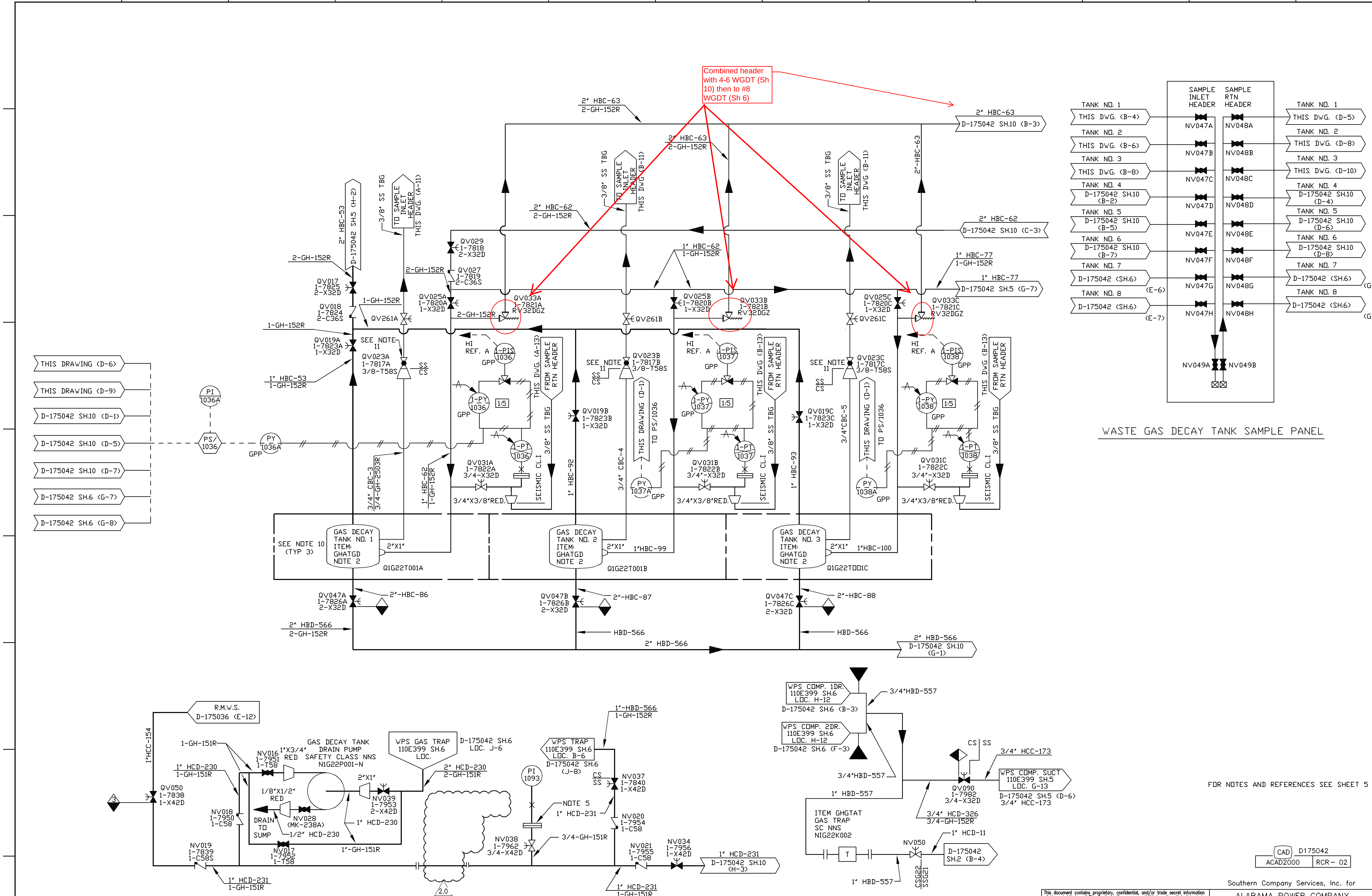
DRAWN: BST | CHECKED: RLB | DESIGNED: N/A

APPROVED: DML | DATE: 5-24-89

APPROVED: CAB NRA | DATE: 5-24-89

APPROVED: JAMES R. CRANE | DATE: 5-24-89

OPERATIONS CRITICAL



CAD D175042
ACAD2000 | RCR- 02

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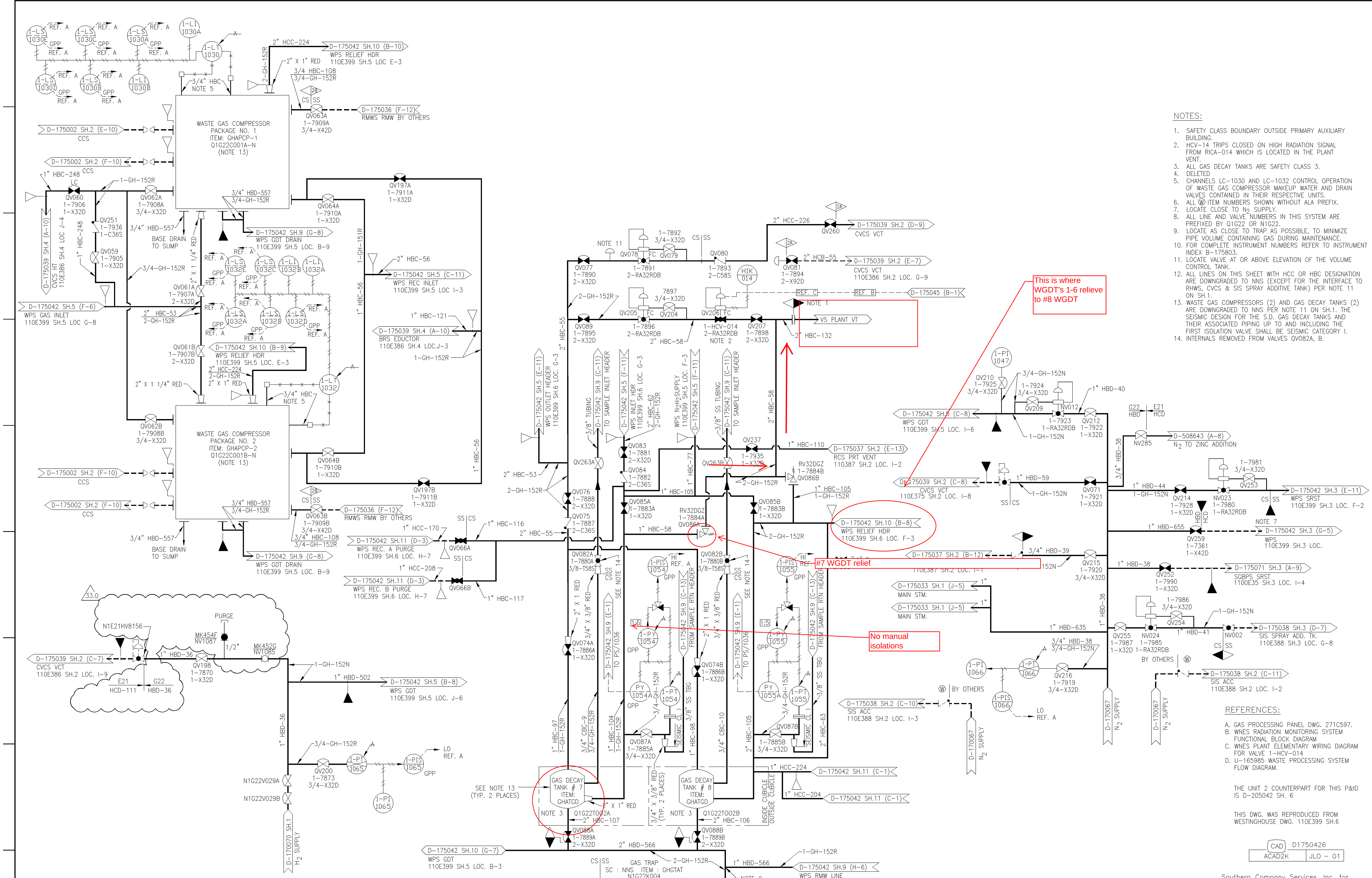
Southern Company Services, Inc. for
ALABAMA POWER COMPANY
PROJECT: J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
SUBJECT: P & I DIAGRAM
WASTE PROCESSING SYSTEM

DRAWN: BST CHECKED: RLB DESIGNED: N/A
APPROVED: D.M. LLOYD DATE 5-24-89
APPROVED: NRA / CAB DATE 5-24-89
APPROVED: JAMES R. CRANE DATE 5-24-89
SCALE: NONE
SHEET 9 OF – SHEETS

VER. NO. 2.0 DATE 06-08-07
REVISED PER ABN 1060929301M003, VERSION 1.0
RCR CFC JMR

D-175042
DWG. CATEGORY CRITICAL



NOTES:

1. SAFETY CLASS BOUNDARY OUTSIDE PRIMARY AUXILIARY BUILDING.
2. HCV-14 TRIPS CLOSED ON HIGH RADIATION SIGNAL FROM RICA-014 WHICH IS LOCATED IN THE PLANT VENT.
3. ALL GAS DECAY TANKS ARE SAFETY CLASS 3.
4. DELETED
5. CHANNELS LC-1030 AND LC-1032 CONTROL OPERATION OF WASTE GAS COMPRESSOR MAKEUP WATER AND DRAIN VALVES CONTAINED IN THEIR RESPECTIVE UNITS.
6. ALL W ITEM NUMBERS SHOWN WITHOUT ALA PREFIX.
7. LOCATE CLOSE TO N_2 SUPPLY.
8. ALL LINE AND VALVE NUMBERS IN THIS SYSTEM ARE PREFIXED BY Q1G22 OR N1G22.
9. LOCATE AS CLOSE TO TRAP AS POSSIBLE, TO MINIMIZE PIPE VOLUME CONTAINING GAS DURING MAINTENANCE.
10. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175803.
11. LOCATE VALVE AT OR ABOVE ELEVATION OF THE VOLUME CONTROL TANK.
12. ALL LINES ON THIS SHEET WITH HCC OR HBC DESIGNATION ARE DOWNGRADED TO NNS (EXCEPT FOR THE INTERFACE TO RHWS, CVCS & SIS SPRAY ADDITIVE TANK) PER NOTE 11 ON SH.1.
13. WASTE GAS COMPRESSORS (2) AND GAS DECAY TANKS (2) ARE DOWNGRADED TO NNS PER NOTE 11 ON SH.1. THE SEISMIC DESIGN FOR THE S.D. GAS DECAY TANKS AND THEIR ASSOCIATED PIPING UP TO AND INCLUDING THE FIRST ISOLATION VALVE SHALL BE SEISMIC CATEGORY I.
14. INTERNALS REMOVED FROM VALVES QV082A, B.

REFERENCES:

A. GAS PROCESSING PANEL DWG. 271C597.
B. WNES RADIATION MONITORING SYSTEM FUNCTIONAL BLOCK DIAGRAM
C. WNES PLANT ELEMENTARY WIRING DIAGRAM FOR VALVE 1-HCV-014
D. U-165985 WASTE PROCESSING SYSTEM FLOW DIAGRAM.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205042 SH. 6

THIS DWG. WAS REPRODUCED FROM WESTINGHOUSE DWG. 110E399 SH.6

CAD D1750426
ACAD2K | JLO - 01

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1

SUBJECT P&ID WASTE PROCESSING SYSTEM

DRAWN CHECKED DESIGNED

APPROVED FOR SIGNATURES DATE

APPROVED SEE MICROFILM DATE

APPROVED COPY DATE

SCALE NONE B/M

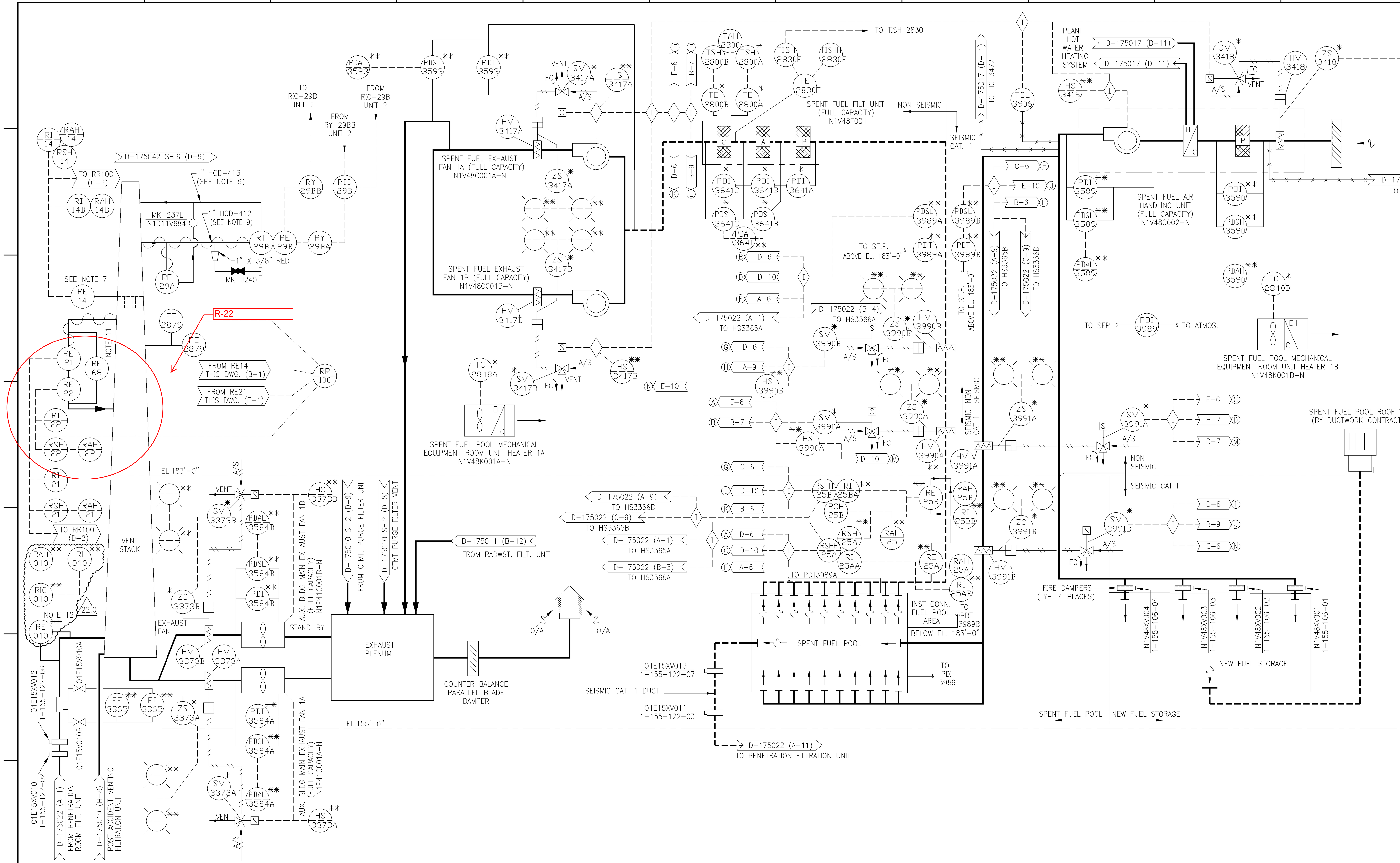
SHEET 6 OF — SHEETS

SUPERSEDES

D-175042

33.0 DATE 4-3-08 REVISED PER ABN 1070168101M003, VER. 1.0

DRAWING CATEGORY: CRITICAL



NOTES:

1. FOR DUCT MATERIAL & CONSTRUCTION SEE (SPEC. SS-1102-54)
2. FOR DAMPER INFORMATION SEE (SPEC. SS-1102-54).
3. FOR EQUIPMENT DESCRIPTION SEE (SPEC. SS-1102-15, SS-1102-45, SS-1102-80).
4. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX D-175803.
5. FOR SEISMIC CLASSIFICATION OF DUCTWORK SEE (SPEC. SS-1102-54).
6. ** ASSOCIATED INSTRUMENTATION TO BE PROVIDED UNDER INSTRUMENTATION & CONTROL CONTRACT.

   * ASSOCIATED INSTRUMENTATION TO BE PROVIDED BY EQUIP. MFGR.

   ALL OTHER INSTRUMENTATION SHALL BE PROVIDED BY DUCTWORK SUBCONTRACTOR.
7. RADIATION INSTRUMENTS (RE 14) SHALL BE FURNISHED BY WESTINGHOUSE.
8. SEE DWG. D-175015 FOR PROCESS FLOW.
9. SMLS OR WELDED SCH 40 PIPE MAY BE USED FOR 1" HCD-412 AND 1" HCD-413.
10. B-508501 - UNIT NO. 1 & SHARED FIRE DAMPER REPORT.
11. FOR DETAILS SEE B-175983.
12. RADIATION INSTRUMENT (RE 0010) SHALL BE FURNISHED BY GENERAL ATOMICS-ESI. 22.0

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205045 SH.1

CAD D1750101
ACAD2K JLO - 03

Southern Company Services, Inc. for
ALABAMA POWER COMPANY

PROJECT: J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT: HVAC - P&ID
SPENT FUEL POOL VENT SYSTEM

FOR SIGNATURES SEE MICROFILM COPY

JLO GDM JMR 22.0 6-29-10 REVISED PER ABN 1071862301J019, VER. 1.0

SCALE: NONE
SHEET 1

D-175045

DRAWING CATEGORY: CRITICAL

1. 073A2.02 051/MOD/FNP 07/MEM 2.7/3.2/073A2.02/N///

Unit 1 is operating at 100% when the following condition occurs:

- R-19, SGBD SAMPLE, fails HIGH.

Which one of the following completes the statements below?

_(1)_ valves will automatically close.
Per SOP-45.0, Radiation Monitoring System, the actions required to allow the shift chemist to obtain a sample of the SGs is to _(2)_ .

A. 1) HV-3328, HV-3329 AND HV-3330, STEAM GEN 1A/1B/1C SAMPLE ISO,

2) pull the INSTRUMENT power fuses for R-19

B.✓ 1) HV-3328, HV-3329 AND HV-3330, STEAM GEN 1A/1B/1C SAMPLE ISO,

2) place R-19 Operations Selector Switch to the RESET position

C. 1) HV-3179A, 3180A, AND 3181A, STEAM GEN 1A/1B/1C LOWER BLOWDOWN,

2) pull the INSTRUMENT power fuses for R-19

D. 1) HV-3179A, 3180A, AND 3181A, STEAM GEN 1A/1B/1C LOWER BLOWDOWN,

2) place R-19 Operations Selector Switch to the RESET position

ARP-1.6, FH1 - R-19 isolates HV-3328, 3329 and 3330.

SOP-45
4.4 Obtaining a Steam Generator Sample with R-19 in Alarm or Inoperable:

4.4.1 Notify Health Physics and Chemistry that R-19 will be inoperable during the time required to obtain a sample

4.4.2 IF in alarm, THEN place the switch for R-19 to the Reset position.

4.4.3 Open the Steam Generator Blowdown sample valves listed below as necessary to obtain a Steam Generator sample:
Q1P15HV3328 1A Steam Generator Blowdown sample valve
Q1P15HV3329 1B Steam Generator Blowdown sample valve
Q1P15HV3330 1C Steam Generator Blowdown sample valve

Distracter analysis

A. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2). Plausible since this is the procedure directed action for a monitor in saturation, but not to allow the chemist to sample the SG..

B. Correct. First part is correct. R-19 failing in the "High Radiation" condition shuts HV-3328, 3329, and 3330.

Second part is correct. Per SOP-45, the Rad monitor switch must be taken to reset to allow SGBD sample valves to be opened.

C. Incorrect. First part is incorrect (See B.1). Plausible if the applicant doesn't recall that R-19 will isolate HV-3328, 3329, and 3330. They may believe that R-19 closes HV-3179A, 3180A, and 3181A which are immediately upstream of the correct valves and closed by the AFW pump start signal and on High Penetration Room DIFFERENTIAL Pressure.

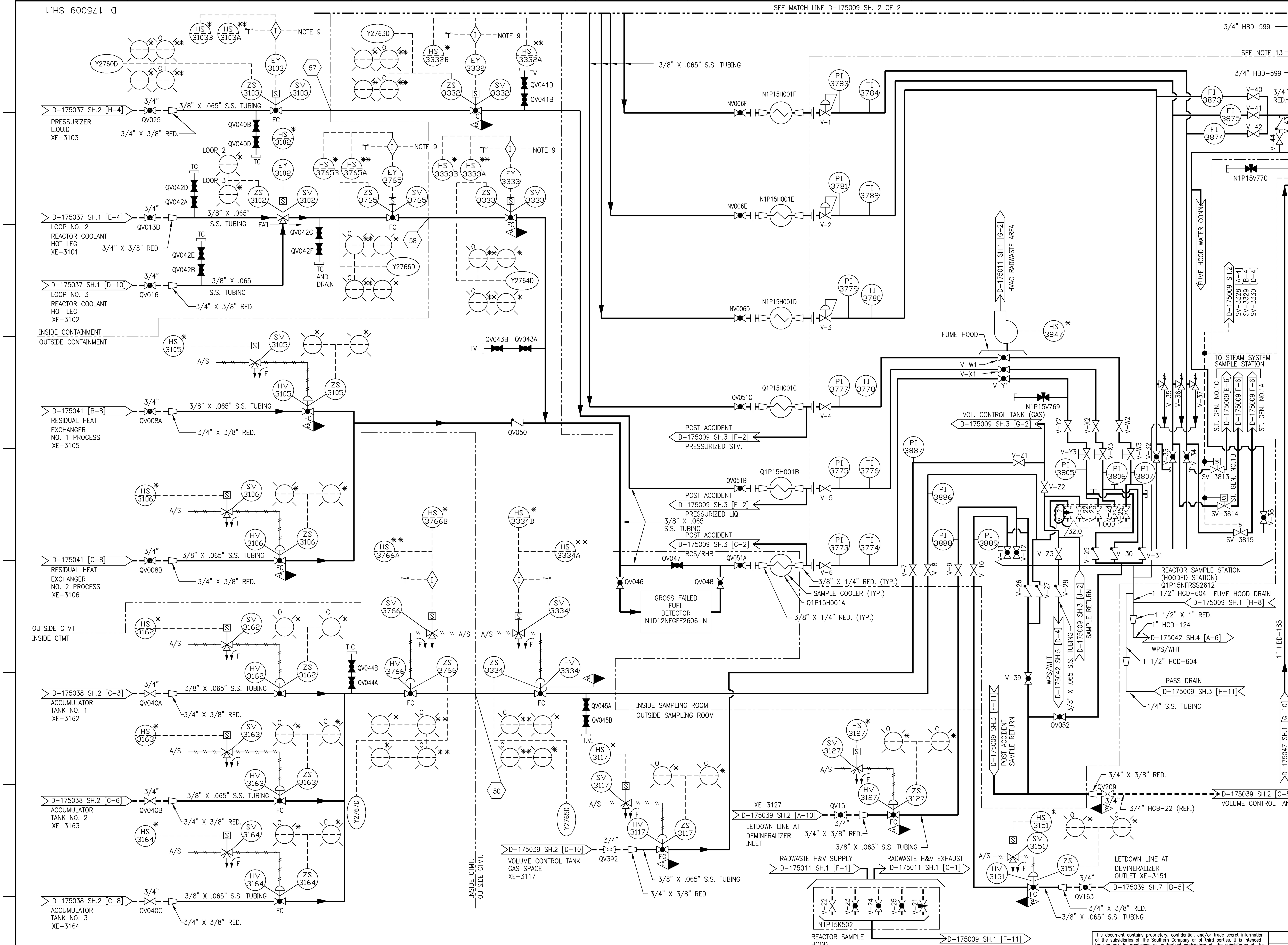
Second part is incorrect (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is correct (See B.2).

NOUN NAME for HV-3179A, 3180A, AND 3181A, 1A/1B/1C SG LOWER BLOWDOWN SAMPLE ISO came from ARP BK1.

K/A: **073A2.02**	Process Radiation Monitoring (PRM) System - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure
Importance Rating:	2.7 3.2
Technical Reference:	FNP-1-ARP-1.6, FH1, RMS HI RAD, Ver 70 FNP-1-SOP-45, Radiation Monitoring System, Ver 46.2
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Radiation Monitoring System components and equipment, to include the following (OPS-40305A07): [...] • Automatic actuation • Protective isolations • Protective interlocks • Actions needed to mitigate the consequence of the abnormality
Question History:	MOD FNP 07
K/A match:	The applicant is required to know the **impact on the SG sample system due to R-19 failing high (Desktop simulator shows a level amp failure - high will alarm R-19)** and that SOP-45 provides **procedural guidance to sample the SGs under this condition.**
SRO justification:	N/A



LEGEND:

"T"-CONTAINMENT ISOLATION ACTUATION SIGNAL, PHASE "A".

AFPSS-TRIP OR CLOSE ON THE START OF AUXILIARY FEEDWATER PUMP
FC-FAIL CLOSED
TC-LEAK RATE TEST CONNECTION
TV-TEST VALVE CONNECTION.
HVAC-HEATING, VENTILATION & AIR CONDITIONING
WPS-WASTE PROCESSING SYSTEM.
RCS-REACTOR COOLANT SYSTEM.
VCT-VOLUME CONTROL TANK.
PRT-PRESSURIZER RELIEF TANK.
RE-RADIATION MONITOR ELEMENT.
* -LOCATED AT SAMPLE STATION.
** -LOCATED IN CONTROL ROOM.
+-EQUIPMENT SUPPLIER ON SAMPLE DRAIN TK. ORDERS.
PH-PH ANALYZER
P -PHOSPHATE ANALYZER

QUICK DISCONNECT COUPLING

MANUAL HIGH PRESSURE REDUCING VALVE

NOTES:

1. ALL VALVE NUMBERS PRECEDED BY PREFIX Q1P15 OR N1P15 UNLESS SHOWN.
2. FOR INSTALLATION AND FABRICATION SPECIFICATIONS SEE SECTION 6 OF INQUIRY SS-1109-2.
3. FOR SAMPLE STATION DETAILS SEE SS-1102-85.
4. ROUTE ALL SAMPLE STATION RELIEF BLOWOFF'S TO RESPECTIVE PANEL DRAIN HEADER.
5. ALL EQUIPMENT LOCATED WITHIN THE SAMPLE ROOM IS TO BE SUPPLIED BY SAMPLE SYSTEM VENDOR. EXCEPT SAMPLE COOLER RELIEF BLOWOFFS.
6. FOR P&ID LEGEND SEE DRAWING D-175016, SHEET 1 & 2.
8. ALL LINES NOT TO ASME SECTION III, SHALL BE DESIGNED IN ACCORDANCE WITH ANSI B 31.1.
9. STEAM GENERATOR BLOWDOWN PRESSURIZER AND REACTOR HOT LEG SAMPLE VALVES AUTOMATICALLY CLOSE ON HIGH PENETRATION ROOM PRESSURE.
10. FOR DETAILED EQUIPMENT DESCRIPTION AND BILL OF MATERIAL SEE U-259780, U-259638, & U-259455.
11. FOR EQUIPMENT LIST SEE DRAWING D-175070.
12. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175803.
13. 2" BLIND FLANGE DRILLED TO SUIT FOR 3/4" LINE.
14. VALVE MK-J-240 CAN BE USED INSIDE THE SAMPLE ROOM TO REPLACE THE KEROTEST GLOBE VALVES (FIG. 9921) USED IN THE REACTOR SAMPLE STATION AND THE STEAM SYSTEM SAMPLE STATION.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205009.

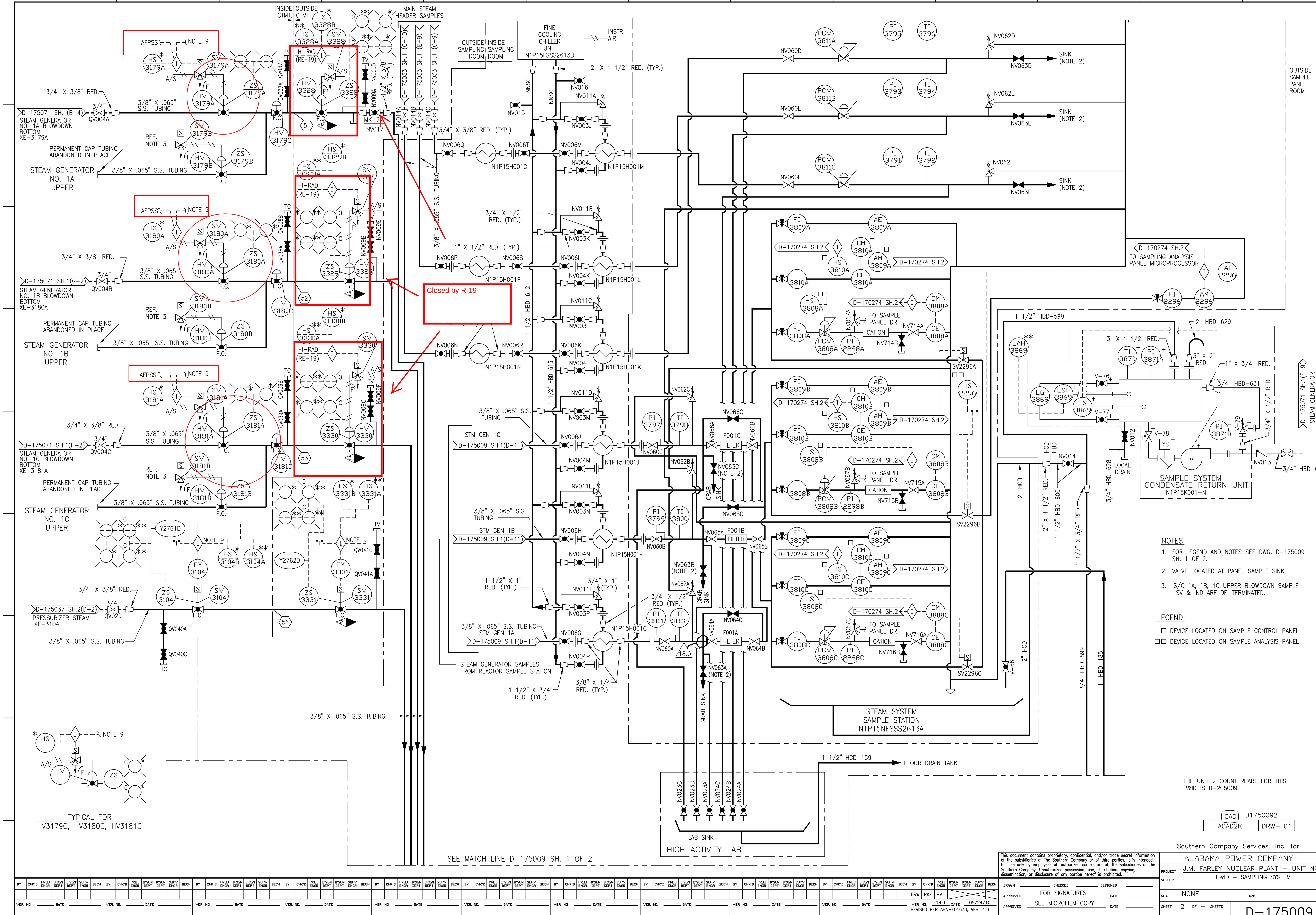
CAD D1750091.DWG
ACAD2002 | DFV-04

Southern Company Services, Inc. for

DFV WMW ASK
VER. NO. 32.0 DATE 8-3-05
REVISED PER ABN 04-0-2806, VER. 2.0

DRAWN ____ CHECKED ____ DESIGNED ____
APPROVED FOR SIGNATURES DATE ____
APPROVED SEE MICROFILM DATE ____
APPROVED COPY DATE ____

ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT P&ID - SAMPLING SYSTEM
SCALE ____ B/M ____
SHEET 1 OF — SHEETS
SUPERSEDES
D-175009 VER. 32.0
DWG CATEGORY: CRITICAL



Closed by R-19
STEAM GENERATOR NO. 1A UPPER
STEAM GENERATOR NO. 1B UPPER
STEAM GENERATOR NO. 1C UPPER
PRESSURIZER STEAM XE-3104
FINE COOLING CHILLER UNIT N1P15FSSS2613B
OUTSIDE SAMPLE PANEL ROOM
STEAM GENERATOR SAMPLES FROM REACTOR SAMPLE STATION
STEAM SYSTEM SAMPLE STATION N1P15NFSSS2613A
SAMPLE SYSTEM CONDENSATE RETURN UNIT N1P15K001-N
HIGH ACTIVITY LAB
LAB SINK
FLOOR DRAIN TANK
TYPICAL FOR HV3179C, HV3180C, HV3181C
SEE MATCH LINE D-175009 SH. 1 OF 2
NOTES:
1. FOR LEGEND AND NOTES SEE DWG. D-175009 SH. 1 OF 2.
2. VALVE LOCATED AT PANEL SAMPLE SINK.
3. S/G 1A, 1B, 1C UPPER BLOWDOWN SAMPLE SV & IND ARE DE-TERMINATED.
LEGEND:
□ DEVICE LOCATED ON SAMPLE CONTROL PANEL
□□ DEVICE LOCATED ON SAMPLE ANALYSIS PANEL
THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205009.
CAD D1750092
ACAD2K DRW- 01
Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
P&ID – SAMPLING SYSTEM
FOR SIGNATURES SEE MICROFILM COPY
18.0 05/24/10
REVISED PER ABN-F01678, VER. 1.0
SCALE NONE
SHEET 2 OF – SHEETS
D-175009
DWG. CATEGORY: CRITICAL

LOCATION BK1

K1
PENE RM TO ATMOS A TRN ΔP HI-LO

SETPOINT:
1. Variable Current/Time
2. HI: 3.0 ± 0.25 inches H_2O
3. LO: -1.5 + 0.5 inches H_2O
   - 0

ORIGIN:
1. Overload Aux. Relay, 49X
2. Diff. Pressure Switch (Q1E15PDSH3367A-A)
3. Diff. Pressure Switch (Q1E15PDSL3367A-A)

PROBABLE CAUSE

1. Piping rupture in the Penetration Room.
2. Penetration Room Filtration Recirculation Damper (Q1E15MOV3361A) tripped on overload.
3. Penetration Room Filtration System improper lineup.

AUTOMATIC ACTION

NOTE:	In addition to the automatic actions listed below for the A Train, see Location BK2 for B Train automatic actions.

1. The following valves will close on Penetration Room High Pressure signal from PDSH 3367A:
   a) Instrument Air Supply Isolation Valve, 1-IA-HV-3825
   b) Nitrogen Supply Isolation Valve, 1-NG-HV-3938A
   c) STEAM GENERATOR 1A LOWER BLOWDOWN, Q1P15HV3179A
   d) STEAM GENERATOR 1B LOWER BLOWDOWN, Q1P15HV3180A
   e) STEAM GENERATOR 1C LOWER BLOWDOWN, Q1P15HV3181A
   f) Pressurizer Steam Sample Isolation Valve, 1-SS-HV-3104
   g) Pressurizer Liquid Sample Isolation Valve, 1-SS-HV-3103
   h) Reactor Loops B&C Sample Isolation Valve, 1-SS-HV-3765

OPERATOR ACTION

1. Check PDI-3367A and PDI-3367B, PENE ROOM TO ATMOS, on the BOP and determine IF penetration room pressure is high OR low.
2. IF pressure is high, THEN perform the following actions:
   2.1 Verify all required automatic actions have occurred.

OPERATOR ACTIONS

1. Check indications on radiation monitoring system console and determine which radiation monitor channel indicates high activity. ☐
2. Insure that any automatic actions, associated with the alarmed channel, have occurred. ☐
3. Perform the following general actions as appropriate.
   - 3.1. Determine the source or cause of the high activity and correct or isolate as required. ☐
   - 3.2 Determine the validity of the high activity indication as follows:
     - 3.2.1 Verify that the instrument is aligned for normal operation and is functioning properly. ☐

NOTE: The following step does not apply to Radiation Monitors R-10, 11 and 12. ☐

     - 3.2.2 IF a known problem exists such that the detector is saturated, THEN momentarily pull the affected detector's fuses (located on the front of the drawer) to clear the condition. ☐
     - 3.2.3 If requested to disable a remote audible alarm, refer to FNP-1-SOP-45.0, P&L 3.6. ☐
     - 3.2.4 Sample or survey the affected system or area as required. {**CMT 0008755**}. ☐
   - 3.3 Do not allow personnel to enter the affected area without the approval of the Health Physics Department. ☐
   - 3.4 IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.3, 3.4.15 and TRM TR 13.3.4. ☐
   - 3.5 IF high activity indication of RCS leakage is present AND accompanied by either decreasing pressurizer level, OR decreasing VCT level, THEN go to FNP-1-AOP-1.0, RCS LEAKAGE. ☐
   - 3.6 IF high activity indication of Steam Generator Tube Leakage is present, THEN go to FNP-1-AOP-2.0, STEAM GENERATOR TUBE LEAKAGE. ☐
   - 3.7. IF ARDA activated and not required, THEN have counting room stop the automated dose assessment per FNP-0-EIP-9.1, AUTOMATED DOSE ASSESSMENT METHOD. ☐
   - 3.8 WHEN radiation levels have decreased below alarm setpoint, THEN reset the appropriate HI radiation alarm on the RAD monitor drawer. ☐

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-12*	Containment Atmosphere (AB 121')	Gas	Beta Scint **(GA-ES)		Perform Step 4.11
R-13	Waste Gas Compressor Suction (AB 100' WGC Valve Room)	Gas	G-M (W)		Perform Step 4.12
R-14 ODCM	Plant Vent Stack (AB Roof)	Gas	G-M (W)	Closes HCV-14	Perform Step 4.13
R-15A ODCM	Condenser Air Ejector Discharge Header (TB 155')	Gas	G-M		Perform Step 4.14
R-15B*	Condenser Air Ejector (Intermediate Range) (TB 189')	Gas	G-M (Eberline)		Perform Step 4.15
R-15C*	Condenser Air Ejector (High Range) (TB 189')	Gas	Ion Chamber (Eberline)		Perform Step 4.15
R-17A	Component Cooling Water (CCW Hx Room)	Liquid	Scint. (W)	Closes CCW surge tank vent (RCV-3028)	Perform Step 4.16
R-17B	Component Cooling Water (CCW Hx Room)	Liquid	Scint.	Closes CCW surge tank vent (RCV-3028)	Perform Step 4.16
R-18 ODCM	Waste Monitor Tank Pump Discharge (AB 121' at the Batching Funnel)	Liquid	Scint. (W)	Closes RCV-18	Perform Step 4.17
R-19	Steam Generator Blowdown/Sample (AB 139')	Liquid	Scint. (W)	Isolates sample lines 3328, 3329, 3330	Perform Step 4.18
R-20A	Service Water from Containment Coolers A and B (AB 121' BTRS Chiller Room)	Liquid	Scint. (W)		Perform Step 4.19
R-20B	Service Water from Containment Coolers C and D (AB 121')	Liquid	Scint. (W)		Perform Step 4.19

*Technical Specification related
**General Atomics Electronic Systems

4.4 Obtaining a Steam Generator Sample with R-19 in Alarm or Inoperable:

4.4.1 **Notify** Health Physics and Chemistry that R-19 will be inoperable during the time required to obtain a sample. ☐

4.4.2 IF in alarm, THEN **place** the switch for R-19 to the Reset position. ☐

4.4.3 **Open** the Steam Generator Blowdown sample valves listed below as necessary to obtain a Steam Generator sample:

Q1P15HV3328	1A Steam Generator Blowdown sample valve	☐
Q1P15HV3329	1B Steam Generator Blowdown sample valve	☐
Q1P15HV3330	1C Steam Generator Blowdown sample valve	☐

4.4.4 WHEN sampling of the Steam Generators is completed, THEN **verify** the switch for R-19 to the OPERATE position. ☐

4.4.5 **Verify** that R-19 is aligned per Section 4.1. ☐

4.4.6 **Notify** Health Physics and Chemistry that R-19 has been returned to service. ☐

OPEN opens the associated valve. Placing the BOP panel handswitch in OPEN opens the associated valve if the sample control panel handswitch is in Neutral. Once a valve is fully open, it will remain open if:

1. Both switches are in Neutral.
2. A phase A containment isolation signal is not present.

Valve position indication lights are above each handswitch.

RHR Heat Exchanger, VCT Gas Space, Letdown Line and Accumulator Sample Isolation Valves

A three-position handswitch (CLOSE/Neutral/OPEN, spring return to Neutral), located on the sample control panel, controls each valve (V3105, V3106, V3117, V3127, V3151, V3162, V3163, & V3164). Valve position indication lights are above each handswitch.

Steam Generator Blowdown Sample Isolation Valves (Figure 8)

Individual three-position handswitches (CLOSE/Neutral/OPEN, spring return to Neutral), located on the sample control panel, control the Steam Generator Blowdown Sample Isolation Valves (V3179A; V3180A; V3181A). In Neutral, the valves will automatically close on a penetration room high differential pressure or an auto start of an MDAFW pump. Valve position indication lights are above each handswitch.

Steam Generator Blowdown Sample Containment Isolation Valves (Figure 7)

A three-position handswitch (STOP/Neutral/START, spring return to Neutral) on the penetration room isolation panel controls Steam Generator Blowdown sample valve (V3179C, V3180C, & V3181C). In Neutral, the valve automatically closes on an auto start of an MDAFW pump or a penetration room high differential pressure. Valve position indication lights are above each handswitch.

Steam Generator Blowdown Sample Containment Isolation Valves (Figure 7)

The Steam Generator Blowdown Sample Containment isolation valves outside containment (V3328, V3329, & V3330) may be controlled from either the sample control panel or the BOP panel. The associated switches are three-position handswitches

(CLOSE/Neutral/OPEN, spring return to Neutral). Once the valve is fully open and both switches are in Neutral, the valve remains open as long as:

1. An MDAFW pump auto start does not occur.
2. A high radiation signal from RE-019 does not actuate.

Valve position lights are above each handswitch.

TECHNICAL REQUIREMENTS MANUAL

Technical Requirement 13.4.1, Reactor Coolant System Chemistry

This Technical Requirement states that the Reactor Coolant System chemistry shall be maintained within the limits specified in Table 13.4.1-1 at all times, except for dissolved oxygen when Tavg ≤ 250 ºF. The table limits are shown below.

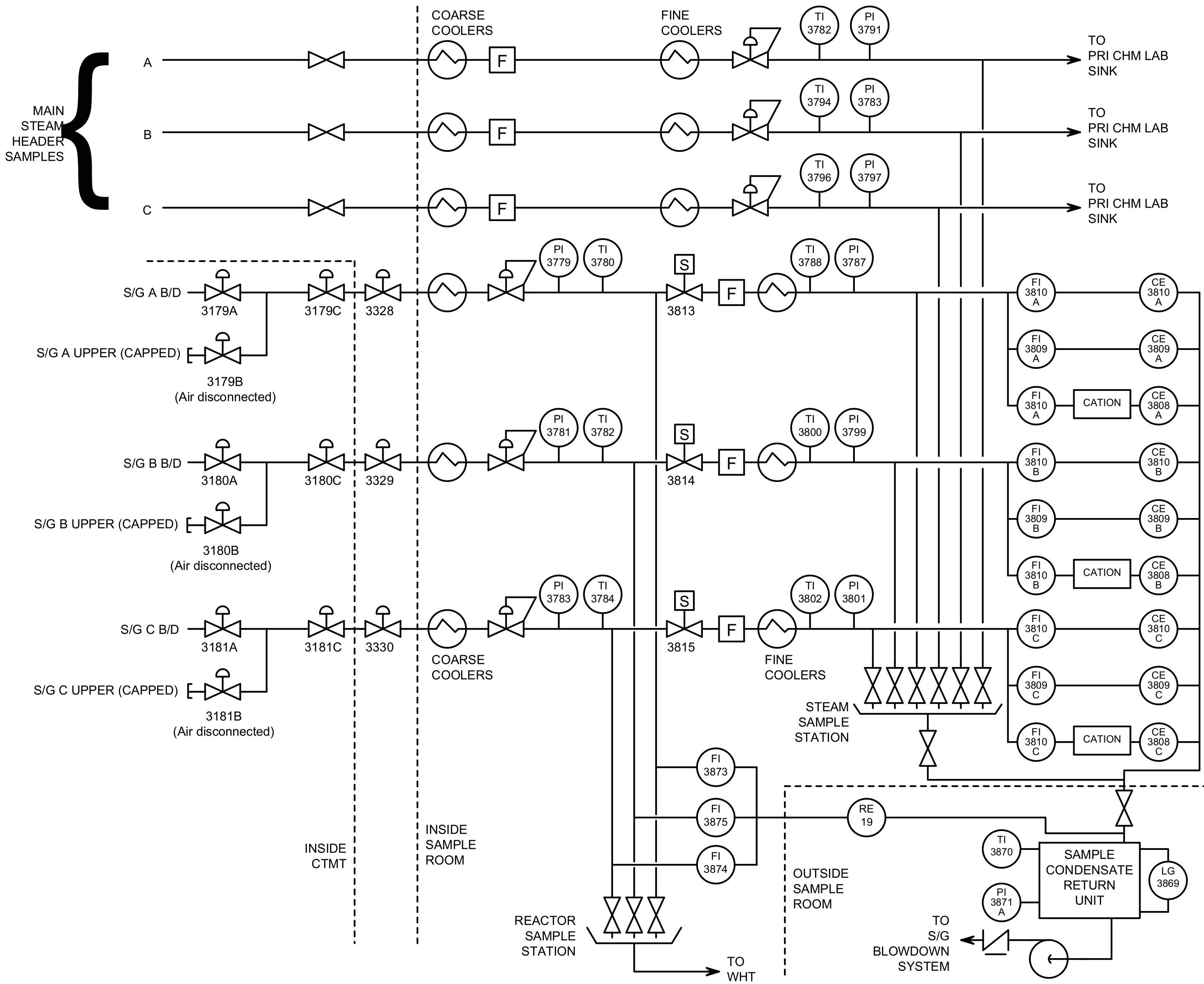
PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

If one or more chemistry parameters are > the steady- state limit and ≤ transient limit in modes 1, 2, 3, or 4 the parameter must be restored to within steady-state limit within 24 hours.

If one or more chemistry parameters are > transient limit in modes 1, 2, 3, or 4 or the above required action and associated completion time is not met, the plant must be placed in mode 3 within 6 hours and in mode 5 within 36 hours.

Consult the Technical Requirements Manual (TRM) for additional requirements in modes 5 and 6.

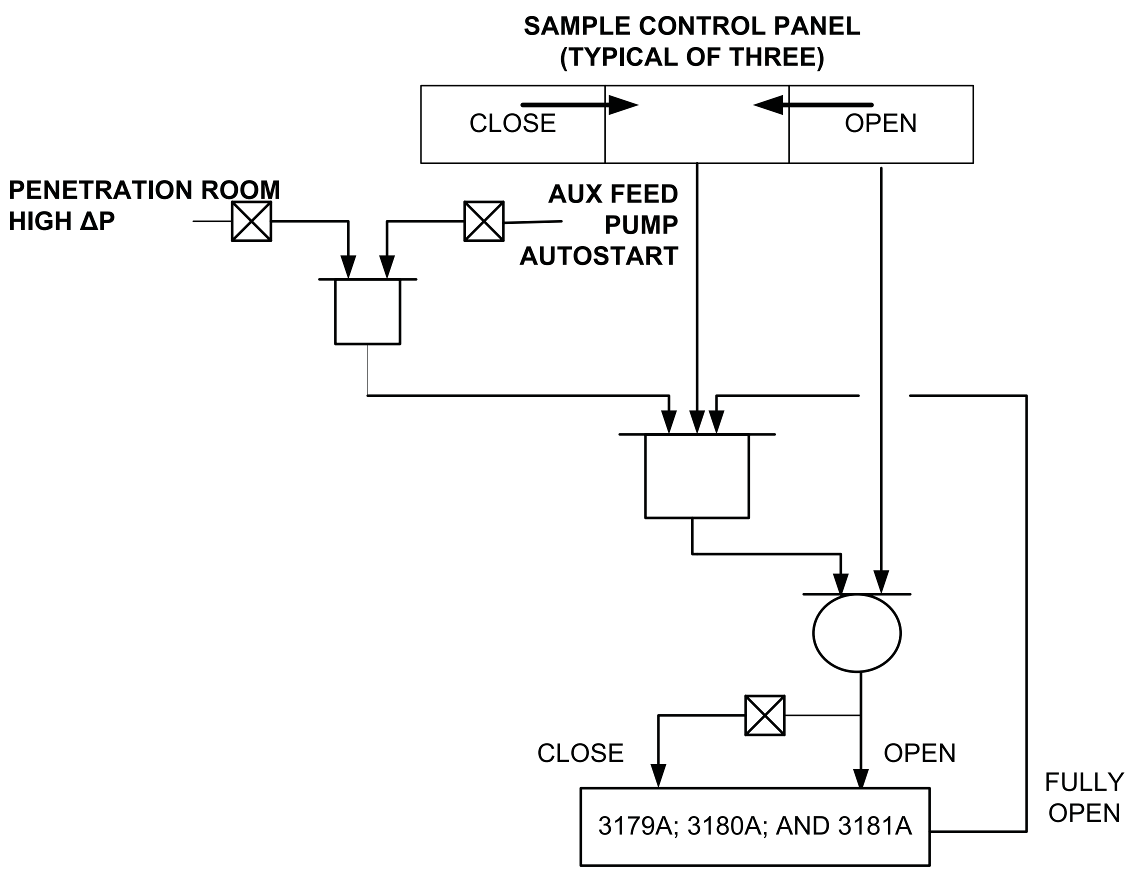
The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant.



Steam Sample System

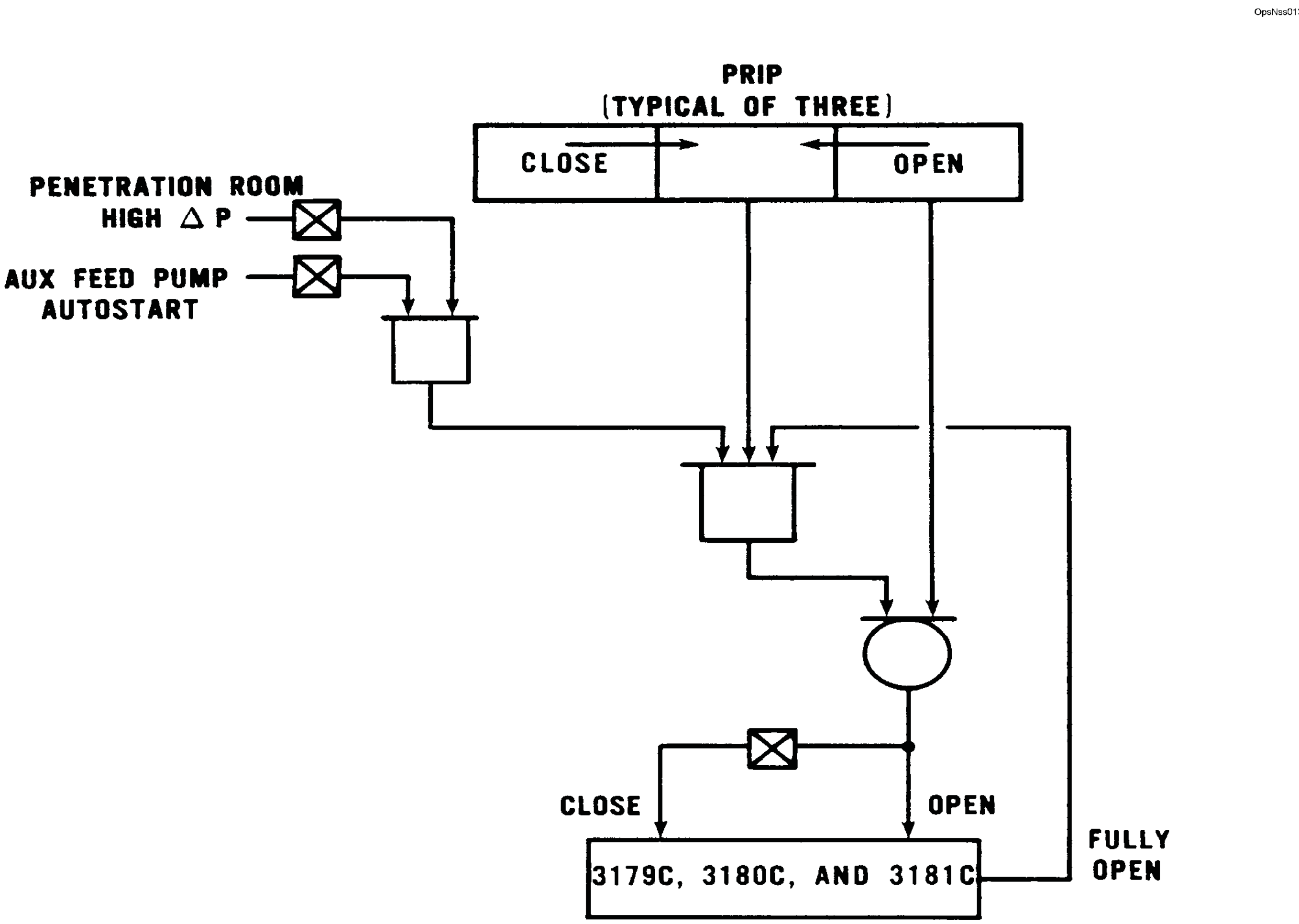
Figure 4

OPSNSS012



Steam Generator Blowdown Sample Isolation Valves
(3179A; 3180A; and 3181A)

Figure 8



Steam Generator Blowdown Sample Containment Isolation Valves
(3179C; 3180C; and 3181C)

Figure 9

1. RMS-52106D03 007/HLT//MEM 3.7/3.7/073A4.02////

Given the following:

- R-19, SGBD SAMPLE, radiation monitor is in alarm and stable above the alarm setpoint.
- The Shift Chemist requests to sample the Steam Generators.

Which ONE of the following correctly describes the actions that will allow the shift chemist to obtain a sample of the SGs IAW SOP-45.0, Radiation Monitoring System?

A. Manually open the sample valves one at a time.

B. Pull the INSTRUMENT power fuses for R-19 to allow opening the sample valves.

C. Pull the DC power fuses to each sample valve solenoid to fail the valve open.

D.✓ Place R-19 Operations Selector Switch to the RESET position, then open the sample valves.

1. 073G2.2.42 052/NEW//MEM 3.9/4.6/073G2.2.42/N///

Unit 1 is operating at 100% power.

- R-12, CTMT GAS has been declared INOPERABLE.

Which one of the following meets the **MINIMUM** reactor coolant leakage detection system(s) that must be in operation and OPERABLE to prevent entering a REQUIRED ACTION STATEMENT of Tech Spec 3.4.15, RCS Leakage Detection Instrumentation?

- R-11 - CTMT PARTICULATE
- CACCLMS - Containment Air Cooler Condensate Level Monitoring System

A. R-11 ONLY.

B. The CACCLMS ONLY.

C. Either R-11 **OR** the CACCLMS.

D.✓ BOTH R-11 **AND** the CACCLMS.

Technical Specifications:
3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment atmosphere particulate radioactivity monitor; and

b. One containment air cooler condensate level monitor or one containment atmosphere gaseous radioactivity monitor.

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

SOP-45
R-11 is the containment atmosphere particulate radioactivity monitor and
R-12 is the containment atmosphere gaseous radioactivity monitor.

Distracter analysis

A. Incorrect. See D. Plausible if applicant believes that the CACCLMS system is not required as long as one radiation monitor is operable.

B. Incorrect. See D. Plausible since this is the requirement per SOP-45 when R-12 is OOS but does NOT meet Tech Specs.

C. Incorrect. See D. Plausible since the Tech spec requires a specific combination of the 3 monitors and the applicant could easily confuse the requirements especially if they thought that grab samples from RE-67 met the requirement while R-12 is OOS.

D. Correct. Per TS - 3.4.15 this is the correct combination.

K/A: **073G2.2.42**	Process Radiation Monitoring (PRM) System - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
Importance Rating:	3.9 4.6
Technical Reference:	Technical Specifications, Ver 190. FNP-1-SOP-45, Radiation Monitoring System, Ver 46.2
References provided:	None
Learning Objective:	Given a set of Plant Conditions ASSESS those conditions and DETERMINE the ability of plant equipment and structures to meet their intended, designated function (OPS-52302A06)
Question History:	NEW
K/A match:	Requires the applicant to know **which RCS leakage detection systems (Process Radiation Monitors) that are required to meet Technical Specifications.**
SRO justification:	N/A

4.1.9.9 At the local radiation monitor skid, **check** the following indications:

4.1.9.9.1 Pump On Light (RED) - LIT ☐

4.1.9.9.2 CH 1 OPER Light (GREEN) - LIT ☐

4.1.9.9.3 CH 1 ALERT Light (AMBER) - NOT LIT ☐

4.1.9.9.4 CH 1 HIGH Light (RED) - NOT LIT ☐

4.1.9.9.5 CH 2 OPER Light (GREEN) - LIT ☐

4.1.9.9.6 CH 2 ALERT Light (AMBER) - NOT LIT ☐

4.1.9.9.7 CH 2 HIGH Light (RED) - NOT LIT ☐

4.1.9.10 At the local radiation monitor skid, **check** the compact cabinet cooler is running. ☐

4.1.9.11 At the RMS Cabinet II in the Control Room, **verify** the NORM/ SUPV Keylock Switch is aligned to NORM. ☐

4.1.9.12 At the RMS Cabinet II in the Control Room, **check** the following indications:

4.1.9.12.1 CH 1 OPER Light (GREEN) - LIT ☐

4.1.9.12.2 CH 1 ALERT Light (AMBER) - NOT LIT ☐

4.1.9.12.3 CH 1 HIGH Light (RED) - NOT LIT ☐

4.1.9.12.4 CH 2 OPER Light (GREEN) - LIT ☐

4.1.9.12.5 CH 2 ALERT Light (AMBER) - NOT LIT ☐

4.1.9.12.6 CH 2 HIGH Light (RED) - NOT LIT ☐

TABLE E

MONITOR	NAME	TYPE	AUTO FUNCTION
R-11	CTMT Particulate (CH 1)	Particulate	No
R-12	CTMT Gas (CH 2)	Gas	No

4.1.9.13 WHEN desired, THEN **remove** R11/12 from operation:

4.1.9.13.1 At the local radiation monitor skid, **place** the PUMP switch to OFF. ☐

4.1.9.14 WHEN desired, THEN **remove** R11/12 flow path from service:

4.1.9.14.1 **Remove** R11/12 form service per step 4.1.9.13. ☐

4.1.9.14.2 **Close** CTMT ATMOS TO R11/12, Q1E14MOV3660. ☐

4.1.9.14.3 **Close** CTMT ATMOS TO R11/12, Q1E14HV3658. ☐

4.1.9.14.4 **Close** R11/12 DISCH TO CTMT Q1E14HV3657. ☐

4.5 Rad Monitor Contingencies for Failed or OOS Rad Monitors:

NOTE

Any radiation monitor that fails or is taken out of service must be evaluated for possible impact on Emergency Action Levels and the Emergency Plan. Emergency Preparedness personnel may be able to provide assistance, if needed. ☐

CAUTION

Auto trips should be defeated, if required for current plant conditions, prior to de energizing Radiation monitors with auto trips (shaded monitors in Table F below). {AI 2007204735} ☐

4.5.1 **Perform** the appropriate contingency per Table F below: ☐

Table F

Monitor	Required Compensatory Action
R-5 (OOS)	Perform portable surveys of the monitored area within 24 hours of the monitor becoming inoperable and once per 24 hours. Refer to TR 13.3.4.
R-11 (OOS)	Obtain gaseous grab from RE-67 once per 24 hours. Refer to Tech. Spec. 3.4.15 for other actions.
R-11 & (CACCLMS) (OOS)	Obtain gaseous grab from RE-67 once per 12 hours. Refer to Tech. Spec. 3.4.15 for other actions.
R-12 (OOS)	Verify Containment Air Cooler Condensate Level Monitoring System (CACCLMS) operable once per 24 hours. Refer to Tech. Spec. 3.4.15 for other actions.
R-12 & (CACCLMS) OOS	Obtain samples from RE-67 once per 24 hours. Refer to Tech. Spec. 3.4.15 for other actions.
R-14 & R-22 (OOS)	Obtain a plant vent stack gas grab once per 8 hours from R-29A, RE-68, or R-29B. Refer to the ODCM for other actions.
R-15A (OOS)	Obtain a gas sample from RE-28 once per 8 hours. Refer to ODCM for other actions.
R-15B & R-15C (OOS)	Verify R-15A is operable. Refer to Tech. Spec. TR 13.3.4 for other actions.
R-18 (OOS)	Prior to discharging a WMT, verify WMT releases using 2 independent samples and analysts per FNP-1-CCP-212, DETAILED GUIDANCE FOR UNIT 1 WMT RELEASES. Refer to ODCM for other actions.
R-23B (OOS)	Obtain a liquid sample of steam generator blowdown in accordance with FNP-1-CCP-643, SAMPLING POINTS FOR POTENTIAL RADIOLOGICAL EFFLUENTS or FNP-1-CCP-652, OPERATION OF THE STEAM SAMPLE STATION, once per 24 hours if DEI of the secondary coolant is 0.01 μCi/gm or once per 8 hours if DEI of the secondary coolant is > 0.01 μCi/gm. Refer to the ODCM for other actions.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

R-11

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment atmosphere particulate radioactivity monitor; and

b. One containment air cooler condensate level monitor or one containment atmosphere gaseous radioactivity monitor.

R-12

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment atmosphere particulate radioactivity monitor inoperable.	A.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>	
	A.1.2 Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>	
	A.2 Restore the containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

---NOTE---
In MODE 3, with RCS pressure > 1000 psig, the accumulators may be inoperable for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

1. 076AA1.04 053/BANK/FNP 08/MEM 3.2/3.4/APE076AA1.04/N///

Unit 1 has been operating at 100% power and the Gross Failed Fuel Detector (GFFD) has been steady at 2000 cpm during the entire fuel cycle.

At 1000:

- FG5, GFFD SYS TRBL, has just come into alarm.

At 1015:

- A Reactor Trip and Safety Injection occurs.

Which one of the following completes the statements below?

The **minimum** GFFD reading that would cause FG5 to come into alarm is _(1)_ above background.

At 1020, flow through the GFFD _(2)_ be isolated.

	(1)	(2)
A.	1 X 10^4 cpm	will NOT
B.✓	1 X 10^4 cpm	WILL
C.	1 X 10^5 cpm	will NOT
D.	1 X 10^5 cpm	WILL

FG5 setpoint 1 x 10^4 cpm ABOVE background.

D175009- Sheet 1 - SV-3333 and SV-3765 close on a T signal (Phase A) which will actuate on a Safety Injection.

Distracter analysis

A. Incorrect. First part is correct (See B.1)

Second part is incorrect (See B.2). Plausible since the RHR to GFFD detector valves do not close on a T signal. In Mode 1, the GFFD will be lined up the RCS not RHR.

B. Correct. First part is correct. 1 X 10^4 cpm > background will cause the alarm.

Second part is correct. On a safety injection, a Phase A is generated isolating the GFFD from the RCS. In Mode 1, the GFFD is aligned to the RCS.

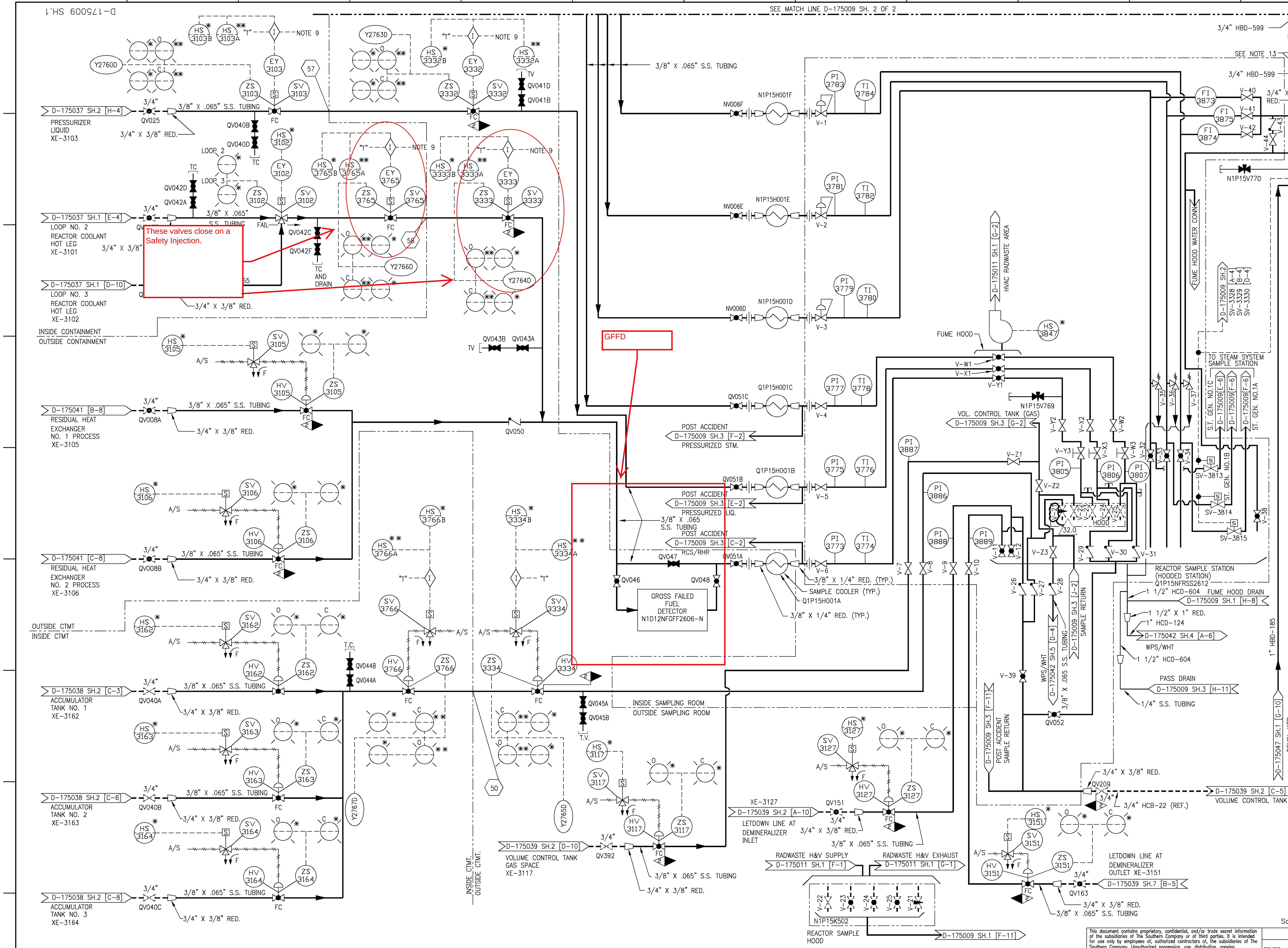
C. Incorrect. First part is incorrect (See B.1). Plausible since this is the setpoint in AOP-32 to reduce power by 25%. The applicant could confuse the two numbers.

Second part is incorrect (See A.2).

D. Incorrect. First part is incorrect (See C.1).

Second part is correct (See B.2).

K/A: **076AA1.04**	High Reactor Coolant Activity - Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment.
Importance Rating:	3.2 3.4
Technical Reference:	FNP-1-ARP-1.6, FG5 GFFD SYS TRBL, Ver 70 D175009, SH 1, Sampling System, Ver 32
References provided:	None
Learning Objective:	RELATE AND IDENTIFY the operational characteristics including design features, capacities and protective interlocks for the components associated with the Gross Failed Fuel Detector, to include the components found on Figure 2, GFFD Failed Fuel Detector System, and Figure 3, Sampling Assembly Flow Diagram (OPS-52106E02).
Question History:	FNP 08
K/A match:	Requires the applicant to be able to **monitor the failed fuel monitoring equipment and determine the minimum level at which the GFFD system trouble alarm actuates which directs the operators to AOP-32, Reactor Coolant High Activity.**
SRO justification:	N/A



LEGEND:

"T"–CONTAINMENT ISOLATION ACTUATION SIGNAL, PHASE "A".

AFPSS–TRIP OR CLOSE ON THE START OF AUXILIARY FEEDWATER PUMP
FC–FAIL CLOSED
TC–LEAK RATE TEST CONNECTION
TV–TEST VALVE CONNECTION.
HVAC–HEATING, VENTILATION & AIR CONDITIONING
WPS–WASTE PROCESSING SYSTEM.
RCS–REACTOR COOLANT SYSTEM.
VCT–VOLUME CONTROL TANK.
PRT–PRESSURIZER RELIEF TANK.
RE–RADIATION MONITOR ELEMENT.
* –LOCATED AT SAMPLE STATION.
** –LOCATED IN CONTROL ROOM.
+–EQUIPMENT SUPPLIER ON SAMPLE DRAIN TK. ORDERS.
PH–PH ANALYZER
P –PHOSPHATE ANALYZER

QUICK DISCONNECT COUPLING

MANUAL HIGH PRESSURE REDUCING VALVE

NOTES:

1. ALL VALVE NUMBERS PRECEDED BY PREFIX Q1P15 OR N1P15 UNLESS SHOWN.
2. FOR INSTALLATION AND FABRICATION SPECIFICATIONS SEE SECTION 6 OF INQUIRY SS–1109–2.
3. FOR SAMPLE STATION DETAILS SEE SS–1102–85.
4. ROUTE ALL SAMPLE STATION RELIEF BLOWOFF'S TO RESPECTIVE PANEL DRAIN HEADER.
5. ALL EQUIPMENT LOCATED WITHIN THE SAMPLE ROOM IS TO BE SUPPLIED BY SAMPLE SYSTEM VENDOR. EXCEPT SAMPLE COOLER RELIEF BLOWOFFS.
6. FOR P&ID LEGEND SEE DRAWING D–175016, SHEET 1 & 2.
8. ALL LINES NOT TO ASME SECTION III, SHALL BE DESIGNED IN ACCORDANCE WITH ANSI B 31.1.
9. STEAM GENERATOR BLOWDOWN PRESSURIZER AND REACTOR HOT LEG SAMPLE VALVES AUTOMATICALLY CLOSE ON HIGH PENETRATION ROOM PRESSURE.
10. FOR DETAILED EQUIPMENT DESCRIPTION AND BILL OF MATERIAL SEE U–259780, U–259638, & U–259455.
11. FOR EQUIPMENT LIST SEE DRAWING D–175070.
12. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B–175803.
13. 2" BLIND FLANGE DRILLED TO SUIT FOR 3/4" LINE.
14. VALVE MK–J–240 CAN BE USED INSIDE THE SAMPLE ROOM TO REPLACE THE KEROTEST GLOBE VALVES (FIG. 9921) USED IN THE REACTOR SAMPLE STATION AND THE STEAM SYSTEM SAMPLE STATION.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D–205009.

CAD D1750091.DWG
ACAD2002 | DFV–04

Southern Company Services, Inc. for

ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
SUBJECT P&ID – SAMPLING SYSTEM

DRAWN CHECKED DESIGNED
APPROVED FOR SIGNATURES DATE
APPROVED SEE MICROFILM DATE
APPROVED COPY DATE
SCALE B/M
SHEET 1 OF – SHEETS
SUPERSEDES

D–175009 VER. 32.0

DFV WMW ASK
VER. NO. 32.0 DATE 8–3–05
REVISED PER ABN 04–0–2806, VER. 2.0

DWG CATEGORY: CRITICAL

Step	Action/Expected Response		Response NOT Obtained
___ 1	Direct Chemistry to sample RCS for activity using FNP-0-CCP-1300, CHEMISTRY AND ENVIRONMENTAL ACTIVITIES DURING A RADIOLOGICAL ACCIDENT.		
___ 2	Check gross failed fuel detector indication - LESS THAN 10^5 cpm ABOVE NORMAL.	2	Reduce reactor power by 25% from existing power level.
___ 3	<u>WHEN</u> RCS activity determined, <u>THEN</u> evaluate continued plant operation using Technical Specification 3.4.16.		
___ 4	Direct HP to survey AUX BLDG for radiation level.		
___ 5	Direct Chemistry to sample all SGs for activity.		
___ 6	Direct Chemistry to sample affected effluent paths.		
___ 7	<u>IF</u> elevated RCS activity <u>NOT</u> caused by failed fuel, <u>THEN</u> reduce RCS activity.	7	Consult Operations Manager to evaluate further plant response.

NOTE: Placing the standby mixed bed demineralizer in service will reduce RCS lithium concentration.

7.1 Align standby mixed be demineralizer for service using FNP-1-SOP-2.5, RCS CHEMICAL ADDITION, VCT GAS CONTROL AND DEMINERALIZER OPERATION.

7.2 Direct Chemistry to sample demineralizer inlet and outlet at least once per 4 hours.

7.3 Raise letdown flow to 120 gpm.

Step 7 continued on next page.

LOCATION FG5

G5
GFFD
SYS
TRBL

SETPOINT:
1. High: 1×10^4 CPM above background
2. Low: 1×10^1 CPM
3. High sample temperature: 135°F

ORIGIN:
1. High Neutron Relay (K101)
2. Low Neutron Relay (K102)
3. Malfunction Alarm Relay (K-2)

PROBABLE CAUSE

NOTE: Low Alarm (1×10^1 CPM) and High Sample Temperature are used for indicating a system malfunction.

1. High RCS Activity due to Fuel Failure.
2. Loss of 118VAC Control Power.
3. Gross Failed Fuel Detector (GFFD) System circuit malfunction.
4. Low flow thru Gross Failed Fuel Detector system.
5. Chemistry Sampling Activities
6. Low Component Cooling Water flow through GFFD Sample Cooler

AUTOMATIC ACTION

NONE

OPERATOR ACTION

NOTE: A high sample temperature condition will illuminate the red local high sample temperature indicator light on the control room GFFD panel.

1. Check actual reading to determine if alarm is due to high activity, instrument malfunction, low flow, or high sample temperature.
2. IF high activity is indicated, THEN perform the actions required by FNP-1-AOP-32.0, REACTOR COOLANT HIGH ACTIVITY.
3. IF a high sample temperature condition exists, THEN dispatch personnel to locally check the GFFD to determine and correct the cause of the problem.
   3.1 WHEN the high condition has cleared, THEN manually reset the alarm by depressing the reset button on the lower left of the digital temperature indicator. (GFFD 121’ PPR)
4. IF an instrument malfunction is indicated, THEN:
   A. Sample and analyze the Reactor Coolant to verify that High Activity does not exist.
   B. Notify appropriate personnel to determine and correct the cause of the malfunction as soon as possible.
5. Refer to the Technical Specifications section on Reactor Coolant System Specific Activity, section 3.4.16.
6. WHEN activity levels have decreased below the alarm setpoint, THEN reset the HI alarm on the drawer by rotating OPERATION SELECTOR switch to RESET and back to OPERATE.

References: A-177100, Sh. 305; U-215440; U-215441

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 3

PHASE A CONTAINMENT ISOLATION

NOTE:
- ATTACHMENT 3, FIGURE 1 provides a listing of component names corresponding to each MLB-2 location.
- ATTACHMENT 10 provides a listing of sequenced loads.

1 **Check all the following MLB-2 indicating lights lit.**

1 Verify associated component status.

	1	2	3	4	5	6	7	8	9	10
1	CTMT ISO PHASE A	3657 CLOSED	3198A CLOSED	3772A CLOSED	8112 CLOSED	LCV1003 CLOSED	7126 CLOSED	CONT RM FILT FAN 1A ON	CONT RM PRZN FAN 1A ON	3622 CLOSED
2	3234A CLOSED	3660 CLOSED	3198D CLOSED	3772B CLOSED	8149A CLOSED	3377 CLOSED	3103 CLOSED	3104 CLOSED	3649A CLOSED	3624 CLOSED
3	P16V515 CLOSED	3318B CLOSED	2866C CLOSED	3772C CLOSED	8149B CLOSED	3380 CLOSED	8033 CLOSED	3765 CLOSED	3649B CLOSED	3626 CLOSED
4	P16V517 CLOSED	3999A CLOSED	2867C CLOSED	3443 CLOSED	8149C CLOSED	8871 CLOSED	8028 CLOSED	3766 CLOSED	3649C CLOSED	3628 CLOSED

	11	12	13	14	15	16	17	18	19	20
1	CTMT ISO PHASE A	3658 CLOSED	3198B CLOSED	3196 CLOSED	8100 CLOSED	7136 CLOSED	3331 CLOSED		CONT RM FILT FAN 1B ON	CONT RM PRZN FAN 1B ON
2	3234B CLOSED		3198C CLOSED	3197 CLOSED	8152 CLOSED	3376 CLOSED	3332 CLOSED		3623 CLOSED	3627 CLOSED
3	P16V514 CLOSED	3318A CLOSED	2866D CLOSED	3067 CLOSED	8880 CLOSED	7150 CLOSED	3333 CLOSED		3625 CLOSED	3629 CLOSED
4	P16V516 CLOSED	3999B CLOSED	2867D CLOSED	3095 CLOSED	8860 CLOSED	8961 CLOSED	3334 CLOSED		8047 CLOSED	3659 CLOSED

2 **Notify control room of phase A containment isolation status.**

-END-

ATTACHMENT 3

FIGURE 1

LOCATION	COMPONENT NUMBER	NAME
6-3	Q1G21HV3380	CTMT SUMP RECIRC (BOP) - CLOSED
6-4	Q1E21HV8871	ACCUM TEST LINE TO RWST ISO - CLOSED
7-1	Q1G21HV7126	RCDT VENT LINE ISO - CLOSED
7-2	Q1P15SV3103	PRZR LIQ SAMPLE ISO (BOP) - CLOSED
7-3	Q1B31HV8033	PRT N2 SUPPLY - CLOSED
7-4	Q1B31HV8028	RMW TO PRT ISO - CLOSED
8-1	QSV49F001A/3A	1A CONT RM FILTRATION RECIRC UNIT (BOP) - ON
8-2	Q1P15SV3104	PRZR STM SAMPLE ISO (BOP) - CLOSED
8-3	Q1P15SV3765	RCS LOOPS 2 & 3 SAMPLE ISO (BOP) - CLOSED
8-4	Q1P15HV3766	ACCUM SAMPLE ISO (BOP) - CLOSED
9-1	QSV49F002A	1A CONT RM PRZN FILTER UNIT (BOP) - ON
9-2	QSV49HV3649A	CONT RM EXH FAN INLET DMPRS (BOP) - CLOSED
9-3	QSV49HV3649B	CONT RM EXH FAN INLET DMPRS (BOP) - CLOSED
9-4	QSV49HV3649C	CONT RM EXH FAN INLET DMPRS (BOP) - CLOSED
10-1	QSV47HV3622	COMPUTER RM HVAC RTN (BOP) - CLOSED
10-2	QSV47HV3624	CONT RM HVAC SUPP (BOP) - CLOSED
10-3	QSV47HV3626	COMPUTER RM HVAC SUPP (BOP) - CLOSED
10-4	QSV49HV3628	CONT RM UTILITY EXH (BOP) - CLOSED
11-1	N/A	CTMT ISO PHASE A - ACTUATED
11-2	Q1N12HV3234B	TDAFWP STM SUPP WARMUP ISO (BOP) - CLOSED
11-3	Q1P16V514	SW TO TURB BLDG ISO B TRN - CLOSED
11-4	Q1P16V516	SW TO TURB BLDG ISO A TRN - CLOSED
12-1	Q1E14HV3658	R-11/12 DISCH TO CTMT (BOP) - CLOSED
12-2	N/A	SPARE

ATTACHMENT 3

FIGURE 1

LOCATION	COMPONENT NUMBER	NAME
12-3	Q1E14MOV3318A	CTMT ΔP ISO (BOP) - CLOSED
12-4	Q1E12HV3999B	RX CAV CLG DMPR (BOP) - CLOSED
13-1	N1P13HV3198B (N1P13V292)	CTMT PURGE DMPRS (HS-3196) - CLOSED
13-2	N1P13HV3198C (N1P13V291)	CTMT PURGE DMPRS (HS-3196) - CLOSED
13-3	Q1P13HV2866D (Q1P13V302)	MINI-PURGE SUPPLY DAMPER - CLOSED
13-4	Q1P13HV2867D (Q1P13V304)	MINI-PURGE EXHAUST DAMPER - CLOSED
14-1	Q1P13HV3196 (Q1P13V283)	CTMT PURGE DPRS (HS-3196) - CLOSED
14-2	Q1P13HV3197 (Q1P13V282)	CTMT PURGE DPRS (HS-3196) - CLOSED
14-3	Q1P17HV3067	CCW FROM EXC LTDN/RCDT HXS - CLOSED
14-4	Q1P17HV3095	CCW TO EXC LTDN/RCDT HXS - CLOSED
15-1	Q1E21MOV8100	RCP SEAL WTR RTN ISO - CLOSED
15-2	Q1E21HV8152	LTDN LINE CTMT ISO - CLOSED
15-3	Q1E21HV8880	ACCUM N2 SUPPLY ISO - CLOSED
15-4	Q1E21HV8860	ACCUM FILL LINE ISO - CLOSED
16-1	Q1G21HV7136	RCDT PUMPS DISCH LINE ISO - CLOSED
16-2	Q1G21HV3376	CTMT SUMP DISCH (BOP) - CLOSED
16-3	Q1G21HV7150	RCDT VENT LINE ISO - CLOSED
16-4	Q1E21HV8961	ACCUM TEST LINE TO RWST ISO - CLOSED
17-1	Q1P15SV3331	PRZR STM SAMPLE ISO (BOP) - CLOSED
17-2	Q1P15SV3332	PRZR LIQ SAMPLE ISO (BOP) - CLOSED
17-3	Q1P15SV3333	RCS LOOPS 2 & 3 SAMPLE ISO (BOP) - CLOSED

1. 076K2.08 054/BANK/FNP 08/MEM 3.1*/3.3*/076K2.08/N///

Unit 1 is stable in Mode 3 following a Reactor Trip when the following conditions occur:

- Power has been lost to 4160V AC buses G, J, and L.

Which one of the following lists the valves that the OATC can close from the MCB to isolate a Service Water rupture in the Turbine Building?

Q1P16V514, SW TO TURB BLDG ISO B TRN
Q1P16V515, SW TO TURB BLDG ISO A TRN
Q1P16V516, SW TO TURB BLDG ISO A TRN
Q1P16V517, SW TO TURB BLDG ISO B TRN

A. MOVs 514 and 517

B. MOVs 514 and 516

C.✓ MOVs 515 and 517

D. MOVs 515 and 516

Electrically:
1N MCC (A Train) - V515 and V517
1T MCC (B Train) - V514 and V516

Mechanically:
A Train - V515 and V516
B Train - V514 and V517

Distracter analysis

A. Incorrect. See B. Plausible since the 4 SW to Turbine Building Isolation valves are powered from and mechanically aligned to different trains and the applicant could easily confuse which valve is powered by which train and which valve is in which mechanical train.

B. Incorrect. See A.

C. Correct. Both of these valves are powered from A train power and in opposite trains mechanically (See Above).

D. Incorrect. See A.

K/A: **076K2.08**	Service Water System (SWS) - Knowledge of bus power supplies to the following: ESF-actuated MOVs
Importance Rating:	3.1* 3.1*
Technical Reference:	A506250, Unit 1 Electrical Load List, Ver 74.0 D-170119, SH 2, Service Water, Ver 47
References provided:	None
Learning Objective:	NAME AND IDENTIFY the Bus power supplies, for those electrical components associated with the Service Water System, to include those items in Table 7- Power Supplies (OPS-40101B04).
Question History:	FNP 08
K/A match:	Apllicant is required to **know the bus power supplies to Service Water ESF actuated MOVs.**
SRO justification:	N/A

DF13
1H 4160V BUS **DB - 155'** **D177018**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R15A0503-A	**1H 4160V BUS <<< DF13 (NORMAL)**	
DH01	**N1R11B0501-N**	**1G 4160/600V SST >>> EG02 (NORMAL)**	**H-2**
DH02	**Q1R11B0502-A**	**1H 4160/600V SST >>> EH02**	**H-44**
DH03	QSP25M0008-A	NO. 8 RW PUMP	
DH04	QSP25M0009-A	NO. 9 RW PUMP	
DH05	QSP25M0010-A	NO. 10 RW PUMP	
DH06	Q1R15BKRDH06	PT COMPARTMENT	
DH07	**QSR43A0503-A**	**1C DIESEL GENERATOR (EMERG) <<<**	
DH08	**Q1R11B0503-A**	**1R 4160/600V SST >>> ER02**	**H-49**

DH08
1R 600V LOAD CENTER **DB - 155'** **F177677**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R16B0508-A	**1R 600V LOAD CENTER**	
ER01	Q1R16BKRER01	PT COMPARTMENT	
ER02	**Q1R11B0503-A**	**1R 4160/600V SST <<< 1-DH08 (ALTERNATE)**	
ER03	**Q1R17B0507-A**	**1N 600/208V MCC >>>**	**H-50**
ER05	**Q2R11B0503-A**	**2R 4160/600V SST <<< 2-DH08 (NORMAL)**	

DH08
ER03
1N 600/208V MCC **DB-155'** **B177556-13**

BKR	TPNS	DESCRIPTIONS	SEE PAGE
	Q1R17B0507-A	**1N 600/208V MCC (600V SECTION) <<< ER03**	
FNA3	QSY41B0523A-A	DG BLDG SWITCHGEAR ROOM A HEATER A	
FNA4	QSY41B0523B-A	DG BLDG SWITCHGEAR ROOM A HEATER B	
FNB2	QSY41B0523C-A	DG BLDG SWITCHGEAR ROOM A HEATER C	
FNB3	Q1P16V0515-A	SERVICE WATER MOV V515	
FNB4	Q1P16V0517-A	SERVICE WATER MOV V517	
FNC2	QSY41C0508B-A	1C GENERATOR ROOM FAN B	
FNC4	N1P26V0505A-N	CWP DISCHARGE MOV V505A	
FNC5	QSY41B0522L-A	1C GENERATOR ROOM HEATER B	
FND2	QSY41B0522M-A	1C GENERATOR ROOM HEATER C	
FND3	Q1P16V0558-A	SERVICE WATER MOV V558	
FND4	QSR43M0508-B	AUX JACKET WATER PUMP - DIESEL 1C	
FND5	Q1P16V0542-A	SERVICE WATER MOV V542	
FNE2	QSY41C0508A-A	1C GENERATOR ROOM FAN A	
FNE3	QSY41B0522K-A	1C GENERATOR ROOM HEATER A	
FNE4	Q1P16V0524-A	SERVICE WATER MOV V524	
FNE5	Q1P16V0525-A	SERVICE WATER MOV V525	
FNF2L	**QSR43L0503-A**	**1C DIESEL 75 KVA 600/208V AUX TRANSFORMER LOCATED IN HN02 >>> 1C D.G. BREAKER PANEL BOARD >>>**	H-52
FNF3	Q1P16V0532-A	SERVICE WATER MOV V532	
FNF4	Q1P16V0533-A	SERVICE WATER MOV V533	
FNF5	QSR43M0503A-A	AIR COMPRESSOR A - DIESEL 1C	
FNF6	QSR43M0503B-A	AIR COMPRESSOR B - DIESEL 1C	
FNG2	Q1P16V0539-A	SERVICE WATER MOV V539	

1G 4160V BUS AB - 121' D177006

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R15A0007-B	**1G 4160V BUS**	
DG01	**N1R11A0501-N**	**1A STARTUP TRANSFORMER (ALTERNATE) <<<**	
DG02	**Q1R15A0506-B**	**1L 4160V BUS >>>**	L-1
DG03	**Q1R11B0005-B**	**1E 4160/600V SST (NORMAL) >>> EE02 >>>**	G-2
DG04	Q1P17M0001A-B	1A CCW PUMP	
DG05	Q1P17M0001B-AB	1B CCW PUMP DISC SWITCH Q1R18A00004B-B >>> 1B CCW PUMP (B TRAIN SUPPLY)	
DG06	Q1E21M0001C-B	1C CHARGING/HHSI PUMP	
DG07	Q1E21M0001B-AB	1B CHARGING/HHSI PUMP DISC SWITCH Q1R18A0001B-B >>> 1B CHARGING/HHSI PUMP (B TRAIN SUPPLY)	
DG08	**Q1R43A0502-B**	**1B DIESEL GENERATOR (EMERG) <<<**	
DG09	Q1E11M0001B-B	1B RHR/LHSI PUMP	
DG10	Q1N23M0001B-B	1B AFW PUMP	
DG11	Q1E13M0001B-B	1B CTMT SPRAY PUMP	
DG12	**Q1R11B0006-AB**	**1F 4160/600V SST DISC SWITCH Q1R18A0003B-B >>> 1F 4160/600V SST >>> 1F LOAD CENTER (B TRAIN SUPPLY) >>>**	F-117
DG13	**Q1R15A0504-B**	**1J 4160V BUS >>>**	J-1
DG14	Q1R15BKRDG14	PT COMPARTMENT	
DG15	**N1R11A0502-N**	**1B STARTUP TRANSFORMER (NORMAL) <<<**	

DG03

1E 600V LOAD CENTER **AB - 121'** **D177011**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R16B0007-B	**1E 600V LOAD CENTER**	
EE01	Q1R16BKREE01	PT COMPARTMENT	
EE02	**Q1R11B0005-B**	**1E 4160/600V SST (NORMAL) <<< DG03**	
EE03	N1C11M0001B-N	1B CRDM MG SET	
EE05	**Q1R42B0001B-B**	**1B BATTERY CHARGER Q1R42E0001B-B >>> 1B 125VDC SWGR >>>**	G-3
EE06	**Q1R42B0001B-B**	**600V AC DISC SW Q1R18B0001B-B >>> 1C BATT CHARGER Q1R42E0001C-AB(B TRAIN SUPPLY) >>> 125V DC DISC SW Q1R18B0002B-B >>> 1B 125V DC SWGR >>>**	G-3
EE07	**Q1R16B0005-B**	**1C 600V LOAD CENTER (ALTERNATE-EMERG) >>> EC10 >>>**	E-5
EE08	Q1E12M0001C-B	1C CONTAINMENT COOLER (EMERG./ LOW SPEED)	
EE09	Q1E22M0001B-B	1B REACTOR CAVITY DILUTION FAN	
EE10	**Q1R17B0002-B**	**1B 600/208V MCC >>>**	G-80
EE11	**QSR17B0007-B**	**1G 600/208V MCC >>>**	G-88
EE12	**Q1R16B0008-AB**	**1F 600V LOAD CENTER (ALTERNATE) <<< EF08**	
EE13	N1T47M0001A-B	1A CRDM COOLER FAN	
EE14	**Q1R17B0510-B**	**1T 600/208V MCC >>>**	G-103
EE15	**Q1R17B0009-B**	**1V 600/208V MCC >>>**	G-109
EE16	Q1E12M0001D-B	1D CONTAINMENT COOLER (EMERG./ LOW SPEED)	

DG03
EE14
FTF5L
1T 600V MCC SECTION (CONT'D) **DB - 155'** **B177556-18B**

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
FTF4	---------------	MCC SPACE HTRS BKR COMPT	
FTF5L	---------------	1T 600/208V MCC XFMR>>>1TMCC 208V SECTION >>>	G-106
FT-M1		INCOMMING LINE COMPARTMENT	
FT-M2		TERMINAL BLOCK COMPARTMENT	
FT-M3	Q1P16V514-B	SERVICE WATER MOV - V514	
FT-M4	Q1P16V516-B	SERVICE WATER MOV - V516	
FT-M5	---------------	SPARE	
FT-N1		TERMINAL BLOCK COMPARTMENT	
FT-N2	Q1P16V518-B	SERVICE WATER MOV - V518	
FT-N3	Q1P16V536-B	SERVICE WATER MOV - V536	
FT-N4	Q1P16V538-B	SERVICE WATER MOV - V538	


TURBINE BUILDING
AUXILIARY BUILDING
VALVE BOX VB-1
VALVE BOX VB-2
VALVE BOX VB-3
VALVE BOX VB-4
DIESEL GENERATOR BUILDING
COOLING TOWER BLOWDOWN
CIRCULATING WATER CANAL
STORAGE POND FLUME
Turbine Building Supply
V-515 and V516 are A Train mechanically
V514 and V517 are B Train mechanically
DILUTION BY-PASS
12" HBC-201
24" HBD-255
24" HBD-254
24" HBC-210
4" HBD-346 ABANDONED
TRAIN "A"
TRAIN "B"
D-175003 SH.1
D-170119 SH.4
Q1P16V514
Q1P16V515
Q1P16V516
Q1P16V517
LOCA SIGNAL
LOSP SIGNAL
SERVICE WATER TRAIN "A"
SERVICE WATER TRAIN "B"
D-170119 SH.1
DETAIL "A"
HEADER
HBC HCC INSTR. CLASS
RADWASTE DISCHARGE
STEAM GEN. BLOWDOWN
TURB BLDG. SUMP DISCH.
SAMPLE LINE AT CIRCULATING WATER PUMP STRUCTURE
CHEMICAL INJECTION NOZZLE
CHEMICAL INJECTION SYS. SUPPLIED BY CONTRACTOR
STANDPIPE SURGE TNK
MANHOLE
TO RIVER
60" KBD-200
FROM UNIT 2
SURFACE MOUNTED ULTRASONIC TRANSDUCERS
N1P16FE570 (ABANDONED IN PLACE)
FOR GENERAL NOTES SEE SH.1
CAD D1701192.DWG
ACAD2010 Vep - 04
Southern Company Services, Inc. for
ALABAMA POWER COMPANY
J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
P&ID - SERVICE WATER SYSTEM
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FOR SIGNATURES
SEE MICROFILM
COPY
SCALE NONE
SHEET 2
D-170119
47.0 2-19-13 REVISED PER ABN SNC461051M002, VER. 1.0
DRAWING CATEGORY: CRITICAL

1. 076K4.03 055/BANK/FNP 07/C/A 2.9*/3.4*/076K4.03/N///

Unit 1 is operating at 100% power with the following conditions:

- 1A Containment Cooler is isolated per SOP-12.1, Containment Air Cooling System.
- The following valves are closed with power available:

  - MOV-3019A, SW TO 1A CTMT CLR AND CTMT FPS
  - MOV-3441A, SW FROM 1A CTMT CLR
  - MOV-3024A, EMERG SW FROM 1A CTMT CLR

- MOV-3023A, 1A CTMT CLR SW DISCH, is OPEN.

Subsequently, a steam break occurs and containment pressure rises to 5 psig.

Which one of the following completes the statement below?

1A Containment Cooler service water flow will be _____.

A. 0 gpm

B. approximately 600 gpm

C. approximately 800 gpm

D.✓ approximately 2000 gpm

FSD-A-181013: Post-accident, the containment coolers provide for long-term containment heat removal. **Following a safety injection signal** and depending upon the availability of offsite power, the containment coolers are restarted on low speed (A loss of off site power (LOSP) would result in one fan from each train being started). Each cooler is nominally operated at a low speed generating 40,000 cfm with a service water **flow rate of approximately 2000 gpm**. During post-accident operation, each cooler provides approximately 80 x 106 Btu/hr of cooling capacity

Distracter analysis

A. Incorrect. See D. Plausible since the applicant may know that an MOVin the service water supply/return to each cooler will not open on an SI (MOV-3023A) and improperly think that it is MOV-3441A and therefore there would be no flow.

B. Incorrect. See D. Plausible since this is the minimum design flow per tech spec bases. The applicant may not be able to recall the proper SW flow.

C. Incorrect. See D. Plausible since this is the normal flow through the 1A containment cooler. The applicant may not be able to recall the proper SW flow

D. Correct. This is the post accident flow through the 1A containment cooler.

K/A: **076K4.03**	Service Water System (SWS) - Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: **Automatic opening** features associated with **SWS isolation valves** to CCW heat exchangers
Importance Rating:	2.9* 3.4*
Technical Reference:	FSD-A181013, Containment Ventilation System, Ver 14
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Service Water System components and equipment, to include the following (OPS-40101B07): [...] Automatic actuation including setpoint (example SI, Phase A, LOSP)
Question History:	FNP 07
K/A match:	Requires the applicant to know the **Containment Cooler Isolation valves open by design on a safety injection and what the resultant SW flow to the coolers for accident conditions will be.** 10/24/12 - Per discussion with Chief Examiner, using SWS Turbine Building Isolation closure was acceptable due to FNP plant design. **Since the SW to TB MOVs were being addressed in a different KA and since this was an auto opening,** Service Water to the Containment Coolers have automatic opening features and more closely fit the K/A based on FNP design.
SRO justification:	N/A

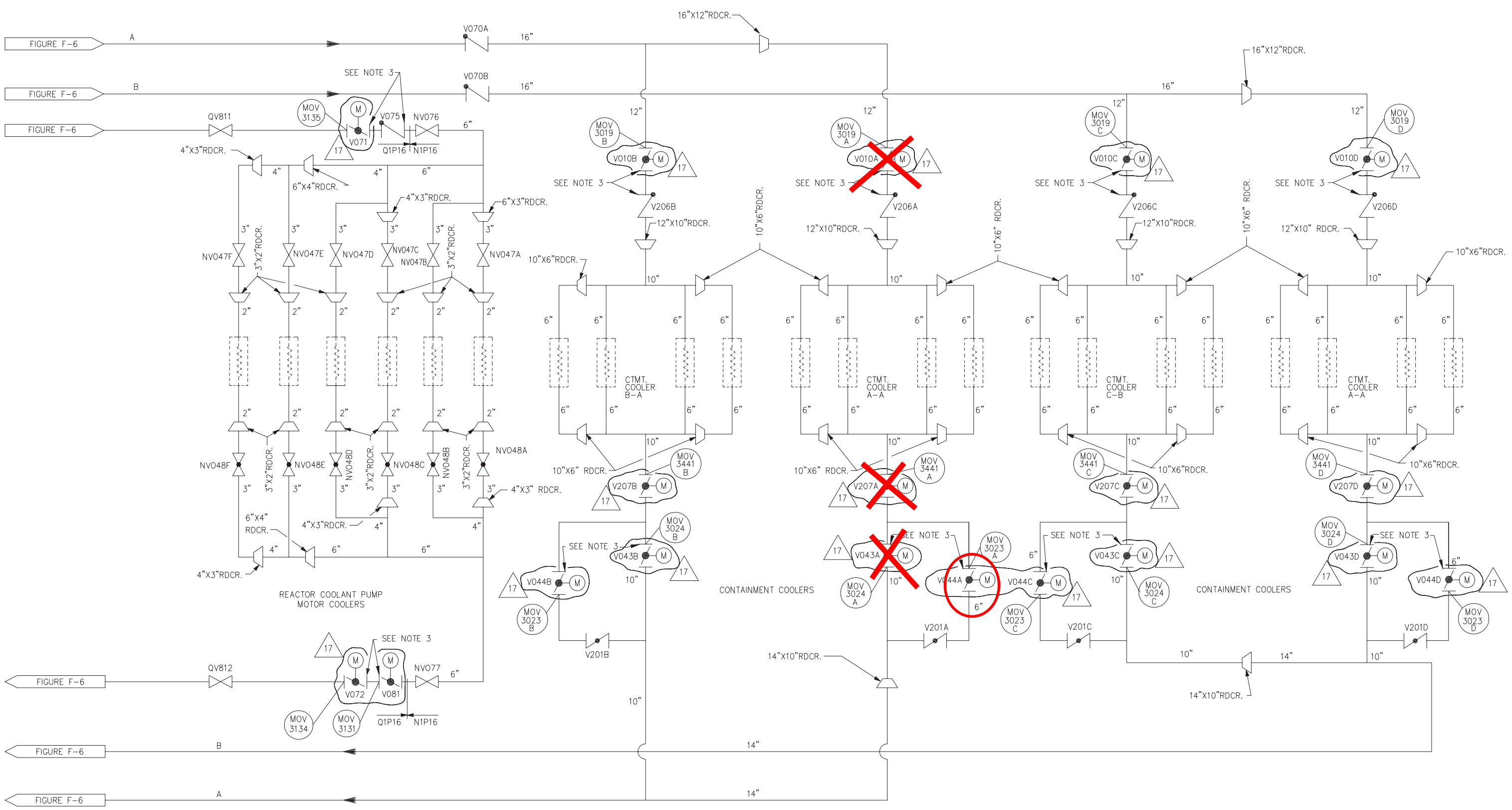


FIGURE F-7
SERVICE WATER SYSTEM
UNIT 1 CONTAINMENT & AUXILIARY BLDG.
SIMPLIFIED SYSTEM BOUNDARY DIAGRAM

NOTES:

1. ALL VALVES ARE PREFIXED BY Q1P16, UNLESS NOTED OTHERWISE.
2. FOR GENERAL NOTES, SEE FIGURE F-1.
3. CONTAINMENT ISOLATION FUNCTIONS OF THESE VALVES IS NOT WITHIN THE SCOPE OF THE SWS FSD.

2.1.1.2 Containment Dome Recirculation Fans

These four fans recirculate and mix the containment dome atmosphere to prevent thermal stratification by drawing air from the upper elevations of the containment and discharging it towards the 155' elevation (inlet elevation of the containment coolers).

Sizing of the containment recirculation system was based on 1/3 to 1/2 of the total flow rate of the containment coolers as allowed by design standards. Total containment cooler flow rate was assumed as 240,000 cfm since 100 percent design capacity was based on three coolers operating. Based on this design criterion, the containment recirculation system is sized for a total flow of 100,000 cfm (Reference 6.3.001).

2.1.2 Post-Accident Operation

2.1.2.1 Containment Coolers and Fans

Post-accident, the containment coolers provide for long-term containment heat removal. Following a safety injection signal and depending upon the availability of offsite power, the containment coolers are restarted on low speed (A loss of off site power (LOSP) would result in one fan from each train being started). Each cooler is nominally operated at a low speed generating 40,000 cfm with a service water flow rate of approximately 2000 gpm. During post-accident operation, each cooler provides approximately 80 x 10^6 Btu/hr of cooling capacity (References 6.3.010, 6.7.001, 6.7.002).

Containment pressure and temperature accident analysis is based on the operation of one containment cooler with a degraded service water flow of 600 gpm. Heat removal by the containment cooler is modeled to begin 115 seconds after the start of the accident (Reference 6.3.002).

2.1.2.2 Containment Dome Recirculation Fans

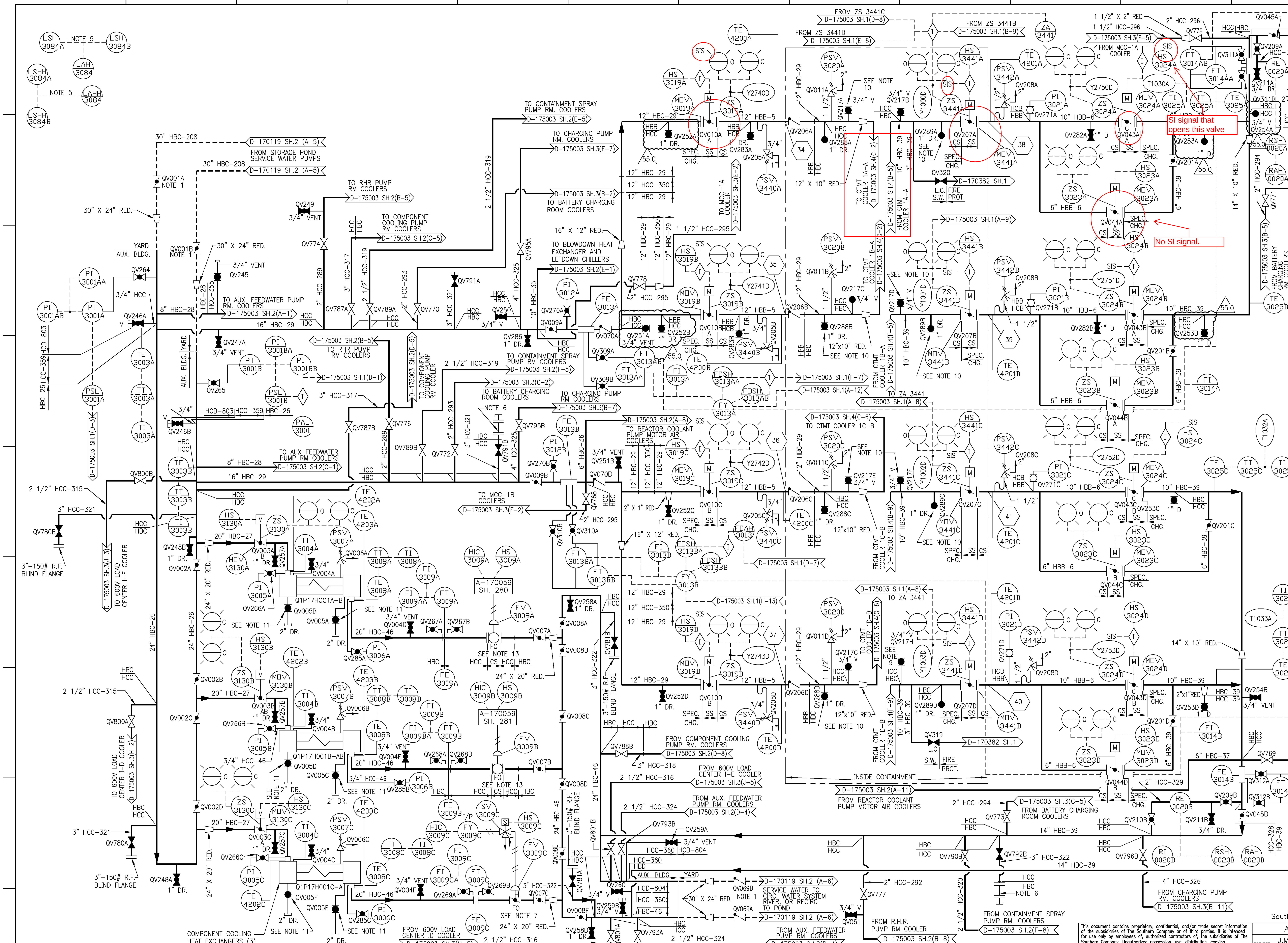
The containment dome recirculation fans are not required for post-accident operation (Reference 6.3.002).

3.1.2.2 For post-LOCA operation, the fans are reduced to low speed due to changes in air density and develop a minimum design flow rate of 40,000 cfm at an external static pressure of 1 in. w.g. (References 6.3.010, 6.5.002). Containment pressure and temperature analysis is based on the operation of one containment cooler with a degraded service water flow of 600 gpm (Reference 6.3.002).

3.1.2.3 Sizing of the containment coolers on the water side was based on a maximum inlet service water temperature of 95 °F with a nominal flow rate of 800 gpm/cooler for normal plant operation and 2000 gpm/cooler for post-LOCA operation (Reference 6.3.010). A total of 69 circuits (equivalent of 2 coils plus 5 circuits) of any containment cooler can be removed from service and not degrade the containment cooling performance to a point that the cooler can not adequately remove the containment analysis heat loads. This analysis was performed using the design basis fouling factor of 0.003 (References 6.3.020, 6.3.021) Coolers have been reconfigured with a waterbox design to allow for individual tube plugging, and cooler coils have been replaced with erosion/corrosion resistant materials (Reference 6.7.052).

Water hammer analysis has identified that the region of the service water piping that is most susceptible to water hammer is the containment cooler return piping. However, this analysis showed that no water hammer will occur in this piping. Due to the elevation differences between the service water pond, the containment coolers, and the river discharge, a substantial flow of service water through the containment coolers will continue in the event of a LOSP. The maximum temperature of the water achievable in the water hammer region of the service water return piping prior to Service Water Pump restart is 119 °F. This temperature, being less than the 164°F required to form a vapor cavity in the return piping, denies the possibility of water hammer when the system is repressurized due to pump restart after a LOSP (Reference 6.3.023, 6.7.053).

3.1.2.4 During post-accident conditions, the air density changes from a nominal value of 0.07 lb/ft3 to 0.19 lb/ft3. This change necessitates a larger motor to have the ability to operate at 80,000 scfm (Δhorsepower is proportional to Δdensity). Therefore, to reduce the power demand on the emergency diesels, the fan speeds are reduced. However, in order to provide additional cooling required during post-accident, the amount of cooling water is increased from a nominal 800 gpm to 2000 gpm (Reference 6.3.010).



NOTES:

1. VALVE IS LOCATED OUTSIDE AUXILIARY BUILDING
2. RELIEF VALVE MUST BE LOCATED AT LEAST 6" ABOVE THE TOP OF THE LOOP SEAL. BOTTOM OF THE LOOP MUST EXTEND AT LEAST 6" BELOW THE CONNECTION TO THE MAIN PIPE.
3. FOR COMPLETE INSTRUMENT NUMBERS, REFER TO INSTRUMENT INDEX B-175803.
4. LINE NUMBERS FOR VENT, DRAIN, AND INSTRUMENT LINES ON EQUIPMENT, UP TO AND INCLUDING THE ROOT VALVES: FOR HEAT EXCHANGERS AND HVAC COOLERS SAME AS SERVICE WATER OUTLET LINE NUMBERS. FOR PIPING DOWNSTREAM OF VENT AND DRAIN ROOT VALVES, SEE SPEC. SS-1109-2, PARA. 11.5.1(b). RELIEF VALVE DISCHARGE LINES SHALL BE IN ACCORDANCE WITH SPEC. SS-1109-2, PARA, 11.5.1(e).
5. COMPONENT COOLING WATER PUMP AREA FLOOD DETECTORS.
6. REMOVABLE FLANGED SPOOL PIECE. REQUIRED FOR SEISMIC 1 INTEGRITY OF PIPING STUB.
7. CCW HX FLOW CONTROL VALVE FV3009C IS LIMITED TO A NOMINAL 35° OPEN POSITION.
8. ALL VALVE AND LINE NUMBERS BEGIN WITH THE PREFIX OF Q1P16 OR N1P16.
9. VALVE Q1P16V217H IS PIPING CLASS HCC PER MDC 1040935001. THE PIPING NIPPLES WHICH CONNECT VALVE Q1P16V217H TO THE PROCESS ARE SCHEDULE 80 STAINLESS STEEL PER MDC 1040935001.
10. Q1P16V217A,D,E,&F; Q1P16V288B&D; Q1P16V289A,B,&C; AND Q1P16V216A ARE PIPING CLASS HCC PER MDC 1051657401. THE PIPING NIPPLES WHICH CONNECT THESE VALVES TO THE PROCESS ARE SCHEDULE 80 SS PER MDC 1051657401.
11. Q1P16V005A, B, C, D, E & F, ARE PIPING CLASS HCC PER MDC 1051657401. THE PIPING NIPPLES WHICH CONNECT THESE VALVES TO THE PROCESS ARE SCHEDULE 80 SS PER MDC 1051657401.
12. PIPE FLANGE ARE 300# FLANGES TO MATCH COOLER FLANGES SUPPLIED BY VENDOR.
13. CCW HX FLOW CONTROL VALVES FV3009A AND B ARE LIMITED TO A NOMINAL 45° OPEN POSITION.

THE UNIT 2 COUNTERPART FOR THIS P&ID IS D-205003, SH. 1

CAD D1750031
ACAD2K | KB - 06

VER. NO. 55.0 DATE 06/25/12
REVISED PER ABN 1092714601M002, VER. 1.0.

DRAWN ____ CHECKED ____ DESIGNED ____
APPROVED FOR SIGNATURES DATE ____
APPROVED SEE MICROFILM DATE ____
APPROVED COPY DATE ____

Southern Company Services, Inc. for
ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT – UNIT NO.1
SUBJECT P&ID – SERVICE WATER SYSTEM
SCALE NONE B/M
SHEET 1 OF – SHEETS
SUPERSEDES

D-175003

DRAWING CATEGORY: CRITICAL

4.5 Refueling Water Surface Ventilation Operation

NOTE

The refueling water surface exhaust may be temporarily routed to a filtration system such as the containment pre-access filtration unit and/or portable filtration unit for ALARA purposes during refueling operations. However, the exhaust flow shall NOT be directed to the containment purge intake. (DCP 97-1-9220)

4.5.1 **Place** refueling water surface ventilation in service as follows:

4.5.1.1 **Start** the refueling water surface exhaust fan 2. ☐

4.5.1.2 **Start** the refueling water surface supply fan 1. ☐

4.5.2 **Remove** the refueling water surface ventilation from service as follows:

4.5.2.1 **Stop** refueling water surface supply fan 1. ☐

4.5.2.2 **Stop** refueling water surface exhaust fan 2. ☐

4.6 Isolating and Restoring Service Water to Individual CTMT Coolers

NOTES

It is permissible to use both the inlet and outlet valves to isolate individual CTMT coolers. Overpressure protection for the CTMT Coolers are provided by relief valves located inside CTMT. See CR 2009114121

Movats testing of CTMT CLR SW MOV's can be performed utilizing this section of the procedure AFTER the desired CTMT cooler has been isolated. This includes CTMT CLR MOV's 3019A,B,C,D and MOVs 3023A,B,C,D.

4.6.1 For the affected CTMT cooler, **close** SW TO CTMT CLR valve:

4.6.1.1 SW TO 1A CTMT CLR Q1P16MOV3019A ☐

4.6.1.2 SW TO 1BCTMT CLR Q1P16MOV3019B ☐

4.6.1.3 SW TO 1C CTMT CLR Q1P16MOV3019C ☐

4.6.1.4 SW TO 1D CTMT CLR Q1P16MOV3019D ☐

4.6.2 IF desired, THEN **close** SW FROM CTMT CLR on the affected cooler:

4.6.2.1 SW FROM 1A CTMT CLR Q1P16MOV3441A ☐

4.6.2.2 SW FROM 1B CTMT CLR Q1P16MOV3441B ☐

4.6.2.3 SW FROM 1C CTMT CLR Q1P16MOV3441C ☐

4.6.2.4 SW FROM 1D CTMT CLR Q1P16MOV3441D ☐

4.6.3 IF desired, THEN **close**, or **verify closed**, EMERG SW FROM CTMT CLR on the affected cooler.

4.6.3.1 EMERG SW FROM 1A CTMT CLR Q1P16MOV3024A ☐

4.6.3.2 EMERG SW FROM 1B CTMT CLR Q1P16MOV3024B ☐

4.6.3.3 EMERG SW FROM 1C CTMT CLR Q1P16MOV3024C ☐

4.6.3.4 EMERG SW FROM 1D CTMT CLR Q1P16MOV3024D ☐

4.6.4 IF desired, THEN **stop** the affected CTMT cooler

4.6.4.1 1A containment cooler ☐

4.6.4.2 1B containment cooler ☐

4.6.4.3 1C containment cooler ☐

4.6.4.4 1D containment cooler ☐

4.6.5 IF stopped in the previous step, THEN **check** CTMT CLR DISCH CLOSED light illuminated:

4.6.5.1 CTMT CLR 1A DISCH 3186A ☐

4.6.5.2 CTMT CLR 1B DISCH 3186B ☐

4.6.5.3 CTMT CLR 1C DISCH 3186C ☐

4.6.5.4 CTMT CLR 1D DISCH 3186D ☐

4.6.6 IF required, THEN **verify** that an unaffected CTMT cooler in that train is selected for auto-start:

4.6.6.1 1A containment cooler ☐

4.6.6.2 1B containment cooler ☐

4.6.6.3 1C containment cooler ☐

4.6.6.4 1D containment cooler ☐

4.6.7 **Refer** to the following for operability requirements:

4.6.7.1 Technical Specifications SR 3.6.6.2 and SR 3.6.6.3 ☐

4.6.7.2 FNP-1-STP-17.0, Containment Cooling System Train A(B) Operability Test ☐

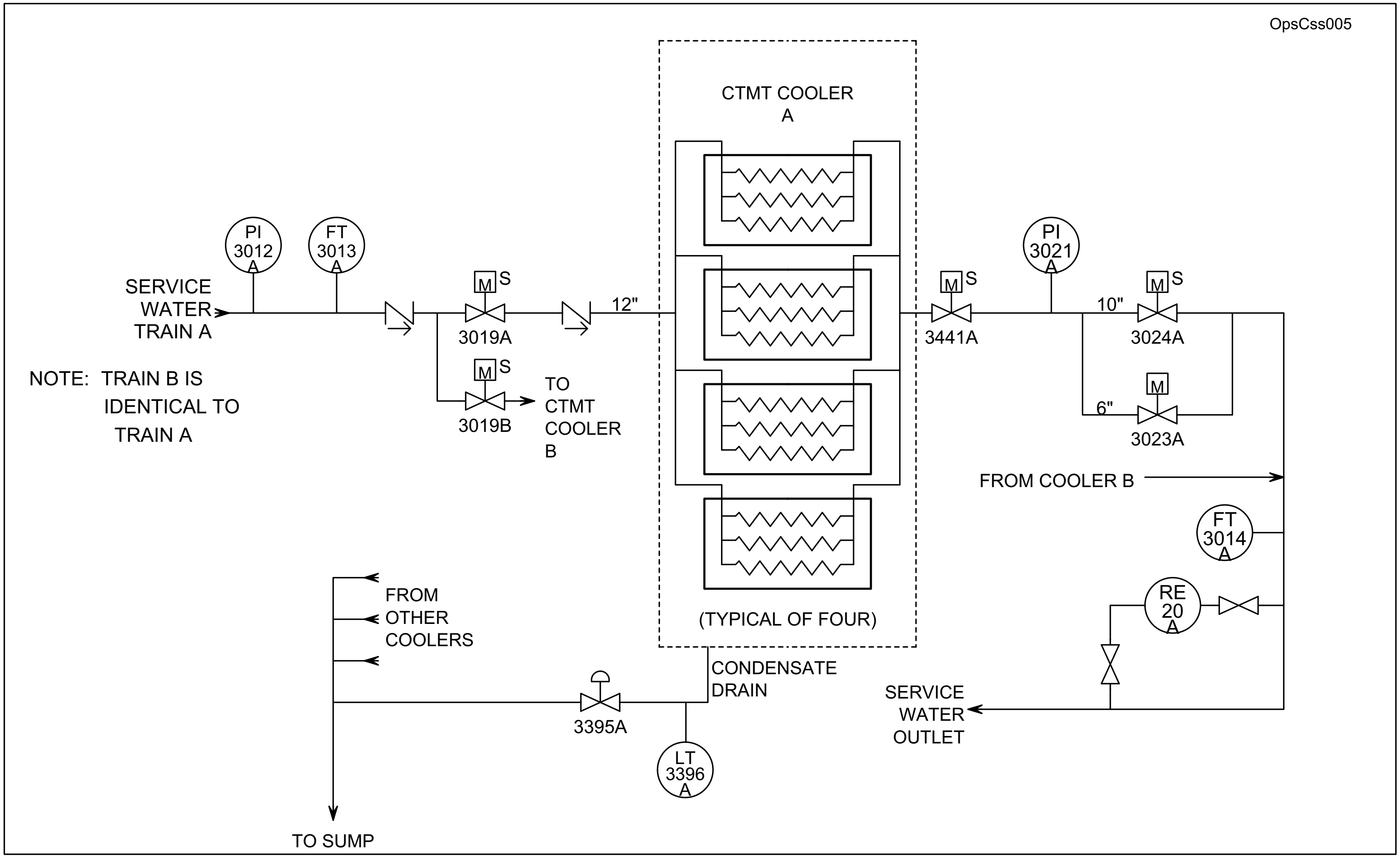


FIGURE 4 - Service Water to Containment Coolers

BACKGROUND

Containment Spray System (continued)

from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Containment Spray System is actuated either automatically by a containment High-3 pressure signal or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low level alarm for the RWST signals the operator to manually align the system to the recirculation mode. The Containment Spray System in the recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Cooling System

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train consists of two fan units supplied with cooling water from a separate train of service water (SW). However, under post-accident conditions, a single fan unit with at least 600 gpm SW flow provides sufficient cooling capacity to meet post accident heat removal requirements. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, and outside the secondary shield in the lower areas of containment.

During normal operation, up to four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. The Containment Cooling System is designed to limit the

(continued)

1. 077AK3.02 056/BANK/FNP EXAM BANK/C/A 3.6/3.9/APE077AK3.02/N///

Unit 1 is at 100% power with the following conditions:

- 1F and 1G 4160V bus voltages are 4220 volts.
- A Power Control Center (PCC) notification has NOT been issued.
- The crew has entered AOP-5.2, Degraded Grid.

Per AOP-5.2, which one of the following action(s) is required by the crew and the reason?

A. Place the Capacitor Bank on service;

to lower grid voltage.

B.✓ Contact Alabama Control Center (ACC);

to obtain voltage control strategies.

C. Reduce bus loads such as securing RW and SW pumps;

to assist in voltage control.

D. Place the unit in Mode 3 within 6 hours using UOP-3.1, Power Operation;

to meet Technical Specification requirements.

AOP-5.2
3. [CA] **Check Status of 4160 V Buses 1F And 1G:**

3.1 Check 4160 V Buses 1F and 1G voltages - LESS THAN 4200V.	3.1 Perform the following: 3.1.1 Consider starting additional bus loads, such as RW Pumps, to assist in voltage control. 3.1.2 Contact Alabama Control Center for voltage control strategies. 3.1.3 [...] 3.1.4 IF a PCC notification has been issued, THEN proceed to step 4. 3.1.5 [...] 3.1.6 Return to step 2

<u>**Distracter analysis**</u>

A. Incorrect.	See C. Capacitor Bank RAISES voltage. Plausible if the applicant confuses the Capacitor Bank with the Shunt Reactor which lowers the bus voltage.
B. Correct.	Correct per step 3.1.2 RNO of AOP-5.2.
C. Incorrect.	See C. Plausible if the applicant doesn't understand the relationship of the plant loads to grid voltage. The procedure has the operator consider ADDING more load not reducing.
D. Incorrect.	See C. Plausible since this is may be required if grid voltage is low and the requirements of TS 3.3.5 are not met. The applicant could confuse the actions of low grid voltage with high grid voltage.

K/A: **077AK3.02**	Generator Voltage and Electric Grid Disturbances - Knowledge of the **reasons** for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: **Actions contained in abnormal operating procedure for voltage and grid disturbances.**
Importance Rating:	3.6 3.9
Technical Reference:	FNP-1-AOP-5.2, Degraded Grid, Ver 15
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing [...] and AOP-5.2, Degraded Grid. (OPS-52521N06)
Question History:	FNP EXAM BANK (AOP-5.1/.2-52521N02)
K/A match:	Applicant is required to know that **when voltage exceeds 4200V, AOP-5.2 requires the operating crew to Contact Alabama Control Center and the reason is to obtain voltage control strategies.**
SRO justification:	N/A

4.8.2 Guidelines to *Raise* and *Lower* System Voltage. The Farley Operators SHALL adjust the main generator reactive load voltage to meet the system requirements as directed by the System Operator, while observing the following guidelines.

4.8.2.1 The reactive load adjustments CANNOT *exceed* 22kV +/- 5% (20.9-23.1kV).

4.8.2.2 The Farley administrative limit is -300 MVARs to prevent the auto adjuster from going to its mechanical stop.

4.8.2.3 The 230kV Shunt Reactor is placed *in* service when the 230kV bus voltage needs to be *lowered*, and the 230kV Capacitor Bank is placed *in* service when the 230kV bus voltage needs to be *raised*. Because the two devices perform opposite functions, they never should be *in* service at the same time, and an interlock scheme is provided on switches 955 and 957 to prevent this from happening.

4.8.2.4 The Capacitor Bank will automatically switch on when bus voltage drops to 234kV (5 second time delay), and automatically switch OFF when bus voltage *exceeds* 240kV (20 second time delay).

4.8.2.5 The ACC will switch the Shunt Reactor *in* service when system load is ≤ 16,000 MW, and it is expected to remain at that load for ≥4 hours.

4.8.2.6 When the system load goes *above* 16,000 MW, and is *increasing*, then the ACC will switch the Shunt Reactor *out* of service.

4.8.2.7 **Placing** the Capacitor Bank in service *raises* the bus voltage ~ 2kV.

4.8.2.8 **Placing** the Shunt Reactor in service *lowers* the bus voltage ~ 3kV.

Step	Action/Expected Response	Response Not Obtained
__ 2	**[CA] Ensure abnormal condition(s) logged in Control Room Log.** • Any abnormal grid condition(s) • Any abnormal 4160V Buses 1F and 1G voltage and/or current • Any PCC notification type	
__ 3	**[CA] Check Status of 4160 V Buses 1F And 1G:**	
3.1	Check 4160 V Buses 1F and 1G voltages - LESS THAN 4200V.	3.1 Perform the following: 3.1.1 Consider starting additional bus loads, such as RW Pumps, to assist in voltage control. 3.1.2 Contact Alabama Control Center for voltage control strategies. 3.1.3 Record 1F and 1G 4160 V bus voltages every four hours on TABLE 1, 1F and 1G 4160 V BUS VOLTAGE LOG. 3.1.4 IF a PCC notification has been issued, THEN proceed to step 4. 3.1.5 WHEN both 4160 V Busses 1F and 1G voltage are less than 4200 V, THEN go to procedure and step in effect. 3.1.6 Return to step 2.

Step 3 continued on next page

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Step	Action/Expected Response	Response Not Obtained
3.2	Check voltages - NOT DEGRADED • 4160 V Buses 1F and 1G voltages - GREATER THAN 3850 V • 4KV BUS OV-OR-UV OR LOSS OF DC annunciators - CLEAR • WE2 for bus 1F AND • VE2 for bus 1G	3.2 Perform the following: 3.2.1 Record 1F and 1G 4160 V bus voltage indication every half hour on TABLE 1, 1F and 1G 4160 V BUS VOLTAGE LOG. 3.2.2 Refer to TS 3.3.5.

NOTE: WHEN transitioning to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION AND at the Shift Supervisors direction, it is ACCEPTABLE for one team member to complete the IOA's of FNP-1-EEP-0, while the other team member verifies the reactor trip, THEN continues with step 3.3 RNO before finishing the Immediate Operator Actions of FNP-1-EEP-0.

Step	Action/Expected Response	Response Not Obtained
3.3	Check NO Loss of Single Phase	3.3 Perform the following: 3.3.1 Trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION 3.3.2 IF 1F 4160V Bus is affected, THEN verify the following 3.3.2.1 BKR DF01 open 3.3.2.2 BKR DF15 open 3.3.3 IF 1G 4160V Bus is affected, THEN verify the following 3.3.3.1 BKR DG01 open 3.3.3.2 BKR DG15 open 3.3.4 De-energize the affected S/U XFMR per FNP-1-SOP-36.1, STARTUP, UNIT AUXILIARY, AND MAIN TRANSFORMERS PREPARATION FOR OPERATION 3.3.5 Go to FNP-1-AOP-5.0, LOSS OF A OR B TRAIN ELECTRICAL POWER

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Step	Action/Expected Response	Response Not Obtained
__ 9	**Contact Alabama Control Center For Voltage Control Strategies**	

NOTE: Step 10 is a continuing action step.

Step	Action/Expected Response	Response Not Obtained
__ 10	**[CA] Check Status of 4160 V Buses 1F And 1G:**	
	10.1 4160 V Buses 1F and 1G voltages - LESS THAN 4200 V	10.1 Return to step 2.
	10.2 4160 V Buses 1F and 1G voltages - LESS THAN 3850 V	10.2 Perform the following: 10.2.1 IF no PCC notification exists, THEN go to procedure and step in effect. 10.2.2 Return to step 2.
__ 11	**Evaluate need for plant shutdown (refer to TS 3.3.5, Action F.1):**	
	11.1 Check reactor - MODE 1 OR 2	11.1 Proceed to Step 12.
	11.2 Check duration of degraded voltage - GREATER THAN ONE HOUR	11.2 Return to Step 2.
	11.3 Initiate actions to place Reactor in MODE 3 within the next 6 Hours using: • FNP-1-UOP-3.1, POWER OPERATION • FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY	

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. ----------NOTE----------- Only applicable to Function 3. ----------------------------- One Alarm Function channel inoperable on one or more trains.	D.1	Verify voltage on associated bus is ≥ 3850 volts.	Once per 4 hours
E. Required Action and associated Completion Time of Condition D not met.	E.1	Restore bus voltage to ≥ 3850 volts.	1 hour
F. Required Action and associated Completion Time of Condition E not met.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	---------------------------NOTES-------------------------------- 1. TADOT shall exclude actuation of the final trip actuation relay for LOP Functions 1 and 2. 2. Setpoint verification not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program

QUESTIONS REPORT
for ILT 36 RO NRC Exam Version 6

1. 078K1.01 057/BANK/FNP 04/MEM 2.8*/2.7*/078K1.01/N///

The following conditions exist on Unit 1:

- A rupture in the Instrument Air system has occurred.
- Instrument Air header pressure is 65 psig and lowering slowly.

Which one of the following completes the statements below?

V-902, AIR DRYER AUTO BYP, will be __(1)__ .

V-904, NON-ESSENTIAL IA HDR AUTO ISO, will be __(2)__ .

	(1)	(2)
A.✓	OPEN	OPEN
B.	OPEN	CLOSED
C.	CLOSED	OPEN
D.	CLOSED	CLOSED

KD2
AUTOMATIC ACTION

3. Pressure downstream of inst air dryers, bypasses dryers (V902) at 70 psig.
4. Pressure downstream of inst air dryers, isolates inst air to service bldg (V904) at 55 psig.

Distracter analysis

A. Correct. First part is correct. V-902 opens at 70 psig.

Second part is correct. V-904 closes at 55 psig.

B. Incorrect First part is correct (See A.1).

Second part is incorrect (See A.2). Plausible since there are numerous setpoints for alarms and automatic valve repositionings in the air system and they are easily confused.

C. Incorrect. First part is incorrect (See A.1). Plausible since there are numerous setpoints for alarms and automatic valve repositionings in the air system and they are easily confused.

Second part is correct (See A.2)

D. Incorrect. First part is incorrect (See C.1).

Second part is incorrect (See B.2).

K/A: **078K1.01**	Instrument Air System - Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Sensor air
Importance Rating:	2.8* 2.7*
Technical Reference:	FNP-1-ARP-1.10, KD2, IA PRESS LO Ver 70.2
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Compressed Air System components and equipment, to include the following (OPS-40204D07): [...] Automatic actuation including setpoints for selective isolation on decreasing header pressure. [...]
Question History:	FNP 04
K/A match:	Requires the applicant to **know the cause and effect relationship between the sensed air header pressure and the automatic operation of isolation valve V-904 and bypass valve 902.**
SRO justification:	N/A

LOCATION KD2

SETPOINT: 75 ± 2 PSIG

ORIGIN: Pressure Switch N1P19PS510-N

D2
IA PRESS LO

PROBABLE CAUSE

1. Air Compressor tripped.
2. Improper valve lineup.
3. Instrument Air Line rupture.
4. Air Dryer malfunction.

AUTOMATIC ACTION

NOTE: IF instrument air pressure falls below 85 psig, THEN LCV-459 and 460 may partially close causing orifice isolation valves 8149A, B, and C to isolate letdown. ☐

1. If air header pressure decreases to 80 psig, Service Air will isolate (V-901)
   a. If Service Air pressure drops to 75 psig and was aligned to containment, Service Air would isolate, (HV-2935B and HV-2935C) close.
2. 1A, 1B and 1C air compressors will start as demanded by the sequence controller.
3. Pressure downstream of inst air dryers, bypasses dryers (V902) at 70 psig.
4. Pressure downstream of inst air dryers, isolates inst air to service bldg (V904) at 55 psig.
5. Pressure downstream of inst air dryers, isolates inst air to turbine bldg (V903) at 45 psig.

OPERATOR ACTION

1. Check indications and determine actual instrument air pressure. ☐
2. Start an additional air compressor, if available, to maintain system pressure. ☐
3. IF a loss of instrument air has occurred, THEN perform the actions required by FNP-1-AOP-6.0, LOSS OF INSTRUMENT AIR. ☐
4. IF an Air System rupture is indicated, THEN notify Plant Personnel to locate and isolate the ruptured piping. ☐
5. IF an improper valve lineup exists, THEN investigate and correct the valve lineup. ☐
6. IF malfunction of air compressor sequencer indicated, THEN operate the air compressors in accordance with FNP-1-SOP-31.0, COMPRESSED AIR SYSTEM. Station appropriate personnel at the air compressors to monitor air header pressure. ☐

1. 079G2.4.34 058/MOD/FNP 05/C/A 4.2/4.1/079G2.4.34/N///

The following plant conditions exist on Unit 1:

- The Reactor has been tripped due to loss of Instrument Air.
- The operating crew is performing the actions of ESP-0.1, Reactor Trip Response.
- SG Atmospheric Relief Valves (ARVs) are aligned per SOP-62.0, Emergency Air System.

Subsequently, the operator applies 18 psig to the valve actuator for PCV-3371A, 1A MS ATMOS REL VLV.

Which one of the following completes the statements below?

PCV-3371A is __(1)__ open.

If PCV-3371A were fully open, __(2)__.

A. 1) IS

2) a High Steam Flow - Lo Lo Tavg Main Steam Isolation may occur

B. 1) IS

2) Technical Specification cooldown limits may be exceeded

C. 1) is NOT

2) a High Steam Flow - Lo Lo Tavg Main Steam Isolation may occur

D.✓ 1) is NOT

2) Technical Specification cooldown limits may be exceeded

This question is not a true 2 + 2 to improve plausibility of distracters.

SOP-62.0
Caution after step 4:
Atmospheric relief valves will start to open at 24 ± 2 psig and will be full open at 45 psig. IF the atmospheric relief is full open, THEN Tech Spec cooldown limits may be exceeded.

<u>Distracter analysis</u>

A. Incorrect. First part is incorrect (See D.1). Plausible if the applicant is not familiar with the actuation pressure of the ARV's.

Second part is incorrect (See D.2). Plausible since a caution exists in numerous procedures that excessive opening of the **STEAM DUMPS** will cause this isolation of the MSIV's. Hi Steam Flow Lo Lo Tavg is 1 of 2 flow instruments on 2 of 3 steam lines.

This isolation closes the MSIV's and not the ARV's. The applicant could think that this isolation in fact does close the ARV's to prevent exceeding a technical specification cooldown.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

D. Correct. First part is correct. The ARV will not open until at least 22 psig of air is applied.

Second part is correct. Per the caution of SOP-62, a fully open ARV may cause tech spec limits to be exceeded.

K/A: **079G2.4.34**	Station Air System - Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.
Importance Rating:	4.2 4.1
Technical Reference:	FNP-1-SOP-62.0, Emergency Air System, Ver 23
References provided:	None
Learning Objective:	DEFINE AND EVALUATE the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Main and Reheat Steam System components and equipment, to include the following (OPS-40201A07): […] • Abnormal and Emergency Control Methods • Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level) • Protective isolations such as high flow, low pressure, low level including setpoint […]
Question History:	MOD FNP 05
K/A match:	Requires the applicant to know the **operational affects of local operator actions** to control the SG ARV's. The operator must use the Emergency Air system locally to control ARV position due to the loss of the Station Air system.
SRO justification:	N/A

SECTION 1.5

STEAM GENERATOR STEAM FLOW/FEED FLOW/STEAM PRESSURE INSTRUMENTATION

AUTOMATIC ACTIONS:

- Main Steam Line isolation on two steam lines(>40% steam flow on 1/2 flow instruments on 2/3 steam lines) coincident with low low Tavg less than 543°F on 2/3.
- MSIV Closure on MSLIA--LO SG PRESSURE <585 psig on 2/3 PT's. (PT 474, 485, 496)
- Safety Injection--steam pressure <585 psig on 2/3 PT's. (PT 474,485,496)
- Safety Injection-1/3 steam lines 100 psig < than the other steam lines on 2/3 protection sets.

Step	Action/Expected Response	Response NOT Obtained

9.1.8 IF main steam lines isolated AND gland steam supplied from Unit 1, THEN break condenser vacuum.

COND VAC BKR
MAN ISO (155 ft, TURB BLDG)
[] N1N51V518A open
[] N1N51V518B open

COND VAC BKR
VLVS
[] N1N51V519A/519B open

9.2 IF RCS temperature greater than 547°F and rising, THEN perform the following.

NOTE: Excessive opening of steam dumps can cause a high steam flow LO-LO TAVG main steam line isolation signal.

9.2.1 IF condenser available, THEN dump steam to condenser at faster rate.

Step 9 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
		1.2 IF RCS temperature greater than 547°F and rising, THEN perform the following.

NOTE: Excessive opening of steam dumps can cause a high steam flow LO-LO TAVG main steam line isolation signal.

1.2.1 IF condenser available, THEN dump steam to condenser at faster rate.

1.2.2 IF heatup continues, THEN dump steam to atmosphere.

1.2.2.1 Direct counting room to perform FNP-0-CCP-645, MAIN STEAM ABNORMAL ENVIRONMENTAL RELEASE.

1.2.2.2 IF normal air available, THEN control atmospheric relief valves to dump steam, IF NOT, dump steam using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

1A(1B,1C) MS ATMOS REL VLV
- [] PC 3371A adjusted
- [] PC 3371B adjusted
- [] PC 3371C adjusted

NOTE: If possible, the RCP left running should accommodate pressurizer spray capability. If available, RCP 1B (preferred) OR both RCP 1A AND 1C should be left running.

1.3 IF dumping of steam is not effective at controlling RCS temperature, THEN minimize running RCP(s).

4.0 Instructions

CAUTION

Atmospheric relief valves will start to open at 24 ± 2 psig and will be full open at 45 psig. If the atmospheric relief is full open, then Tech Spec cooldown limits may be exceeded.

NOTE

The compressors are rated at 41.6 Ft^3/min at 100 psig.

4.1 Placing the Emergency Air Compressors in Operation

4.1.1 **Place** emergency air compressor 1A hand switch in ON. ☐

4.1.2 **Place** emergency air compressor 1B hand switch in ON. ☐

4.1.3 With power available to the atmospheric relief valve, MCB controllers, **proceed** to Step 4.3.

- 1A S/G atmospheric relief valve controller PC 3371A ☐
- 1B S/G atmospheric relief valve controller PC 3371B ☐
- 1C S/G atmospheric relief valve controller PC 3371C ☐

4.1.4 With no power available to the atmospheric relief valve, MCB controllers, **proceed** to Step 4.4. ☐

4.2 Removing the Emergency Air Compressors from Operation

4.2.1 **Place** emergency air compressor 1A hand switch in OFF. ☐

4.2.2 **Place** emergency air compressor 1B hand switch in OFF. ☐

1. MN&RHT STM-40201A07 027/HLT//C/A 3.1/3.5/035K6.02////

The following plant conditions exist on Unit 2:

- The reactor has been tripped due to loss of instrument air 45 minutes ago.
- No station air compressors are running.
- The MSIVs have been closed.
- 2A emergency air compressor has been started and is running.
- Tavg is stable at approximately 555°F.
- SG Pressures are cycling on the lowest set pressure SG code safeties.
- SG Atmospheric Relief Valves (ARVs) are being lined up locally in the lower equipment room to control SG pressure.

The air regulator for the 2A ARV fails and 50 psig is inadvertently admitted to the valve actuator.

Which one of the following completes the statements below?

2A SG ARV will _________ , the 2A SG pressure will ______________

ARV response	SG pressure/RCS temperature response
A.✔ fully open	lower rapidly and a large RCS cooldown will commence.
B. partially open	lower slowly and a small RCS cooldown will commence.
C. remain closed	remain constant and RCS temperature will remain at the current value.
D. remain closed	lower slowly and RCS temperature will slowly decrease.

1. 103K4.06 059/MOD/VOGTLE 02/MEM 3.1/3.7/103K4.06/N///

Unit 1 has experienced a large break LOCA and the following conditions exist:

- PT-953, CTMT PRESS is 28 psig and rising.

Which one of the following completes the statement below?

A MINIMUM of _(1)_ OR _(2)_ PHASE B CTMT ISO CS ACTUATION handswitch(es) is(are) required to actuate a Phase B isolation.

A. 1) 1 additional Containment pressure channel reaching 16.2 psig

2) TWO

B. 1) 2 additional Containment pressure channels reaching 16.2 psig

2) ONE

C✔ 1) 1 additional Containment pressure channel reaching 27 psig

2) TWO

D. 1) 2 additional Containment pressure channels reaching 27 psig

2) ONE

Not a true 2 + 2 for plausibility.

FSD-A181007 2.7.1
Phase B isolation is initiated by containment pressure High-3 (27 psig) on 2 of 4 b/s or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

Distracter analysis

A. Incorrect. First part is incorrect (See C.1). Plausible since this is the correct number of channels but the wrong setpoint. 16.2 psig is HI-2 main steam line isolation and NOT HI-3 Phase B isolation.

Second part is correct (SeeC.2).

B. Incorrect. First part is incorrect (See C.1). Plausible since there are 4 bistables and the applicant could reason that 3 of 4 are required to actuate Phase B. Additionally, the setpoint is incorrect as discussed in A.1.

Second part is incorrect (See C.2). Plausible since Phase A and SI only require 1 handswitch to actuate. The applicant may confuse them.

C. Correct. First part is correct. Per the FSD, High-3 Containment Isolation Phase B coincidence is 2 of 4 bistables.

Second part is correct. Per the FSD, 2 handswitches are require to actuate Phase B Containment Isolation.

D. Incorrect. First part is incorrect (See C.1). Plausible since there are 4 bistables and the applicant could reason that 3 of 4 are required to actuate Phase B. Permissives such as P-8 and P-9 require 2 of 4 to enable and 3 of 4 to disable. There are many coincidences in the reactor protection system and they are easily confused.

Second part incorrect (See B.2)

K/A: **103K4.06**	Containment System - Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Containment isolation system
Importance Rating:	3.1 3.7
Technical Reference:	FSD-A181007, Reactor Protection System, Ver 18
References provided:	NONE
Learning Objective:	SELECT AND ASSESS the following instrument/equipment response expected when performing Containment Structure and Isolation System evolutions including the fail condition, alarms, and trip setpoints (OPS-52102A05): • PT-950, PT-953
Question History:	VOGTLE 02 - Changed to containment phase B isolation to meet K/A vs ctmt spray actuation.
K/A match:	Requires the applicant to have **knowledge of the design feature of the Phase B Containment Isolation System** in that 2 of 4 bistables or 2 of 2 handswitches are required for actuation.
SRO justification:	N/A

3. Ensure the integrity of the containment

The output of each of the initiation circuits consists of a master relay which drives slave relays for contact multiplication as required. The logic, master, and slave relays shall be mounted in the solid state logic protection cabinets designated Train A, and Train B respectively, for the redundant counterparts. The master and slave relay circuits operate various pump and fan circuit breakers or starters, motor operated valve contactors, solenoid operated valves and emergency generator starting circuits. Table T-8 lists the various interfaces associated with the master and slave relays.

The ESF functions are accomplished by some or all of the following: starting and loading emergency diesel generators injecting borated water, isolating containment, initiating auxiliary feedwater, steam line isolation, feedwater isolation, reactor trip, turbine trip, containment spray actuation. (References 6.1.014, 6.7.003)

1. Safety Injection

Safety Injection (SI) provides two primary functions;

1. Primary side mass addition to ensure maintenance or recovery of reactor vessel water level (cover the active fuel for heat removal and clad integrity, Peak Clad Temperature < 2200^{o}F)

2. Boration to ensure recovery and maintenance of shutdown margin (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks both inside and outside containment.

The following signals are used for initiation of safety injection:

	SI Signals	Set Point & Coincidence
a.	Low steam line pressure	≤ 585 psig, 1/1 on 2/3 steam lines, rate compensated (If not manually blocked below P-12)
b.	High steam line differential pressure	100 psid, (one steam line 100 psig less than other two) on 2/3 protection set
c.	Low Pressurizer pressure	≤ 1850 psig on 2/3 (If not manually blocked below P-11)

d.	High-1 containment pressure	≥ 4 psig on 2/3
e.	Manual	1/2 Manual Switches

Safety Injection is the most complex protection function. When either train senses a condition requiring safety injection, the following actions shall occur:

a. The reactor is automatically tripped

b. The turbine is automatically tripped

c. The diesel generators are started

d. The motor-driven auxiliary feedwater pumps are started

e. The main feedwater is isolated

f. The containment is isolated (Phase A and containment ventilation)

g. The emergency mode of control room ventilation is initiated

h. The required fluid system valves are repositioned for safety injection.

i. The safeguards sequencers are energized.

The safeguards sequencers are timing devices that selectively start the large current drawing ESF loads.

(References 6.1.022, 6.4.007, 6.4.015, 6.4.082, 6.7.012)

2. Containment Isolation

The containment isolation function limits the release of fission products to the environment in the event of a large break LOCA. Containment isolation is provided by three isolation categories; Phase A, Phase B and Containment Ventilation Isolation. Phase A isolation is initiated by completion of the Safety Injection logic, or by manual actuation. Each function that initiates SI will also result in the isolation of all non essential process containment penetrations. Phase B isolates the remaining process lines, except

for the required engineered safety features lines. Phase B isolation is initiated by containment pressure High-3 (27 psig) or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

The Containment Ventilation Isolation isolates the containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on the completion of the SI logic, high radioactivity levels in the purge exhaust, or by manual initiation of either Phase A Containment Isolation or Phase B Isolation/Containment Spray Actuation. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.080)

3. Main Steam Line Isolation

Isolation of the Main Steam lines limits the effects of an uncontrolled release of steam either inside or outside the containment. For a break upstream of the isolation valves (MSIV) in the steamlines, valves closure will limit the release to the blowdown of the one affected steam generator. A break downstream of the valves is limited to the depressurization of the pipe volume downstream of the valves. This results in a rapid termination of the event and significantly reduces the mass lost from the secondary.

The Main Steam Line Isolation is initiated by the following:

a. High steam line flow with low-low T_{avg} , 1/2 steam flow channels above setpoint (40% of full steam flow between 0-20% load and increasing linearly to 110% at full load) on 2/3 steam lines with $T_{avg} \leq$ P-12

b. Low steam pressure; ≤ 585 psig on 2/3 S.G.

c. High-2 containment pressure; ≥ 16.2 psig on 2/3

d. Manual. By closing each MSIV by operating individual hand switches. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main Feed line Isolation:

for the required engineered safety features lines. Phase B isolation is initiated by containment pressure High-3 (27 psig) or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

The Containment Ventilation Isolation isolates the containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on the completion of the SI logic, high radioactivity levels in the purge exhaust, or by manual initiation of either Phase A Containment Isolation or Phase B Isolation/Containment Spray Actuation. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.080)

3. Main Steam Line Isolation

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The Main Steam Line Isolation is initiated by the following:

a. High steam line flow with low-low T_{avg} , 1/2 steam flow channels above setpoint (40% of full steam flow between 0-20% load and increasing linearly to 110% at full load) on 2/3 steam lines with $T_{avg} \leq$ P-12

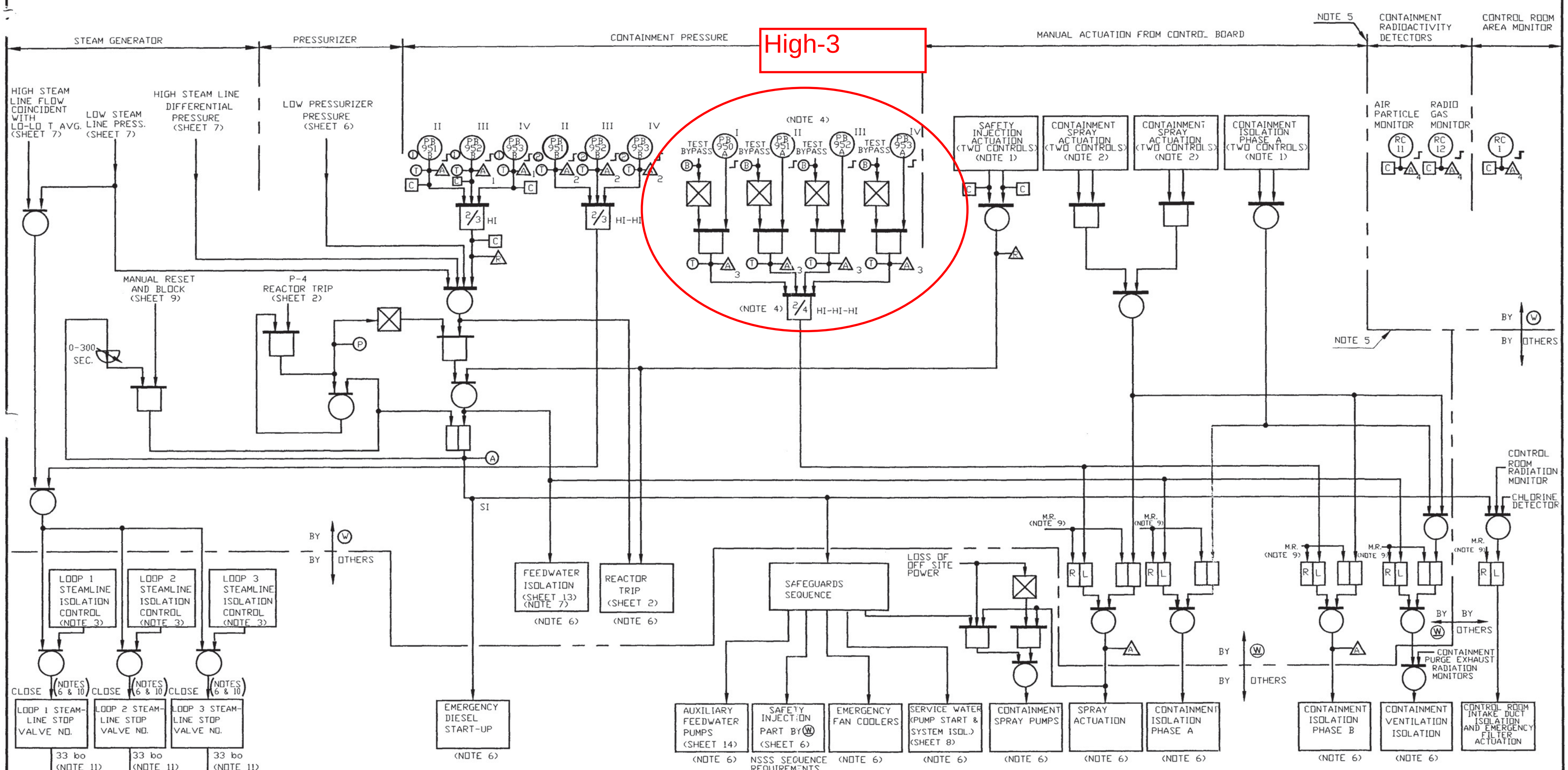
b. Low steam pressure; $\leq$ 585 psig on 2/3 S.G.

c. High-2 containment pressure; $\geq$ 16.2 psig on 2/3

d. Manual. By closing each MSIV by operating individual hand switches. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main Feed line Isolation:



NOTES:

1. TWO MOMENTARY CONTROLS: OPERATING EITHER CONTROL WILL ACTUATE.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE MOMENTARY CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE TO ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. ENCLOSED CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM AND IS NOT REDUNDANT.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. FEEDWATER ISOLATION INCLUDES THE TRIPPING OF ALL MAIN FEEDWATER PUMPS.
8. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
9. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
10. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
11. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAM LINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.

SAFEGUARDS ACTUATION SIGNALS

FIGURE F-2 SHEET 8

RPSFSD08.DWG

PERMISSIVES

	Permissive	Source	Setpoint	Coincidence & Light Status	Function
3.	P-7 Low Power Rx Trip Block Permissive	P-10 P-13	10% Rx Power 10% Turbine Power	Lit < setpoint permission to trip RCPs, etc. (2/4 > setpoint) or (1/2 > setpoint)	Prevents unnecessary reactor trips at low power by auto blocking following Rx trips below setpoint. 1. Low Rx Coolant Flow 2. RCP U.V. 3. RCP Bus U.F. 4. PZR Low Pressure 5. PZR High Level Auto unblocks above Rx Trips above setpoint.
4.	P-8 Single Loop Loss of Flow Permissive	NIS 41, 42, 43, and 44	30% Rx Power	2/4 > setpoint Lit < setpoint Permission to stop 1 pump	Prevents a Rx trip from loss of flow <u>OR</u> RCP undervoltage in a single loop. Auto reinstates above Rx trips.
5.	P-9 Turbine Trip Permissive	NIS 41, 42, 43, and 44	35% Rx Power	2/4 > setpoint 3/4 < setpoint Lit < setpoint Permission to trip turbine	Auto reinstates a Rx trip from turbine trip. Prevents a Rx trip from a turbine trip.

QUESTIONS REPORT
for 103K4.06 VNP 02

1. 026A1.01 002/T2G2/T2G1//M (3.9/4.2)/N/VG02301/C/LMF

Given the following conditions on Unit 2:

- Following a LOCA, containment pressure is rising.
- Containment pressure has reached 20 psig on three channels and 22 psig on one channel.

Which one of the following are the MINIMUM actions that would result in containment spray actuation?

A. When 1 more containment pressure channel indicates 22 psig, or when 1 manual handswitch is actuated.

B.✔ When 1 more containment pressure channel indicates 22 psig, or when 2 manual handswitches are actuated.

C. When 2 more containment pressure channels indicate 22 psig, or when 1 manual handswitch is actuated.

D. When 2 more containment pressure channels indicate 22 psig, or when 2 manual handswitches are actuated.

1. G2.1.18 060/NEW//MEM 3.6/3.8/G2.1.18/N///

The electronic log is malfunctioning. The control room has shifted to manual logs and the following entries have been made:

1000 Q1E21V061A, HHSI to 1C RCS loop CL iso, as left position; $1^{3/16}$ turns OPEN.
1012 Started 1B CCW Pump.

At 1030:

- The OATC recognizes that an error was made on the 1000 log entry.
- Q1E21V061A should have been logged as throttled to $3^{1/16}$ turns OPEN.

Per SOP-0.11, Watch Station Tours and Operator Logs, the OATC is required to correct the 1000 log entry by which one of the following methods?

A. • Circle the incorrect entry in red.

• Enter the correct information next to the incorrect information and record the date and initial.

B. • Circle the incorrect entry in red.

• At 1030 make a log entry with the correct information and designate it as a LATE ENTRY.

C✓ • Draw a single line through the incorrect entry.

• Enter the correct information next to the incorrect information and record the date and initial.

D. • Draw a single line through the incorrect entry.

• At 1030 make a log entry with the correct information and designate it as a LATE ENTRY.

SOP- 0.11 Pg 13
IF an error is made when recording hand written entries, THEN a single line will be drawn through the incorrect entries AND the correct entries recorded. The person making the correction must initial AND date the change.

<u>Distracter analysis</u>

A. Incorrect. First part is incorrect (See C.1). Plausible because NMP-OS)007-001, Conduct of Operations Standards and Expectations, Step 6.13.2.3 requires out of specification reading in manual logs to be circled. The applicant could confuse these requirements.

Second part is correct (See C.2).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See C.2). Plausible since the correct data would be placed in the log. The SOP-0.11 uses a LATE ENTRY to add additional information to the log and NOT to correct errors.

C. Correct. First Part is correct. IF an error is made when recording hand written entries, THEN a single line will be drawn through the incorrect entries AND the correct entries recorded. The person making the correction must initial AND date the change.

Second part is correct. The person making the correction must initial AND date the change.

D. Incorrect. First Part is correct (See C.1).

Second part is incorrect (See B.2).

K/A: **G2.1.18**	Ability to make accurate, clear, and concise logs, records, status boards, and reports.
Importance Rating:	3.6 3.8
Technical Reference:	FNP-0-SOP-0.11, Watch Station Tours and Operator Logs, Ver 26.4 NMP-OS-007-001, Conduct of Operations Standards and Expectations, Ver 13
References provided:	None
Learning Objective:	Identify the required entries into the Plant Operator's Logbook and the position with overall responsibility for maintenance of the Reactor Operator's Logbook. (OPS52303O01)
Question History:	NEW
K/A match:	Applicant is required to have the **ability to correct log errors to ensure the operator logs are accurate, clear, and concise.**
SRO justification:	N/A

5.2 **Requirements Common To All Narrative Logs:**

Significant events (e.g., trips, ESF actuations) will be included in sufficient detail so that the event is basically described.

IF an instrument is removed from service to perform a TS, TRM, or ODCM test, the out of service time will be tracked in accordance with FNP-0-SOP-0.13.

The narrative log will be shown in chronological order. WHEN necessary to insert additional information into a log that has been archived, THEN the entry will be designated as a late entry AND be noted with the actual date/time of the event in the active log.

Narrative log entries must be kept current with clear, concise, and complete entries, using the appropriate log entry type when applicable.

The "NOTES" feature of the computerized log may be used to enter miscellaneous information for shift turnover or other purposes. This is not a part of the official log.

The Shift Clerk should maintain at least the previous seven days logs in the Shift Clerk's office for review by the operating crew should the electronic logs become unavailable. Then the logs shall be forwarded to Document Control for filing in accordance with FNP-0-AP-4, CONTROL OF PLANT DOCUMENTS AND RECORDS.

CAUTION: **Once the log is printed, no further entries are allowed until the log is archived. The printed log is the official document. Only the active log is printed.**

At least once per day, the Shift Supervisor will perform the following:

- Print the narrative log (including all sublogs)
- Archive the Control Room Log
- Review the printed log

In case of the unavailability of the electronic log, or as deemed necessary by the Shift Manager, the Control Room logs will be recorded on Figure 2 or in a logbook. For manual logs using Figure 2 in lieu of electronic logs, start a new page at the beginning of each shift.

Upon completion of the shift when using manual logs or upon the return to using electronic logs, the completed log sheets be maintained at that workstation for review by on-coming shifts for at least three days after the electronic logs are again being used or until loaded into Control Room Log. When entering the information from the manual logs into Control Room Log, an entry will be made stating that the logs are being entered after being recorded initially manually. This is required since the person entering the logs may not be the same person who recorded the information on Figure 2.

To ensure consistency, operating logs shall be formatted as follows:

- Each oncoming shift's entries start with the operating status, e.g., operating mode, tests in progress, or special condition.
- Shift entries shall be per Section 5.3 and 5.4 as applicable.

For any narrative logs using logbooks:

- Start a new page at the beginning of each shift. Remove the original of each page completed the previous shift from the logbook. After local review, the page copies are forwarded to the Service Building to file.
- Operator logbooks with completed duplicate pages of all used pages are kept in the Control Room or at their operating station until all pages in the book are used. The latest completed operator logbook may remain in the Control Room or at their operating station after completion, until the current log is completed.
- Completed logbooks that are no longer required in the Control Room or on operating stations as reference (usually, six weeks for reference) should be forwarded to Document Control for filing.

IF an error is made when recording hand written entries, THEN a single line will be drawn through the incorrect entries AND the correct entries recorded. The person making the correction must initial AND date the change. The date entered must be the date on which the correction is made (i.e., don't backdate the correction). IF necessary additional comments may be added to explain the difference between the two dates. The use of chemical correctants OR other alterations is NOT permitted.

2.66 WHEN 1A Charging Pump has operated at least 2 minutes at stable conditions, THEN **record** the following:

2.66.1 Discharge pressure at PI-151B (use Test gauge) ____________ psig

2.66.2 Suction pressure at PI-151A (use Test gauge) ______________ psig

2.67 **Calculate** the ΔP (Step 2.66.1 – Step 2.66.2). _____________ psid

2.68 **Record** the ΔP on the Pretest Data Sheet for 1A HHSI Pump in the Table space for Flow (Q) equal 450 GPM. _____

NOTES

Non Technical Specification vibration readings should be compared to the previous reading shown on the vibration instrument display. Significant difference from the previous reading may be indicative of a problem with the vibration probe or probe alignment.

Vibration data collected on the gearbox will readout in "G's" vs. in/s for use by the Vibration Group.

2.69 ***Measure** vibration on the motor, gearbox and pump bearing housings in the horizontal, vertical and axial planes as shown on Figure 1 of the STP. _____

2.70 ***Record** the Tech. Specification required measurements in the Vibration Data Sheet 2 of Attachment 4A. _____

2.71 **Have** I&C close the test flow instrument isolation valves and open the equalization valve on N1E21FE975, N1E21FE976, and N1E21FE977.(Indicate completion with check mark)

Test Flow Instrument	Flow Element		
	N1E21FE975	N1E21FE976	N1E21FE977
HP Iso vlv closed			
Equalization vlv open			
LP Iso vlv closed			

NOTE

The as left throttle valve positions will be the starting point for setting the HHSI TO RCS COLD LEGS flow path in the body of STP-40.7.

The HHSI TO RCS COLD LEGS flow path should be the first tested by STP-40.7 if this Attachment was performed.

2.72 Using the Throttle Valve Data Sheets provided to track number of turns for each valve **record** the as left data on the sheet and log in the Reactor Operators Log. _____

Attachment 4A
Throttle Valve Data Sheet

HHSI TO RCS COLD LEG Flow Path

THROTTLE VALVE	INITIAL POSITION (Turns open)	RECORD ALL ADJUSTMENT					AS LEFT POSITION
		6th	7th	8th	9th	10th	
HHSI to 1A RCS loop CL iso Q1E21V061C	N/A						

THROTTLE VALVE	INITIAL POSITION (Turns open)	RECORD ALL ADJUSTMENT					AS LEFT POSITION
		6th	7th	8th	9th	10th	
HHSI to 1B RCS loop CL iso Q1E21V061B	N/A						

THROTTLE VALVE	INITIAL POSITION (Turns open)	RECORD ALL ADJUSTMENT					AS LEFT POSITION
		6th	7th	8th	9th	10th	
HHSI to 1C RCS loop CL iso Q1E21V061A	N/A						

6.13 Logkeeping

6.13.1 Standard

Logs are maintained to accurately reflect plant status and preserve an account of operation. Narrative logs document the timeline of key activities. Logs are used for verification of proper operation and trending.

6.13.2 Expectations

6.13.2.1 General

Keeping an accurate log is an integral part of an operator's responsibility on shift. Logs are established for key shift positions in order to enhance turnover of information, aid in trending and troubleshooting potential problems, and improve plant monitoring.

Logs may be either hand-written or electronic. Electronic logs are preferred but handwritten logs may be kept as an alternative. Hand-written operations logs are recorded using dark ink.

Shift management reviews logs frequently in order to confirm appropriateness and accuracy.

Operations management reviews logs periodically to ensure that high standards are met.

Corrections to hand-written or printed logs or records are made using a single line for deletion, inserting any correct information next to the deletion, and initialing and dating the change.

Completed logs are stored for long-term preservation and are retrievable.

6.13.2.2 Narrative Logs

Log entries are made in a timely manner. Entries provide enough detail for events to be reconstructed at a later date. A log is a legal document and is therefore accurate, factual, complete and clear.

Log entries document all major equipment and plant configuration changes, including time of occurrence, such as:

- Watchstation turnover/relief times
- Reactor mode change or condition change (reduced inventory, pressurizer solid, etc.)
- Reactor declared critical or subcritical
- Reactivity changes

- Major equipment or system status changes
- Protected Train/Division and Protected Equipment changes
- Main Turbine roll, rated speed, or trip
- Main feedpump startup or shutdown
- Main generator on line or off line
- Operator adjustable controller setpoint changes
- Power System Stabilizer placed in service or removed from service
- Transmission lines placed in service or removed from service
- Switchyard transformers placed in service or removed from service
- Surveillance testing and results
- Technical Specification Action statement entry or exit
- Entry into AOPs/ABs or EOPs
- Implementation of the Emergency Plan

During transients, control of the plant takes precedence over logkeeping. When a required log entry is discovered to have been missed, it is annotated in a manner that indicates it is a late entry.

6.13.2.3 Watch Station Rounds

The recording of operating logs is performed during Watch Station Rounds in accordance with an approved schedule. Required readings are taken unless the associated equipment or system is out of service. Readings not taken are noted with a documented reason.

Abnormal readings are explained in the "remarks" section and reported to the SS, OATC, UO, or SSS. Corrective actions taken to address out-of-specification conditions are also documented. In the case of written logs, the out-of-specification is circled.

Building rounds are normally commenced within an hour following the beginning of shift control room crew briefing. If rounds are waived, then a Condition Report is submitted to document the reason.

When a recorder chart requires removal, the operator annotates the chart with the unit number, date removed, recorder number, and equipment name. If a chart is adjusted, then the operator documents the date and time on the new chart.

6.14 Shift Briefings and Updates

6.14.1 Standard

Briefings are conducted to enhance communication and ensure the receipt of relevant information among the operating crew.

1. G2.1.5 061/NEW//MEM 2.9*/3.9/G2.1.5/N///

Both Units are operating at 100% power with the following conditions:

- A non-licensed Fire Protection Administrator who is qualified as a Shift Communicator is on shift.

Which one of the following completes the statements below?

Per EIP-0.0, Emergency Organization, a **minimum** of __(1)__ licensed Plant Operators is required to staff the shift.

The **maximum** number of hours that a Plant Operator may work in any 24 hour period is __(2)__ .

	(1)	(2)
A.	3	12
B.✓	3	16
C.	4	12
D.	4	16

EIP-0.0 Table 1 requires:
1 OATC per Unit - Total of 2
1 UO Shared - Total of 1
Shift Communicator (Least affected UO) - 1

NMP-AD016-003
6.1.1 The following work hour **ceiling** limits apply to covered individuals regardless of unit status:
• No more than 16 work hours in any 24-hour period
• No more than 26 work hours in any 48-hour period
• No more than 72 work hours in any 7-day/168-hour period

Distracter analysis

A. Incorrect. First part is correct (See B.2).

Second part is incorrect (See B.2). Plausible since this is the normal number of hours work and the applicant could not be able to recall the correct limit.

B. Correct. First part is correct. Per EIP-0.0, 3 Licensed operators are required to man the shift since a shift communicator is also on shift.

Second part is correct. The following work hour **ceiling** limits apply to covered individuals regardless of unit status:
• No more than 16 work hours in any 24-hour period

C. Incorrect. First part is incorrect (See B.2). Plausible since without a non-licensed shift communicator, this would be a correct answer.

Second part is incorrect (See A.2).

D. Incorrect. First part is incorrect (See C.2).

Second part is correct (See B.2).

K/A: **G2.1.5**	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
Importance Rating:	2.9* 3.9
Technical Reference:	FNP-0-EIP-0.0, Emergency Organization, Ver 29 NMP-AD-016-003, Scheduling and Calculating Work Hours, Ver 5
References provided:	None
Learning Objective:	Given the plant mode for each unit, STATE AND EXPLAIN the minimum manning requirements for manning one or both units (OPS40502H04).
Question History:	NEW
K/A match:	Requires the applicant to have the **ability to determine minimum crew manning as well as maximum hours that the operator may work.** Since this question asks for the reactor operator position it is deemed to be an RO question and since an RO objective exists for this knowledge requirement.
SRO justification:	N/A

Number of nuclear power units operating[2]	Position	One Unit	Two units		Three units	
		One control room	One control room	Two control rooms	Two control rooms	Three control rooms
None	Senior Operator	1	1	1	1	1
	Operator	1	2	2	3	3
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
Two	Senior Operator		2	3	[3]3	3
	Operator		3	4	[3]5	5
Three	Senior Operator				3	4
	Operator				5	6

[1]Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.
[2]For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.
[3]The number of required licensed personnel when the operating nuclear power units are controlled from a common control room are two senior operators and four operators.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

(3) Licensees who cannot meet the January 1, 1984 deadline must submit by October 1, 1983 a request for an extension to the Director of the Office of Nuclear Regulation and demonstrate good cause for the request.

(n) The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or construction permit pursuant to § 50.36 of this part.

(o) Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be

4.6.7 Chemistry Technician, Sampling

- Perform sampling as required for the emergency that is in progress.
  - Assigned assembly area is the OSC.
  - This position may also fill the position of FMT Communicator.

4.7 Security Force

4.7.1 The Nuclear Security Captain will normally be located in the TSC during emergencies and will be available to support the ED, TSC Manager, or Shift Manager if required.

4.7.2 IF there is a security event, the Nuclear Security Captain will follow the appropriate Security Procedures which supersede this procedure.

4.7.3 In addition to the requirements of FNP-0-EIP-7.0, the Security Force is required to assist with accountability in the Control Room and the TSC if the plant emergency alarm is sounded.

4.7.4 One Security Force Member (SFM) may be assigned to perform the ERO callout as directed by the Shift Manager or the Operations Shift Communicator using the ERO Activation package contained in the Shift Manager's Book.

4.8 *Minimum* Emergency Plan Staffing table

4.8.1 Table 1 provides a list of the *minimum* on shift staff that must be met at all times to ensure compliance with the Emergency Plan.

4.8.2 The shift crew composition may be one *less* than the Table 1 requirements for a period of time not to *exceed* 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the Table 1 requirements.

- This provision does NOT permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

4.8.3 Operations Shift Supervision is responsible for verifying that the *minimum* shift staffing is met for each shift In Accordance With (IAW) Table 1.
CAR193408

- Discrepancies in meeting the requirements of Table 1 *SHALL be documented by initiating a Condition Report.*

4.8.4 Documentation that the *minimum* staffing has been met for each position can be maintained in a hardcopy or electronic log.

TABLE 1

MINIMUM SHIFT STAFFING REQUIREMENTS
Based on the Emergency Plan table 3 and FNP-0-EIP-0.0

Position		Person Filling Position	Function
Operations			
Shift Manager	1		Emergency Direction and Control (Emergency Director)
			Plant Operations and Assessment of Operational Aspects. Shift SRO
Unit 1 Shift Supervisor	1		Plant Operations and Assessment of Operational Aspects. SS (SRO)
Unit 2 Shift Supervisor	1		Notification / Communication. The SS of the least affected unit will assume the role of the ENN/ENS communicator
SSS, STA Qualified	1		Shift Technical Advisor Core/Thermal Hydraulics, Electrical, Mechanical
SSS, Fire Brigade Qualified	1		Fire Brigade per the FSAR (Fire Brigade Chief)
OATC Unit 1	1		Plant Operations and Assessment of Operational Aspects
OATC Unit 2	1		Plant Operations and Assessment of Operational Aspects
Unit Operator Unit 1/2	1		Plant Operations and Assessment of Operational Aspects
Shift Communicator	1		Least affected unit UO will assume the role of the Shift Communicator
System Operator Operations	3	SO #1________________ SO #2________________ SO #3________________	Plant Operations and Assessment of Operational Aspects
Systems Operators Fire brigade	4	SO #4________________ SO #5________________ SO #6________________ SO #7________________	Fire Brigade per the FSAR (Fire Brigade members)

41.0 SHIFT MANNING AND POSITION CHANGES

41.1 The on-shift complement will be shown by name on the Shift Organization Chart located near the entrance to the ATC area.

41.2 Deviations in shift complement may be made so long as the minimum manning and license requirements of Technical Specifications are met.

41.3 Additional Operations Group personnel may be required on shift because of unusual plant conditions or operational needs. Shift Supervision shall obtain the additional personnel as necessary. Activities requiring additional personnel will not be undertaken until the shift is properly manned.

41.4 Shift positions may be swapped, by mutual consent of both affected parties, as long as plant proficiency is not affected and Shift Manager approves the swap.

41.5 Personnel expecting to be late or unable to report for shift duty at the scheduled time shall at the earliest possible moment inform Shift Supervision of the situation. The Shift Supervision shall make necessary arrangements for obtaining a replacement.

41.6 Refer to T.S. 5.2.2 for specifics concerning Technical Specification minimum manning.

41.7 Refer to FNP-0-EIP-0.0, EMERGENCY ORGANIZATION, Table 1 for minimum shift manning.

41.8 Refer to Table 6 for normal shift complement.

42.0 PERSONNEL RECALL RESPONSIBILITIES

42.1 If Maintenance or Facilities personnel must be called out for work, the Shift Manager will have designated maintenance supervisory personnel initiate the call out. The designated maintenance supervisory personnel will call out maintenance personnel and maintenance supervision as required.

42.2 If Chemistry and Environmental personnel must be called out for work, the Shift Manager will have the Shift Chemist call out Chemistry and Environmental personnel as required.

42.3 If Security Force personnel must be called out for work, the Security Foreman will normally call out security personnel as required. He/she will ensure that the Shift Manager is kept informed of any problems with minimum staffing.

43.0 ADMINISTRATIVE OVERTIME / EXCESS STRAIGHT TIME GUIDELINES

43.1 Refer to NMP-AD-016 for guidance concerning work hour controls.

6.1.1 The following work hour **ceiling** limits apply to covered individuals regardless of unit status:
- No more than 16 work hours in any 24-hour period
- No more than 26 work hours in any 48-hour period
- No more than 72 work hours in any 7-day/168-hour period

The periods of "24-hours," "48-hours," and "7-days/168-hours" are rolling time periods. Rolling means the period is not re-zeroed or reset following a day off. The period continues to roll.

6.1.2 The following **break** requirements apply to covered individuals regardless of unit status:
- At least a 10-hour break between successive work periods (an 8-hour break is acceptable only when a break of less than 10 hours is necessary to accommodate a crew's scheduled transition between work schedules or shifts).
- A 34-hour break in any 9-calendar day/216-hour period (this limit may be coincident with the tables in 6.1.3 and 6.1.4)

NOTE: For the purpose of calculating an average number of days off, the duration of the shift cycle may not exceed six (6) weeks.

6.1.3 During **online** operations and without issuance of a waiver, an individual's required average **minimum days off** (MDO) shall adhere to the requirements listed in the table below (averaged over the shift cycle):

PERSONNEL	8-HOUR SHIFT	10-HOUR SHIFT	12-HOUR SHIFT
Maintenance	1 day off per week	2 days off per week	2 days off per week
Ops, HP, Chem, Fire Brigade	1 day off per week	2 days off per week	2.5 days off per week
Security	1 day off per week	2 days off per week	3 days off per week

6.1.4 During outage operations and without issuance of a waiver, an individual who performs outage activities may apply the **minimum day off** requirements listed in the table below (not an average):

PERSONNEL	8-HOUR SHIFT	10-HOUR SHIFT	12-HOUR SHIFT
Maintenance	1 day off per week	1 days off per week	1 days off per week
Ops, HP, Chem, Fire Brigade	3 days off in each successive (i.e., non-rolling)15-day period	3 days off in each successive (i.e., non-rolling)15-day period	3 days off in each successive (i.e., non-rolling) 15-day period
Security	4 days off in each successive (i.e., non-rolling) 15-day period	4 days off in each successive (i.e., non-rolling) 15-day period	4 days off in each successive (i.e., non-rolling) 15-day period

6.1.4.1 The table above applies to the first 60 days of a unit outage; after this 60-day period expires, normal online work hour limits will apply. The 60-day period may be extended seven days for an individual for each 7-day block during which they

1. G2.2.20 062/NEW//C/A 2.6/3.8/G2.2.20/N///

Unit 1 is operating at 100% power when the following occurs:

- 1B CCW pump failed to start while attempting to "bump" it after scheduled maintenance.

Which one of the following are the required actions per SOP-0.0, General Instructions to Operations Personnel?

A. Obtain Shift Manager's permission, THEN take the handswitch to START a second time.

B. Take the handswitch to START a second time, THEN write a Condition Report to document the action.

C.✓ Write a condition report documenting the event and contact Maintenance.

D. The System Operator will rackout and perform a visual inspection of the circuit breaker and write a condition report.

SOP-0.0
15.1.3. For handswitches on the MCB, EPB, BOP, and HSDP, if the associated component fails to actuate (pump—start, valve—move in open or closed direction, et cetera) when operating a handswitch, a second actuation may NOT be attempted until the cause can be thoroughly investigated.

15.2.4 IF a breaker has malfunctioned (i.e., failed to close, open, trip, or charge when expected) contact appropriate Maintenance personnel for involvement in troubleshooting prior to attempting restoration efforts. [...]

<u>Distracter analysis</u>

A. Incorrect.	See C. Plausible since this is correct when backing up ESF equipment actuation on the third, fourth attempt etc per step 15.1.5
B. Incorrect.	See C. Plausible since this is correct when backing up ESF equipment actuation per step 15.1.4 and 15.1.5
C. Correct.	A second attempt is not allowed under normal operating conditions.
D. Incorrect.	See C. Per SOP-0.0, Step 15.2.4, racking out a circuit breaker that has malfunctioned is NOT allowed. Plausible since all breaker malfunctions are investigated by the Systems Operators (without racking the breaker out) and the applicant could think that a visual inspection of a racked out breaker is appropriate before calling maintenance.

K/A: **G2.2.20**	Knowledge of the process for managing troubleshooting activities.
Importance Rating:	2.6 3.8
Technical Reference:	FNP-0-SOP-0.0, General Instructions to Operations Personnel, Ver 152.2
References provided:	None
Learning Objective:	Using plant procedures, describe the work control process and associated program interfaces, including Toolpouch Work (for example, tagging, radiation protection, foreign material exclusion, fire protection, and industrial safety). (OPS-40502N09).
Question History:	NEW
K/A match:	The applicant is **required to know what actions are required to support troubleshooting activities for a circuit breaker that failed to close. The actions are the same regardless of if the applicant assumes the breaker failed to shut or the handswitch failed to actuate.**
SRO justification:	N/A

15.0 COMPONENT OPERATION

Instructions for component operation given either verbally or in procedures rely on the standard use of certain terminology, i.e., open, close, check open, verify closed, etc. Guidance on the correct usage of this terminology is contained in Appendix G.

15.1 HANDSWITCH OPERATION

15.1.1 Observe handswitches when they are operated to ensure the handswitch is functioning normally.

15.1.2 Maintain control of the handswitch with your hand until it comes to stop in neutral/normal position. Operate the handswitches gently, they are sensitive and require careful operation. Deliberate operation with appropriate force of the hand switch to the desired position should be utilized to ensure adequate contact development.

15.1.3 For handswitches on the MCB, EPB, BOP, and HSDP, if the associated component fails to actuate (pump—start, valve—move in open or closed direction, et cetera) when operating a handswitch, a second actuation may NOT be attempted until the cause can be thoroughly investigated. Second actuation attempts without investigation should be authorized only for situations where immediate cooling or other important functions for supported SSCs is required.

15.1.4 Second attempts can be made without authorization as described in SOP-0.8 when ESF components fail to actuate when required during an emergency (with the exception of starting a DG or closing the output breaker, which requires the procedure to be used to ensure load shed is verified). The Shift Supervisor should be informed as soon as possible after initiating the manual action.

15.1.5 Any handswitch which requires a second actuation attempt will have a CR written for further evaluation and maintenance, even if proper operation is demonstrated on the second actuation, and will be documented on the STRS, if performing an STP. Prior to additional actuation attempts, i.e., third, forth, et cetera, notify the Shift Manager.
[AIs 2005203550 & 2005205400]

15.2 RESPONSE TO A BREAKER THAT HAS TRIPPED OR MALFUNCTIONED

NOTES

The following is not applicable for amber light due to a load shed signal. ☐

The Shift Supervisor may waive the following requirements based on plant conditions or specific event circumstances. ☐

15.2.1 Components that have tripped due to potential overload or failure condition should not be re-started until the cause can be thoroughly investigated. Do not reset trip flags, thermal overloads handswitch amber lights until directed by appropriate personnel after investigation has started. Before reclosing a tripped circuit breaker, an investigation shall be performed by qualified personnel to determine the cause of the trip and if the circuit breaker can be safely reclosed.

15.2.2 Mechanical relay flags (for example, on 4160V breaker cubicle doors) are prone to drop out due to vibrations. If a breaker is clearly not tripped and no associated alarms have occurred, the system operator may attempt to reset the flag. The relay flag provides indication only. If a relay has actuated it should have resulted in an annunciator or breaker trip, and would not reset. If the relay flag will not reset, or it is apparent that the flag is repetitively dropping, then a condition report should be submitted.

15.2.3 Handswitch breaker disagreement lights should be investigated and the cause evaluated. The light can then be reset by taking the handswitch to the Trip/Open/Off position as applicable to clear the disagreement. This paragraph would apply to clearing the light after a load shed signal. (CR2010106081)

15.2.4 IF a breaker has malfunctioned (i.e., failed to close, open, trip, or charge when expected) contact appropriate Maintenance personnel for involvement in troubleshooting prior to attempting restoration efforts. This is to allow for investigation of the as-found condition of the breaker. In addition, write a condition report clearly stating the sequence of events and state what was heard and seen (i.e., breaker changed state, light illuminated, etc.) This is necessary for documentation of the problem and trending purposes. A condition report should be initiated for failure of a breaker to close, open, or trip when expected or any observed problem which had the potential for breaker malfunction. (IR 1-96-215 and OR 2-97-250)

1. G2.2.25 063/BANK/FNP EXAM BANK/MEM 3.2/4.2/G2.2.25/N///

Which one of the following completes the statements below for Technical Specification 2.1.1, Reactor Core Safety Limits?

Nucleate boiling __(1)__ acceptable.

Reactor Thermal power, Pressurizer Pressure and the highest RCS operating loop __(2)__ must be maintained within limits.

	(1)	(2)
A.✓	IS	Tavg
B.	IS	Thot
C.	is NOT	Tavg
D.	is NOT	Thot

T.S. 2.1.1 Bases: Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime.

COLR
2.10 Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)
2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, **Reactor Coolant System (RCS) highest loop average temperature**, and pressurizer pressure shall not exceed the safety limits specified in Figure 4.

Distracter analysis

A. Correct. First part is correct. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime.

Second part is correct. the combination of THERMAL POWER, **Reactor Coolant System (RCS) highest loop average temperature**, and pressurizer pressure shall not exceed the safety limits specified in Figure 4.

B. Incorrect. First part is correct (See A.1).

Second part is incorrect (See A.2) Plausible because this temperature is higher than Tavg so the applicant may believe it is a more limiting temperature.

C. Incorrect. First part is incorrect (See A.1). Plausible if the applicant recalls that NO DNB is allowed and confuses it with nucleate boiling.

Second part is correct (See A.2).

D. Incorrect. First part is incorrect (See C.1)

Second part is incorrect (See B.2).

K/A: **G2.2.25**	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
Importance Rating:	3.2 4.2
Technical Reference:	Technical Specifications Bases, Ver 58 Core Operating Limit Report, Unit 1 Cycle 25, Ver 1
References provided:	None
Learning Objective:	RECALL AND APPLY the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Plant Design and ESF components/Emergency Core Cooling System/ Tavg and attendant equipment alignment, to include the following (OPS-52102J01): (OPS-52201J10) (OPS-52102B01) 2.1.1 Reactor Core Safety Limit
Question History:	FNP EXAM BANK
K/A match:	Requires the applicant to know the **bases for Tech Spec 2.1.1, Reactor Core Safety Limits.**
SRO justification:	N/A

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$ (Specification 3.2.2)

2.7.1 $F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} * \left(1 + PF_{\Delta H} * (1 - P)\right)$

where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

2.7.2 $F_{\Delta H}^{RTP} = 1.70$

2.7.3 $PF_{\Delta H} = 0.3$

2.8 Axial Flux Difference (Specification 3.2.3)

2.8.1 The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 3.

2.9 Boron Concentration (Specification 3.9.1)

2.9.1 The boron concentration shall be greater than or equal to 2000 ppm.[1]

2.10 Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 4.

2.11 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Setpoint Parameter Values for Table 3.3.1-1 (Specification 3.3.1)

2.11.1 The Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for TS Table 3.3.1-1 are listed in COLR Tables 2 and 3.

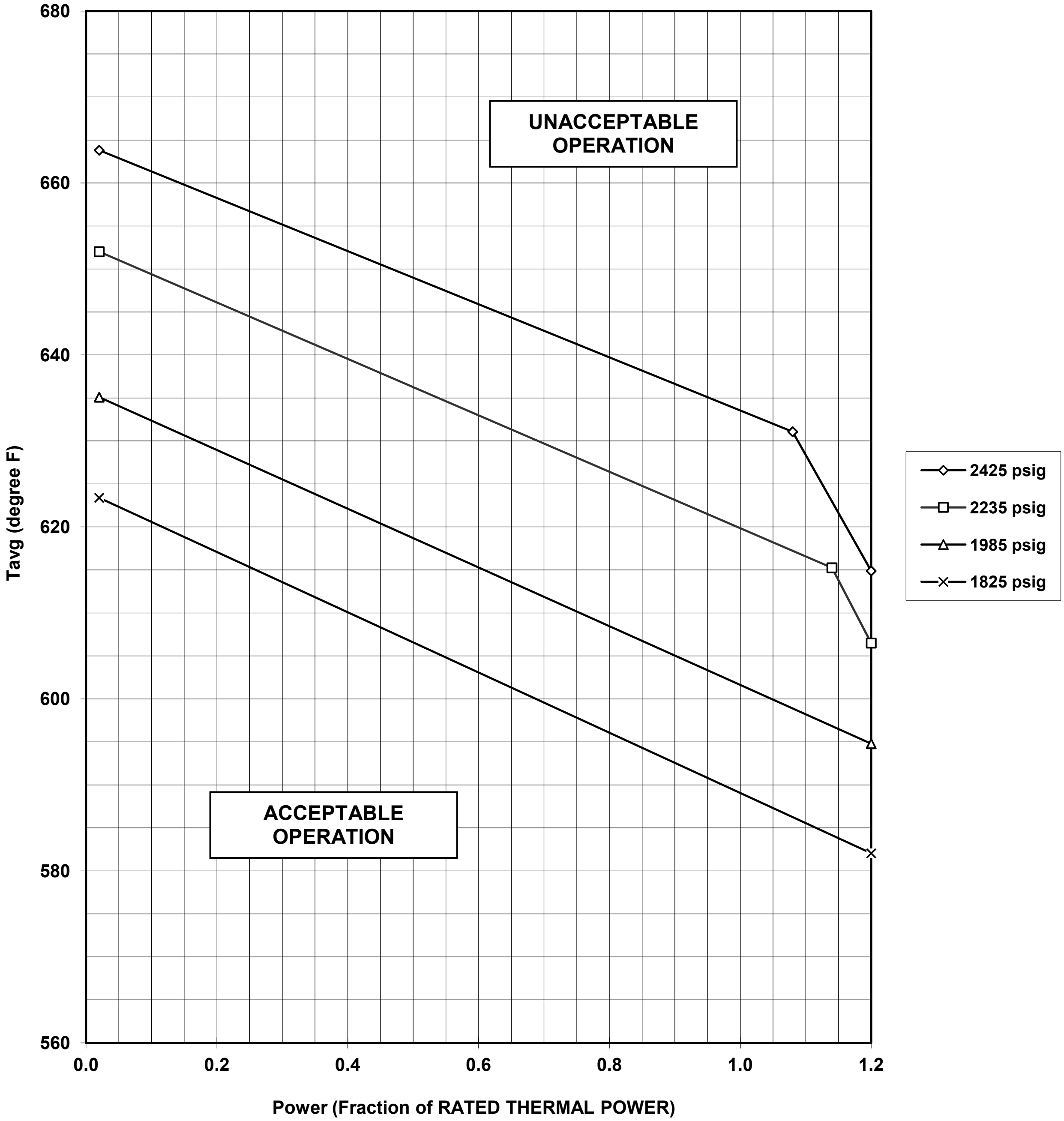
2.12 RCS DNB Parameters for Pressurizer Pressure, RCS Average Temperature, and RCS Total Flow Rate (Specification 3.4.1)

2.12.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2209 psig;
b. RCS average temperature ≤ 580.3°F; and
c. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method and ≥ 264,200 GPM when using the elbow tap method.

[1] This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

Figure 4
Reactor Core Safety Limits



B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rod and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. G2.3.12 064/MOD/SUMMER 11/MEM 3.2/3.7/G2.3.12/N///

Unit 1 is in Mode 6 for a refueling outage.

- Two Plant Operators are required to enter a room that is posted as a **Locked High Radiation Area (LHRA)** to perform work.

Which one of the following completes the statements below?

The radiation level at which this posting is required is __(1)__.

The LHRA key is obtained from __(2)__.

A. 1) > 100 mrem/hr

2) Health Physics Supervision

B. 1) > 100 mrem/hr

2) the Shift Support Supervisor (SSS)

C.✓ 1) > 1000 mrem/hr

2) Health Physics Supervision

D. 1) > 1000 mrem/hr

2) the Shift Support Supervisor (SSS)

RCP-0
5.2.5.1 A LHRA means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 1 Rem/hr at 30 centimeters (11.81 inches or ~ 12 inches) from the radiation source or 30 centimeters from any surface that the radiation penetrates.

RCP-0.1 APP A:
2.1 Maintain Individual Locked High Radiation Area keys under the control of HP Supervision.

Distracter analysis

A. Incorrect. First part is incorrect (See C.1). Plausible because this is the limit for a high radiation area and the applicant could confuse the two limits.

Second part is correct (See C.2).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See C.2). Plausible since a LHRA Master Key is locked within a key storage cabinet located in the Control Room. The key is available for issue by the OPS Shift Supervisor to support mitigation activities associated with an NMP-EP-110 Emergency.

Additionally, the SSS issues numerous keys to personnel during plant operation and the applicant could assume this is one of them.

C. Correct. First Part is correct. 1000 mrem/hr is a Locked HRA.

Second part is correct. Individual Locked High Radiation Area keys are maintained under the control of HP Supervision.

D. Incorrect. First part is correct (See C.1).

Second part is incorrect (See B.2). This would be a correct answer if a declared emergency were in progress and emergency actions were required. The Shift Supervisor could issue a key from the SSS office.

K/A: **G2.3.12**	Knowledge of **radiological safety principles** pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
Importance Rating:	3.2 3.7
Technical Reference:	FNP-0-RCP-0, General Guidance and Special Instructions to Health Physics Personne, Ver 68 FNP-0-RCP-0.1, Key Control Program and Health Physics Guidance for Control of High Radiation areas, Locked High Radiation Areas, and very High Radiation Areas, Ver 18
References provided:	None
Learning Objective:	List four types of areas posted based on radiation levels and the radiation levels/distances that require them to be posted (OPS30401A22)
Question History:	MOD SUMMER 11
K/A match:	Requires the applicant to **know the radiological safety principle (value at which the locked high radiation is posted) and the requirements to enter a locked high radiation area.**
SRO justification:	N/A

5.2.4 High Radiation Area

5.2.4.1 A High Radiation Area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mRem in 1 hour at 30 centimeters (11.81 inches or ~ 12 inches) from the radiation source or 30 centimeters from any surface that the radiation penetrates.

NOTE:	**Barricaded means blocking or obstructing a route or access into the area of concern (e.g. door, rad rope, swing gate, orange snow fence, etc.).**

5.2.4.2 IF an area meets the above criteria, it shall be **barricaded** and conspicuously **posted** with a sign(s) that has a triangular appearance bearing the radiation symbol and the following words as a minimum:

NOTE:	**The word CAUTION may also be used instead of DANGER with the HIGH RADIATION AREA sign.**

DANGER

HIGH RADIATION AREA

Health Physics Escort Required

Or

Alarming Dosimeter Required

RWP Required for Entry

5.2.5 Locked High Radiation Area (LHRA)

5.2.5.1 A LHRA means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 1 Rem/hr at 30 centimeters (11.81 inches or ~ 12 inches) from the radiation source or 30 centimeters from any surface that the radiation penetrates.

NOTE: A HP Supervisor may designate areas as LHRA if in their judgment the radiation levels in the area poses a radiological risk of large or unanticipated exposures but do not meet the definition of an LHRA (e.g. radiation levels < 1 Rem/hr @ 30 cm). Areas designated as such shall meet the same posting and precautionary requirements as LHRAs.

5.2.5.2 A LHRA shall be a **locked** area, EXCEPT during entry, UNLESS it meets the criteria in step 5.2.5.5.

5.2.5.3 A LHRA shall be conspicuously **barricaded** and posted with a sign(s) that has an octagonal appearance bearing the radiation symbol and the following words as a minimum:

DANGER

LOCKED HIGH RADIATION AREA

Health Physics Escort Required

Alarming Dosimeter Required

RWP Required for Entry

NOTE: A LHRA is considered LOCKABLE if the area has an enclosure that *can* be secured with a physical barrier such as a door, a chain link fence that is ~6 feet in height, or a floor plug in which HP can secure with a LHRA locking mechanism. Other physical barriers may be used if approved by a HP Supervisor. However, this approval needs to carefully consider the ability of a person to easily circumvent the barrier such as cutting through with a pocket knife.

The heavy gauge orange snow fence has been determined insufficient for use to establish a LOCKABLE LHRA.

5.2.5.4 A LHRA that is **LOCKABLE** requires the entry point(s) to be locked except during entry.

OR

For Locked High Radiation Areas accessible to personnel that are located within large areas, such as FNP's Containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the areas, then that area shall be controlled by utilizing the alternate method of posting stated in FNP-0-RCP-0. This method basically utilizes flashing lights and barricade material to help personnel to recognize the radiological hazard that exist in their normal work environment.

OR

An HP Technician or HP representative will be stationed in the immediate area of an entrance to a Locked High Radiation Area to prevent any unauthorized entry into the area until required controls can be implemented.

4.4 Entry Requirements for Locked High Radiation Areas

NOTE:
- **A Locked High Radiation or Very High Radiation Area entry is defined as a physical entry of a major portion of the individual's whole body past the plane of the Locked High Radiation Area or Very High Radiation Area boundary.**
- **In no case, will any entry into a dose field with radiation levels such that a major portion of the body could receive in one hour a dose > 1,000 mRem be allowed without the proper administrative controls.**

4.4.1 An approved SRWP is required for entry into a Locked High Radiation Area unless the entry is to save a life. The SRWP will be **prepared** AND **used** for entry into the Locked High Radiation Area and will specify the following requirements:

NOTE:
- **Whenever stay times/jumpsheets are required, and a worker reaches their time limit without reaching their exposure limit, the HP Technician may have the worker come to a low dose area, check the reading on the individuals DAD, and if the individual still has sufficient dose margin, get the HP Foreman or Shift Coordinator to recalculate a new stay time and allow the worker to go back to work with the new stay time in the Locked High Radiation Area.**
- **Consider using maximum possible dose rate that could be encountered when calculating stay time for conservatism.**

4.4.1.1 Stay times for entries into a Locked High Radiation Area can be **controlled** by the digital alarming dosimeter's alarm vs timing an individual's entry if controls are established that eliminate the potential for personnel to actively work in whole body dose rates ≥ 2000 mRem/hr and dose is limited to < 200 mRem/entry.

4.4.1.2 IF there's a potential for personnel to actively work in areas where whole body dose rates are ≥ 2000 mRem/hr or an individual's dose in a single entry is expected to result in ≥ 200 mRem, THEN a Stay Time Calculation Sheet, HP Form 545 or equivalent is required, which specifies the expected dose rates (estimated or actual) in the immediate work area and the maximum allowable staytime for each individual in that area based upon their dose margin. The maximum allowable stay time for each individual will be based on the approved dose for the job and their allowable dose margin, not to exceed 80% of their annual administrative dose margin.

4.4.1.3 Be accompanied by an HP Technician who uses a radiation exposure rate monitoring device and is responsible for providing positive exposure control over the activities within the area and who will provide continuous surveillance,

OR

Direct or remote continuous surveillance using a qualified HP Technician in conjunction with an alarming dosimeter preset to exposure limits based upon their dose margin and expected/allowable dose rates.

4.4.2 Entry into Locked High Radiation Areas shall be by approval of HP Supervision. The job or entry that was approved and HP Supervision approving it shall be written in the HP Shift log. This approval can be given via the phone. Once specific approval is granted for a Locked High Radiation Area entry, multiple entries are authorized for continuing work on the same job with the same scope of work without repeated HP Supervision approval (e.g. entry into the CVCS valve compartment after a resin transfer).

4.4.3 The HP Foreman or Shift Coordinator or their representative will check to ensure that an individual is not allowed a Locked High Radiation Area entry without evaluation of any previously allowed entries (e.g. dose received, dose margin remaining).

APPENDIX A

KEY CONTROL FOR LOCKED HIGH RADIATION AREAS

1.0 Purpose

To provide positive control of keys to Locked High Radiation Areas and prevent unauthorized entries into these areas.

2.0 Procedure

NOTE: A BEST C Keyway Grand Master Key is locked within a "breakable" glass key storage cabinet located in the Control Room. HP maintains key control to this cabinet and periodically inventories this cabinet. This key is available for issue by the OPS Shift Supervisor to support mitigation activities associated with an NMP-EP-110 Emergency

CAUTION: Clamshell door knob locking devices are prohibited for securing access to Locked High Radiation Areas.

2.1 **Maintain** Individual Locked High Radiation Area keys under the control of HP Supervision.

CAUTION: Locked High Radiation Area keys shall only be identified by an inventory number. The prohibition to labeling keys with the corresponding locks physical location permits an additional barrier to inadvertent high radiation area entry when a key is lost or temporarily misplaced.

2.2 **Maintain** Locked High Radiation Area keys in a locker which is separate from general issue keys. The key inventory list will be maintained at the locker for cross referencing the key number to the Locked High Radiation Area access point physical location.

2.3 **Maintain** continuous positive control of the key to the Locked High Radiation Area key locker by the on-shift Health Physics Foreman or Shift Coordinator at all times. The only exceptions to this control would be to a member of Health Physics Supervision above the foreman level (e.g. Supervisor, Manager, etc.).

2.4 At no time will the Locked High Radiation Area key locker be left unattended while unlocked or opened.

2.5 HP Supervision must authorize issuing Locked High Radiation Area keys.

2.6 A Locked High Radiation Area key may only be issued to a Qualified HP Technician and the key will be signed out on HP Form 118A.

2.7 Multiple key signouts for Locked High Radiation Areas entries will be performed by using a single line entry for each individual key issued on HP Form 118A (e.g. 8 keys used - 8 line entries on form).

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics personnel) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the health physics supervision in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels, as measured at 30 cm from the radiation source or from any surface that the radiation penetrates, such that a major portion of the body could receive in one hour a dose greater than 1000 mrem, shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked

(continued)

44. Given the following plant conditions:

- Plant is in an outage in Mode 6.
- A system restoration valve lineup is required in a room that is posted **"Locked High Radiation Area"**.

Which ONE (1) of the choices below identifies the following:

1) The radiation level at which this posting is required.

2) A requirement for the operator to conduct the lineup.

A. 100 mrem/hr.

Must be accompanied by an HP representative with a radiation monitor.

B. 100 mrem/hr.

Must be accompanied by another operator who performs the radiation protection function with a radiation monitor.

C. 1000 mrem/hr.

Must be accompanied by an HP representative with a radiation monitor.

D. 1000 mrem/hr.

Must be accompanied by another operator who performs the radiation protection function with a radiation monitor.

1. G2.3.13 065/NEW//MEM 3.4/3.8/G2.3.13/N///

Which one of the following completes the statements below for entry into the Dry Cask Storage Radiation Controlled Area (RCA)?

Per AP-42, Access Control, the operator __(1)__ required to log in on the normal Auxiliary Building Access Control System (ACS) terminal outside the HP office prior to entering the Dry Cask Storage Area RCA.

The operator is required to __(2)__ when exiting from the Dry Cask Storage Area.

A. 1) IS

2) frisk out at the Dry Cask Storage Area exit ONLY

B.✓ 1) IS

2) frisk out at the Dry Cask Storage Area exit AND use the Auxiliary Building RCA exit portal monitors

C. 1) is NOT

2) frisk out at the Dry Cask Storage Area exit ONLY

D. 1) is NOT

2) frisk out at the Dry Cask Storage Area exit AND use the Primary Access Point (PAP) exit portal monitors

Not a true 2 + 2 to improve distracter plausibility.

AP-42 rev 49.2:
6.0 ENTRY INTO RCAS

6.2 Radiation workers authorized entry into any RCA will ensure they have on their person, personnel monitoring device(s) assigned to them by Health Physics (Dosimetry), that being their dosimetry badge, and a self-issued digital alarming dosimeter prior to entry into that RCA.

6.3 Entry into any RCA requires a Radiation Work Permit and issued personnel dosimetry. Routine access to the main RCA will be through the hallway adjacent to the Health Physics Office.

6.3.4 Prior to entering any RCA, each individual is responsible for ensuring that they meet the requirements of the RWP under which they are entering.

6.3.5 Prior to entry into any RCA each worker will either log in on ACS terminal or log in using an alternate method which will be determined by Health Physics

(e.g., manually logging personnel into and out of the RCA).

6.3.6 Upon exit from the RCA each worker will log out at a ACS terminal except as noted below.

6.3.7 Personnel who are required to enter other RCA's **where no ACS terminal exist** (e.g. outside RCA's, temporary RCA's in Turbine Building, etc.), **will either be required to use the normal Auxiliary Building ACS terminal** or if available, a terminal which is more convenient to the outside RCA. This may require individuals to transit back and forth while logged inside a RCA.

6.3.8 Health Physics will implement an alternate method of control when the ACS is inoperable.

6.4 Entry into the LLRB and other RCAs outside the Protected Area:

6.4.1 Personnel with Vital Area access will log into the RCA per step 6.3.

6.4.2 Upon completion of radiation work, personnel must either return to the HP Office, log out per step 6.3 and check out via the Auxiliary Building RCA exit portal monitor, leaving digital dosimeters at appropriate locations (e.g., at the RCA exit) and other personnel monitoring devices as directed in step 6.2 or they must log out and be monitored as directed by Health Physics.

*** Per the sign on the Dry Cask Storage Area access, frisking is required upon exit from that RCA.**

Distracter analysis

A. Incorrect. First part is correct (See B.1).

Second part is incorrect (See B.2). Plausible since frisking is required per the posting at the Dry Cask RCA but it is NOT the only requirement.

B. Correct First part is correct. Per step 6.3.5, the worker will use the Aux Building ACS terminal.

Second part is correct. Per the posting at the Dry Cask RCA, frisking is required upon exit and per 6.4.2, personnel must log out of the RCA and use the Aux Bldg exit portal monitor.

C. Incorrect. First part is incorrect (See B.1). Plausible if the applicant believes that since they are not entering the Aux building Rad Side, the ACS terminal entry is not required.

Second part is incorrect (See A.2). Logical connection to the first

part since the applicant thought logging into the RCA was not required, this could be an acceptable method since it detect contamination.

D. Incorrect. First part is incorrect (See C.1).

Second part is correct (See B.2). Plausible and logical connection to the first part if the applicant assumes they can use the Primary Access Point (PAP) portal monitors since everyone who leaves the protected area passes through them. If they think that logging in to the RCA is not required and are unfamiliar with the posting at the Dry Cask , this would be a viable way to detect contamination. The PAP portal monitors are closer to the dry cask area than the Aux Bldg exit portal monitor. I tactually would make more sense to use this monitor since it is closer.

K/A: **2.3.13** Knowledge of radiological **safety procedures** pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Importance Rating: 3.4 3.8

Technical Reference: FNP-0-AP-42, Access Control, Ver 49.2

References provided: None

Learning Objective: Outline the requirements and process for entry into an RCA (OPS40502M03).

Question History: NEW

K/A match: Requires the applicant to **know the radiological safety procedure requirements (in this case AP-42, access control) to enter and exit an out building classified as a radiation controlled area.**

SRO justification: N/A

6.0 ENTRY INTO RCAS

6.1 The main RCA comprises most of the Unit 1 and 2 Auxiliary Buildings including the Spent Fuel Storage Areas and Containments. RCAs may also be established at the Low Level Radwaste Building (LLRB), the Solidification and Dewatering Facility (SDF), the Turbine Building or in other areas as required.

6.2 Radiation workers authorized entry into any RCA will ensure they have on their person, personnel monitoring device(s) assigned to them by Health Physics (Dosimetry), that being their dosimetry badge, and a self-issued digital alarming dosimeter prior to entry into that RCA.

6.3 Entry into any RCA requires a Radiation Work Permit and issued personnel dosimetry. Routine access to the main RCA will be through the hallway adjacent to the Health Physics Office.

6.3.1 Optional access to the main RCA can be gained through doors 505 and 2505 (Auxiliary Bldg. roof access to the spent fuel areas).

6.3.2 Under special circumstances access to the main RCA can be gained through door 2484, on Unit 2, with the approval of the Shift Manager, the Health Physics Manager, and the Security Manager.

6.3.3 The Shift Manager must authorize access to the main RCA from any other point. (RESP. 1521). This authorization can be for extended periods of time.

6.3.4 Prior to entering any RCA, each individual is responsible for ensuring that they meet the requirements of the RWP under which they are entering.

6.3.5 Prior to entry into any RCA each worker will either log in on ACS terminal or log in using an alternate method which will be determined by Health Physics (e.g., manually logging personnel into and out of the RCA).

6.3.6 Upon exit from the RCA each worker will log out at a ACS terminal except as noted below.

6.3.7 Personnel who are required to enter other RCA's where no ACS terminal exist (e.g. outside RCA’s, temporary RCA’s in Turbine Building, etc.), will either be required to use the normal Auxiliary Building ACS terminal or if available, a terminal which is more convenient to the outside RCA. This may require individuals to transit back and forth while logged inside a RCA.

6.3.8 Health Physics will implement an alternate method of control when the ACS is inoperable.

6.4 Entry into the LLRB and other RCAs outside the Protected Area:

6.4.1 Personnel with Vital Area access will log into the RCA per step 6.3.

6.4.2 Upon completion of radiation work, personnel must either return to the HP Office, log out per step 6.3 and check out via the Auxiliary Building RCA exit portal monitor, leaving digital dosimeters at appropriate locations (e.g., at the RCA exit) and other personnel monitoring devices as directed in step 6.2 or they must log out and be monitored as directed by Health Physics.

6.4.3 The appropriate group will be responsible for obtaining dosimetry for personnel without Vital Area access. Personnel without Vital Area access will monitor as directed by Health Physics upon exiting from the RCA.

6.5 Entry and/or Exit via Equipment Hatch

6.5.1 Health Physics and Security representatives will be present at the Equipment Hatch Access Enclosure.

6.5.2 Personnel with containment access will log in per step 6.3 at the equipment hatch.

6.5.3 Upon completion of work, personnel exiting the containment will log out and monitor as directed by Health Physics upon exit from the RCA.

7.0 ENTRY INTO CONTAINMENT AT POWER

7.1 Prior to entry into Containment, Health Physics will review the most recent analysis of the Containment atmosphere and they will compare the current readings on the following radiation monitors with the monitor readings at the time of the most recent analysis:

- R-2 Containment Area Radiation Monitor
- R-11 Air Particulate Radiation Monitor
- R-12 Radioactive Gas Monitor

7.2 IF the current readings on the monitors indicate that the Containment atmosphere has not changed significantly, entry may be made based on the most recent analysis. However, if the monitors do indicate a marked change, the following analysis of the atmosphere shall be made prior to entry.

- Gamma-isotopic (iodine and particulates)
- Tritium

7.2.1 Containment entries will be controlled per Health Physics procedures FNP-1-RCP-11 or FNP-2-RCP-11.

1. G2.3.4 066/NEW//MEM 3.2/3.7/G2.3.4/N///

Which one of the following completes the statement below?

An employee who is a fully documented radiation worker and DOES NOT declare her pregnancy has an annual FNP Administrative TEDE limit of _____ .

A. 450 mRem

B. 500 mRem

C.✔ 2000 mRem

D. 5000 mRem

HP manual Step 4.1.3.3
Any employee who discloses that she is or may be pregnant will complete the election form (DOS Form 931 in FNP-0-DOS-2) to accept or decline a prenatal radiation exposure limit of 500 mrem (0.5 rem) for the embryo or fetus for the term of the pregnancy as recommended in 10CFR20.1208.

FNP Admin Annual Dose Guidelines -
Fully documented radiation worker - 2000 mRem per year.

<u>Distracter analysis</u>

A. Incorrect. See C. Plausible since the is the Admin limit for the woman during the term of the pregnancy of a declared pregnant woman.

B. Incorrect. See C. Plausible since this is the Federal Limit for the Embryo for the term of the pregnancy.

C. Correct. Since the woman has not declared her pregnancy, her admin exposure limit is 2000 mRem.

D. Incorrect. See C. Plausible because this is the federal annual limit for and undeclared pregnancy.

K/A: **G2.3.4**	Knowledge of radiation exposure limits under normal or emergency conditions.
Importance Rating:	3.2 3.7
Technical Reference:	FNP-0-M-001, SNC FNP Health Physics Manual, Ver 18
References provided:	None
Learning Objective:	List FNP Admin Limits for various categories of dose (OPS30401A20).
Question History:	NEW
K/A match:	Requires the applicant to **know the normal exposure limits** for an un-declared pregnant woman.
SRO justification:	N/A

For purposes of evaluating internal exposures which do occur, the following options should be considered.

- Continued tracking of an individuals internal exposure until it is verified the internal exposure is below 10% of the ALI.
- If feasible based on the isotopes involved, a bioassay may be performed. The results of the bioassay can then be used to determine any applicable dose.
- Appropriate calculations may be performed to determine the applicable dose. This approach is particularly useful when alpha emitting isotopes are involved which may not be detected by Whole Body Count bioassays.

4.1.3 Radiation Exposure Limits

The following exposure limits are based upon federal regulations, industry standards, and Company Policy. The maximum permissible radiation dose an occupational worker may receive in the course of their duties shall be limited to the following (this applies to ionizing radiation).

Annual Dose Limits (Whichever is More Limiting)		Maximum Dose in Rem Per Calendar Year
1)	Total Effective Dose Equivalent (TEDE)	5
2)	Total Organ Dose Equivalent (TODE) Sum of Deep Dose Equivalent and Committed Dose Equivalent to an individual organ or tissue other than lens of eye.	50
3)	Lens dose equivalent (LDE)	15
4)	Shallow dose equivalent to skin of the whole body (SDE-WB) or the skin of any extremity (SDE-EX).	50

4.1.3.1 During any calendar year the dose to the whole body of any individual shall not exceed 5 rem TEDE or 50 rem TODE and the lifetime exposure limits of 4.1.3.7 must be complied with.

4.1.3.2 Individuals under 18 years of age shall not be badged as radiation workers at Farley Nuclear Plant.

4.1.3.3 Any employee who discloses that she is or may be pregnant will complete the election form (DOS Form 931 in FNP-0-DOS-2) to accept or decline a prenatal radiation exposure limit of 500 mrem (0.5 rem) for the embryo or fetus for the term of the pregnancy as recommended in 10CFR20.1208.

4.1.3.4 Members of the general public, including Escorted Visitors that are not considered radiation workers, shall not be allowed to receive more than 100 mrem of exposure from reactor produced sources for the calendar year.

4.1.3.5 A worker must be appropriately respirator qualified, and medically tested if the use of a respiratory protection device is to be utilized. Appropriate measures shall be taken (e.g. trending of DAC-hours, bioassay analysis) to ensure personnel do not exceed the limits set forth in 10CFR20, Appendix B, Table I for intake of radionuclides.

4.1.3.6 Annual Administrative Exposure Guidelines

In order to maintain the occupational exposure of personnel ALARA, the following administrative exposure guidelines shall apply. To exceed these guidelines, a dose extension request form must be obtained, completed and signed by the worker and a member of their supervision and authorized by the appropriate personnel as listed in the following table.

Category	Annual Dose Guidelines (mrem)[1]				Approvals Required for Dose Extension
	TEDE	TODE	LDE	SDE-SK, & SDE-EXT	
Escorted Visitor (Non- Radworker)[2]	100	1,000	300	1,000	None Allowed
Concurrently Badged[8,10]	450	4,500	1,350	4,500	HP Superintendent
Declared Pregnant Woman[6,7,8]	450/term	—	—	—	None Allowed
Escorted Radiation Worker (ERW)[3,4,8,9]	500	5,000	1,500	5,000	None Allowed
Radiation Workers Less Than Fully Documented[4,8,9] Radiation Workers Fully Documented[5,8,9]	2,000	20,000	6,000	20,000	As Noted Below
	>2,000	>20,000	>6,000	>20,000	HP Manager or Health Physics and Chemistry Manager (or designee)
	>4,000	>40,000	>12,000	>40,000	Assistant General Manager (Ops.) or the Nuclear Plant General Manager (or designee)
	>4,500	>45,000	>13,500	>45,000	Project Vice President

FOOTNOTES:

1. The annual dose for an individual shall include any dose that was occupationally received while being employed by any other facility during that year unless the individual is considered an Escorted Visitor at Farley Nuclear Plant.
2. Escorted Visitor (Non-Radiation Worker) includes Visitors, Guests who have not been Radiation Worker trained and whose access to Radiation Control Areas and exposure margins will be limited.
3. Escorted Radiation Workers (ERW) will have been briefed as a radiation worker but have not received Radiation Worker Training.
4. "Less Than Fully Documented" means the individual has disclosed their current year and lifetime cumulative dose on a written, signed statement from the individual or the individual's most recent employer or on an up-to-date NRC Form-4. This individual is not eligible for Planned Special Exposure (PSE) jobs.

1. G2.4.20 067/MOD/FNP EXAM BANK/C/A 3.8/4.3/G2.4.20/N///

Unit 1 is performing the actions of EEP-3.0, Steam Generator Tube Rupture, due to a tube rupture in the 1B SG.

- The 1B SG Narrow range level is 36% and rising.

Which one of the following completes the statements below?

The 1B SG narrow range level __(1)__ adequate to begin the initial RCS cooldown.

The operational implication of having sufficient level in the 1B SG prior to the cooldown is to __(2)__.

A. 1) is NOT

2) ensure a secondary side heat sink

B. 1) is NOT

2) prevent SG depressurization during the RCS cooldown

C. 1) IS

2) ensure a secondary side heat sink

D.✓ 1) IS

2) prevent SG depressurization during the RCS cooldown

EEP-3 Note prior to Step 4

[CA] Maintaining ruptured SG(s) narrow range level greater than 31%{48%} prevents SG depressurization during RCS cooldown.

FNP-0-EEB-3.0 version 2

ERG Step Text: *Check Ruptured SG(s) Level*

Purpose: 1. To reduce feed flow to the ruptured steam generators to minimize the potential for steam generator overfill.

2. **To establish and maintain a water level in the ruptured steam generators above the top of the U-tubes in order to promote thermal stratification to prevent ruptured steam generator depressurization.**

Basis:

It is also important to maintain the water level in the ruptured steam generator above the top of the U-tubes. When the primary system is cooled in subsequent steps, the steam generator tubes in the ruptured steam generator will approach the temperature of the reactor coolant, particularly if reactor coolant pumps continue to run. If the steam space in the ruptured steam generator expands to contact these colder tubes, condensation will occur which would decrease the ruptured steam generator pressure. As previously demonstrated (see Step 3), this would reduce the reactor coolant subcooling margin and/or increase primary-to secondary leakage, possibly delaying SI termination or causing SI reinitiation. Consequently, the water level must be maintained above the top of the tubes to insulate the steam space. **In addition to insulating the steam space, this ensures a secondary side heat sink in the event that no intact steam generator is available** and also provides protection against misdiagnosis of the ruptured steam generator due to an imbalance of feed flow.

Distracter analysis

A. Incorrect. First part in incorrect (See D.1). Plausible since the applicant may apply adverse numbers of 48% which would make this part correct.

Second part in incorrect (See D.2). Plausible since this is another reason for having sufficient level in the ruptured SG ONLY if there are NO intact SGs available. This is not the case in this question.

B. Incorrect. First part in incorrect (See A.1).

Second part is correct (See D.1).

C. Incorrect. First part in correct (See D.1).

Second part in incorrect (See A.2).

D. Correct. First part in correct. Ruptured SGWL must be > 31%.

Second part is correct. Prevents SG depressurization during RCS cooldown.

K/A: **G2.4.20**	Knowledge of the operational implications of EOP warnings, cautions, and notes.
Importance Rating:	3.8 4.3
Technical Reference:	FNP-1-EEP-3, Steam Generator Tube Rupture, Ver 27 FNP-0-EEB-3.0, Specific Background Document for FNP-1/2 EEP-3, Ver 2
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with EEP-3, Steam Generator Tube Rupture. (OPS-52530D03).
Question History:	MOD FNP EXAM BANK
K/A match:	Requires the applicant to **know the operational implications of not meeting the Note and Caution of EEP-3 by having the improper SG water level prior to RCS cooldown during a tube rupture event.**
SRO justification:	N/A

Section: Procedure

Unit 1 ERP Step: 4 **Unit 2 ERP Step:** 4 ***ERG Step No:*** *4*

ERP StepText: WHEN ruptured SG(s) narrow range level greater than 31%{48%}, THEN perform the following.

ERG StepText: *Check Ruptured SG(s) Level*

Purpose:

1. To reduce feed flow to the ruptured steam generators to minimize the potential for steam generator overfill.
2. To establish and maintain a water level in the ruptured steam generators above the top of the U-tubes in order to promote thermal stratification to prevent ruptured steam generator depressurization.

Basis: Following a steam generator tube rupture, primary-to-secondary leakage into the affected steam generator will exceed steam flow and lead to an accumulation of water in the steam generator. Feed flow will increase the rate of accumulation and reduce the time at which steam generator overfill would occur. Hence, feed flow to the ruptured steam generator should be minimized.

It is also important to maintain the water level in the ruptured steam generator above the top of the U-tubes. When the primary system is cooled in subsequent steps, the steam generator tubes in the ruptured steam generator will approach the temperature of the reactor coolant, particularly if reactor coolant pumps continue to run. If the steam space in the ruptured steam generator expands to contact these colder tubes, condensation will occur which would decrease the ruptured steam generator pressure. As previously demonstrated (see Step 3), this would reduce the reactor coolant subcooling margin and/or increase primary-to-secondary leakage, possibly delaying SI termination or causing SI reinitiation. Consequently, the water level must be maintained above the top of the tubes to insulate the steam space. In addition to insulating the steam space, this ensures a secondary side heat sink in the event that no intact steam generator is available and also provides protection against misdiagnosis of the ruptured steam generator due to an imbalance of feed flow.

Knowledge:

1. The operator should stop feed flow as early as permitted to minimize the potential for steam generator overfill.
2. In most cases, the ruptured steam generator level will continue to increase even after feed flow has been completely terminated. However, for some multiple failure events, such as an unisolable SGTR (i.e., ruptured steam generator cannot be isolated from any intact steam generator), level may decrease during RCS cooldown due to steaming. Consequently, level in the ruptured SG should be monitored periodically to ensure that it remains above the tubes unless the ruptured steam generator is also faulted. In addition to ensuring heat sink if no intact steam generator is available, this also minimizes radiological releases.
3. The operator should continuously monitor ruptured SG level. If feed flow to the ruptured SG is stopped due to level being in the narrow range and later the level drops below the narrow range, feed flow to the ruptured SG should be reinitiated to reestablish level in the narrow range.

References:

Step	Action/Expected Response	Response NOT Obtained

3.7 Verify at least one SG main steam isolation and bypass valve for ruptured SG(s) - CLOSED.

Ruptured SG	1A	1B	1C
1A(1B,1C) SG MSIV - TRIP Q1N11HV	[]3369A []3370A	[]3369B []3370B	[]3369C []3370C
1A(1B,1C)SG MSIV - BYPASS Q1N11HV	[]3368A []3976A	[]3368B []3976B	[]3368C []3976C

3.7 Perform the following.

3.7.1 Place associated test switch to TEST position.

Ruptured SG	1A	1B	1C
1A(1B,1C) SG MSIV - TEST Q1N11HV	[]3369A/ 70A	[]3369B/ 70B	[]3369C/ 70C

3.7.2 IF at least one main steam isolation and one bypass valve for ruptured SG closed, THEN proceed to step 4 IF NOT go to FNP-1-ECP-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.

****

CAUTION: [CA] To prevent excessive RCS cooldown, AFW flow to any ruptured SG that is also faulted, should remain isolated during subsequent recovery actions unless the SG is needed for RCS cooldown.

****

NOTE: [CA] Maintaining ruptured SG(s) narrow range level greater than 31%{48%} prevents SG depressurization during RCS cooldown.

____ 4 **[CA] WHEN ruptured SG(s) narrow range level greater than 31%{48%}, THEN perform the following.**

Step 4 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Establishment of ruptured SG level > 31%{48%} must be complete before continuing with this procedure.

****

NOTE: The potential for reinitiation of ruptured SG(s) auxiliary feedwater and possible TDAFWP steam release should be anticipated if the 2/3 Low Low SG level TDAFWP auto start setpoint is reached during RCS cooldown.

___ **6 Perform RCS cooldown.**

6.1 Determine required core exit temperatures for cooldown based on ruptured SG pressure.

RUPTURED SG PRESSURE (psig)	REQUIRED CORE EXIT TEMPERATURE
1151 - 1200	536°F {522°F}
1101 - 1150	531°F {516°F}
1051 - 1100	525°F {510°F}
1001 - 1050	519°F {504°F}
951 - 1000	513°F {498°F}
901 - 950	507°F {491°F}
851 - 900	500°F {484°F}
801 - 850	494°F {477°F}
751 - 800	487°F {469°F}
701 - 750	479°F {461°F}
651 - 700	471°F {453°F}
601 - 650	463°F {443°F}
551 - 600	454°F {434°F}
501 - 550	445°F {423°F}
451 - 500	434°F {412°F}
401 - 450	423°F {400°F}
351 - 400	411°F {386°F}
301 - 350	398°F {370°F}
251 - 300	383°F {353°F}
- 250	365°F {332°F}

Step 6 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

___ 7 **Check intact SG level.**

7.1 Check any intact SG narrow range level - GREATER THAN 31%{48%}.

RNO 7.1 Verify total AFW flow to intact SGs greater than 395 gpm.

AFW FLOW TO
1A(1B,1C) SG
[] FI 3229A
[] FI 3229B
[] FI 3229C

AFW
TOTAL FLOW
[] FI 3229

7.2 [CA] WHEN all intact SG(s) narrow range level greater than 31%{48%}, THEN maintain intact SG narrow range level 33%-65%{48%-65%}.

RNO 7.2 IF any SG narrow range level rising in an uncontrolled manner, THEN stop RCS cooldown AND return to step 1.

7.2.1 Control MDAFWP flow.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

Intact SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[] 3227A in MOD	[] 3227B in MOD	[] 3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

Step 7 continued on next page.

1. EEP-3-52530D03 005/HLT/LOCT//MEM 4.2/4.5/EPE038EK3.06///LOCT/

Which one of the following is an adverse effect of allowing ruptured SG levels to fall to below the **minimum** required level of EEP-3, Steam Generator Tube Rupture?

A. A rapid rise in ruptured SG level due to "swell" when cooldown is commenced.

B. Ruptured SG depressurization due to increased thermal stratification that will occur.

C. A rapid rise in ruptured SG pressure if the leaking tube is uncovered during cooldown.

D✓ Ruptured SG depressurization due to steam space contact with the colder SG tubes and subsequent condensation that will occur.

1. G2.4.23 068/BANK/FNP 08/MEM 3.4/4.4/G2.4.23/N///

FRP-Z.1, Response to High Containment Pressure, has the following caution:

IF ECP-1.1, Loss of Emergency Coolant Recirculation, is in effect, THEN Containment Spray should be operated as directed in ECP-1.1.

Which one of the following describes the bases for giving priority to ECP-1.1?

ECP-1.1 directs the operation of the Containment Spray (CS) pumps to ensure _____.

A.✔ RWST level is conserved

B. adequate NPSH for the RHR pumps is available

C. the maximum available Containment heat removal systems are running

D. automatic swapover of the CS pumps to the Containment sump is prevented

FRP-Z.1 Caution prior to step 3
IF FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, THEN containment spray should be operated as directed in FNP-1-ECP-1.1.

FRB-Z.1 Step 3 Basis
Guideline ECA-1.1 uses a less restrictive criteria, which permits reduced spray pump operation depending on RWST level, containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to **conserve RWST water**, if possible, by stopping containment spray pumps

Distracter analysis

A. Correct. Per above basis statement: The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to conserve RWST water, if possible, by stopping containment spray pumps

B. Incorrect. See A. Plausible since ECP-1.1 is Loss of Emergency Coolant Recirculation and the applicant may think that ECP-1.1 operates the spray pumps to maximize sump level to allow the RHR pumps to get a proper suction for alignment to sump recirculation.

C. Incorrect. See A. Plausible since this is the goal of FRP-Z.1 and the applicant could confuse the two procedures.

D. Incorrect. See A. Plausible since there is an auto swap over for RHR sump suction valves but not for Containment Spray pumps. The applicant could confuse these and think that ECP-1.1's mitigation addressed this issue.

K/A: G2.4.23	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.
Importance Rating:	3.4 4.4
Technical Reference:	FNP-1-FRP-Z.1, Response to High Containment Pressure, Ver 15 FNP-0-FRB-Z.1, Specific Background Document for FNP-1/2-FRP-Z.1, Ver 1
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with (1) FRP-Z.1, Response to High Containment Pressure; [...]. (OPS-52533M03)
Question History:	FNP 08
K/A match:	Requires the applicant to **know the basis for operating the Containment Spray pumps per ECP-1.1 versus FRP-Z.1 (prioritizing emergency procedure implementation during emergency operations).**
SRO justification:	N/A

RESPONSE TO HIGH CONTAINMENT PRESSURE
Plant Specific Background Information

Section: Procedure

Unit 1 ERP Step: 3 CAUTION-1 **Unit 2 ERP Step:** 3 CAUTION-1 ***ERG Step No:*** *3 CAUTION-1*

ERP StepText: IF FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, THEN containment spray should be operated as directed in FNP-1-ECP-1.1.

ERG StepText: *If ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray should be operated as directed in ECA-1.1 rather than step 3 below.*

Purpose: To ensure containment spray pumps are operated as directed in ECA-1.1 instead of this guideline, if ECA-1.1 is in effect

Basis: This caution warns the operator that the operation of the containment spray pumps indicated in guideline ECA-1.1 takes precedence over that noted in Step 3 of this guideline. This guideline specifies maximum available heat removal system operability in order to reduce containment pressure. Guideline ECA-1.1 uses a less restrictive criteria, which permits reduced spray pump operation depending on RWST level, containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to conserve RWST water, if possible, by stopping containment spray pumps.

Knowledge: N/A

References:

Justification of Differences:

1 Changed to make plant specific.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: IF FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, THEN containment spray should be operated as directed in FNP-1-ECP-1.1.

****

___ 3 **[CA] Check if containment spray is required.**

3.1 Containment pressure - HAS RISEN TO GREATER THAN 27 psig.	3.1 Return to procedure and step in effect.
3.2 Verify PHASE B CTMT ISO - ACTUATED. [] MLB-3 1-1 lit [] MLB-3 6-1 lit	
3.3 Verify containment spray pumps - RUNNING.	

Step 3 continued on next page.

___Page Completed

1. G2.4.29 069/NEW//C/A 3.1/4.4/G2.4.29/N///

Unit 1 is operating at 100% power and the following conditions exist:

- #1 Waste Monitor Tank (WMT) release is in progress.
- The Unit 1 Rad Side SO is at the RCA exit preparing to enter the portal monitors.

Subsequently, the plant emergency alarms sounds and an announcement is made declaring a Site Area Emergency.

Which one of the following completes the statements below?

The Rad Side SO will go to the designated assembly area __(1)__.

The designated assembly area for the Rad Side SO is the __(2)__.

A. 1) after securing the #1 WMT release

2) Operations Support Center (OSC)

B. 1) after securing the #1 WMT release

2) Control Room

C. 1) immediately

2) Operations Support Center (OSC)

D.✔ 1) immediately

2) Control Room

EIP-0.0
4.4.5 System Operators (2), plant operations.
- Assigned assembly area is the Control Room.

4.4.6 Other System Operators (as required by Technical Specifications), plant operations.
- Assigned assembly area is the Control Room.

EIP-10
4.10 During outages and normal Monday through Friday day shifts, individuals NOT described in section 4.1 thru 4.8 above will report to their assembly area as follows:
[...]
OPS Group on shift or qualified for a Shift position, and NOT in Training - Control Room

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since this would stop a release to the environment. However, the release is monitored and will stop if R-18 alarms and the pump will trip on low level which would render the system safe.

Second part is incorrect (See D.2). Plausible since this is an assembly area per procedure and the applicant may not recall the proper assembly areas for on shift staff. The OSC is extremely close to the Control Room and where all other personnel assemble. Prior to the new protected area, onshift staffing did assemble in the OSC so this was a normal assembly location for on shift OPS personnel in the recent past.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

C. Incorrect. First part is correct (See D.1).

Second part is incorrect (See A.2).

D. Correct. First part is correct.NMP-EP-111-001 (pg 17) page announcement has a section to give directions to personnel out in the field if the intent is to allow them to remain in the field. The stem does not indicate that this specific direction is given so the SO will immediately go to his/her assembly area.

Second part is correct. The control room is the proper assembly area per EIP-0.0 and EIP-10.

K/A: **G2.4.29**	Knowledge of the emergency plan.
Importance Rating:	3.1 4.4
Technical Reference:	NMP-EP-111-001, Emergency Notification Network Communicator Instructions - Farley, Ver 3.2 FNP-0-EIP-0.0, Emergency Organization, Ver 29
References provided:	None
Learning Objective:	IDENTIFY AND EXPLAIN the actions to be taken by an individual following an evacuation announcement (OPS40501B04).
Question History:	NEW
K/A match:	Require the applicant to have **knowledge of an individual's responsibilities when the Emergency Plan** is activated.
SRO justification:	N/A

3.6 After the Emergency Response Facilities (ERFs) have been made functional, the ERF staff is expected to comply with all requirements listed in the EIPs. During setup, prior to the facility becoming functional, some latitude can be used by the staff to facilitate the setup of the ERF (SAER 96 EP 16-1, Comment 5).

4.0 ON-SHIFT RESPONSE TO A PLANT EMERGENCY

4.1 The Shift Manager will be the Emergency Director (ED) who will function in that role in the event of an emergency until the Plant Manager OR other designated ED arrives ON SITE AND relieves the Shift Manager of the ED function.

- The Shift Manager may delegate any of his ED duties *except* those items in NMP-EP-110 listed as items that CANNOT be delegated.

4.1.1 In addition to the ED function, the Shift Manager will fill the Emergency Plan Senior Reactor Operator (SRO) role of plant operations and operational assessment IF other Shift SROs are NOT available.

4.1.2 In the event that the Shift Manager is NOT available, the least affected unit Shift Supervisor SHALL perform all of the duties and have all of the responsibilities of the ED as described above, until properly relieved of those duties by the Shift Manager or ON-CALL ED.

4.2 The affected unit Shift Supervisor SHALL direct the operational activities of the plant to combat the plant emergency.

4.3 The least affected unit Shift Supervisor, in addition to maintaining oversight responsibility for the least affected unit, will be responsible for communicating emergency information to the state and local agencies (ENN Communicator) and to the NRC (ENS Communicator).

- The ENS and ENN Communicator function may also be filled by other qualified individuals that are ON SHIFT and NOT required to be performing other functions.

4.4 The *minimum* ON-SHIFT operations staff required to support the Emergency Plan, other than the Shift Supervisors, will include the following positions, with the indicated responsibilities and assembly areas:

4.4.1 Shift Technical Advisor, core thermal hydraulic evaluation.

- Assigned assembly area is the Control Room.

4.4.2 Shift Support Supervisor, Fire Brigade.

- Assigned assembly area is the Control Room.
- This individual may be used as a communicator or for plant operations in the event that there is NO fire brigade required for the emergency.

4.4.3 Control Room Operators (2) (*affected* unit) plant operations.

- Assigned assembly area is the Control Room.

4.4.4 Control Room Operators (*opposite* unit) as required by Technical Specifications plant operations.

- Assigned assembly area is the Control Room.

4.4.5 System Operators (2), plant operations.

- Assigned assembly area is the Control Room.

4.4.6 Other System Operators (as required by Technical Specifications), plant operations.

- Assigned assembly area is the Control Room.

4.4.7 Shift Communicator (1), assigned to perform ENN roll call, confirm receipt of the electronic ENN form, or to read the ENN form over the ENN if the electronic ENN method fails.

- This position will also perform the ERO callout per ERO Activation package.
- This position is typically filled by the least affected unit's Unit Operator (UO) but can also be filled by any Shift Communicator qualified person that is NOT assigned to fill one of the above required positions.
- Assigned assembly area is the TSC or the Control Room.

4.5 The *minimum* ON-SHIFT maintenance staff will include the following positions, with the indicated responsibilities and assembly areas:

4.5.1 Maintenance Supervisor, supervise repair and corrective action.

- Assigned assembly area is the OSC.

4.5.2 Mechanical Journeyman, repair and corrective action.

- Assigned assembly area is the OSC.

4.5.3 Electrical Journeyman, repair and corrective action.

- Assigned assembly area is the OSC.

4.5.4 I&C Journeyman, repair and corrective action.

- Assigned assembly area is the OSC.

4.5.5 An individual NOT filling any of the above positions that is qualified to drive the fire brigade tanker truck.

TABLE 1

MINIMUM SHIFT STAFFING REQUIREMENTS Based on the Emergency Plan table 3 and FNP-0-EIP-0.0			
Position		Person Filling Position	Function
Operations			
Shift Manager	1		Emergency Direction and Control (Emergency Director) Plant Operations and Assessment of Operational Aspects. Shift SRO
Unit 1 Shift Supervisor	1		Plant Operations and Assessment of Operational Aspects. SS (SRO) Notification / Communication. The SS of the least affected unit will assume the role of the ENN/ENS communicator
Unit 2 Shift Supervisor	1		
SSS, STA Qualified	1		Shift Technical Advisor Core/Thermal Hydraulics, Electrical, Mechanical
SSS, Fire Brigade Qualified	1		Fire Brigade per the FSAR (Fire Brigade Chief)
OATC Unit 1	1		Plant Operations and Assessment of Operational Aspects
OATC Unit 2	1		Plant Operations and Assessment of Operational Aspects
Unit Operator Unit 1/2	1		Plant Operations and Assessment of Operational Aspects
Shift Communicator	1		Least affected unit UO will assume the role of the Shift Communicator
System Operator Operations	3	SO #1____________________ SO #2____________________ SO #3____________________	Plant Operations and Assessment of Operational Aspects
Systems Operators Fire brigade	4	SO #4____________________ SO #5____________________ SO #6____________________ SO #7____________________	Fire Brigade per the FSAR (Fire Brigade members)

3.3 ALTERNATE ASSEMBLY AREAS

NOTE

EXTRA PERSONNEL FROM THE OSC MAY BE RELOCATED TO THE OPS READY ROOM, THE SE CORNER OF THE CONTROL ROOM, AND THE HP OFFICE AREA, BY THE OSC MANAGER. OTHER ALTERNATE ASSEMBLY AREAS MAY BE USED AT THE EMERGENCY DIRECTOR'S (ED's) DISCRETION.

Assembly Area 12	Contractor Parking Lot	
Assembly Area 13	Switchhouse Parking Lot	
Assembly Area 14	Area between the 2A and 2B Cooling Towers	
Assembly Area 15	Utility Building	2427
Assembly Area 16	SE Corner of Control Room	2306
Assembly Area 17	OPS Ready Room	2405
Assembly Area 18	HP Office Area	2334

4.0 ASSEMBLY AREA ASSIGNMENTS

4.1 On-Call Personnel assigned to a position with the emergency response organization (**ERO**) will report to their designated **emergency response facility** as described in FNP-0-EIP-6.0, instead of assembling with their work group.

4.2 Personnel that are on site and assigned to a position with the **ERO** on the on-call memo, but not currently in an on-call status, will also report to their designated **emergency response facility** unless otherwise directed by FNP-0-EIP-6.0 (e.g. FEOC Liaison Alternates report to the TSC).

4.2.1 After it has been determined if these individuals are needed to augment the facility staff or will be required to be available for a long term relief, they may be relocated or evacuated offsite as necessary.

4.3 On-shift personnel with a specific function assigned in FNP-0-EIP-0.0 will report to the area described, **or** perform the described function instead of assembling with their work groups.

4.4 On shift **security** will remain on station until relocation is required.

4.4.1 IF, due to a personal hazard, security personnel must evacuate their station, security supervision will be notified as soon as possible to implement the guidance of FNP-0-EIP-14.0.

4.5 Personnel who have been assigned to augment the TSC or OSC staffs will remain at that location for accountability.

4.5.1 The *senior* individual in their normal assembly area should be notified as soon as possible, when time permits.

4.6 **NRC** Inspectors will assemble in **any** one of the **assembly areas** as appropriate for plant and emergency conditions.

4.7 Escorted **visitors** will remain with their escort and report to the same assembly area as their escort.

4.8 Other **visitors/contractors** on site will report to the assembly area designated for the on-site work group with whom they are working.

4.9 On Back Shifts, Weekends, and Holidays during **non** outage periods the following assembly areas will be used.

4.9.1 All personnel on site that meet **ALL** of the following requirements have an assigned assembly area of the **Operations Support Center**.

- Individuals with unescorted access.
- Individuals not assigned to the Control Room as an assembly area.
- Individuals not assigned to another Emergency Response Facility as an assembly area.
- Individuals qualified as a radiation worker.
- Individuals not engaged in recreational activities on site such as hunting or fishing.

4.9.2 All personnel on site that are not assembling in the OSC, the Control Room, or another Emergency Response Facility will assemble in the **Support Building Auditorium**.

4.10 During outages and normal Monday through Friday day shifts, individuals NOT described in section 4.1 thru 4.8 above will report to their assembly area as follows:

STATUS	ASSEMBLY AREA
HP Group or a contractor assigned to HP, Except ADM Assistant.	**OSC**
Chemistry Group, Except ADM Assistant.	**OSC**
OPS Group on shift or qualified for a Shift position, and NOT in Training.	**Control Room**
OPS Students in initial training, assigned to shift.	**Control Room**
Individual whose normal work location is the **Support Building.**	**Support Building Cafeteria**
Students participating in Training Activities in the **Support Building.**	**Support Building Cafeteria**
HP Group or HP contractors involved in training anywhere on site.	**Support Building Cafeteria**
Chemistry or CHM contractors involved in training anywhere on site.	**Support Building Cafeteria**
Individual whose normal work location is the **Service Building** or **Service Building Annex**, excluding Maintenance Staff.	**Support Building Cafeteria**
Ionics or other **Chemistry Group Contractor**	**Support Building Cafeteria**
Maintenance Staff whose normal work location is the Service Building or Service Building Annex.	**Support Building Auditorium**
Facilities Group or a contractor assigned to the Facilities Group.	**Support Building Auditorium**
Painter or Material Handler.	**Support Building Auditorium**
Individual in a non-work status, engaged in **sporting** or other **recreational activities.**	**Support Building Auditorium**
Individual on site and **not previously listed** in this procedure.	**Support Building Auditorium**
Turbine Group **Siemans Personnel and Subcontractors**	**Support Building Auditorium**
Security member NOT on an assigned post.	**NON "Protected Area", Entrance Side of The PAP.**

ATTACHMENT 1
PRE-RELEASE RE-18 TEST

> **NOTE**
>
> This attachment may be used by the Control Room Operator in conjunction with Section 4.3 which is maintained by the Radside Systems Operator.

7.0 PRE-RELEASE RE-18 TEST

7.1 IF LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018 drawer is in alarm, THEN **reset** the alarm. (Section 4.3, Step 4.3.3) ☐

7.1.1 IF background radiation levels prevent resetting LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018, THEN **raise** the pot setting to a value above background AND **reset** the alarm. ☐

7.2 **Notify** the SO to open WMT DISCH TO ENVIRONMENT, N1G21RCV18 (N1G21V113). (Section 4.3, Step 4.3.4) ☐

7.3 **Lower** the trip setting of LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018, drawer until a high radiation alarm actuates per Section 4.3, Step 4.3.5. ☐

7.4 **Notify** the SO to verify WMT DISCH TO ENVIRONMENT, N1G21RCV18 (N1G21V113), closes per Section 4.3, Step 4.3.6. ☐

7.5 **Notify** the SO to operate the control switch for WMT DISCH TO ENVIRONMENT, N1G21RCV18 (N1G21V113), from the LWPP to verify that the valve cannot be opened from the panel per Section 4.3, Step 4.3.7. ☐

7.6 **Adjust** the pot setting of LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018, to normal value. (Section 4.3, Step 4.3.8) ☐

7.6.1 IF background radiation levels prevent resetting LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018, to normal value, THEN **raise** the pot setting to a value sufficiently above background to prevent spurious trips while maintaining the setpoint less than allowed by the release permit. ☐

7.7 **Reset** the high radiation alarm on LIQUID WASTE DISCHARGE RADIATION MONITOR, N1D11RE0018. (Section 4.3, Step 4.3.9) ☐

APPENDIX 1
WASTE MONITOR TANK 1
RELEASE TO THE ENVIRONMENT
(Page 9 of 13)

4.4.13.3 Slowly **throttle open** #1 WMT PUMP DISCH, N1G21V108B (1-LWP-V-7443B) such that pump discharge pressure does not drop below the pressure recorded in Step 4.4.13.2. ______

4.4.14 **Record** the following information on release permit.

- Release start time ______
- Release flow rate ______

NOTE

Local indication may be used from LIQUID RADWASTE DISCHARGE INDICATOR, N1D11RI018, located on the liquid waste panel.

4.4.15 IF R-18 is operable, THEN **record** monitor RE-18 count rate 10 minutes into discharge on release permit. ______

4.4.16 **Verify** pump tripped at low level set point or secure pump. ______

4.4.17 **Place** #1 WMT PUMP N1G21P006B-LWPP H/S in PULLED-TO-LOCK ______

4.4.18 **Close** WMT DISCH TO ENVIRONMENT, N1G21RCV18 (N1G21V113). ______

4.4.19 **Perform** the following:

4.4.19.1 **Record** the following information on the release permit.

- Release stop time ______
- SW dilution totalizer ______
- WMT totalizer ______
- Final tank level ______

4.4.19.2 **Record** the following information on the WMT flow recorder paper.

- Release stop time. ______
- Release date. ______

4.4.20 **Close** and **lock** #1 WMT DISCH Q1G21V111 (1-LWP-V-7446) ______

4.4.21 **Close** #1 WMT PUMP DISCH, N1G21V108B (1-LWP-V-7443B). ______

Checklist 1 - Page Announcement (page 1 of 1)

> **NOTE**
>
> **The completion of an initial plant page announcement to activate the ERO is expected to be completed within 5 minutes of the declaration of an Alert or higher. All subsequent announcements should be completed as soon as practicable.**

1. Obtain copies of the appropriate site specific document

Farley	Hatch	Vogtle
NMP-EP-111-001	NMP-EP-111-002	NMP-EP-111-003

2. Select the appropriate page announcement script from the site specific document _______
3. Sound the emergency tone for approximately ten (10) seconds (Alert or higher) _______
4. Make an announcement with the plant page public address system: _______
5. <u>REPEAT</u> the above tone and announcement _______
6. For an Alert or higher, perform announcement(s) on the following frequencies:
   a. Repeat the announcement approximately every thirty (30) minutes during the first (2) hours of the declared emergency and track time of announcement below: _______

      Initial Page Announcement Time: ________________
      30 minute repeat (approx.) Time: ________________
      60 minute repeat (approx.) Time: ________________
      90 minute repeat (approx.) Time: ________________
      120 minute repeat (approx.) Time: ________________

   b. After the first two (2) hours, repeat the announcement as directed by the ED, SM, or SS and track time of announcement below: _______

      Time: ________________
      Time: ________________
      Time: ________________
      Time: ________________

INSTRUCTION 6 – Initial/Upgrade Non-Security Page Announcement

(Page 1 of 1)

NOTE

The following scripts are provided to assist in the development of page announcements during emergency conditions. Utilizations of this script is encouraged. Verbatim usage of this script is NOT required. Modification of this script may be necessary to ensure that the proper message is delivered based on the event(s) in progress.

[Select one]

ATTENTION ALL PERSONNEL - THIS IS A

☐ DRILL MESSAGE ☐ ACTUAL EMERGENCY MESSAGE

[Select one]

☐ A NOTIFICATION OF UNUSUAL EVENT ☐ A SITE AREA EMERGENCY

☐ AN ALERT EMERGENCY ☐ A GENERAL EMERGENCY

[Select one]

HAS BEEN DECLARED BASED ON EVENTS AFFECTING

☐ UNIT 1 ☐ UNIT 2 ☐ THE SITE

[Event description]

BASED ON____

NOTE

1. **For Alert declarations or higher, complete** a **and** b **as applicable.**
2. Steps **a** and **b** are **not** required **if** assembly and accountability are complete.

[Announce as applicable]

a. "PERSONNEL WORKING ON [*site specific tasks as specified by the ED*] CALL CONTROL ROOM AT EXTENSION [*site specific phone number*] & CONTINUE WORK."

b. "EMERGENCY RESPONSE PERSONNEL REPORT TO YOUR EMERGENCY RESPONSE FACILITY. NON EMERGENCY RESPONSE PERSONNEL, CONTRACTORS AND VISITORS SECURE YOUR WORK AREA, AND REPORT TO YOUR ASSEMBLY AREA

[Select one]

THIS IS A

☐ DRILL MESSAGE

☐ ACTUAL EMERGENCY MESSAGE

1. W/E04EA1.1 070/NEW//C/A 4.0/4.0/W/E04EA1.1/N///

The crew has transitioned to ECP-1.2, LOCA Outside Containment.

- Step 2 of ECP-1.2 is in progress and the first flow path has been isolated.

The following conditions exist:
- Aux Building radiation levels are rising slowly.
- Safety Injection flow is stable.
- Aux Building sump levels are rising slowly.
- PT-402 and 403, RCS 1A/1C LOOP RCS NR PRESS, is rising.

Which one of the following completes the statements below per ECP-1.2?

The first flow path that was isolated was _(1)_ injection.

The intersystem LOCA _(2)_ been isolated.

	(1)	(2)
A.	RCP seal	HAS
B.	RCP seal	has NOT
C.✓	RHR cold leg	HAS
D.	RHR cold leg	has NOT

ECP-1.2
Step 2: Try to identify and isolate break.
2.1 Isolate A train RHR cold leg injection path.
2.2 Check RCS pressure - RISING.

2.5 Isolate B train RHR cold leg injection path.
2.6 Check RCS pressure - RISING.

Distracter analysis

A. Incorrect. First part is incorrect (See C.1). Plausible since this is isolated during ECP-1.2 but not first.

Second part is correct (See C.2).

B. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See C.2). Plausible if the applicant does not recall which parameter is used to check leak isolation. Aux building sump levels and radiation levels could cause the applicant to believe that the leak is not isolated. Once the leak was isolated, sump levels could continue to rise as well as radiation levels as the isolated piping drains.

C. Correct. First part is correct. Per step 2 of ECP-1.2, LHSI (RHR cold leg injection) is isolated first.

Second part is correct. Per ECP-1.2, RCS pressure rising is the parameter monitored for verifying the leak is isolated.

D. Incorrect. First part is incorrect (See A.1).

Second part is incorrect (See B.2).

K/A: **W/E04EA1.1**	LOCA Outside Containment - Ability to operate and / or monitor **Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features** as they apply to the (LOCA Outside Containment)
Importance Rating:	4.0 4.0
Technical Reference:	FNP-1-ECP-1.2, LOCA Outside Containment, Ver 8
References provided:	NONE
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing ECP-1.2, LOCA Outside Containment. (OPS-52532E06) ANALYZE plant conditions and DETERMINE the successful completion of any step in ECP-1.2, LOCA Outside Containment. (OPS-52532E07)
Question History:	NEW
K/A match:	Requires the applicant to know which **components are operated and** be able to determine from listed instrumentation (**monitor) if the leak has stopped during the leak isolation phase of ECP-1.2, LOCA Outside of Containment.**
SRO justification:	N/A

Step	Action/Expected Response	Response NOT Obtained
1.5	Verify charging pump to regenerative heat exchanger valves - CLOSED. CHG PUMPS TO REGENERATIVE HX [] Q1E21MOV8107 [] Q1E21MOV8108	
1.6	Verify containment sump pump isolation valves - CLOSED. (BOP) CTMT SUMP DISCH [] Q1G21HV3376 [] Q1G21HV3377 CTMT SUMP RECIRC [] Q1G21HV3380	
___ 2	**Try to identify and isolate break.**	
2.1	Isolate A train RHR cold leg injection path. 1A RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A closed RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A closed	
2.2	Check RCS pressure - RISING. 1C(1A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	2.2 Proceed to step 2.4.
2.3	Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.	

Step 2 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
2.4	Restore A train RHR cold leg injection path. 1A RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A open RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A open	
2.5	Isolate B train RHR cold leg injection path. 1B RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888B closed RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887B closed	
2.6	Check RCS pressure - RISING. 1C(1A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	2.6 Proceed to step 2.8.
2.7	Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.	
2.8	Restore B train RHR cold leg injection path. 1B RHR HX TO RCS COLD LEG ISO [] Q1E11MOV8888B open RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887B open	

Step 2 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

2.9 Verify CCW being supplied to RCP thermal barriers.

2.9.1 Verify at least one CCW PUMP in on service train - STARTED.

[] Train A (1C or 1B)
[] Train B (1A or 1B)

2.9.2 Verify flow indicated in the On-Service train.

HX 1A(1B,1C)
CCW FLOW
[] FI 3043AA
[] FI 3043BA
[] FI 3043CA

2.9.3 Verify CCW to RCP thermal barriers - ALIGNED.

CCW TO
SECONDARY HXS
[] Q1P17MOV3047 open

CCW TO
RCP CLRS
[] Q1P17MOV3052 open

CCW FROM
RCP THRM BARR
[] Q1P17HV3184 open
[] Q1P17HV3045 open

2.10 Isolate RCP seal injection.

2.10.1 Close BKR FEG3. (100 ft, AUX BLDG lower equipment room)

2.10.2 Close seal water injection isolation valve.

RCP SEAL WTR
INJ ISO
[] Q1E21MOV8105

Step 2 continued on next page.

1. W/E05EG2.4.2 071/MOD/FNP EXAM BANK/C/A 4.5/4.6/W/E05EG2.4.2/N///

Unit 1 was operating at 100% power when a Reactor Trip and SI occurred due to a steam line break in containment. The following conditions exist:

- The operating crew is performing the actions of EEP-2.0, Faulted Steam Generator Isolation.
- The maximum total AFW flow rate that can be achieved is 350 GPM.
- Containment pressure is 6 psig and falling.
- SG Narrow range levels are:
  - 1A - Off Scale Low
  - 1B - 32% and decreasing slowly
  - 1C - 34% and decreasing slowly

Which one of the following completes the statement below?

Secondary heat sink _(1)_ adequate because _(2)_.

A. 1) IS

2) 1B and 1C SG levels are sufficient

B.✓ 1) is NOT

2) Neither SG levels nor AFW flow capability is sufficient

C. 1) IS

2) AFW flow capability is sufficient

D. 1) is NOT

2) 1B and 1C SG levels are sufficient but AFW flow capability is NOT sufficient

CSF-0/0.3 Heat Sink: To have adequate heat sink -

SG Narrow Range levels in at least ONE SG greater than 31%{48%}

OR

Total AFW to all SG's > 395 gpm

Distracter analysis

A. Incorrect. See B. Plausible if the applicant does not recognize that adverse numbers apply in this scenario then this would be correct.

B. Correct. One SG NR level must be >48% OR AFW flow must be >395 gpm to satisfy the heat sink criteria.

C. Incorrect. See B. Plausible if the applicant does not recall the minimum AFW flow required to meet heat sink and confuses it with 350 gpm which is the design flow rate of one AFW pump.

D. Incorrect. See B. Plausible if the applicant believes that BOTH AFW flow and SG NR levels are required to meet heat sink and does not recognize that adverse numbers apply in this scenario.

K/A: **W/E05EG2.4.2** Loss of Secondary Heat Sink - Knowledge of **system set points**, interlocks and automatic actions **associated with EOP entry conditions**.

Importance Rating: 4.5 4.6

Technical Reference: FNP-1-CSF-0, Critical Safety Function Status Trees, Ver 17

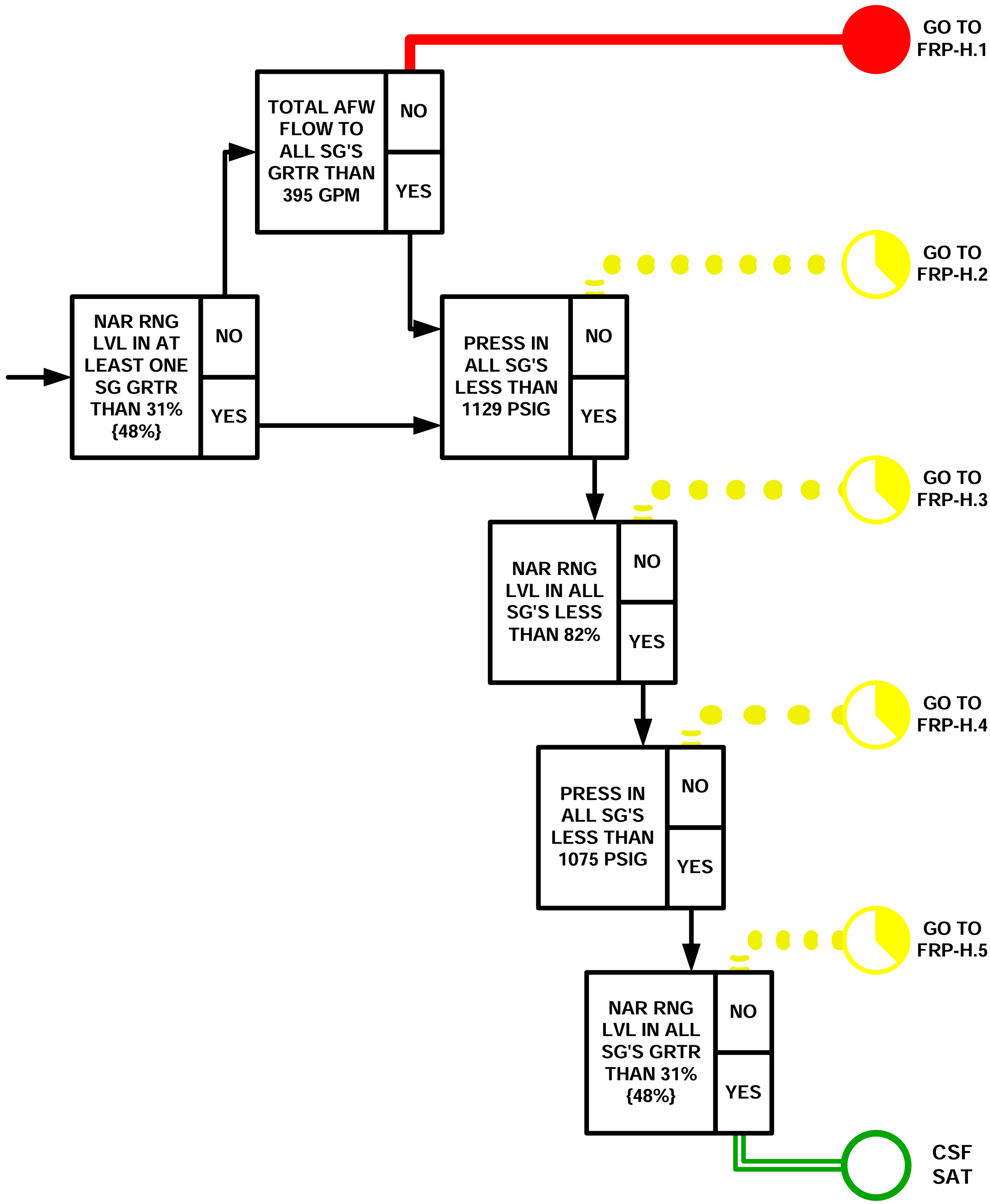
References provided: None

Learning Objective: EVALUATE plant conditions and DETERMINE if entry into (1) FRP-H.1, Response to Loss of Secondary Heat Sink; [..] is required. (OPS-52533F02)

Question History: MOD FNP EXAM BANK

K/A match: Requires the applicant to know the **setpoints of CSF-0** and recognize that heat sink does not exist and the **setpoints which are met for entry into FRP-H.1, Response to Loss of Secondary Heat Sink.**

SRO justification: N/A


NAR RNG LVL IN AT LEAST ONE SG GRTR THAN 31% {48%}
NO
YES
TOTAL AFW FLOW TO ALL SG'S GRTR THAN 395 GPM
NO
YES
GO TO FRP-H.1
PRESS IN ALL SG'S LESS THAN 1129 PSIG
NO
YES
GO TO FRP-H.2
NAR RNG LVL IN ALL SG'S LESS THAN 82%
NO
YES
GO TO FRP-H.3
PRESS IN ALL SG'S LESS THAN 1075 PSIG
NO
YES
GO TO FRP-H.4
NAR RNG LVL IN ALL SG'S GRTR THAN 31% {48%}
NO
YES
GO TO FRP-H.5
CSF SAT

1. FRP-H-52533F02 002/HLT/LOCT//C/A 3.4/4.4/W/E05EA2.1///LOCT/

A Unit 1 Reactor Trip with an SI has occurred due to a steam line break downstream of the MSIVs. The following conditions exist:

- BOTH MSIVs on 1A SG are OPEN and will **NOT** close.
- The maximum total AFW flow rate that can be achieved is 200 GPM.

	1A SG	1B SG	1C SG
• SG narrow range levels are:	offscale low	28%	34%

Which one of the following describes the impact of this configuration on secondary heat removal capability?

Secondary heat sink __(1)__ adequate because __(2)__.

	(1)	(2)
A.✓	is	ONLY SG level(s) is(are) sufficient
B.	is **NOT**	NEITHER SG level(s) NOR AFW flow is sufficient
C.	is	BOTH AFW flow capability AND SG level(s) is(are) sufficient.
D.	is **NOT**	AFW flow capability is NOT sufficient

QUESTIONS REPORT

for ILT 36 RO NRC Exam Version 6

1. W/E08EG2.4.6 072/BANK/FNP 08/C/A 3.7/4.7/W/E08EG2.4.6/N///

Unit 2 has experienced a large steam break inside containment and the following conditions exist:

- RCS cold leg temperature has decreased to 250°F in the past 30 minutes and is decreasing slowly.
- RCS pressure is 1500 psig.

Which one of the following completes the statements below?

The RCS cooldown must __(1)__.

An RCS depressurization __(2)__ required.

	(1)	(2)
A.	continue	is NOT
B.	continue	IS
C.	be stopped	is NOT
D.✓	be stopped	IS

FRP-P.1:

Step 3: Check cold leg temperature STABLE OR RISING. | 3. [CA] Stop the cooldown.

Step 18: Reduce RCS pressure.

FRB-P.1

Step 3: [..] It is important to terminate, if possible, any cooldown in progress to limit the extent of possible vessel damage due to excessive thermal stresses. [...]

Step 18: The RCS pressure reduction is intended to decrease pressure stress on the vessel wall as much as possible. [...]

Distracter analysis

A. Incorrect. First part is incorrect (See D.1). Plausible since a cooldown will be performed after the 1 hour soak but the overall strategy of FRP-P.1 is stop the cooldown.

Second part is incorrect (See D.1). Plausible since FRP-P.1 requires the RCS pressure stable for the soak and the applicant could confuse this with current conditions in the stem.

B. Incorrect. First part is incorrect (See A.1).

Second part is correct (See D.2).

C. Incorrect. First part is correct (See D.2).

Second part is incorrect (See A.2)

D. Correct. First part is correct. Per the background document - [...] It is important to terminate, if possible, any cooldown in progress to limit the extent of possible vessel damage due to excessive thermal stresses.

Second part is correct. Per the background document - The RCS pressure reduction is intended to decrease pressure stress on the vessel wall as much as possible.

K/A: **W/E08EG2.4.6**	Pressurized Thermal Shock - Knowledge of EOP mitigation strategies.
Importance Rating:	3.7 4.7
Technical Reference:	FNP-2-FRP-P.1, Response to Imminent Pressurized Thermal Shock Conditions, Ver 23 FNP-0-FRB-P.1, Specific Background Document For FNP1/2-FRP-P.1, Ver 2
References provided:	NONE
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with (1) FRP-P.1, Response to Imminent Pressurized Thermal Shock Condition; [...] (OPS-52533K03) EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing (1) FRP-P.1, Response to Imminent Pressurized Thermal Shock Condition; [...] (OPS-52533K06).
Question History:	FNP 08
K/A match:	Requires the applicant to **recognize that the FRP-P.1 entry requirements have been met. Also, they must know the mitigation strategy of FRP-P.1, Response to Imminent Pressurized Thermal Shock Conditions, and select the appropriate actions to take under the given conditions.**
SRO justification:	N/A

Section: **Procedure**

Unit 1 ERP Step: 3 **Unit 2 ERP Step:** 3 ***ERG Step No:*** *2*

ERP StepText: Check RCS cold leg temperatures - STABLE OR RISING.

ERG StepText: *Check RCS Cold Leg Temperatures - STABLE OR INCREASING*

Purpose: To determine if RCS cold leg temperature is still decreasing and, if so, to attempt to stop the decrease

Basis: Cold leg temperature is the best available indication of vessel downcomer temperature. It is important to terminate, if possible, any cooldown in progress to limit the extent of possible vessel damage due to excessive thermal stresses. If the RCS cold leg temperatures are decreasing the operator is instructed to eliminate any secondary-side or RHR System instigated RCS cooldown. The items checked in this step are in a preferred order such that the most probable causes of the cooldown are checked first. Therefore, any valves that dump steam are verified to be closed. Next, any cooldown from the RHR System terminated. A cooldown caused by overfeeding the intact SGs is stopped by controlling feed flow consistent with minimum secondary heat sink requirements. The operator checks for any faulted SGs and isolates them. Finally, if a faulted SG is necessary for RCS temperature control or if all SGs are faulted, feed flow to those SGs is controlled at a minimum measurable value to minimize the effects of the RCS cooldown due to the secondary side depressurization.

Knowledge: Thermal stresses caused by rapid RCS temperature transients are not immediately removed if cooldown is stopped or RCS heatup occurs. Therefore, additional steps to minimize RCS pressure and to perform an RCS temperature "soak" are necessary.

References:

Justification of Differences:

1 Changed to make plant specific.

2 Isolation of a faulted SG is presented in an attachment. Use of an attachment facilitates delegation to personnel as resources are available while enhancing procedure flow.

Section: **Procedure**

Unit 1 ERP Step: 18 **Unit 2 ERP Step:** 18 ***ERG Step No:*** *16*

ERP StepText: Reduce RCS pressure.

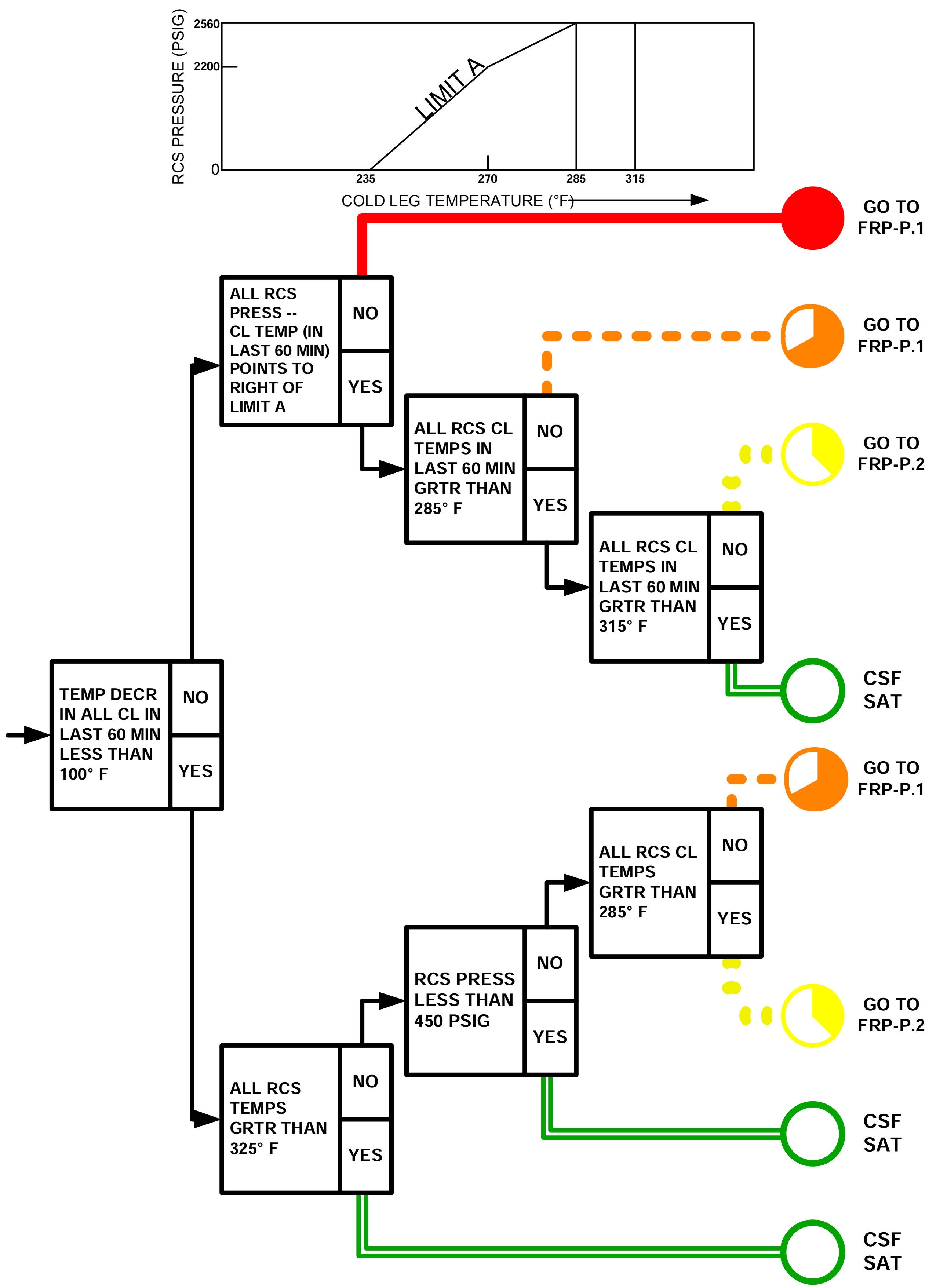
ERG StepText: *Depressurize RCS To Decrease RCS Subcooling*

Purpose: To decrease RCS pressure to the lowest pressure possible without losing subcooling

Basis: The RCS pressure reduction is intended to decrease pressure stress on the vessel wall as much as possible. The RCS should be depressurized until RCS subcooling is less than (R.08)°F [(R.09)°F for adverse containment]. If a PORV is used and RCS subcooling decreases to less than (R.01)°F [(R.02)°F for adverse containment] before a PORV is closed or isolated, the operator should allow adequate time for the PORV or its associated block valve to close (i.e., the time necessary for the valve to stroke) before manually operating SI pumps as necessary to restore subcooling per the previous continuous action step. If normal PRZR spray is not available, and the RCS cannot be depressurized using any PRZR PORV, then the operator is instructed to use auxiliary spray. This preferred order of the means to depressurize the RCS takes into account that letdown has not been established yet to heat the auxiliary spray flow and minimize the thermal shock to the spray nozzle. Once letdown has been established, using auxiliary spray for depressurization is preferred before using a PRZR PORV. If the operator is directed to return to this step after letdown has been established, and normal PRZR spray is not available, auxiliary spray should be used for depressurization. A second criterion, in addition to subcooling, for stopping the pressure reduction is PRZR level greater than (D.08)% [(D.09)% for adverse containment]. Limiting PRZR level ensures a substantial steam bubble which facilitates further pressure control. A third criterion for stopping the pressure reduction is RCS pressure less than (B.10) psig [(B.11) psig for adverse containment]. For certain postulated accidents, it is possible to enter FR_P.1, Step 16 with a low RCS pressure (less than approximately 200 psig) and greater than the required 10°F subcooling. It may be difficult to reduce RCS pressure any further per the RNO column. Since the intent of the step has been met, no further pressure reduction is necessary.

Knowledge: 1. RCS depressurization should be stopped when RCS subcooling based on core exit TCs is between (R.01)°F [(R.02)°F for adverse containment] and (R.08)°F [(R.09)°F for adverse containment]. 2. It is possible that this pressure reduction could result in loss of the normal conditions for RCP operation, i.e., minimum number one seal differential pressure or minimum number one seal leakoff flow. If either condition is lost, the affected RCP should be stopped. This action is not specifically included in this step, but is inferred by the pressure reduction. 3. If subcooling decreases below the setpoint for reinitiating SI during the depressurization, the operator should take the appropriate actions such as closing the PORV or the block valve for a stuck open PORV, and wait to see if the actions are successful (i.e., allow adequate time for valves to stroke closed), before reinitiating SI. If the actions stop the depressurization and subcooling is restored, SI reinitiation is not necessary.

INTEGRITY


2560
2200
0
RCS PRESSURE (PSIG)
LIMIT A
235
270
285
315
COLD LEG TEMPERATURE (°F)
TEMP DECR IN ALL CL IN LAST 60 MIN LESS THAN 100° F
NO
YES
ALL RCS PRESS -- CL TEMP (IN LAST 60 MIN) POINTS TO RIGHT OF LIMIT A
NO
YES
GO TO FRP-P.1
ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 285° F
NO
YES
GO TO FRP-P.1
ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 315° F
NO
YES
GO TO FRP-P.2
CSF SAT
ALL RCS TEMPS GRTR THAN 325° F
NO
YES
RCS PRESS LESS THAN 450 PSIG
NO
YES
ALL RCS CL TEMPS GRTR THAN 285° F
NO
YES
GO TO FRP-P.1
GO TO FRP-P.2
CSF SAT
CSF SAT

Step	Action/Expected Response	Response NOT Obtained

4 Isolate all faulted SGs.

4.1 Verify all faulted SG atmospheric reliefs - MANUALLY CLOSED.

Faulted SG	2A	2B	2C
2A(2B,2C) MS ATMOS REL VLV PC	[]3371A	[]3371B	[]3371C

4.1 Locally unlock and close one isolation valve for any failed atmospheric relief. (127 ft, AUX BLDG main steam valve room)

Faulted SG	2A	2B	2C
Q2N11V	[]004A []004B	[]004C []004D	[]004E []004F
Key	Z-479 V-99	V-100 V-101	V-153 V-103

4.2 Verify all faulted SG main feed stop valves - CLOSED.

Faulted SG	2A	2B	2C
MAIN FW TO 2A(2B,2C) SG STOP VLV Q2N21MOV	[]3232A	[]3232B	[]3232C

4.2 Locally close all faulted SG main feed stop valves. (127 ft, AUX BLDG main steam valve room)

Faulted SG	2A	2B	2C
MAIN FW TO 2A(2B,2C) SG STOP VLV Q2N21MOV	[]3232A	[]3232B	[]3232C

4.3 Verify blowdown from all faulted SGs - ISOLATED.

Faulted SG	2A	2B	2C
2A(2B,2C) SGBD ISO Q2G24HV	[]7614A closed	[]7614B closed	[]7614C closed

4.3 Locally isolate blowdown. (121 ft, AUX BLDG rad side at PRIP)

Faulted SG	2A	2B	2C
2A(2B,2C) SGBD PENE RM ISO Q2G24HV	[]7697A stopped []7697B stopped	[]7698A stopped []7698B stopped	[]7699A stopped []7699B stopped

Step 4 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Check RCS cold leg temperatures - STABLE OR RISING.** RCS COLD LEG TEMP [] TR 410	3 [CA] Stop cooldown. 3.1 Verify atmospheric reliefs closed. 2A(2B,2C) MS ATMOS REL VLV [] PC 3371A [] PC 3371B [] PC 3371C 3.2 Verify steam dumps closed. STM DUMP INTERLOCK [] A TRN in OFF RESET [] B TRN in OFF RESET 3.3 IF RHR system in service in cooldown mode, THEN adjust RHR heat exchanger flow to stop cooldown. 2A(2B) RHR HX BYP FLOW [] FK 605A [] FK 605B 2A(2B) RHR HX DISCH VLV [] HIK 603A [] HIK 603B

Step 3 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

___15 **Check RCS hot leg temperatures.**

15.1 Check RCS hot leg temperatures - STABLE OR RISING.

RCS HOT LEG TEMP
[] TR 413

RESPONSE NOT OBTAINED: 15.1 WHEN actions of step 3 complete, THEN proceed to step 16.

15.2 [CA] Maintain RCS hot legs temperatures - STABLE AT EXISTING TEMPERATURES.

15.2.1 Control MDAFWP flow to intact SGs.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
2A/2B/2C SG
B TRN
[] FCV 3227 in MOD

Intact SG	2A	2B	2C
MDAFWP TO 2A(2B,2C) SG Q2N23HV	[] 3227A in MOD	[] 3227B in MOD	[] 3227C in MOD
MDAFWP TO 2A(2B,2C) SG FLOW CONT HIC	[] 3227AA adjusted	[] 3227BA adjusted	[] 3227CA adjusted

Step 15 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: The PRT may rupture causing abnormal containment conditions while using pressurizer PORVs.

****

****

CAUTION: To prevent pressurizer PORV failure, cycling of pressurizer PORVs should be minimized.

****

****

CAUTION: The following step may result in a loss of minimum No. 1 seal differential pressure or minimum No. 1 seal leakoff flow. If either support condition is lost, the associated RCPs should be stopped.

****

NOTE: Reactor vessel upper head voiding may occur during pressure reduction while on natural circulation, causing a rapid rise in pressurizer level.

___ **18** **Reduce RCS pressure.**

18.1 Begin pressure reduction.

18.2 WHEN SUBCOOLED MARGIN MONITOR indication less than 26°F{55°F} subcooled in CETC mode

OR

pressurizer level greater than 73%{66%}

OR

RCS pressure less than 150 psig{200 psig},

THEN stop RCS pressure reduction.

Step 18 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
___25	**[CA] Maintain RCS pressure - STABLE.**	

****

CAUTION: Excessive reactor vessel stress may result from raising RCS pressure. RCS temperature and pressure should be maintained stable until completion of any required soak.

****

25.1 IF normal pressurizer spray available, THEN control normal pressurizer spray to maintain RCS pressure stable.

2A(2B) LOOP
SPRAY VLV
[] PK 444C
[] PK 444D

RNO:

25.1 IF normal letdown established, THEN control auxiliary spray to maintain RCS pressure stable.

25.1.1 Manually open both normal pressurizer spray valves.

2A(2B) LOOP
SPRAY VLV
[] PK 444C
[] PK 444D

25.1.2 Open auxiliary spray valve.

RCS PRZR
AUX SPRAY
[] Q2E21HV8145

25.1.3 Verify both charging line valves closed.

RCS NORMAL
CHG LINE
[] Q2E21HV8146

RCS ALT
CHG LINE
[] Q2E21HV8147

Step 25 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
		25.1.4 WHEN RCS pressure reduction required, THEN operate the following valves as required to control pressurizer pressure. CHG FLOW [] FK 122 manually open 2A(2B) LOOP SPRAY VLV [] PK 444C manually open/closed [] PK 444D manually open/closed RCS PRZR AUX SPRAY [] Q2E21HV8145 open/closed RCS NORMAL CHG LINE [] Q2E21HV8146 open/closed RCS ALT CHG LINE [] Q2E21HV8147 open/closed
25.2	IF normal pressurizer spray NOT available AND normal letdown NOT established, THEN control only one PRZR PORV to maintain RCS pressure stable.	
25.3	Control pressurizer heaters to maintain RCS pressure stable. PRZR HTR GROUP VARIABLE [] 2C PRZR HTR GROUP BACKUP [] 2A [] 2B [] 2D [] 2E	

Step	Action/Expected Response	Response NOT Obtained
___26	**Verify adequate RCS pressure reduction.** • SUB COOLED MARGIN MONITOR indication - LESS THAN OR EQUAL TO 26°F{55°F} SUBCOOLED IN CETC MODE. OR • RCS pressure - LESS THAN 150 psig{200 psig}	26 Return to Step 17.
___27	**Determine if RCS soak required.**	
27.1	Check RCS cold leg cooldown - GREATER THAN 100°F IN ANY 60 MINUTE PERIOD. RCS COLD LEG TEMP [] TR 410	27.1 Go to procedure and step in effect.
___28	**Establish RCS soak.**	
28.1	[CA] Maintain RCS temperature and pressure stable for 1 hour. RCS COLD LEG TEMP [] TR 410 2C(2A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	
28.2	IF RCS soak will not be affected, THEN perform actions of other procedures in effect.	

Step 28 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
28.3	WHEN RCS soak complete, THEN perform the following.	
28.3.1	[CA] IF cooldown required, THEN maintain RCS parameters within the limits of FIGURE 3{FIGURE 4}.	
28.3.2	[CA] IF cooldown required, THEN maintain RCS cold leg cooldown rate less than 50°F in any 60 minute period. RCS COLD LEG TEMP [] TR 410	
___29	**Go to procedure and step in effect.**	

-END-

1. W/E11EK2.1 073/BANK/FNP 10/C/A 3.6/3.9/W/E11EK2.1/N///

A Dual Unit LOSP with a LOCA on Unit 1 has occurred and the following conditions exist:

- EEP-1.0, Loss of Reactor or Secondary Coolant, is in progress.

At 1000:

- WA2, 1-2A DG GEN FAULT TRIP, comes into alarm.

At 1015:

- The following alarms are received:
  - CF3, 1A OR 1B RHR PUMP OVERLOAD TRIP
  - CH2, RWST LVL A TRN LO
  - CH3, RWST LVL B TRN LO

Which one of the following states:

1) the status of Unit 1 emergency recirculation capability

and

2) the action(s) that the applicable procedure(s) direct?

A. 1) One train ONLY of emergency recirculation capability has been lost.
2) Transfer to Cold Leg **AND** Containment Spray recirculation at this time.

B. 1) One train ONLY of emergency recirculation capability has been lost.
2) Transfer to Cold Leg recirculation ONLY.

C.✓ 1) Both trains of emergency recirculation capability have been lost.
2) Minimize HHSI flow to the minimum required to remove decay heat while attempting to restore at least one train of emergency recirculation.

D. 1) Both trains of emergency recirculation capability have been lost.
2) Secure HHSI pumps while attempting to restore at least one train of emergency recirculation.

This is not a true 2+2 question to improve distracter plausibility.

EEP-1:

13.1 Verify cold leg recirculation capability - AVAILABLE.	13.1 IF cold leg recirculation capability can NOT be verified, THEN go to FNP-2-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

13.1.1 Train A equipment available:
- 2A RHR Pump
- CTMT SUMP TO 2A RHR PUMP
Q2E11MOV8811A
- CTMT SUMP TO 2A RHR PUMP
Q2E11MOV8812A
- 2A RHR HX TO CHG PUMP
SUCT Q2E11MOV8706A
- CCW TO 2A RHR HX
Q2P17MOV3185A
OR

13.1.2 Train B equipment
available:
- 2B RHR Pump
- CTMT SUMP TO 2B RHR PUMP
Q2E11MOV8811B
- CTMT SUMP TO 2B RHR PUMP
Q2E11MOV8812B
- 2B RHR HX TO CHG PUMP
SUCT Q2E11MOV8706B
- CCW TO 2B RHR HX
Q2P17MOV3185B

ECP-1.1
Purpose - This procedure provides actions to restore emergency coolant recirculation capability, to delay depletion of the RWST by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow.

Distracter Analysis

A. Incorrect. First part is incorrect (See C.1). Plausible if the applicant doesn't recognize that the DG trip results in the loss of the 1A RHR pump.

Second part is incorrect (See C.2). Plausible if the applicant thinks that one train of recirc capability is available because this would be partially correct. The containment spray is not transferred to sump recirc until the RWST is less than 4.5 ft. The applicant could be unfamiliar with the procedure and believe that both cold leg and containment spray are required to be transferred to sump recirc when RWST is at 12.5 ft.

B. Incorrect. First part is incorrect (See A.1)

Second part is incorrect (See C.2). Plausible since this would be the correct answer if recirc capability existed.

C. Correct. First part is correct. Since neither RHR pumps are available so

there is no recirculation capability.

Second part is correct. This is the correct strategy for ECP-1.1.

D. Incorrect. First part is correct (See C.1).

Second part is correct (See C.2). Plausible if the applicant recognizes that recirculation capability is lost but incorrectly believes that CH2 and CH3 being in alarm indicates that the RWST is less than 4.5 ft which would make this the correct answer per step 34 of ECP-1.1.

K/A: **W/E11EK2.1** Loss of Emergency Coolant Recirculation - Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Importance Rating: 3.6 3.9

Technical Reference: FNP-1-EEP-1.0, Loss of Reactor or Secondary Coolant, Ver 31.
FNP-1-ECP-1.1, Loss of Emergency Coolant Recirculation, Ver 30

References provided: None

Learning Objective: EVALUATE plant conditions and DETERMINE if entry into (1) ECP-1.1, Loss of Emergency Coolant Recirculation; and/or (2) ECP-1.3, Loss of Emergency Coolant Recirculation, Caused by Sump Blockage is required. (OPS-52532D02)

Question History: FNP 10

K/A match: Applicant is required to **know the interrelation between failure modes of the RHR pumps and the Loss of Emergency Coolant Recirculation procedure.**

SRO justification: N/A

LOCATION CH2

SETPOINT: 12′7″ ± 1″ above Tank Bottom (150,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT-501 through a comparator card bistable designated LSL503 in BOP Cabinet J.

H2
RWST LVL A TRN LO

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. IF an ECCS actuation signal is present, THEN refer to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION.
2. Determine actual tank level as indicated by LI-4075A & B, on the MCB OR the local level indicator on the side of the RWST.
3. IF an ECCS Actuation Signal is NOT present OR the tank is NOT being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. IF required, THEN restore RWST level to normal per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 177; A-170750, Pg. 95; D-173497; Technical Specifications

LOCATION CH3

H3
RWST LVL B TRN LO

SETPOINT: 12′7″ ± 1″ above Tank Bottom (150,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT502 through a comparator card bistable designated LSL504 in BOP Cabinet K.

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. If an ECCS actuation signal is present, THEN refer to FNP-1-ESP-1.3. TRANSFER TO COLD LEG RECIRCULATION.
2. Determine actual tank level as indicated by LI-4075A & B, on the MCB OR the local level indicator on the side of the RWST.
3. IF an ECCS Actuation signal is NOT present OR the tank is NOT being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. IF required, THEN restore RWST level to normal per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 178; A-170750, Pg. 95; B-170058, Sh. 72; D-173497; Technical Specifications

LOCATION CH4

H4
RWST LVL A TRN LO-LO

SETPOINT: 4′7″ ± 1″ above Tank Bottom (50,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT501 through a comparator card bistable designated LSL505 in BOP Cabinet J.

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NOTE: The automatic opening of the containment sump to LHSI valves uses separate RWST level switches. These switches (Q1F16LS507 for Q1E11MOV8811A and Q1F16LS508 for Q1E11MOV8812A) are set at the same setpoint as this alarm. Since different level switches accomplish the valve opening action, failure of the instrumentation associated with this alarm would not affect the valve opening function.

1. IF `S' Signal present, THEN ECCS valve Switchover occurs.

OPERATOR ACTION

1. IF an ECCS actuation signal is present, THEN refer to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION, for transfer of containment spray to sump recirculation.
2. Determine actual tank level as indicated BY LI-4075A & B, on the MCB.
3. IF an ECCS Actuation signal is not present OR the tank is not being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. Restore RWST level to normal, IF required per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 179; A-170750, Pg 95a; D-173497; B-170058, Sh. 72;Technical Specifications

LOCATION CH5

SETPOINT: 4′7″ ± 1″ above Tank Bottom (50,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT502 through a comparator card bistable designated LSL506 in BOP Cabinet K.

H5
RWST LVL B TRN LO-LO

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NOTE: The automatic opening of the containment sump to LHSI valves uses separate RWST level switches. These switches (Q1F16LS515 for Q1E11MOV8811B and Q1F16LS516 for Q1E11MOV8812B) are set at the same setpoint as this alarm. Since different level switches accomplish the valve opening action, failure of the instrumentation associated with this alarm would not affect the valve opening function.

1. IF ‘S’ signal present, THEN ECCS valve Switchover occurs.

OPERATOR ACTION

1. IF an ECCS actuation signal is present, THEN refer TO FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION, for transfer of containment spray to sump recirculation.
2. Determine actual tank level as indicated by LI-4075A & B, on the MCB.
3. IF an ECCS Actuation signal is not present OR the tank is not being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. Restore RWST level to normal, IF required per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 180; A-170750, Pg. 95a; D-173497; B-170058, Sh. 72; Technical Specification

A. **Purpose**

This procedure provides actions to restore emergency coolant recirculation capability, to delay depletion of the RWST by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow.

B. **Symptoms or Entry Conditions**

I. This procedure is entered when emergency coolant recirculation capability is lost; from the following:

a. FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 13, when cold leg recirculation capability cannot be verified.

b. FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION, step 7, when at least one flow path from the containment sump cannot be established or maintained.

c. FNP-1-ECP-1.2, LOCA OUTSIDE CONTAINMENT, step 2, when a LOCA outside containment cannot be isolated.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: SI or spray pump damage will occur if suction is lost and the pump is not secured.

****

NOTE:
- IF both trains of RHR have lost emergency coolant recirculation capability AND ECCS sump level is approximately 4.6 ft or less, THEN the loss may be due to insufficient NPSH or air entrainment (vortexing) due to the low ECCS sump level.
- Erratic pump parameters (flow, discharge pressure, amps, etc.) are indications of pump cavitation.
- Step 1 is a continuing action.

___ 1 **Verify ECCS pumps not affected by sump blockage.**

1.1 [CA] Monitor ECCS pump suction conditions - NO INDICATION OF CAVITATION.

CHG PUMP
- [] 1A
- [] 1B
- [] 1C

RHR PUMP
- [] 1A
- [] 1B

CS PUMP
- [] 1A
- [] 1B

1 (RNO) IF both trains are affected such that at least one train of SI recirculation flow cannot be maintained, THEN go to FNP-1-ECP-1.3, LOSS OF EMERGENCY COOLANT RECIRCULATION CAUSED BY SUMP BLOCKAGE.

___ 2 **[CA] WHEN emergency coolant recirculation capability is restored, THEN go to procedure and step in effect.**

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Check cold leg recirculation equipment - AVAILABLE.** 3.1 Train A equipment available: • 1A RHR Pump • CTMT SUMP TO 1A RHR PUMP Q1E11MOV8811A • CTMT SUMP TO 1A RHR PUMP Q1E11MOV8812A • 1A RHR HX TO CHG PUMP SUCT Q1E11MOV8706A • CCW TO 1A RHR HX Q1P17MOV3185A <u>OR</u> 3.2 Train B equipment available: • 1B RHR Pump • CTMT SUMP TO 1B RHR PUMP Q1E11MOV8811B • CTMT SUMP TO 1B RHR PUMP Q1E11MOV8812B • 1B RHR HX TO CHG PUMP SUCT Q1E11MOV8706B • CCW TO 1B RHR HX Q1P17MOV3185B	3 Perform the following. a) [CA] Continue attempts to restore at least one train of recirculation equipment. b) Proceed to Step 4.
___ 4	**Verify SI - RESET.** [] MLB-1 1-1 off (A TRN) [] MLB-1 11-1 off (B TRN)	4 Perform the following: 4.1 <u>IF</u> any train will <u>NOT</u> reset using the MCB SI RESET pushbuttons, <u>THEN</u> place the affected train S821 RESET switch to RESET. (SSPS TEST CAB.) 4.2 <u>IF</u> a failure exists in SSPS such that SI cannot be reset, <u>THEN</u> reset SI using FNP-1-SOP-40.0, RESPONSE TO INADVERTENT SI <u>AND</u> INABILITY TO RESET <u>OR</u> BLOCK SI, Appendix 2.
___ 5	**Check PHASE B CTMT ISO - RESET.** [] MLB-3 1-1 not lit [] MLB-3 6-1 not lit	5 Reset PHASE B CTMT ISO.

Step	Action/Expected Response	Response NOT Obtained

NOTE: The intent of step 17 is to establish some (minimum) SI flow from the RWST (to delay emptying the RWST) concurrent with efforts to restore recirculation capability.

___17	**Establish one train of ECCS equipment.**	
	17.1 Check RCS pressure - LESS THAN 275 psig{435 psig}. 1C(1A) LOOP RCS NR PRESS [] PI 402B [] PI 403B	17.1 Perform the following. 17.1.1 IF started CHG PUMP suction aligned to RHR, THEN stop affected CHG PUMP. 17.1.2 Stop both RHR PUMPs. 17.1.3 Close RHR to charging pump suction valves. 1A(1B) RHR HX TO CHG PUMP SUCT [] Q1E11MOV8706A [] Q1E11MOV8706B 17.1.4 Open RWST to charging pump valves. RWST TO CHG PUMP [] Q1E21LCV115B [] Q1E21LCV115D 17.1.5 Establish only one CHG PUMP started. 17.1.6 Proceed to step 18.

Step 17 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 21 is a continuing action.

___21	**[CA] Check SI termination criteria.**	
21.1	Check REACTOR VESSEL LEVEL. • IF any RCP started, THEN check REACTOR VESSEL LEVEL greater than 72% UPPER PLENUM. OR • IF no RCP started, THEN check REACTOR VESSEL LEVEL greater than 0% UPPER PLENUM.	21.1 Proceed to step 27.

Step 21 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

NOTE: TABLE 1 provides minimum SI flow to remove decay heat vs. time elapsed after shutdown.

Step	Action/Expected Response	Response NOT Obtained
21.2	Check SUB COOLED MARGIN MONITOR - GREATER THAN 66°F {95°F} SUBCOOLED IN CETC MODE.	21.2 Establish minimum SI flow. 21.2.1 IF charging pump suction aligned to RHR, THEN stop all CHG PUMPs. 21.2.2 Verify both RHR PUMPs stopped. 21.2.3 Open miniflow valve for available charging pump. 1A(1B,1C) CHG PUMP MINIFLOW ISO [] Q1E21MOV8109A [] Q1E21MOV8109B [] Q1E21MOV8109C 21.2.4 Open common miniflow isolation valve. CHG PUMP MINIFLOW ISO [] Q1E21MOV8106 21.2.5 Verify RWST to charging pump valves open. RWST TO CHG PUMP [] Q1E21LCV115B [] Q1E21LCV115D 21.2.6 Close RHR supply to A AND B train charging pump suction. 1A(1B) RHR HX TO CHG PUMP SUCT [] Q1E11MOV8706A [] Q1E11MOV8706B

Step 21 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

21.2.7 Start CHG PUMP with miniflow valve open.

21.2.8 Maintain core exit T/C temperature stable or falling.

- IF core exit T/C temperatures rising AND started charging pump aligned to A train, THEN establish A train SI flow.

  HHSI TO
  RCS CL ISO
  [] Q1E21MOV8803A open
  [] Q1E21MOV8803B open

- IF core exit T/C temperatures rising AND started charging pump aligned to B train, THEN establish B train SI flow.

  CHG PUMP RECIRC
  TO RCS COLD LEGS
  [] Q1E21MOV8885 open

21.2.9 Open and close HHSI isolation valves to control SI flow to keep core exit T/C temperatures stable or falling.

HHSI TO
RCS CL ISO
[] Q1E21MOV8803A
[] Q1E21MOV8803B

CHG PUMP RECIRC
TO RCS COLD LEGS
[] Q1E21MOV8885

Step 21 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 27 is a continuing action.

___27 **[CA] Verify RCS makeup flow - ADEQUATE.**

27.1 Check REACTOR VESSEL LEVEL.

- IF any RCP started, THEN check REACTOR VESSEL LEVEL greater than 72% UPPER PLENUM.

OR

- IF no RCP started, THEN check REACTOR VESSEL LEVEL greater than 0% UPPER PLENUM.

Response NOT Obtained:

27.1 Increase RCS makeup flow to maintain required REACTOR VESSEL LEVEL.

- CHG FLOW

[] FK 122 adjusted

OR

- HHSI TO RCS CL ISO

[] Q1E21MOV8803A open
[] Q1E21MOV8803B open

OR

- CHG PUMP RECIRC TO RCS COLD LEGS

[] Q1E21MOV8885 open

Step 27 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
27.2	Check core exit T/Cs stable or falling.	27.2 Increase RCS makeup flow to maintain core exit T/Cs stable or falling. • CHG FLOW [] FK 122 adjusted OR • HHSI TO RCS CL ISO [] Q1E21MOV8803A open [] Q1E21MOV8803B open OR • CHG PUMP RECIRC TO RCS COLD LEGS [] Q1E21MOV8885 open

If SI termination criteria is nor met, some form of makeup is established to remove decay heat.

Step	Action/Expected Response	Response NOT Obtained
___**31**	**[CA] Monitor RCP support conditions.**	31 IF any support condition NOT satisfied, THEN stop affected RCP(s).
31.1	Verify seal leakoff flow - WITHIN FIGURE 1 LIMITS.	
31.2	Verify No. 1 seal differential pressure - GREATER THAN 200 psid. #1 SEAL PRESS [] 1A RCP - PI 156A [] 1B RCP - PI 155A [] 1C RCP - PI 154A	
31.3	Check RCS - WITHIN FIGURE 2, RCP PRESSURE-TEMPERATURE LIMITS.	
___**32**	**Check RCS temperatures - GREATER THAN 200°F.** RCS HOT LEG TEMP [] TR 413 RCS COLD LEG TEMP [] TR 410	32 Proceed to step 42.
___**33**	**Check RWST level - LESS THAN 4.5 ft.** RWST LVL [] LI 4075A [] LI 4075B	33 Return to step 3.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: The remainder of this procedure should only be performed if RWST level is less than 4.5 ft and cold leg recirculation is not available.

****

___34 **Stop all safeguards pumps. taking suction from the RWST.**

CHG PUMP
[] 1A
[] 1B
[] 1C

RHR PUMP
[] 1A
[] 1B

CS PUMP
[] 1A
[] 1B

___35 **[CA] Establish makeup to RCS from any available source.**

35.1 Consult TSC staff for alternate method of RCS makeup such as normal makeup.

OR

Step 35 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

NOTE: Unless a known problem exists with components required for cold leg recirculation or their power supplies, it is assumed cold leg recirculation capability is available. Transition to FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, however, should be made upon discovery of inability to establish at least one train of recirculation.

13.1 Verify cold leg recirculation capability - AVAILABLE.

13.1.1 Train A equipment available:

- 1A RHR Pump
- CTMT SUMP TO 1A RHR PUMP Q1E11MOV8811A
- CTMT SUMP TO 1A RHR PUMP Q1E11MOV8812A
- 1A RHR HX TO CHG PUMP SUCT Q1E11MOV8706A
- CCW TO 1A RHR HX Q1P17MOV3185A

<u>OR</u>

13.1.2 Train B equipment available:

- 1B RHR Pump
- CTMT SUMP TO 1B RHR PUMP Q1E11MOV8811B
- CTMT SUMP TO 1B RHR PUMP Q1E11MOV8812B
- 1B RHR HX TO CHG PUMP SUCT Q1E11MOV8706B
- CCW TO 1B RHR HX Q1P17MOV3185B

13.1 <u>IF</u> cold leg recirculation capability can <u>NOT</u> be verified, <u>THEN</u> go to FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

13.2 Begin taking ECCS logs.

Step 13 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
___15	**Check when to transfer to cold leg recirculation.**	
15.1	Check RWST level - LESS THAN 12.5 ft. RWST LVL [] LI 4075A [] LI 4075B	15.1 Return to step 13.

NOTE: Step 15.1 must be complete before continuing with this procedure.

Step	Action/Expected Response	Response NOT Obtained
15.2	Go to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION.	
___16	**Check when to isolate SI accumulators.**	
16.1	Verify power to discharge valves - AVAILABLE. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	
16.2	[CA] WHEN at least two RCS hot leg temperatures less than 385°F, THEN close all SI accumulator discharge valves. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	16.2 Vent any SI accumulator that cannot be isolated. ACCUM N2 VENT [] HIK 936 open

SI ACCUM	1A	1B	1C
1A(1B,1C) ACCUM N2 SUPP/VT ISO Q1E21HV	[] 8875A open	[] 8875B open	[] 8875C open

Step	Action/Expected Response	Response NOT Obtained
___10	**[CA] WHEN RWST level less than 4.5 ft, THEN align containment spray for recirculation.**	
	10.1 Reset PHASE B CTMT ISO. [] MLB-3 1-1 not lit [] MLB-3 6-1 not lit	
	10.2 Open containment spray pump containment sump suction isolation valves. CTMT SUMP TO 1A(1B) CS PUMP [] Q1E13MOV8826A [] Q1E13MOV8827A [] Q1E13MOV8826B [] Q1E13MOV8827B	10.2 IF unable to open a containment sump suction isolation valve, THEN perform the following: 10.2.1 Secure containment spray pump in affected train. [] CS RESET TRN A(B) containment spray signals - RESET (Annunciator EE4 clear). [] CTMT SPRAY PUMP 1A(B) - STOPPED in affected train 10.2.2 Verify closed BOTH containment sump suction isolation valves in affected train. CTMT SUMP TO 1A CS PUMP [] Q1E13MOV8826A [] Q1E13MOV8827A CTMT SUMP TO 1B CS PUMP [] Q1E13MOV8826B [] Q1E13MOV8827B

Step 10 continued on next page.

QUESTIONS REPORT
for ILT 36 RO NRC Exam Version 6

1. W/E12EK1.1 074/NEW//C/A 3.4/3.8/W/E12EK1.1/N///

The crew is responding to a Steam Line Break on Unit 1.

- Due to equipment failures, ECP-2.1, Uncontrolled Depressurization of All Steam Generators, has been entered.
- All SG Narrow Range levels are 25% and lowering.

Which one of the following completes the statement below?

Per ECP-2.1, AFW flow will be adjusted to __(1)__ .

A. at least 20 gpm to **each** SG to prevent AFW pump damge

B.✓ at least 20 gpm to **each** SG to prevent dryout of the SGs

C. at least 20 gpm **total** AFW flow to minimize thermal stress to the SGs

D. at least 395 gpm **total** AFW flow to maintain adequate heat sink

ECP-2.1
Step 4: [CA] Maintain at least 20 gpm AFW flow to SGs with narrow range level less than 31%{48%}.

ECB-2.1
Basis: If feed flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable feed flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

Distracter Analysis

A. Incorrect. See B. Plausible if the applicant thought that without some minimum flow through the AFW pump. it would overheat. The AFW pumps have miniflows.

B. Correct. Per ECP-2.1, the operator is required to maintain at least 20 gpm AFW flow to SGs with narrow range level less than 31%{48%} to prevent dryout of the SGs.

C. Incorrect. See B. Plausible if the applicant confuses AFW to EACH versus Total AFW flow. If this were 20 gpm to EACH SG it would be a correct answer.

D. Incorrect. See B. Plausible since this meets the Heat Sink Critical Safety Function Status Tree. FRP-H.1, Response to Loss of Secondary Heat Sink, has a caution that says the following: This procedure should not be performed if total AFW flow is less than 395 gpm due to operator action.

K/A: **W/E12EK1.1**	Uncontrolled Depressurization of all Steam Generators - Knowledge of the **operational implications** of the following concepts as they apply to the (Uncontrolled Depressurization of all Steam Generators): Components:, **capacity, and function** of emergency systems.
Importance Rating:	3.4 3.8
Technical Reference:	FNp1-ECP-2.1, Uncontrolled Depressurization of All Steam Generators, Ver 24 FNP-0-ECB-2.1, Specific Background Document for FNP-1/2-ECP-2.1, Ver 1
References provided:	None
Learning Objective:	EVALUATE plant conditions and DETERMINE if any system components need to be operated while performing ECP-2.1, Uncontrolled Depressurization of All SGs. (OPS-52532F06)
Question History:	NEW
K/A match:	Requires the applicant to know the AFW flow rate for ECP-2.1 and the **operational implication of lowering AFW flow** (AFW is an emergency system) to 20 gpm (reducing pump capacity) during an Uncontrolled Depressurization of all Steam Generators. Each flow rate has a different operational implication to improve plausibility.
SRO justification:	N/A

Section: **Procedure**

Unit 1 ERP Step: 4 **Unit 2 ERP Step:** 4 ***ERG Step No:*** *2 CAUTION-1*

ERP StepText: Maintain at least 20 gpm AFW flow to SGs with narrow range level less than 31%{48%}.

ERG StepText: *A minimum feed flow of (S.04) gpm must be maintained to each SG with a narrow range level less than (M.02)% [(M.03)% for adverse containment].*

Purpose: To alert the operator to maintain a minimum feed flow to minimize any subsequent thermal shock to SG components

Basis: If feed flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable feed flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

Knowledge: N/A

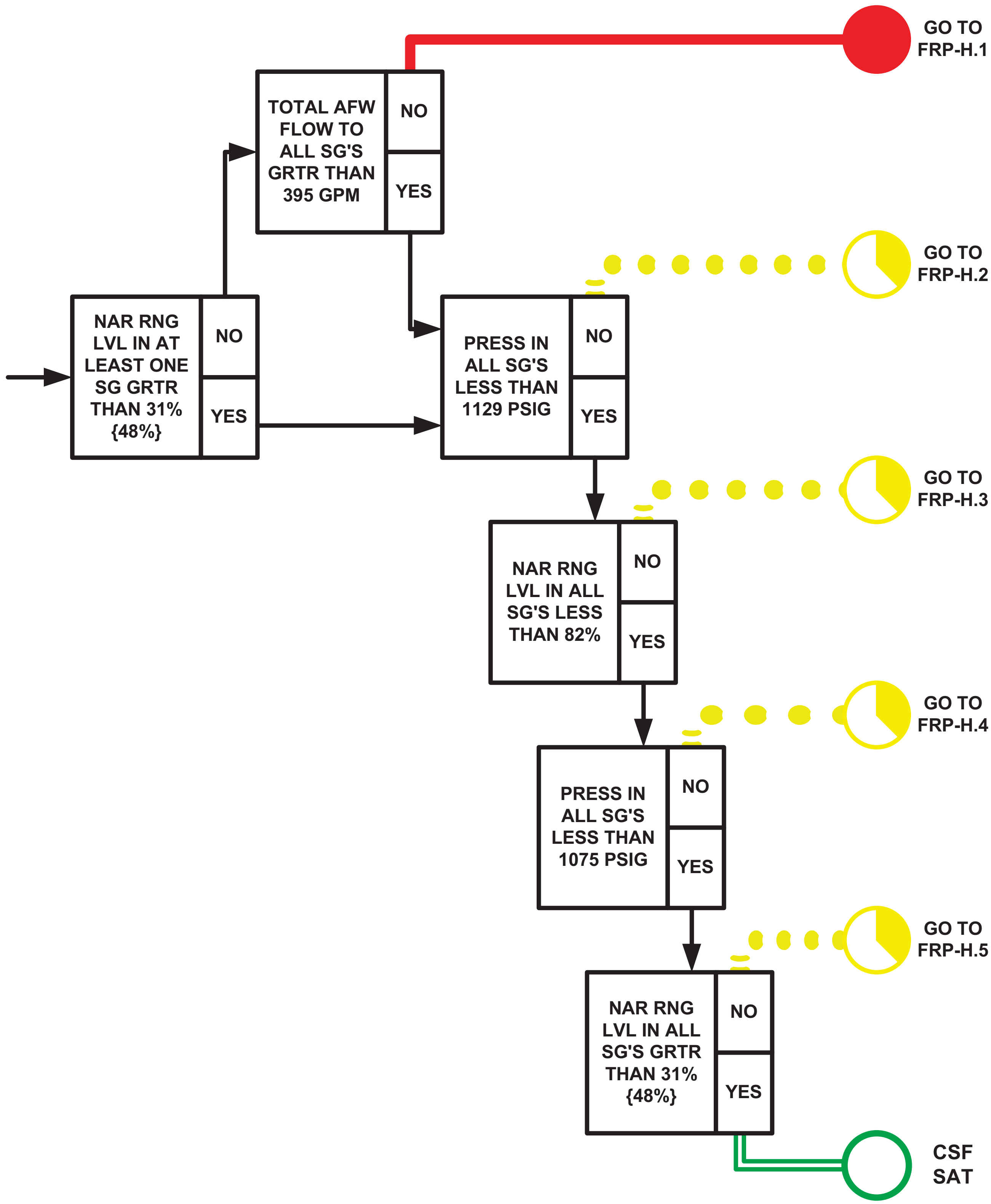
References:

Justification of Differences:

1 Changed to make plant specific.

2 Changed ERG caution to a step since it implies a directed action.

8/29/2007 08:33 FNP-1-CSF-0.3	HEAT SINK	Revision 17



Step	Action/Expected Response	Response NOT Obtained

3.2 Verify AFW flow control valve handswitches - IN THE MODULATE POSITION.

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

MDAFWP
TO 1A(1B,1C) SG
[] Q1N23HV3227A in MOD
[] Q1N23HV3227B in MOD
[] Q1N23HV3227C in MOD

TDAFWP
TO 1A(1B,1C) SG
[] Q1N23HV3228A in MOD
[] Q1N23HV3228B in MOD
[] Q1N23HV3228C in MOD

___ **4** **[CA] Maintain at least 20 gpm AFW flow to SGs with narrow range level less than 31%{48%}.**

AFW FLOW TO
1A(1B,1C) SG
[] FI 3229A
[] FI 3229B
[] FI 3229C

Step	Action/Expected Response	Response NOT Obtained

NOTE: Throttling AFW flow in the following RNO step may result in a Red Path. However, transition to FNP-1-FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK should only be made when AFW flow of 395 gpm is unavailable, and NOT due to deliberate operator action.

___ 5 **[CA] Control AFW flow to minimize RCS cooldown.**

Action/Expected Response	Response NOT Obtained
5.1 [CA] Check cooldown rate in RCS cold legs - LESS THAN 100°F IN ANY 60 MINUTE PERIOD. RCS COLD LEG TEMP [] TR 410	5.1 Reduce AFW flow to 20 gpm to each SG and proceed to step 5.3. MDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3227AA adjusted [] HIC 3227BA adjusted [] HIC 3227CA adjusted TDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3228AA adjusted [] HIC 3228BA adjusted [] HIC 3228CA adjusted
5.2 [CA] Check narrow range level in all SGs - LESS THAN 65%.	5.2 Control AFW flow to maintain narrow range level in all SGs less than 65%. MDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3227AA adjusted [] HIC 3227BA adjusted [] HIC 3227CA adjusted TDAFWP TO 1A(1B,1C) SG FLOW CONT [] HIC 3228AA adjusted [] HIC 3228BA adjusted [] HIC 3228CA adjusted

Step 5 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: This procedure should not be performed if total AFW flow is less than 395 gpm due to operator action.

****

****

CAUTION: To minimize thermal stress, feed flow should not be reestablished to any faulted steam generator if a nonfaulted steam generator is available.

****

___ 1 **Check secondary heat sink - REQUIRED.**

Action/Expected Response	Response NOT Obtained
1.1 Check RCS pressure - GREATER THAN ANY NON-FAULTED SG PRESSURE. 1C(1A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	1.1 Go to procedure and step in effect.
1.2 Check RCS hot leg temperatures - GREATER THAN 350°F{350°F}. RCS HOT LEG TEMP [] TR 413	1.2 Perform the following. 1.2.1 Place RHR system in service using FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM while continuing with this procedure. 1.2.2 WHEN adequate cooling established with RHR system, THEN go to procedure and step in effect.

1. W/E15EA1.3 075/BANK/SUMMER 11/MEM 2.8/3.0/W/E15EA1.3/N///

Which one of the following is the **first** Major Action Category in FRP-Z.2, Response To Containment Flooding, and reason for this in accordance with the background document?

A.✓ Identify unexpected sources of water in the sump since flooding could damage critical plant equipment.

B. Evaluate the ECCS system status to determine a strategy to transition to simultaneous cold and hot leg recirculation.

C. Have chemistry evaluate sump level, chemistry, and activity level to determine a strategy to transfer excess water out of containment.

D. Notify the TSC of sump chemistry, and activity level to determine potential changes in the planned transition to simultaneous cold and hot leg recirculation.

FRP-Z.2
Step 1: Try to identify source of water into sump.

• Check indications for components supplied with service water.

• Check indications for components supplied with CCW.

• Check indication of Reactor Makeup Water Storage Tank level.

• Check indication of Demineralized Water Storage Tank level.

FRB-Z.2 Background:
Step 1 Basis: This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable.

Distracter analysis

A. Correct. Step 1 of FRP-Z.2 directs evaluating potential sources of flooding. The background document states - This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable.

B. Incorrect. See A. Plausible since the ECCS system does enter containment and the applicant may improperly think that this is a source of flooding. If the ECCS system were damaged, then determining a strategy for going on to simultaneous cold and hot leg recirculation would be a plausible reason for this step. There is no step to evaluate ECCS as a source of flooding as it is designed to put water into the recirculation sump via the RCS break.

C. Incorrect. See A. Plausible since this is Step 2 and the basis for this step in FRP-Z.2. The applicant may not be familiar with the procedure and believe that this is the first step.

D. Incorrect. See A. Plausible since Step 3 does have the TSC evaluate sump chemistry, and activity level but not for this reason.

K/A: **W/E15EA1.3**	Containment Flooding - Ability to operate and / or monitor the following as they apply to the (Containment Flooding): Desired operating results during abnormal and emergency situations.
Importance Rating:	2.8 3.0
Technical Reference:	FNP-1-FRP-Z.2, Response To Containment Flooding, Ver 6 FNP-0-FRB-Z.2, Specific Background Document for FNP-1/2-FRP-Z.2, Ver 1
References provided:	None
Learning Objective:	STATE AND EXPLAIN the basis for all Cautions, Notes, and Actions associated with [...] ; (2) FRP-Z.2, Response to Containment Flooding; [...]. (OPS-52533M03)
Question History:	SUMMER 11
K/A match:	Requires to applicant to **monitor containment sump flooding sources and recognize the undesired operating results of not isolating flooding.**
SRO justification:	N/A

SAFETY RELATED

FARLEY NUCLEAR PLANT

SPECIFIC BACKGROUND DOCUMENT

FOR

FNP-1/2-FRP-Z.2

RESPONSE TO CONTAINMENT FLOODING

PROCEDURE USAGE REQUIREMENTS PER FNP-0-AP-6	SECTIONS
Continuous Use	
Reference Use	
Information Use	**ALL**

Approved:

Jim L. Hunter (for)

Operations Manager

Date Issued: 04/10/07

TABLE OF CONTENTS

Section: Title

Unit 1 ERP Step: **Unit 2 ERP Step:** ***ERG Step No:***

ERP StepText: RESPONSE TO CONTAINMENT FLOODING

ERG StepText: *RESPONSE TO CONTAINMENT FLOODING*

Purpose:

Basis:

Knowledge:

References:

Justification of Differences:

1 No differences.

RESPONSE TO CONTAINMENT FLOODING
Plant Specific Background Information

Section: Purpose

Unit 1 ERP Step:	**Unit 2 ERP Step:**	***ERG Step No:***

ERP StepText: This procedure provides actions to respond to containment flooding.

ERG StepText: *This guideline provides actions to respond to containment flooding.*

Purpose:

Basis:

Knowledge:

References:

Justification of Differences:

1 Changed to make plant specific.

Section: Symptoms

Unit 1 ERP Step: **Unit 2 ERP Step:** ***ERG Step No:***

ERP StepText: This procedure is entered from the Containment Critical Safety Function Status Tree on an Orange condition.

ERG StepText: *This guideline is entered from F_0.5, Containment Critical Safety Function Status Tree on an Orange condition.*

Purpose:

Basis:

Knowledge:

References:

Justification of Differences:

1 Changed to make plant specific.

RESPONSE TO CONTAINMENT FLOODING

Plant Specific Background Information

Section: **Procedure**

Unit 1 ERP Step: 1 **Unit 2 ERP Step:** 1 ***ERG Step No:*** *1*

ERP StepText: Try to identify source of water into sump.

ERG StepText: *Try To Identify Unexpected Source Of Water To Sump:*

Purpose: To identify unexpected source of water in sump.

Basis: This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable. A water level greater than the design basis flood level provides an indication that water volumes other than those represented by the emergency stored water sources (e.g., RWST, accumulators, etc.) have been introduced into the containment sump. Typical sources which penetrate containment are service water, component cooling water, primary makeup water and demineralized water. All possible plant specific sources which penetrate containment should be included in this step. These systems provide large water flow rates to components inside the containment and a major leak or break in one of these lines could introduce large quantities of water into the sump. Identification and isolation of any broken or leaking water line inside containment is essential to maintaining the water level below the design basis flood level.

Knowledge: N/A

References:

Justification of Differences:

1 Changed to make plant specific.

Section: **Procedure**

Unit 1 ERP Step: 2 **Unit 2 ERP Step:** 2 ***ERG Step No:*** *2*

ERP StepText: Direct Chemistry to sample containment sump for radioactivity, chromates and boron concentration using FNP-0-CCP-1300, CHEMISTRYÿAND ENVIRONMENTAL ACTIVITIES DURING A RADIOLOGICAL ACCIDENT.

ERG StepText: *Check Containment Sump Activity Level:*

Purpose: To determine the radioactivity level of the sump fluid.

Basis: The step instructs the operator to determine the activity level in the containment sump water in order to provide information concerning the possible transfer of containment sump water to plant storage tanks outside the containment. The transfer of containment sump water from the containment to other plant storage tanks may be desirable in order to minimize the potential for flooding of critical plant components inside the containment. However, the ultimate disposition of this water outside the containment will depend, in large part, on the level of radioactivity in the water. The method of sampling the containment sump water is plant dependent. Appropriate precautions should be observed due to the potential for high radioactivity.

Knowledge: N/A

References:

Justification of Differences:

1 Changed to make plant specific.

RESPONSE TO CONTAINMENT FLOODING

Plant Specific Background Information

Section: **Procedure**

Unit 1 ERP Step: 3 **Unit 2 ERP Step:** 3 ***ERG Step No:*** *3*

ERP StepText: Notify TSC staff of sump level and activity level to obtain recommended action.

ERG StepText: *Notify Plant Engineering Staff Of Sump Level And Activity Level To Obtain Recommended Action.*

Purpose: To notify plant engineering staff of sump level and activity level.

Basis: The step instructs the operator to provide the plant engineering staff with information concerning the containment sump level and information on the radioactive content of the water. The plant specific design and layout will affect the options available to the plant engineering staff regarding the potential transfer of containment sump water outside containment. The design considerations include:
1) location of critical plant components in relation to containment sump water level, 2) location, size and shielding of outside containment storage tanks, and 3) pump and line routing from the containment sump to various storage tanks.

The plant engineering staff should evaluate the event and provide specific recommendations to the operators concerning the high containment sump water levels.

Knowledge: N/A

References:

Justification of Differences:

1 Changed to make plant specific.

RESPONSE TO CONTAINMENT FLOODING

Plant Specific Background Information

Section: **Procedure**

Unit 1 ERP Step: 4 **Unit 2 ERP Step:** 4 ***ERG Step No:*** *4*

ERP StepText: Go to procedure and step in effect.

ERG StepText: *Return To Guideline And Step In Effect*

Purpose: To direct the operator to the proper guideline following successful completion of the steps in this guideline.

Basis: Now that the guideline steps have been completed, the operator should continue plant recovery operations by returning to the guideline and step that was in effect at the time FR-Z.2 was entered.

Knowledge: It should be noted that once all the actions of this guideline are completed and the operator is returned to the guideline and step in effect, this particular Containment function may not be restored to a GREEN and the Containment Status Tree may continue to display a RED or ORANGE priority. If this is the case, the appropriate Function Restoration Guideline should not be implemented again since all necessary actions have already been performed.

References:

Justification of Differences:

1 Changed to make plant specific.

FNP-1-FRP-Z.2
09-21-2000
Revision 6

FARLEY NUCLEAR PLANT

FUNCTION RESTORATION PROCEDURE

FNP-1-FRP-Z.2

RESPONSE TO CONTAINMENT FLOODING

SAFETY RELATED

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	**ALL**
Reference Use	
Information Use	

Approved:

C.D. Collins
Operations Manager

Date Issued: 9-27-00

LIST OF EFFECTIVE PAGES

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2	2	X		X	X							

FNP-1-FRP-Z.2	RESPONSE TO CONTAINMENT FLOODING	Revision 6

A. Purpose

This procedure provides actions to respond to containment flooding.

B. Symptoms or Entry Conditions

I. This procedure is entered from the Containment Critical Safety Function Status Tree on an Orange condition.

FNP-1-FRP-Z.2	RESPONSE TO CONTAINMENT FLOODING	Revision 6

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Try to identify source of water into sump.** • Check indications for components supplied with service water. • Check indications for components supplied with CCW. • Check indication of Reactor Makeup Water Storage Tank level. • Check indication of Demineralized Water Storage Tank level.	
___ 2	**Direct Chemistry to sample containment sump for radioactivity, chromates and boron concentration using FNP-0-CCP-1300, CHEMISTRY AND ENVIRONMENTAL ACTIVITIES DURING A RADIOLOGICAL ACCIDENT.**	
___ 3	**Notify TSC staff of sump level and activity level to obtain recommended action.**	
___ 4	**Go to procedure and step in effect.**	

-END-