HL-18 NRC Exam 2013-301 Examination KEY

1. 002K5.10 001/2/2/RCS - PWR-TEMP/C/A 3.6/4.1/NEW/HL-18 NRC/RO/SRO/KAJ

Given the following plant conditions:

- The plant is at 100% power based on all available indications.
- All control systems are in automatic.

A narrow range RTD fails on Unit 1 and the following alarms illuminate:

- ALB10-C03 OVERPOWER DELTA T ROD BLOCK AND RUNBACK ALERT
- ALB10-E03 OVERTEMP DELTA T ROD BLOCK AND RUNBACK ALERT

- ALB12-A03 RC LOOP DELTA T/AUCT DELTA T HI-LO- DEV
- ALB12-A04 RC LOOP TAVG/AUCT TAVG HI-LO DEV
- ALB12-A06 OVERTEMP DELTA T ALERT
- ALB12-B06 OVERPOWER DELTA T
- ALB12-C06 TERR (TAVG-TREF) LO

With no operator action, which ONE of the following answers the following statement?

Based on the given indications, a loop T_{cold} RTD failed ___(1)___,

and

the highest <u>indicated</u> power level ___(2)___ exceed 100%.

A. (1) high

(2) will

B. (1) high

(2) will NOT

C.✓ (1) low

(2) will

D. (1) low

(2) will NOT

002K5.10 Reactor Coolant System (RCS)

Knowledge of the operational implications of the following concepts as they apply to the RCS:
(CFR 41.5 / 45.7)

Relationship between reactor power and RCS differential temperature.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a Tcold instrument has failed, the candidate has to determine from given annunciator windows which direction the instrument has failed and whether indicated reactor power based on delta T will exceed 100%.

DISTRACTOR ANALYSIS:

A. Incorrect - The first half is incorrect, the given annunciators indicate a Tcold has failed in the low direction. The 2nd half is correct as the loop Delta T will indicate greater than 100%.

B. Incorrect - The first half is incorrect, the given annunciators indicate a Tcold has failed in the low direction. The 2nd half is incorrect as the loop Delta T will indicate greater than 100%.

C. Correct - The first half is correct, the given annunciators indicate a Tcold has failed in the low direction. The 2nd half is correct as the loop Delta T will indicate greater than 100%.

D. Incorrect - The first half is correct, the given annunciators indicate a Tcold has failed in the low direction. The 2nd half is incorrect as the loop Delta T will indicate greater than 100%.

REFERENCES:

V-LO-TX-16001, pg 65-67
ALB10-C03 OVERPOWER DELTA T ROD BLOCK AND RUNBACK ALERT
ALB10-E03 OVERTEMP DELTA T ROD BLOCK AND RUNBACK ALERT
ALB12-A03 RC LOOP DELTA T/AUCT DELTA T HI-LO- DEV
ALB12-A04 RC LOOP TAVG/AUCT TAVG HI-LO DEV
ALB12-A06 OVERTEMP DELTA T ALERT
ALB12-B06 OVERPOWER DELTA T
ALB12-C06 TERR (TAVG-TREF) LO

OBJECTIVES:

V-LO-PP-16101-06 Given that a narrow range temperature instrument has failed and the response of Tavg and delta T to the failure, determine whether a hot or cold leg instrument has failed, and in what direction (i.e., high or low).

core and is a direct correlation of reactor power. In the formula below **Q** is the heat transfer rate or reactor power. Any change in T_{hot} or T_{cold} can only be the result of a change in reactor power.

$$Q = mC_p(T_{hot}-T_{cold})$$

An example of why Tavg cannot be used as an indication of power. Assume due to Feed water transient which causes T_{cold} to drop from 557°F to 553°F. Tavg would actually lower from 586.4°F to 584.4°F which would appear that reactor power had lowered slightly. Using the formula above would prove otherwise.

615.8°F - 557°F = 58.8°F = <u>100% ΔT</u>

615.8°F - 553°F = 62.8°F = <u>106.8% ΔT</u>

As you can see actual reactor power increased substantially to unacceptable level, which could not be directly measured by Tavg. The example maybe exaggerated slightly but the point can still be made.

The loop ΔT meters have scales that range from 0% to 150% power. The instrument tag numbers are as follows TDI-411A, TDI-421A, TDI-431A, and TDI-441A for each loop respectively. Indicators are located on the "C" panel in the control room. Loop ΔT also can be retrieved from the IPC.

16.52 Temperature Instrument Failures

Narrow range temperature instrument failures cannot be directly identified from the control board indicators. However, the symptoms can be diagnosed to determine which instrument has failed. Narrow range T_{hot} and T_{cold} are not displayed any where on the control boards or on the IPC. I&C can determine from readings taking from the individual temperature transmitter, the failed instrument. In the examples below, an operator can determine if the failure was a T_{hot} or T_{cold} failure and which loop is affected.

DIAGNOSTIC TOOL

	Tavg	**ΔT**
NR T_{hot} fails high	Hi	Hi
NR T_{hot} fails low	Lo	Lo
NR T_{cold} fails high	Hi	Lo
NR T_{cold} fails low	Lo	Hi

Keep in mind that a single failure of hot leg RTD will not be automatically removed from the calculated T_{hot}. So, T_{hot} output should not fail off scale in any direction unless its instrument power is lost.

EXAMPLE #1

Unit 1 was at 100% power at normal operating temperature and pressure with control systems in auto. During the operator shift relief, several alarms were received. The operator observed the following:

Loop 1 Tavg = **586.5°F** ΔT = **100%**

Loop 2 Tavg = **586.0°F** ΔT = **99%**

Loop 3 Tavg = **571.0°F** ΔT = **52.5%**

Loop 4 Tavg = **586.5°F** ΔT = **101%**

In Loops 1, 2, and 4 Tavg and ΔT appear to be normal. It is easy to determine which loop has an instrument problem. There is a distinct difference between Loop 3 and the other temperature instruments. Using the diagnostic tool above you can determine the instrument that has failed. Since Loop 3 Tavg has failed low we know that either T_{hot} or T_{cold} has failed low. The ΔT indication is low also. Loop 3 ΔT indicating low can only be from either T_{hot} failing low or T_{cold} failing high. So, what is common about the two instrument indications?

Answer: Loop 3 T_{hot} failed low.

EXAMPLE #2

Unit 1 was at 100% power at normal operating temperature and pressure with control systems in auto. During the operator shift relief, several alarms were received with rod control automatically inserting rods at 72 steps per minute. The operator observed the following:

Loop 1 **Tavg = 586.5°F** **ΔT = 100%**

Loop 2 **Tavg = 591.0°F** **ΔT = 120%**

Loop 3 **Tavg = 586.0°F** **ΔT = 99%**

Loop 4 **Tavg = 586.5°F** **ΔT = 101%**

In Loops 1,3, and 4 Tavg and ΔT appear to be normal. It is easy to determine which loop has an instrument problem. There is a distinct difference between Loop 2 and the other temperature instruments. Using the diagnostic tool above you can determine the instrument that has failed. Since Loop 2 Tavg has failed high we know that either T_{hot} or T_{cold} has failed high. The ΔT indication is high also. Loop 2 ΔT indicating high can only be from either T_{hot} failing high or T_{cold} failing low. So, what is common about the two instrument indications?

Answer: Loop 2 T_{hot} failed high.

16-53 Protection / Control

Protection Circuits that take input from narrow range temperature instruments:

Over Temperature Differential Temperature (OTΔT) compares actual Tavg to full-load Tavg. If Tavg increases above the full load Tavg, the set point will be reduced. This set point is compared to its actual loop ΔT. If the set point gets within 3% of its ΔT on 2 out of 4 loops a "**C-3 OTΔT rod stop and turbine runback**" will occur. If the set point continues to lower to its actual ΔT power level on 2 out of four loops, a reactor trip will be generated. OTΔT circuit protects the RCS from Departure from Nucleate Boiling (DNB). OTΔT circuit has two other inputs that affect its set point which are Pressurizer pressure and Power range Δ flux. OTΔT set point at low power levels can be as high as 121.6% on U-1 and 118.7% on U-2, but at 100% rated power the set point drops to 119.4% on U-1 and 116.5% on U-2%. The OTΔT set point increases as the plant is cooled down due to the Tavg input lower. This temperature reward is limited or clamped at 585.4°F.

Over Power Differential Temperature (OPΔT) also compares actual Tavg to full-load Tavg. If Tavg increases above full-load Tavg, OPΔT set point lowers from its normal set point of 110%. This set point is compared to its actual loop ΔT. If the set point gets within 3% of its ΔT on 2 out of 4 loops a "**C-4 OPΔT**

rod stop and turbine runback" will occur. If the set point continues to lower to its actual ΔT power level on 2 out of four loops, a reactor trip will be generated. This circuit is very similar to OTΔT but for different reasons. OPΔT circuit protects the Fuel from being damaged due to overpower, based on KW/ft being produced. Its set point is capped at 108.9% in which Tavg is its only input to penalize.

"P-12 Low Low Tavg Steam Dump Interlock" occurs when 2 out 4 Tavg loops drop to **550°F.** This interlock protects the RCS from a cool down accident by preventing the steam dumps from opening. This interlock can be bypassed to allow normal cool down of the plant.

"Low Tavg FWI" occurs when 2 out of 4 Tavg loops drop to **564°F in conjunction with P-4 (Rx Trip).** The Feed Water Isolation protects the RCS from a cool down accident due to over feeding the Steam Generators.

Control Circuits that take input from Narrow Range Temperature Instruments:

Auctioneered high Tavg circuitry compares Tavg values from each of the four RCS loops. The most limiting value from the loops (highest Tavg) is selected for conservatism in calculating control set points. Control Systems that utilize Auctioneer High Tavg are as follows:

1. Rod Control System
2. Pressurizer Level Control
3. Steam Dump System

Auctioneered Low Tavg circuitry compares Tavg from, all loops (lowest Tavg) is selected for conservatism in calculating actuation/control set points. Two things receive input from Auctioneered Low Tavg is:

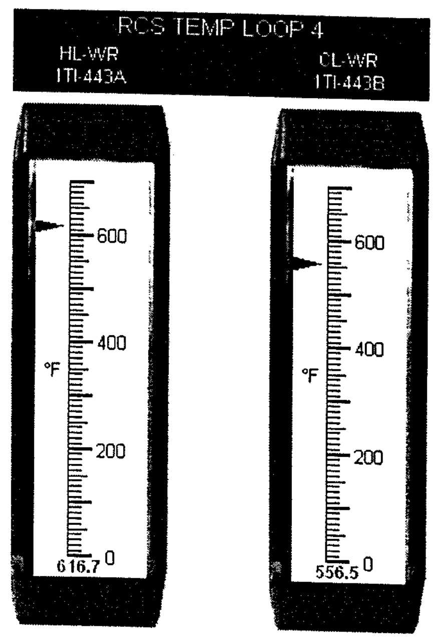
1."**C-16 Low Tavg Turbine Stop Loading" Tavg ≤ 553°F or Tavg ≥ 20°F below Tref.**
Protects the RCS from cooling down below the minimum temperature for criticality. This interlock prevents the Main Turbine load increase that can be bypassed for testing purposes only.

2. Tavg/Tref deviation meter (TI-412A) on "C" panel in the control room.

Auctioneered High ΔT provides input for the Rod Insertion Limit (RIL) computer, which generates an alarm set point based on power level and rod height.

Both loop ΔT and Tavg inputs into control circuits can be defeated by the operator at the control panel if Narrow Range temperature instrument is to be removed from service.

Tavg defeat switch TS-412T is located on the "C" panel in the control room. It allows input from a single Tavg channel to be defeated from various control circuits. (1)Defeats input into auctioneer low Tavg calculation such as C-16 and the Tavg / Tref deviation meter, and (2) defeats selected channel input into auctioneered high Tavg output circuitry for rod control, steam dump control, pressurizer level control, Tavg / Tref Deviation alarm, and Auctioneer Tavg Hi alarm.



ΔT defeat switch TS-411T is located on the "C" panel in the control room. It allows defeat of a single ΔT channel into auctioneered high ΔT calculation. Defeats input into Rod insertion limit computer.

16-54 RCS WIDE RANGE TEMPERATURE INSTRUMENTATION

RCS wide range temperature transmitters measure both the hot and cold legs of the RCS just like the narrow range instruments. The differences in the two are: (1) only one RTD per leg, (2) the thermowells are dry (RTDs do not contact the fluid), (3) they are scaled from 0°F to 700°F, (4) there is no installed spare RTD, (5) located both in control

WINDOW C03

ORIGIN	SETPOINT	
1-TB-411H 1-TB-421H 1-TB-431H 1-TB-441H	Variable (ΔT reactor trip setpoint -3%)	OVERPOWER ΔT ROD BLOCK AND RUNBACK ALERT

1.0 PROBABLE CAUSE

Instrument malfunction or test.

2.0 AUTOMATIC ACTIONS

NONE

> NOTE
>
> A 2 out of 4 coincidence will inhibit rod withdrawal and cause a Turbine runback.

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Check** TSLB-3 and **determine** which channel caused the alarm.
2. **Check** the loop temperature indications and **Go To** 18001-C, "Primary Systems Instrumentation Malfunction" or 18002-C, "Nuclear Instrumentation System Malfunction", as applicable.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AA02-229, 1X6AU01-163, 164, 165, 166, 1X6AX01-104, PLS

WINDOW E03

ORIGIN	SETPOINT	
1-TB-411D 1-TB-421D 1-TB-431D 1-TB-441D	Variable (ΔT reactor trip setpoint-3%)	OVERTEMP ΔT ROD BLOCK AND RUNBACK ALERT

1.0 **PROBABLE CAUSE**

Instrument malfunction or test.

2.0 **AUTOMATIC ACTIONS**

NONE

> NOTE
>
> A 2 out of 4 coincidence will inhibit rod withdrawal and cause a Turbine runback.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Check** TSLB-3 and determine which channel caused the alarm.

2. **Check** pressurizer pressure, Tavg and power range instrumentation and **Go To** 18001-C, "Primary Systems Instrumentation Malfunction" or 18002-C, "Nuclear Instrumentation System Malfunction", as applicable.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AA02-229, 1X6AU01-163, 164, 165, 166, 1X6AX01-104, PLS

WINDOW A03

ORIGIN	SETPOINT
1-TE-0411 A/B 1-TE-0421 A/B 1-TE-0431 A/B 1-TE-0441 A/B	+5%

RC LOOP
ΔT/AUCΔT
HI-LO DEV

NOTE

Setpoint compares the auctioneered high ΔT loop with other ΔT loops.

1.0 PROBABLE CAUSE

1. Unbalanced steam/feed flows in the Steam Generators.
2. Loss of flow in a reactor coolant loop.

2.0 AUTOMATIC ACTIONS

1. Reactor trip will occur due to low flow in one Reactor Coolant Pump loop if above 48 percent rated thermal power.
2. Reactor trip will occur due to low flow in 2 of 4 Reactor Coolant Pump loops if above 10 percent rated thermal power.

3.0 INITIAL OPERATOR ACTIONS

1. **Check** loop temperature and flow indications and IF a loss of flow has occurred, **initiate** 18005-C, "Partial Loss Of Flow."
2. IF instrument failure in indicated, **initiate** 18001-C, "Primary Systems Instrumentation Malfunction."

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF the temperature deviations are due to unbalanced steam/feed flows, **take manual control** of the Steam Generator feed flows as necessary to **correct** the deviation.
2. IF a cooldown is in progress using the Steam Generator atmospheric reliefs, attempt to **equalize** steaming of generators if possible.

WINDOW A04

ORIGIN	SETPOINT	
1-TE-0411 A/B 1-TE-0421 A/B 1-TE-0431 A/B 1-TE-0441 A/B	4°F	RC LOOP TAVG/AUCT TAVG HI-LO DEV

NOTE

Setpoint compares the auctioneered high Tavg loop with other Tavg loops.

1.0 PROBABLE CAUSE

1. Unbalanced steam/feed flows in the Steam Generators.
2. Loss of flow in a reactor coolant loop.

2.0 AUTOMATIC ACTIONS

1. Reactor trip will occur due to low flow in one Reactor Coolant Pump loop if above 48 percent rated thermal power.
2. Reactor trip will occur due to low flow in 2 of 4 Reactor Coolant Pump loops if above 10 percent rated thermal power.

3.0 INITIAL OPERATOR ACTIONS

1. **Check** loop temperature and flow indications and IF a loss of flow has occurred, **initiate** 18005-C, "Partial Loss Of Flow".
2. IF instrument failure is indicated, **initiate** 18001-C, "Primary Systems Instrumentation Malfunction."

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF the temperature deviation is due to unbalanced steam/feed flows, **take manual control** of the Steam Generator feed flows to **correct** the deviation.
2. IF a cooldown is in progress using the Steam Generator atmospheric reliefs, attempt to **equalize** steaming of generators if possible.

WINDOW A06

ORIGIN	SETPOINT	
1-TB-0411C 1-TB-0421C 1-TB-0431C 1-TB-0441C	Variable	OVERTEMP ΔT ALERT

1.0 **PROBABLE CAUSE**

Pressurizer pressure, Tavg or power range instrument malfunction or test.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** TSLB-3 and **determine** which channel caused the alarm.

2. **Check** Pressurizer pressure, Tavg and power range instrumentation and **Go To** 18001-C, "Primary Systems Instrumentation Malfunction" or 18002-C, "Nuclear Instrumentation System Malfunction" as applicable.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB113, 1X6AA02-229, 1X6AU01-163, 164, 165, 166, PLS

WINDOW B06

ORIGIN	SETPOINT	
1-TB-0411G 1-TB-0421G 1-TB-0431G 1-TB-0441G	Variable	OVERPOWER ΔT ALERT

1.0 **PROBABLE CAUSE**

Instrument malfunction or test.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** TSLB-3 and determine which channel caused the alarm.

2. **Check** the loop temperature indications and **Go To** 18001-C, "Primary Systems Instrumentation Malfunction," or 18002-C, "Nuclear Instrumentation System Malfunction" as applicable.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB113, 1X6AA02-229, 1X6AU01-163, 164, 165, 166, PLS

WINDOW C06

ORIGIN	SETPOINT	
1-TB-0412V 1-PT-0505	≤553°F OR TAVG 20° BELOW TREF	TERR (TAVG-TREF) LO

1.0 PROBABLE CAUSE

1. Loading Turbine too quickly.

2. Excessive steam demand.

2.0 AUTOMATIC ACTIONS

C-16 will stop Turbine loading.

3.0 INITIAL OPERATOR ACTIONS

1. **Check** the loop temperature indications and Tref.

2. IF instrument failure is indicated, **initiate** 18001-C, "Primary System Instrumentation Malfunction."

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF the deviation is due to loading the Turbine too quickly, **reduce** Turbine load to **balance** Tavg and Tref.

2. IF the deviation is due to excessive steam demand, **check** Steam Dumps and Atmospheric Relief Valves closed.

3. **Refer** to Technical Specification LCO 3.4.2.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X6AU01-188, 179, 1X6AA02-240, PLS

2. 003A2.05 001/2/1/RCP - SEAL LEAKOFF/C/A 2.5/2.8/NEW/HL-18 NRC/RO/SRO/AML

Initial conditions at 10:00

- Unit 1 is in Mode 4.
- RCS pressure is 230 psig.
- RCPs # 1 and # 3 are running.
- VCT pressure is 22 psig.
- LT-0112, VCT Level, fails low.

Current conditions:

- VCT pressure is 37 psig.

Which one of the following completes the following statement?

With no operator action, RCP seal # 1 leakoff will ___(1)___, and per 13003-1, "Reactor Coolant Pump Operation", the RCPs ___(2)___ required to be stopped.

	__(1)__	__(2)__
A.✓	decrease	are
B.	increase	are
C.	decrease	are NOT
D.	remain constant	are NOT

003A2.05 Reactor Coolant Pump System (RCPS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Effects of VCT pressure on RCP seal leakoff flows.

K/A MATCH ANALYSIS:

This question meets the KA by listing a VCT level malfunction and making the students determine the effect of the malfunction as well as any procedural guidance we may have as a result of the affected instrument. With the VCT level failure, auto makeup will occur raising VCT pressure which will cause seal leakoff to lower. RCS pressure minus VCT pressure will cause Delta P to be lower than 200 psig which is an immediate RCP trip criteria. The candidate will have to make the correlation between RCS pressure and VCT pressure.

DISTRACTOR ANALYSIS:

A. Correct

B. Incorrect - Plausible that the students may confuse a higher pressure with a greater flow rate. However, it is the DP that affects the flow and this DP has gone down due to the greater VCT pressure.

C. Incorrect - First part is correct however the RCPs must be secured procedurally when DP is less than 200 psig.

D. Incorrect - Plausible that the students may not correlate VCT pressure with seal DP. Additionally the RCPs must be secured procedurally when DP is less than 200 psig.

REFERENCES:

13003-1 pg 6 and Figure 2
LO-TX-09101 pg 11, 12

VEGP learning objectives:

LO-PP-09100-03 - State the purpose and describe the control signals, setpoints, and any interlocks for the following:

h. VCT divert valve LV-112A

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

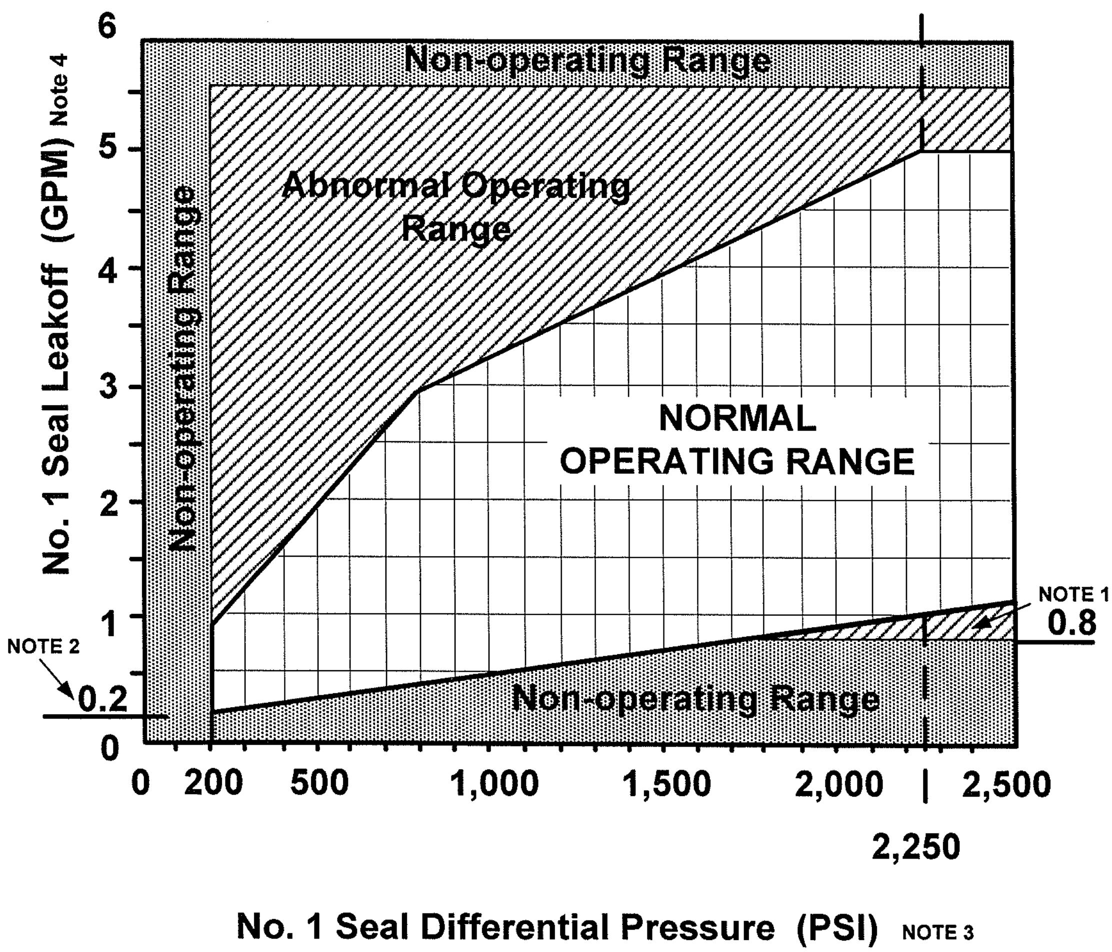
2.2.10 An RCP shall be stopped if any of the following conditions exist.

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

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

FIGURE 2

NO. 1 SEAL NORMAL OPERATING RANGE



1. If the No. 1 seal leak rates are outside the normal (1.0-5.0 gpm) but within the operating limits ((0.8-5.5 gpm), continue pump operation. VERIFY that seal injection flow exceeds No. 1 seal leak rate for the affected RCP. Closely monitor pump and seal parameters and contact Engineering for further instructions.

2. Minimum startup requirements are 0.2 gpm at 200 PSID differential across the No. 1 seal. For startups at differential pressures greater than 200 PSID, the minimum No. 1 seal leak rate requirements are defined in the NO. 1 SEAL NORMAL OPERATING RANGE (e.g., at 1000 psi differential pressure, do not start the RCP with less than 0.5 gpm).

3. No.1 Seal Differential Press = RCS WR Press – VCT Press.

4. Per Westinghouse Technical Bulletin ESBU-TB-93-01-R1, total #1 seal leakoff is the sum of #1 seal leakoff and #2 seal leakoff. #1 seal leakoff is read directly at the MCB and #2 seal leakoff can be obtained from instrumentation in Containment.

Letdown Temperature Divert Valve TV-129

TIS-129 measures the temperature of the letdown flow downstream of the letdown heat exchanger. If letdown temperature increases to 132.5°F, a control room alarm actuates. Automatically, three way valve TV-129 will bypass the mixed bed demineralizers by diverting letdown flow to the VCT. Letdown temperature can be checked by control room indication TI-130. Another input to TV-129 is BTRS demin inlet temperature switch TIS-382. If BTRS is in service, an inlet high temperature of 168.5°F will cause TV-129 to divert to the VCT. Both of these actuations serve to protect the demineralizer resin. HS-129 is a two positioned valve - VCT(maintained position) and DEMINERALIZER, from which it spring returns to AUTO.

Mixed-Bed Demineralizers

One of the two flushable mixed-bed demineralizers is normally used during plant operation to maintain reactor coolant purity. A lithium-type cation resin and hydroxyl-type anion resin are initially charged into the demineralizers. Both forms of resin remove fission and corrosion products. A mixed bed demineralizer contains 28 ft^3 of resin capable of accepting a maximum letdown flow of 120 gpm while reducing (by at least a factor of ten) the concentration of ionic isotopes (except cesium, yttrium, and molybdenum)in the flow stream. Each load of resin has a minimum design life of one core cycle based on an assumption of 1% defective fuel. The demineralizer vessels are fabricated from austenitic stainless steel and are provided with resin retention screens, an inflow deflector, and mesh screens on drain connections. The mixed-bed demineralizers are located in the auxiliary building.

Cation-Bed Demineralizer

A flushable cation resin bed demineralizer in the hydroxyl form is located downstream of the mixed-bed demineralizers and is used intermittently to control the concentration of Li-7 which is produced in the core due to irradiation of boron in the coolant. Its size is based upon the estimated production of Li-7 in the core region due to the B-10 (n, alpha) Li-7 reaction.

The resin also has sufficient capacity to control the concentration of cesium-137 below 1.0 micro curie per cubic cm with 1% defective fuel. The demineralizer contains 21 ft^3 of resin and will accept maximum letdown flow of 75 gpm. It is located next to and is similar in construction to the mixed-bed demineralizers.

Reactor Coolant Filter

The reactor coolant filter is designed to collect resin fines and particulate matter 0.2 microns or larger from the letdown flow. It is located in the system between the demineralizers and the volume control tank. It is a disposable cartridge filter. Connections are provided for venting and draining of the unit for element replacement. The filter unit is designed to accept maximum letdown flow of 120 gpm. It is located in the auxiliary building in a shielded area.

Letdown to Volume Control Tank Valve LV-112A

VCT level control divert valve (LV-112A) is an air-operated valve controlled from the QMCB and receives signals from the VCT level controllers (LIC-185 and LI-112). It is a three way valve which diverts letdown flow to the recycle holdup tank on VCT high level. It fails to the VCT position. The valve has three

positions: AUTO, VCT, and HUT. Positions VCT and HUT are maintained positions. The volume control tank is designed to accept reactor coolant from pressurizer level changes. When the level in the tank reaches the high level set point, the remainder of the letdown is diverted to the recycle holdup tanks. The chart below shows how LIC-185 and LI-112 input to LV-112A position when it is in the AUTO position.

LI-0112	VCT LEVEL	LI-0185
Trip open 112A	97%	Modulate 112A full divert (if LIC-0185 pot @8.70)
Hi level alarm	92%	
112A Trip Open signal Resets	87%	112A starts to divert (if LIC-0185 pot @8.70)
Auto Makeup stops	50%	
Auto Makeup starts	30%	
Low level alarm	20%	Low level alarm
RWST auto swap over	5.7%(2 of 2)	RWST auto swap over

VCT level transmitter failures have been analyzed for failures that could lead to loss of NPSH resulting in cavitation or gas binding. Discuss SOER 97-1 (item B) in the Operating Experience section.

Volume Control Tank

The Volume Control Tank also performs the following functions:

* Introduces hydrogen into the coolant to control and scavenge oxygen produced by radiolysis of water in the core during normal operation
* Provides a means of degassing the reactor coolant
* Provides sufficient net positive suction head (NPSH) for the charging pumps
* Provides a location to accept makeup water to adjust Reactor Coolant System boron concentration
* Provides backpressure for the #1 reactor coolant pump seals

The tank is fabricated from austenitic stainless steel, complete with relief protection, sampling, hydrogen and nitrogen connections. It has a capacity of 400 cubic ft (3000 gal). The tank is located in the auxiliary building and is shielded for personnel protection.

The tank also provides a means for introducing hydrogen into the coolant to maintain the required equilibrium concentration of 25 to 50 cm^3 hydrogen/kg water and is used for degassing the reactor coolant. A spray nozzle located inside the tank on the letdown line provides liquid-to-gas contact between the incoming fluid and the hydrogen atmosphere in the tank. At power operation, hydrogen from the Auxiliary Hydrogen Gas System header in the Auxiliary Building is continuously supplied to the volume control tank via a pressure control valve, PCV-8156. Another penetration in the VCT allows for the VCT gas space to be aligned to the gaseous waste processing system(WPSG). This permits continuous removal of any gaseous fission products which are stripped from the reactor coolant and collected in this tank. Pressure in the VCT is maintained at approximately 25 psig during power operations. If VCT pressure lowers to <18 psig, the VCT isolation valve to the WPSG PV-0115 will automatically close. This automatic feature ensures that least 15 psig backpressure is maintained on the #1 seal return.

3. 003AK1.07 001/1/2/DROP ROD - SDM/C/A 3.1/3.9/NEW/HL-18 NRC/RO/SRO/KAJ

Initial conditions:

- Loop 1 delta T = 76.9%
- Loop 2 delta T = 76.8%
- Loop 3 delta T = 75.8%
- Loop 4 delta T = 76.0%

Current conditions:

- CBD rod D4 drops to the bottom of the core.

- Loop 1 delta T = 77.8%
- Loop 2 delta T = 78.2%
- Loop 3 delta T = 77.2%
- Loop 4 delta T = 70.7%

Which one of the following completes the following statement?

Based on the given conditions, shutdown margin will ___(1)___

and

the setpoint for the ALB10-D04 ROD BANK LO-LO LIMIT alarm will ___(2)___ .

A. (1) increase

(2) increase

B. (1) increase

(2) decrease

C✓ (1) remain the same

(2) increase

D. (1) remain the same

(2) decrease

003AK1.07 Dropped Control Rod

Knowledge of the operational implications of the following concepts as they apply to the Dropped Control Rod: (CFR: 41.8 / 41.10/ 45.3)

Effect of dropped rod on insertion limits and SDM.

K/A MATCH ANALYSIS:

Question meets the KA by testing the students knowledge on the concepts associated with shutdown margin and rod insertion limits and how changes in rod position would affect each.

DISTRACTOR ANALYSIS:

A. Incorrect - While part 1 is correct, part 2 is incorrect since the rod drop has added the reactivity at the time of movement that would otherwise be added at the time of trip, the net effect would be the same.

B. Correct - The RIL will initially lower due to compensate for the changes in power defect and SDM remain the same due to a zero net effect on the total amount of reactivity.

C. Incorrect - RIL lowers due to power defect not xenon reactivity since we offset any changes in reactivity from xenon with boron. SDM remains the same due to a zero net effect on the total amount of reactivity.

D. Incorrect - RIL lowers due to power defect not xenon reactivity since we offset any changes in reactivity from xenon with boron.

REFERENCES:

LO-TX-33800 pg 50, 51
LO-TX-33500 pg 19, 20
LO-PP-16101, slide # 20
17010-1 ARP window D04, ROD BANK LO-LO LIMIT

VEGP learning objectives:

LO-LP-33500-12 Discuss rod insertion limits.
LO-LP-33800-27 Evaluate changes in shutdown margin due to changes in plant parameters.

WINDOW D04

ORIGIN	SETPOINT	ROD BANK LO-LO LIMIT
IPC Calculated	Rod Insertion Limit	
UD0366		

1.0 PROBABLE CAUSE

RCS Boron concentration too low for present reactor power level due to:

1. Plant transient.
2. Xenon transient.
3. IPC failure

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

1. **Check** indications and determine if actual control bank rod position is below the Lo-Lo insertion limit by referring to the COLR and Technical Specification LCO 3.1.6.

2. IF actual control bank position is below the Lo-Lo Insertion Limit, **perform** the following:

   a. Within 1 hour:

      Verify shutdown margin is within the limits specified in the COLR per 14005-1 “Shutdown Margin Calculation”; **Refer To** TR 13.1.1 for applicability.

      OR

      Initiate and **maintain** Emergency Boration per 13009-1, “CVCS Reactor Makeup Control System”, until the Control Banks Lo-Lo Limit Annunciator clears.

   b. **Restore** the affected control bank(s) above the limit within 2 hours.

WINDOW D04
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. Take necessary action to **restore** control banks above the RIL Lo-Lo limit as soon as possible.

    a. IF an RCS dilution is in progress, **stop** dilution.

    b. IF a plant transient is in progress, attempt to **stabilize** load.

    c. **Verify** that CVCS blended flow is at the correct boron concentration.

2. **Restore** stable plant conditions until the cause of the alarm is identified and corrected.

3. As soon as the control banks RIL Lo-Lo limit annunciator clears, **stop** Emergency Boration.

4. **Continue boration** and reactor power and temperature adjustments as required to **restore** control bank(s) above the RIL Low Limit (annunciator clears).

5. **Refer To** Technical Specification LCO 3.1.6, Control Rod Insertion Limits for LCO requirements.

6. IF alarm CANNOT be reset, begin recording demand position and insertion limit in accordance with 14915-1, "Special Condition Surveillance Logs" (This satisfies Technical Specification SR 3.1.6.2).

WINDOW D04
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

1. **Initiate** Data Sheet 5 of 14915-1, "Special Conditions Surveillance Logs"

2. **Log** corrective actions to repair the disabled annunciator or reasons for no action on 10018-C, "Annunciator Control", Figure 2.

3. **Log** compensatory actions on 10018-C, "Annunciator Control", Figure 5.

END OF SUB-PROCEDURE

REFERENCE: DCP 1081846201

SHUTDOWN MARGIN

Shutdown margin (SDM) is the instantaneous amount of reactivity that core is, or can be made, subcritical from its present condition with most reactive control rod fully withdrawn from core at any time during core cycle

By the definition a SDM exists at all times for a core. Technical Specifications require shutdown margin with the most reactive rod withdrawn from the core.

The required shutdown margin varies depending on the mode of operation of the plant and can be as high as 1.3% $\Delta k/k$.

DETERMINING SDM WHEN THE PLANT IS SHUT DOWN

When the plant is shutdown, the SDM is usually equal to the amount by which the core is actually subcritical. As a result, changes to the plant such as temperature changes or boron concentration changes inevitably change the SDM.

Specific details of how SDM is calculated when the reactor is shutdown vary. Some plants approach the issue very broadly in a way that is intended to make life easy for the operators. Reactor engineers determine what the minimum boron concentration would be at specific times in core life in order to meet Tech. Spec. requirements. They then provide a curve to the operators who simply confirm on a regular (usually daily) basis that the boron concentration is greater than or equal to this value, thereby demonstrating adequate SDM. This method is very conservative.

Other plants require that a reactivity balance be performed by the operator to determine that adequate SDM exists. This method is much more flexible and often results in a calculated required boron concentration that is much less than the conservative value calculated by the other method. It generally requires more work and more vigilance on the part of the operators to ensure that SDM requirements are always met.

Some variability exists, when shutdown, as to how the one stuck rod criterion is dealt with. With all the rods fully inserted, some plants will allow full credit for all the rods, while other plants will still assume that the most reactive rod is fully withdrawn. It depends on how the Tech. Spec. is written as to what a plant can do.

DETERMINING SDM WHEN THE PLANT IS CRITICAL

When the plant is critical there is obviously no net reactivity in the core. SDM assumes that the rods will all go in from their current condition, except that the most reactive rod will remain fully withdrawn from the core. The reactivity associated with the most reactive rod must be determined by the reactor engineers and provided to the operators so that SDM can be calculated.

If rods become inoperable in a way that it is unclear whether or not they will insert on a reactor trip, then they must be added to the amount of reactivity that will NOT be inserted upon a trip occurring.

The negative reactivity of the rods is partially offset by the positive reactivity that results from a reactor trip. The amount of positive reactivity added due to power reduction depends on the time in core life and the power level of the reactor. Reactor engineers must provide this information to the operator to allow calculations of SDM while at power.

The SDM is the net negative reactivity after the trip. Thus, the rods must have reactivity equal to the power defect PLUS the SDM requirement (assuming the most reactive rod remains fully withdrawn).

$$SDM = \rho_{Rods} - \rho_{Power\ Defect}$$

Equation 8-13

Changes to other reactivity effects are not considered in the SDM. For example: It is assumed that Xenon reactivity will remain the same after a trip as it was immediately before the trip. We can assume this, because we know that Xenon will in fact start to increase in magnitude, thus creating a larger SDM, in the short term. (Later on it will decrease but that is the concern of the operators when it happens, not to us when we are at power attempting to determine the AT-POWER SDM).

During power operation, the effect of xenon building up to equilibrium value is offset by boron dilution. The boron concentration that exists just before the trip is assumed to be the boron concentration after the trip – which is a very good assumption.

Changes that happen to all reactivities after the trip will need to be dealt with when they occur.

When the reactor is still at power, some reactivity changes will affect the SDM of the reactor while others will not. Rod motion, generally, will not change the SDM since it adds reactivity at the time of movement that would otherwise be added at the time of trip. Boron concentration changes almost always affect SDM.

Common exam questions require an ability to determine if SDM has been affected.

Which ONE of the following will cause the SDM of an operating reactor to decrease:

a. Rods are inserted 4 steps to lower T_{ave} 2°F

b. Boron concentration is increased to lower T_{ave} 2°F

c. T_{ave} goes up 2°F due to a decrease in Xenon concentration

d. T_{ave} drifts down 2°F over a period of 1 day due to fuel burnup

Example 8-30

In this example each possible answer needs to be analyzed.

Rods are inserted and T_{ave} goes down: The rod insertion reduces the height from which the rods will insert, thus reducing the amount of negative reactivity that they will add on a reactor trip. However, the result of the insertion at power is a reduction in the temperature. Thus there will be a smaller cooldown following the trip and the amount of positive reactivity added by the power defect will be smaller. These two reactivities in fact are exactly the same magnitude by in opposite directions. Thus there is no change in SDM in this answer.

Boron concentration is increased to lower Tave: To analyze this, consider where the reactor will end up on trip, before and after the boron concentration change. Before the concentration change, the reactor would end up at normal operating temperature for zero power with x boron concentration. The second situation would result in the same temperature etc., but with the additional boron in the core. Thus, this move would increase the SDM. Thus it is not the answer.

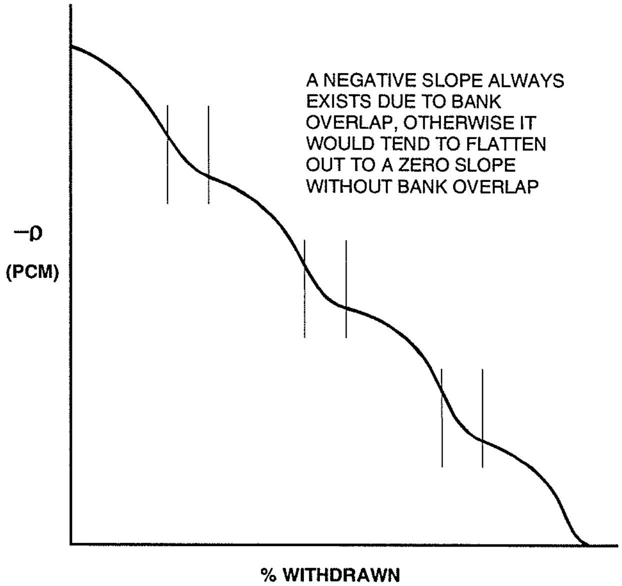


Figure 5-19 Control Bank Overlap Effect on Integral Rod Worth

ROD INSERTION LIMITS

Although the control rods may be positioned axially anywhere in the core, the rods must be above a specified height during reactor operations. This height is referred to as the rod insertion limit. During reactor operations, the rods must be maintained above the rod height specified by Figure 5-20 for the given power level.

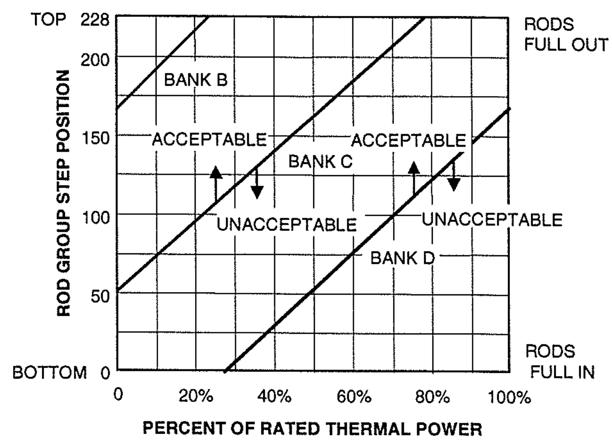


Figure 5-20 Rod Height vs. Power Level

Note: The example shown here is from Westinghouse PWR employing a magnetic jack stepping mechanism with each step measuring 5/8 inch. Hence, 228 steps correspond to about 12 feet.

The rod insertion limits minimize the consequences of an ejected rod accident, guarantee a sufficient shutdown margin from a given power level, and produce an axial flux distribution which prevents high local peak power levels. Maintaining the control rods high in the core, at full power conditions, prevents an ejected control rod from inserting too much positive reactivity. With the control rods in their almost fully withdrawn or fully withdrawn position in a PWR, the amount of reactivity inserted by this event should tend to be small enough so as not to create an uncontrolled power excursion. Instead, such an event should actually create a small-break loss-of-coolant-accident

(SBLOCA), due to being initiated by a rupture of a control rod drive housing.

When the reactor trips, positive reactivity is added by the power defect as well as any subsequent temperature decrease below no-load T_{avg}. Rod insertion limits ensure that the control rods have sufficient negative reactivity to shutdown the reactor from a given power level with a sufficient shutdown margin to maintain the reactor in a safe shutdown condition.

If the rods are inserted too far into the core, the power production in the core is suppressed in the top of the core, raising the power production in the bottom of the core. The higher power in the bottom of the core could cause abnormally high fuel temperatures and fuel damage.

AXIAL FLUX DIFFERENCE

The axial flux difference ($\Delta\phi$ or ΔI) is the difference in the power level (difference in currents, ΔI) between excore power range detectors monitoring the upper and lower halves of the core (Figure 5-21.)

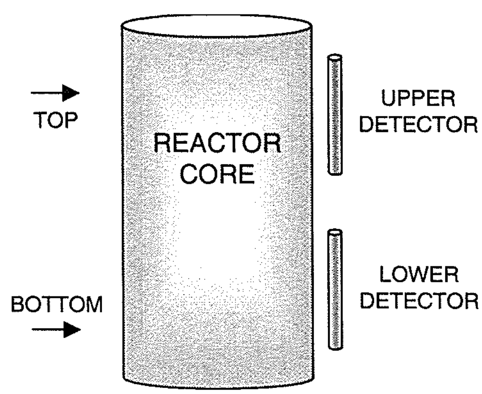


Figure 5-21 Power Range Detectors Used to Determine Axial Flux Difference

This difference is also proportional to the difference in axial neutron flux between the upper and lower halves of the core and may be expressed as:

$$\Delta I = I_{top} - I_{bottom}$$

Equation 5-9

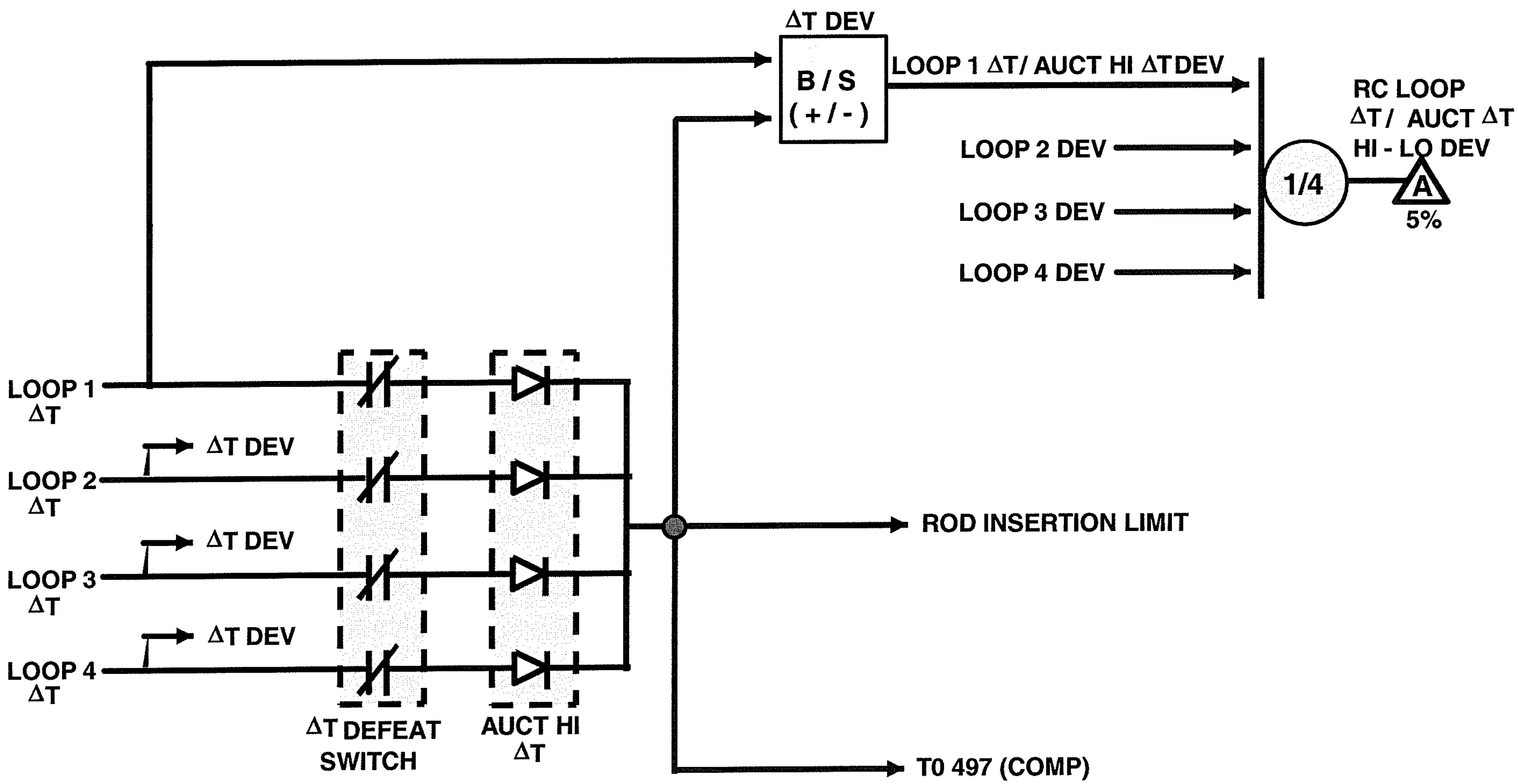
or

$$\Delta\phi = \phi_{top} - \phi_{bottom}$$

Equation 5-10

The axial flux difference is maintained in a specified band to ensure a more uniform axial flux distribution by preventing a high peak power in either the top or bottom of the core. A high peak power results in a high fission product concentration in that location. The decay heat generated by these fission products could overheat the fuel during a loss of coolant accident. Under most operating conditions, the axial flux difference limitation is more restrictive than the rod insertion limits. Control rod position is used to maintain the axial flux difference within the allowed operating range.

LOP 16101 Slide # 20



DELTA T CONTROL

4. 003K5.02 001/2/1/RCP - COASTDOWN/MEM 2.8/3.2/BANK - HL-17 AUDIT/HL-18 NRC/RO/SRO/AML

Initial conditions:

- Unit 1 is starting up following an outage.
- The crew is in 12004DF-1, "Power Operation (Mode 1)."
- Feedwater is being transferred from AFW to MFW.

Current conditions:

- After transfer of SG #1 and # 2 to MFW, RCP #2 trips.

Per 18005-C, "Partial Loss of Flow", which ONE of the following completes the following statement?

Following the RCP trip and as RCS flow lowers during coastdown, the DNBR will ___(1)___

and

the crew will be procedurally directed to ___(2)___.

A.✓ (1) decrease

(2) initiate a reactor shutdown per 12004-C

B. (1) decrease

(2) immediately trip the reactor per 18005-C

C. (1) increase

(2) initiate a reactor shutdown per 12004-C

D. (1) increase

(2) immediately trip the reactor per 18005-C

003K5.02 Reactor Coolant Pump System (RCPS)

Knowledge of the operational implications of the following concepts as they apply to the RCPS: (CFR: 41.5 / 45.7)

Effects of RCP coastdown on RCS parameters.

K/A MATCH ANALYSIS:

correct DNBR response following the RCP trip and coastdown. The candidate must also determine whether an immediate reactor trip is required for the present power level or a controlled shutdown is required.

DISTRACTOR ANALYSIS

A. Correct. The DNBR ratio will decrease, with power less than 15%, a reactor trip is not required per step # 1 of 18005-C.

B. Incorrect. DNBR has decreased. If reactor power was >15%, the correct response would be to TRIP the reactor per step # 1 of 18005-C, a reactor trip is not required per 18005-C at this power level.

C. Incorrect. DNBR has decreased, not increased. If reactor power was >15%, the correct response would be to TRIP the reactor per step # 1 of 18005-C, a reactor trip is not required per 18005-C at this power level.

D. Incorrect. DNBR has decreased, not increased. If reactor power was >15%, the correct response would be to TRIP the reactor per step # 1 of 18005-C, a reactor trip is not required per 18005-C at this power level.

REFERENCES

18005-C, "Partial Loss of Flow"
Tech Spec 3.4.4 Bases, RCS Loops Modes 1 and 2
LO-TX-34700, pg 7
HL-17 Audit Question #3

Learning Objectives

LO-LP-60301-03 Describe how the plant will respond to RCS loop low flow conditions under the following:

1) PWR less than P-7.

LO-LP-60305-08 Given conditions and/or indications, determine the required AOP to enter (including subsections, as applicable).

LO-LP-60305-09 Given the entire AOP, describe:

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

BASES

APPLICABLE SAFETY ANALYSES (continued)	All of the accident/safety analyses performed at full rated thermal power assume that all four RCS loops are in operation as an initial condition. Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to conservatively bound lower modes of operation. The events which assume only two RCPs in operation include the uncontrolled RCCA (Bank) withdrawal from subcritical and the rod ejection events. While all accident/safety analyses performed at full rate thermal power assume that all the RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced RCS flow, reactor coolant pump rotor seizure, and reactor coolant pump shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assumed thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down. By ensuring that the plant operates with all RCS loops in operation in MODES 1 and 2, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops — MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power. An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

(continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check Reactor power - LESS THAN OR EQUAL TO 15%.	1. Perform the following: a. Trip the Reactor. b. Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
2. Stop any power changes in progress.	
3. Initiate the Continuous Actions Page.	
*4. **Check affected loop SG NR Level - TRENDING TO 65%.**	*4. Control feed flow to maintain affected loop SG NR level between 60% and 70%.
5. Check Tavg - TRENDING TO PROGRAM.	5. Adjust control rods to restore Tavg.
6. Verify PRZR level - TRENDING TO PROGRAM.	
7. Verify PRZR pressure - TRENDING TO 2235 PSIG.	
8. Check RCP 1 and RCP 4 – RUNNING.	8. Close the affected loop spray valve: Loop 1: PIC-0455C Loop 4: PIC-0455B
9. Initiate shutdown to Mode 3 by initiating 12004-C, POWER OPERATION (MODE 1). (TS 3.4.4)	
10. Determine and correct the cause of the pump trip.	
11. Check shutdown to Mode 3 – COMPLETE.	11. Return to Step 9.

DEPARTURE FROM NUCLEATE BOILING RATIO

One PWR thermal limit is DNBR, departure from nucleate boiling ratio. DNBR is the ratio of critical heat flux to actual heat flux. DNBR is maintained greater than 1.3 during all modes of operation, ensuring DNB is not reached. When DNBR is maintained greater than or equal to 1.3, there is 95% confidence that 95% of the fuel rods have not exceeded the heat flux resulting in DNB. When DNBR is maintained greater than or equal to 1.74, this confidence factor increases to 99.9%.

$$DNBR = \frac{\text{Critical Heat Flux at Specific Location}}{\text{Actual Heat at Same Location}}$$

Equation 8-1

In PWR operation at low power levels, nucleate boiling does not occur and heat transfer is by convection. During fully developed nucleate boiling at high power levels, the heat transfer coefficient increases.

To keep DNBR above 1.3, the operator monitors RCS temperature, pressure, flow rate, and reactor power level. During full power operation, the hottest primary coolant is subcooled by approximately 30°F. The heat transfer regime in the core is in the bubbly flow region with some subcooled nucleate boiling. If hot leg temperature were increased inadvertently, by boron dilution or rod withdrawal, saturated boiling could occur in the upper regions of the core, causing DNBR to decrease to a less conservative value. To prevent DNBR from falling below 1.3, the control room operators must keep RCS temperatures within their prescribed limits.

If temperature is maintained and RCS pressure reduced, DNBR will decrease. A reduction in pressure shifts the boiling curve to the left. Thus, operating at lower pressures allows DNB to occur at lower temperatures. Pressure could be inadvertently reduced by a pressurizer pressure controller failure, a stuck open spray valve, or PORV (pressure operated relief valve).

At any power level, a reduction in RCS flowrate will result in an increase in coolant temperature, again reducing DNBR.

The fourth factor that reduces DNBR is high local power densities. High local power densities produce higher heat flux, and higher coolant and cladding temperatures. As a result, the heat transfer conditions more closely approach actual CHF conditions and the DNBR is reduced.

1. 003K5.02 001/2/1/RCP-DNB ON TRIP/F -/BANK-N ANNA 2010/HL-17 AUDIT/RO/SML/

Initial conditions:

- Unit 1 is starting up following a mid-cycle forced outage.
- The crew is in 12004-C, "Power Operation (Mode 1)".
- Feedwater is being transferred from AFW to MFW.

Current conditions:

- After transfer of S/G 1 and 2 to MFW, RCP 2 trips.

Per 18005-C, "Partial Loss of Flow", which one of the following correctly completes the following statement?

Following the RCP trip, the DNBR will ___(1)___

and

the crew will be procedurally driven to ___(2)___ .

A✓ (1) decrease

(2) initiate a reactor shutdown per 12004-C

B. (1) decrease

(2) immediately trip the reactor per 18005-C

C. (1) increase

(2) initiate a reactor shutdown per 12004-C

D. (1) increase

(2) immediately trip the reactor per 18005-C

5. 004K5.27 001/2/1/CVCS - N2 PURGE/MEM - 2.6/3.2/NEW/HL-18 NRC/RO/SRO/TNT

Unit 1 is preparing for a refueling outage.

Which ONE of the following completes the following statement?

The VCT gas space will be purged with ___(1)___ in order to ___(2)___ in the 24 hour period preceding a reactor shutdown.

	__(1)__	__(2)__
A.	hydrazine	reduce oxygen to minimize corrosion
B.	hydrazine	allow partial removal of dissolved hydrogen
C.	nitrogen	reduce oxygen to minimize corrosion
D✓	nitrogen	allow partial removal of dissolved hydrogen

004K5.27 Chemical and Volume Control System

Knowledge of the operational implications of the following concepts as they apply to the CVCS: (CFR: 41.5 / 45.7)

Reason for nitrogen purge of CVCS.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a unit is about to be shutdown and degassing of the RCS will be performed prior to shutdown. The candidate is required to have knowledge of which gas is added to VCT gas space and the reason for the gas addition.

DISTRACTOR ANALYSIS:

A. Incorrect. Per 13007-1, section 4.7, Nitrogen is used to allow removal of dissolved hydrogen preceding reactor shutdown. Hydrazine is plausible as a choice as it is frequently added to the RCS for oxygen scavenging. It is plausible the candidate may confuse the chemical added.

B. Incorrect. Per 13007-1, section 4.7, Nitrogen is used to allow removal of dissolved hydrogen preceding reactor shutdown. Hydrazine is plausible as a choice as it is frequently added to the RCS for oxygen scavenging. It is plausible the candidate may confuse the chemical added or think the purpose is to remove hydrogen which is the chemical we want to remove.

C. Incorrect. Per 13007-1, section 4.7, Nitrogen is used to allow removal of dissolved hydrogen preceding reactor shutdown. It is plausible the candidate may confuse the chemical that we are removing since we also have limits on O2 concentrations and both H2 and O2 have explosive atmosphere concerns.

D. Correct. Nitrogen is added to the VCT for degassing to remove Hydrogen.

REFERENCES:

13007-1, "VCT Gas Control and RCS Chemical Addition"

VEGP learning objectives:

LO-PP-46101-09 Describe how VCT purge is established and maintained.

LO-PP-46101-13 Describe how the waste gas system is used during a unit shutdown to degas the RCS.

INITIALS

4.7 DISSOLVED HYDROGEN REDUCTION PRIOR TO REACTOR SHUTDOWN

NOTE

- This section should be used to dilute the VCT gas space with nitrogen to allow partial removal of dissolved hydrogen in the 72 hour period (lower limit: 25 cc/kg) and in the 24 hour period (lower limit: 15 cc/kg) preceding reactor shutdown. ☐
- After shutdown Section 4.9 is used to degas to <5 cc/kg dissolved hydrogen as rapidly as practical. ☐
- The target VCT gas space volume % hydrogen is 55% to reach 15 cc/kg at 20 to 25 psig VCT pressure. At higher VCT pressures a lower volume % hydrogen will be required to approach 15 cc/kg. ☐
- IVs in this section should be documented on Checklist 3 ☐

4.7.1 **Maintain** VCT level at 40 to 50% as indicated by 1-LI-0112, except WHEN adjusting VCT level during nitrogen dilution. ________

4.7.2 **Request** Chemistry sample and analysis of VCT gas space volume % hydrogen. ________

6. 005K2.01 001/2/1/RHR - PUMPS/C/A - 3.0/3.2/NEW/HL-18 NRC/RO/SRO/TNT

Initial Unit 1 conditions:

- RCS temperature is 340°F.
- 'B' RHR in service for RCS cooling.
- 'A' RHR is aligned for ECCS injection.
- The SAT is aligned to 1BA03.

Current conditions:

- An LOSP occurs to both RATs.
- DG1B does NOT start.
- Plant Wilson is unaffected.

Which ONE of the following is correct regarding the status of the RHR pumps' respective 4160V bus breakers?

	'A' RHR PUMP	'B' RHR PUMP
A.	open	open
B.✓	open	closed
C.	closed	open
D.	closed	closed

005K2.01 Residual Heat Removal System (RHRS)

Knowledge of bus power supplies to the following: (CFR: 41.7)

RHR pumps.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where one train of RHR is aligned for shutdown cooling and one train aligned for ECCS injection. The SAT is aligned to provide power to the RHR train providing shutdown cooling. The candidate has to determine which RHR pump 4160V bus output breakers are open / closed after an LOSP condition, load shed, etc.

DISTRACTOR ANALYSIS:

A. Incorrect. RHR pump A output breaker will be open and RHR pump B output breaker will remain closed as it is powered via the SAT and will be unaffected by the LOSP. DG1B does NOT start since 1BA03 will remain energized via the SAT. It is plausible the candidate will think both pump output breakers are open due to the LOSP to both RATs. The candidate may think RHR pump B breaker will be open due to NOT starting on the LOSP.

B. Correct. RHR pump A breaker will remain open and NOT close on the LOSP. RHR pump B breaker will remain closed since 1BA03 is powered via the SAT and a load shed will NOT occur with 1BA03 remaining energized.

C. Incorrect. It is plausible the candidate may think that RHR pump A loads on the UV sequence when DG1A starts, this is a common misconception among initial students. The part for RHR pump B output breaker being open is correct.

D. Incorrect. It is plausible the candidate may think that RHR pump A loads on the UV sequence when DG1A starts, this is a common misconception among initial students. The student may also realize the SAT will still provide power to 1BA03 and RHR pump B output breaker will remain closed.

REFERENCES:

1X3D-AA-K02A, Diesel Generator 1A and Train "A" AC Busses Loading Table
LO-PP-28201, Sequencer

VEGP learning objectives:

LO-PP-12101-02	State the sources for: RHR pump motor power
LO-PP-12101-03	State the auto start signal(s) for the RHR pumps.
LO-PP-01101-01	List all offsite electrical power sources.

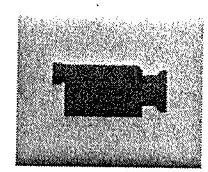
EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION	PROJECT CLASSIFICATION (ELECTRICAL SYSTEM)	PROCUREMENT SPECIFICATION	BUS NUMBER	BUS VOLTAGE	NAMEPLATE RATING HP (OR KW)	REFERENCE DWG. 1X3D-	LOSS OF COOLANT ACCIDENT SI PREFERRED OFFSITE SOURCE AVAILABLE					LOSS OF OFFSITE POWER & SUBSEQUENT SI					LOSS OF OFFSITE POWER					EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION
										MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SECONDS) TO START (NOTE 7)	RUNNING TIME AFTER START (NOTE 4)			MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.b)	RUNNING TIME AFTER START (NOTE 4)			MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.c)	RUNNING TIME AFTER START (NOTE 4)		
11202P4001M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04A			S	25.5	C			S	25.5	C			S	25.5	C	11540B7003M01	NSCW TWR CAB TUN VENT FAN
11202P4003M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04C			S	25.5	C			S	25.5	C			S	25.5	C	1HV-8104 *	BORIC ACID TANK TO CHARGE PUM
11202P4005M01	NUC SERV CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	700	BD-K04E			NOTE. 5	30.5	—			NOTE 5	30.5	—			NOTE 5	30.5	—	1HV-8116 *	CHARGING PUMP 1A DISCHARGE
11203P4001M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01A			S	20.5	C			S	20.5	C			S	20.5	C	1HV-8106	CHG PUMP TO RCS ISO
11203P4003M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01C			S	20.5	C			S	20.5	C			S	20.5	C	1HV-1668B	NUC SERV CL TWR A BYPASS VALVE
11203P4005M01	COMP CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	300	BD-L01E			NOTE 5	25.5	—			NOTE 5	25.5	—			NOTE 5	25.5	—	1HV-8804A	RHR HEXCH TO CHG PUMP
11204P6003M01	SAFETY INJ PP	11E	X6	1AA02	4160V	450	BD-D01C			S	5.5	C			S	5.5	C			M	>30.5	C	1HV-8809A	RHR TNA TO SIS COLD LEG ISO
11205P6001M01	RESID HT REMVL PP	11E	X6	1AA02	4160V	400	BD-E01A			S	10.5	C			S	10.5	C			M	>30.5	C	1HV-8835 *	SIS COLD LEG LOOP INLET HDR ISO
11206P6001M01	CONTAINMENT SPRAY PP	11E	X6	1AA02	4160V	400	BD-J01A			NOTE 6	15.5	C			NOTE 6	15.5	C			M	>30.5	C	1HV-1668A	NUC SERV CL TWR A ISO VALVE
11208P6002M01	CENTRIFUGAL CHARG PP	11E	X6	1AA02	4160V	600 **	BD-C01A			S	0.5	C			S	0.5	C			S	0.5	C	1LV-0112B	VOL CONT TANK OUT ISO
11217P4001M01	AUX COMP CLG WTR PP	11E	X4	1AA02	4160V	600	BD-L03A			M	>30.5	C			M	>30.5	C			S	15.5	C	1PV-2550A	PIPING PEN RM A TO ATMOS
11302P4003M01	AUX FDWTR PP-MOTOR DRIVEN	11E	X4	1AA02	4160V	900	BC-F04A			S	20.5	C			S	20.5	C			S	20.5	C	1HV-1975 *	AUX CCW RETRN ISO
11592C7001M01	ESF CHILLER (COMPRESSOR)	11E	X4	1AA02	4160V	400	BG-G02A			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV-1979 *	AUX CCW SUPPLY ISO
																							DISTR PANEL 1AYB1	DISTRIBUTION PANEL
11805S3B04	LOAD CENTER TRANS. 1AB04	11E	X3	1AB04	4160 -480V	1000KVA	BA-D02F			—	—	C			—	0	C			—	0	C	1PV-3000	ATMOSPHERIC DUMP VALVE
	SECONDARY BREAKER 1AB04	11E	X3	1AB04	480V	—	BA-E02A			—	—	C			S	0.5	C			S	0.5	C	1PV-3030	ATMOSPHERIC DUMP VALVE
11501A7001M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01A			NOTE 10	NA	NA			NOTE 10	NA	NA			S NOTE 15	50.5	C	1HV-11600	NSCW PUMP P4001 OUT
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03F			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	1HV-5137	AUX FD PP P4003 DISCH MOV
11501A7002M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01B			NOTE 10	NA	NA			NOTE 10	NA	NA			S NOTE 15	50.5	C	1HV-5139	AUX FD PP P4003 DISCH MOV
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03G			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	11540B7005M01 *	TB AB ELEC TUNNEL VENT FAN
11501A7005M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01E			NOTE 10	NA	NA			NOTE 10	NA	NA			S	30.5	C	PNL 1AYC1	120/240 DISTRIB PANEL
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03K			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	11532B7001M01	CBSF BATT RM EXH FAN
11501A7006M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01F			NOTE 10	NA	NA			NOTE 10	NA	NA			S	30.5	C	11532B7003M01	CBSF BATT RM EXH FAN (STBY)
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03L			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	11532A7001M01	CBSF ELEC EQUIP RM A/C
																							11539A7001M01 NOTE 19	CB AUX RELAY RM A C
11805S3B05	LOAD CENTER TRANS. 1AB05	11E	X3	1AB05	4.16 -480V	1000KVA	BA-D02L			—	—	C			—	0	C			—	0	C	11807Y310	REG XFMR 1ABC09RX
	SECONDARY BREAKER 1AB05	11E	X3	1AB05	480V	—	BA-E02P			—	—	C			S	0.5	C			S	0.5	C		
11566B7001M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01B			I	>30.5	C			I	>30.5	C			I	>30.5	C		
11566B7003M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01C			P	>30.5	C			P	>30.5	C			P	>30.5	C		
11531N7001H01	CB CONTROL ROOM HTR	11E	X4	1AB05	480V	118KW	BG-C05R			N NOTE 11	>30.5	C			N NOTE 11	>30.5	C			N NOTE 11	>30.5	C	11808T3116	LTG ISOLATION XFMR 1ABC23RX
11531N7001M01	CB CONTROL ROOM FIL UNIT	11E	X4	1AB05	480V	125	BG-C01E			P	>30.5	C			P	>30.5	C			P	>30.5	C		
11513H7001000	CTB HYDROGEN RECOMBINER	11E	X6	1AB05	480V	75KW	BG-B02U			M	>30.5	C			M	>30.5	C			M	>30.5	C		
11805S3B15	LOAD CENTER TRANS. 1AB15	11E	X3	1AB15	4.16 -480V	1000KVA	BA-D02K			—	—	C			—	0	C			—	0	C	11807Y3RX25	ROD POS IND CAB ISO XFMR 1ABC20RX
	SECONDARY BREAKER 1AB15	11E	X3	1AB15	480V	—	BA-E02R			—	—	C			S	0.5	C			S	0.5	C	1HV-2584A	CTB COOLING UNIT (DAMPER)
11202W4001M01	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03A			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV-2584B	CTB COOLING UNIT (DAMPER)
11202W4001M02	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03B			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV-12742	CBSF BATTERY RM EXH
11202W4001M03	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03C			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV-2628A	CTB NORM PURGE EXH ISO
11202W4001M04	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03D			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV-12748	CBSF BATTERY RM FAN (STBY)
11213P6002M01	SPENT FUEL PIT PUMP	11E	X6	1AB15	480V	100	BD-H01B			M	>30.5	C			M	>30.5	C			M	>30.5	C	1HV-8808C	ACCUM LOOP NO. 3
11561N7001M01	PIPE PEN RM FILTER/EXH FAN	11E	X4	1AB15	480V	75	BG-D01D			S	15.5	C			S	15.5	C			S NOTE 17	15.5	C	1HV-19055 *	THERM BARR CW RCP 003
11561N7001H01	PIPE PEN FLTR/EXH UNIT HTR	11E	X4	1AB15	480V	80KW	BG-D01C			I,K NOTE 17	15.5	C			I,K NOTE 17	15.5	C			I,K NOTE 17	15.5	C	1HV-19057 *	THERM BARR CW RCP 004

UNDERVOLTAGE SEQUENCE

LO-PP-28201
Sequencer
Power Point lesson

- Sequencer detects UV condition on the 1E bus
- Sequencer sends start signal to train related D/G (Emergency Start)
- Sequencer sends signal to Non-Sequenced loads
- Sequencer performs a LOAD SHED on train related 4160 VAC bus
- Sequencer generates BLOCK AUTO/MANUAL signal
- Sequencer sends a D/G OUTPUT BRKR closure permissive when 1E bus LOAD SHED completed
- Sequencer runs the following load sequence:

– STEP 1	0.5 sec	CCP & 1E SWGR
– STEP 2	5.5 sec	NONE
– STEP 3	10.5 sec	Stub Bus (NB01/NB10)
– STEP 4	15.5 sec	ACCW
– STEP 5	20.5 sec	CCW & AFW
– STEP 6	25.5 sec	NSCW & spare CCW
– STEP 7 **	30.5 sec	CNMT Coolers (fast) & spare NSCW



- 2 of 4 coolers start at step 7, next 2 start after 20 second time delay

7. 006A3.08 001/2/1/ECCS - AUTO XFER/C/A - 4.2/4.3/NEW/HL-18 NRC/RO/SRO/KAJ

Given the following plant conditions:

- A large break LOCA is in progress on Unit 1.

- RWST SI TEST LIGHT 'A' white light is lit on the QMCB 'A' panel.

- RWST SI TEST LIGHT 'B' white light is NOT lit on the QMCB 'A' panel.

Current condition:

- RWST level is 28% on all channels and slowly lowering.

Assuming no additional operator actions, which ONE of the following completes the following statement?

1-HV-8811A, CNMT SUMP TO RHR PMP-A SUCTION, will indicate ___(1)___,

and

1-HV-8811B, CNMT SUMP TO RHR PMP-B SUCTION, will indicate ___(2)___ on the main control board.

A. (1) open

(2) open

B✓ (1) open

(2) closed

C. (1) closed

(2) open

D. (1) closed

(2) closed

006A3.08 Emergency Core Cooling System (ECCS)

Ability to monitor automatic operation of the ECCS, including:
(CFR: 41.7 / 45.5)

Automatic transfer of ECCS flowpath.

K/A MATCH ANALYSIS:

The candidate is presented a plausible scenario during a DBA LOCA where one of the RWST SI TEST lights on the QMCB fail to illuminate. The candidate has to determine when RWST level lowers below the auto swapover setpoint whether or not the HV-8811A/B Containment Sump Suctions to the RHR pumps open. This also requires the candidate to know the RWST auto swapover setpoint.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. With RWST level < 28% the train A sump suction valve will open. With the Train B RWST SI Test light not lit, the train B sump suction valve will not open.

B. Correct. With RWST level < 28% the train A sump suction valve will open. With the Train B RWST SI Test light not lit, the train B sump suction valve will not open.

C. Incorrect. This choice is plausible if the candidate inverts the effect of the RWST SI TEST lights.

D. Incorrect. This choice is plausible if the candidate inverts the effect of the RWST SI TEST lights and does not correlate the Containment Sump level with the automatic swapover setpoint.

REFERENCES:

1X4-DB-122, Residual Heat Removal
1X3D-BD-E03F, HV-8811A elementary
V-LO-TX-28101-09, Reactor Protection System

VEGP learning objectives:

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

5. RELAY 8811AX IS AN ISOLATION RELAY. COIL CKT IS TRAIN A & OUTPUT CONTACT CKT'S ARE TRAIN C.

6. THE TEST LIGHT SHOULD BE LOCATED NEAR THE SAFETY INJECTION RWST RESET SWITCH WHICH IS SHOWN ON THE SOLID STATE INTERCONNECTION DIAGRAM. PART OF 2LB-40135

7. JUNCTION BOX 1ARJB3642 IS ALSO SHOWN IN DWG 1X3D-BD-DO2L.EO2E & G.

8. HS-8811A IS LOCATED AT (B1). INTERNAL CONNECTION BETWEEN (B1) AND (B) IS BY WESTINGHOUSE. THE PREFAB CABLE CONNECTOR IS LOCATED AT (B).

.SP6.SP7 (LB) YBBY

9. ALSO SHOWN ON DWG. 1X3D-BD-D02L & E02G.

B

10. WIRE NOS C4 & C5 ARE SHOWN ON 1X3D-BD-D02L.

11. CONDUCTORS NOS. 1,13,16,17 SHOWN SPARE ARE DAMAGED NOT TO BE USED IN FUTURE.

12. CONDUCTORS NOS. 6,7,8 SHOWN SPARE ARE DAMAGED NOT TO BE USED IN FUTURE.

Q

2.
7 (LC) YBBY SEE NOTE 12

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
RESIDUAL HEAT REMOVAL SYSTEM
1HV-8811A

A

-99	ELC	EOG	JGM	
MICROFILM FOR SIGNATURES				
E	DR	CHK	APPV	DTL

SCALE: NONE	DRAWING NO.	REV.
JOB NO.10604	1X3D-BD-E03F	9

SIZE F 34X44 CAD NAME

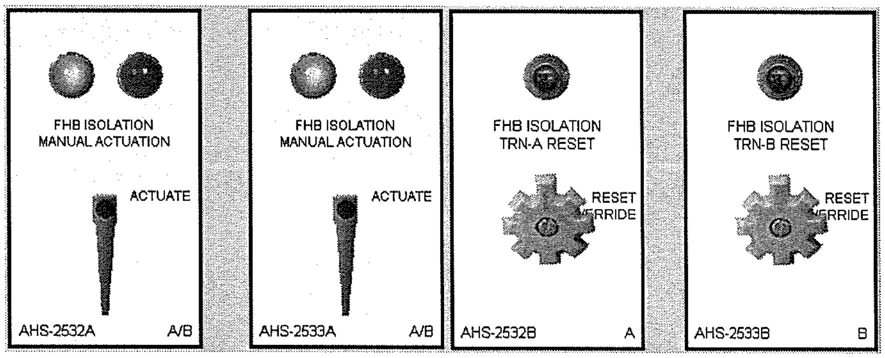
*** Radiation Monitors can be blocked by the use of the Test Block Switch located in the Unit QESF panel.**

Equipment affected by the Fuel Handling Building Ventilation Isolation Actuation

1) Fuel Handling Building Post Accident Filter Units

2) Fuel Handling Building Normal HVAC isolates (the normal supply and exhaust units trip on low flow).

Resetting Fuel Handling Ventilation Isolation Actuation



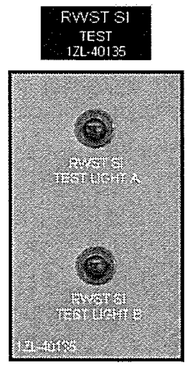
There are two Fuel Handling Building Ventilation Isolation Reset/Override switches located on the Main Control HVAC panel.

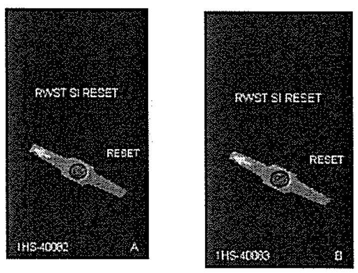
Both reset/override switches must be used to reset or override the actuation signals.

The white light on the Reset/Override switches indicate that the Actuating signal is still present but has been overridden. This allows the Fuel Handling Building Post Accident units to be secured and the Normal HVAC realigned with the actuating signal present.

RWST Safety Injection

The purpose for the RWST Safety Injection signal is to arm the RHR pump suction "Semi Auto Swap Over" from the RWST to the Containment Sump. The reason it is called Semi is because only one valve in each train is automatically re-positioned. The RWST SI is activated on the receipt of a SI signal. This can be verified by the illumination of the "RWST SI test lights"* located on the "A" panel in the control room. This circuit is necessary to ensure that the Semi Auto Swap over is still active even after the safety injection signal is reset. The set points for the "semi auto swap over" are the RWST SI signal and 2 out of the 4 RWST level channels lower to ≤ 29 %. When the "Semi Auto Swap Over" set points have been satisfied, the RHR Containment Suction Valves automatically open. The operator is required to complete the swap over by manually closing the RHR suction valves from the RWST. This is performed once the suction source from the Containment is verified. The emergency operating procedure will guide the operators through this process.





To Reset the RWST SI signal:

- Safety Injection is Reset
- And the RWST SI Reset switches are momentarily taken to the “Reset” Position.

- **Called RWST Test Lights because they can be lamp tested by depressing each lens cover.**

Commitments:

FF 89.021

Recovery from ESF Actuation

1 Procedure 11886-1

  a. Recovery from SI
  b. Recovery from LOSP
  c. Recovery from CIA
  d. Recovery from CRI
  e. Recovery from CVI
  f. Recovery from a SGBD isolation

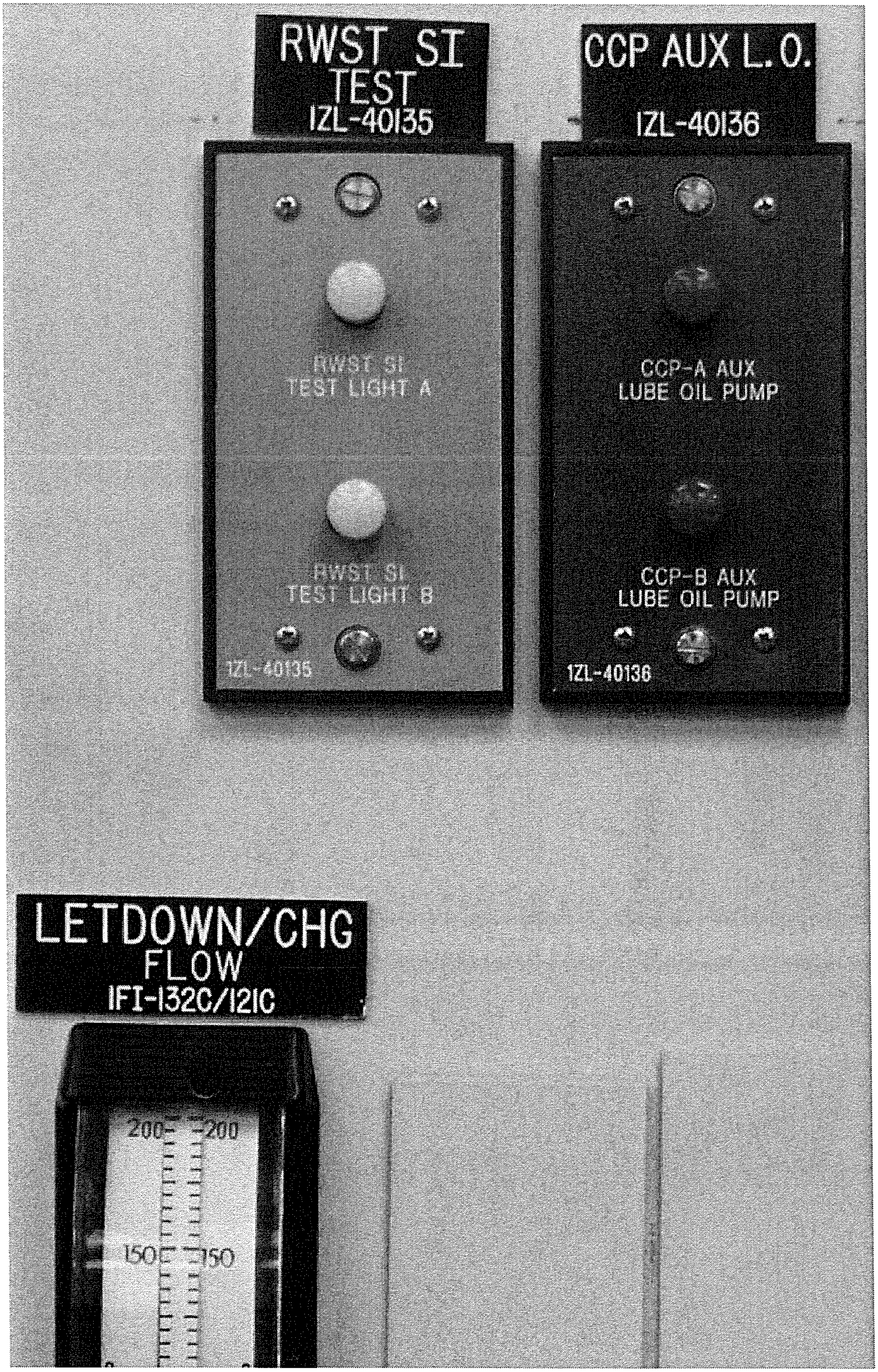
2 Each of the subsections of the procedure provide guidance for restoring normal plant equipment status following associated ESF actuations

3 Procedure Entry

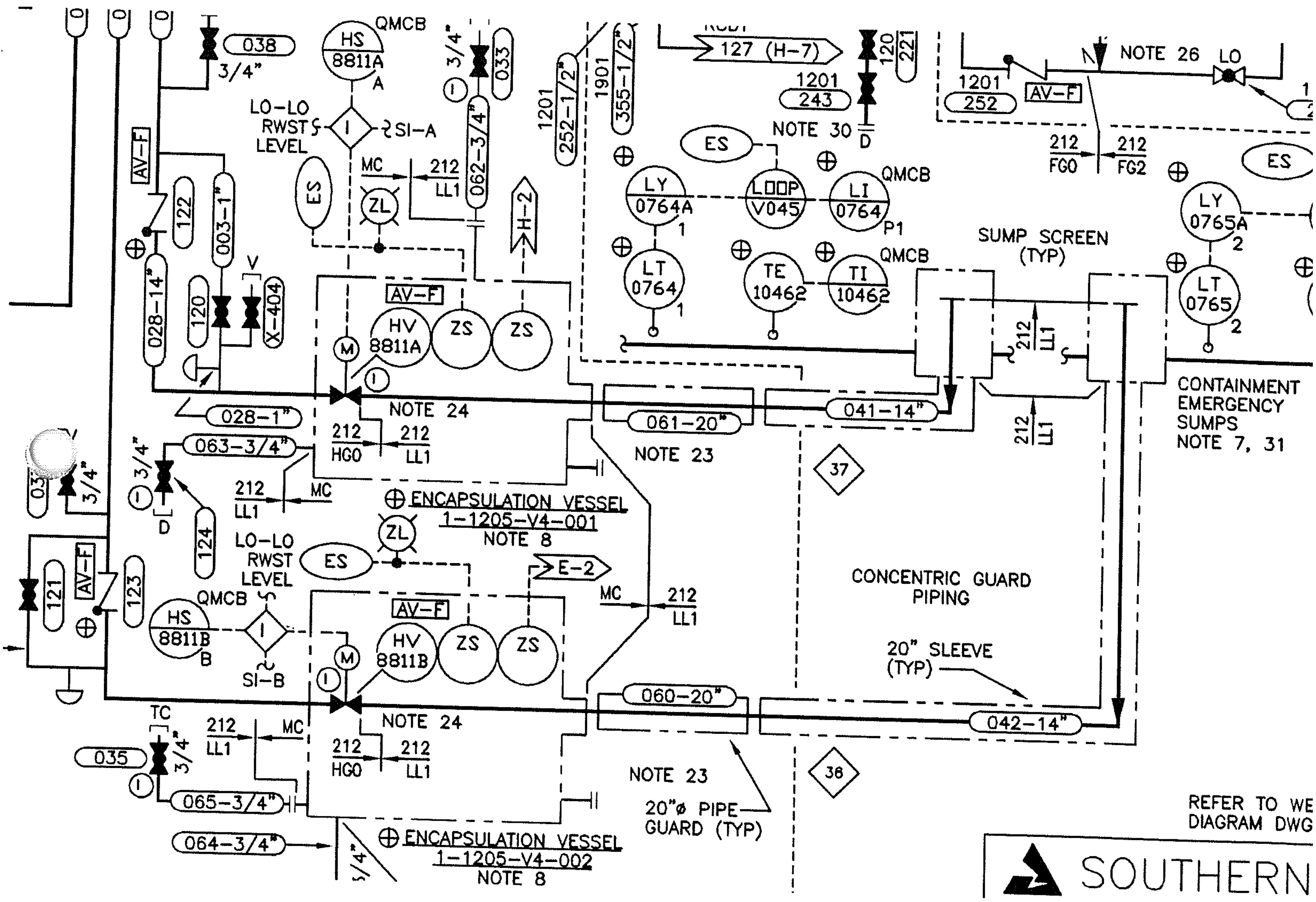
  a. Recovery from safety injection entry from 19011 “SI Termination”
  b. the remaining subsections currently have no direct procedural entry path.

This procedure entry shall occur when the recovery from the associated ESF actuation is desired.

FF 86.008


RWST SI
TEST
1ZL-40135
CCP AUX L.O.
1ZL-40136
RWST SI
TEST LIGHT A
RWST SI
TEST LIGHT B
1ZL-40135
CCP-A AUX
LUBE OIL PUMP
CCP-B AUX
LUBE OIL PUMP
1ZL-40136
LETDOWN/CHG
FLOW
1FI-132C/121C
200
200
150
150

1X4DB122
RHR Sump Suction Valves



8. 007A3.01 001/2/1/PRT - COMPONENTS/MEM - 2.7/2.9/NEW/HL-18 NRC/RO/SRO/TNT

Unit 1 is in Mode 5.

- RHR is in the shutdown cooling mode of operation.

The following alarm illuminates:

- ALB12-E02 PRZR REL TANK HI PRESS

Which one of the following completes the following statement?

The ___(1)___ Relief Valve lifting will cause the PRT High Pressure condition,

and

with no operator action to mitigate the condition, PI-469, PRT Pressure, on the QMCB will reach a maximum of ___(2)___ psig.

A. (1) RHR pump discharge

(2) 115

B. (1) RHR pump suction

(2) 115

C. (1) RHR pump discharge

(2) 100

D✓ (1) RHR pump suction

(2) 100

007A3.01 Pressurizer Relief Tank / Quench Tank System (PRTS)

Ability to monitor automatic operation of the PRTS, including: (CFR: 41.7 / 45.5)

Components which discharge to the PRT.

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario where a relief valve is leaking to the PRT causing a high temperature condition. The candidate has to determine which relief valve from a list can cause the condition and the maximum pressure that the PRT

will reach with no operator action. This meets the KA since the candidate has to monitor or discern components which can discharge to the PRT and the highest pressure which the PRT will reach.

DISTRACTOR ANALYSIS:

A. Incorrect. Both parts are incorrect. The RHR pump discharge, and not suction, relieves to the RHUT. The PRT will rupture between 86 to 100 psig as read on the QMCB meter. It is plausible the candidate may think the PRT pressure will go to 115 psig due to confusing psig with psia. Also, the PRT high temperature alarm is 115F.

B. Incorrect. The RHR pump suction is correct. The second part is incorrect. The PRT will rupture between 86 to 100 psig as read on the QMCB meter. It is plausible the candidate may think the PRT pressure will go to 115 psig due to confusing psig with psia. Also, the PRT high temperature alarm is 115F.

C. Incorrect. The first part is incorrect. RHR pump discharge, and not suction, relieves to the RHUT. Per the SOP and ARP, the PRT will fail between 86 psig to 100 psig.

D. Correct. RHR pump suction relieves to the PRT. Per the SOP and ARP, the PRT will fail between 86 psig to 100 psig.

REFERENCES:

17012-1, E02 for PRT High Pressure.

13004-1, "Pressurizer Relief Tank Operation", Limitation 2.2.2

1X4DB121, Safety Injection System print for cold leg injection.

1X4DB122, Residual Heat Removal print for suction from RCS loops.

VEGP learning objectives:

LO-PP-09100-04 For each of the following relief valves, state the lift setpoint and discharge point.

a. high pressure letdown relief valve upstream of PV-131, PSV 8117
b. low pressure letdown relief valve downstream of PV-131, PSV 8119

LO-PP-16301-01 List the sources that input into the PRT.

LO-PP-16301-11 Describe the reason for the PRT rupture discs and state the approximate rupture pressure.

LO-PP-12101-06 State the lift setpoints for the RHR suction and discharge relief valves and the discharge flow paths for each.

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 Equipment that has been exposed to air must be purged before it is connected to the Vent Header to prevent explosive mixtures of hydrogen and oxygen. ________

2.1.2 A nitrogen gas blanket should be maintained in the PRT to exclude air and prevent the formation of an explosive hydrogen and oxygen mixture. ________

2.1.3 Before opening the PRT to atmosphere, verify that the gas space hydrogen content is less than 4% and the oxygen content is less than 5%. ________

2.2 LIMITATIONS

2.2.1 If the PRT High Temperature Alarm (115°F) is annunciated, the contents of the PRT shall be cooled. ________

2.2.2 The PRT rupture disk will fail at 86 to 100 PSIG. ________

3.0 PREREQUISITES AND INITIAL CONDITIONS

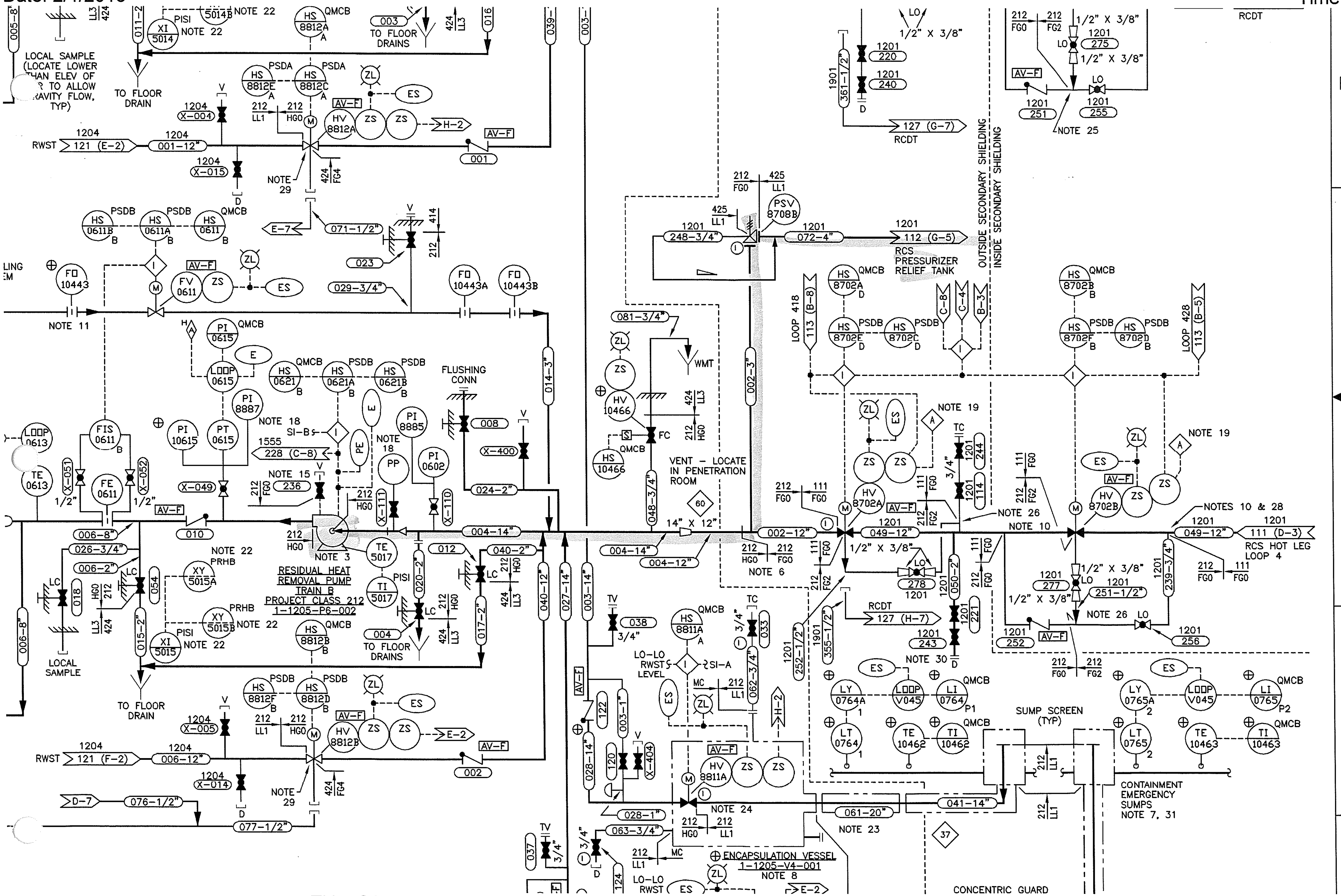
3.1 The Gaseous Waste Processing System is available to provide processing of gases from the PRT. ________

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

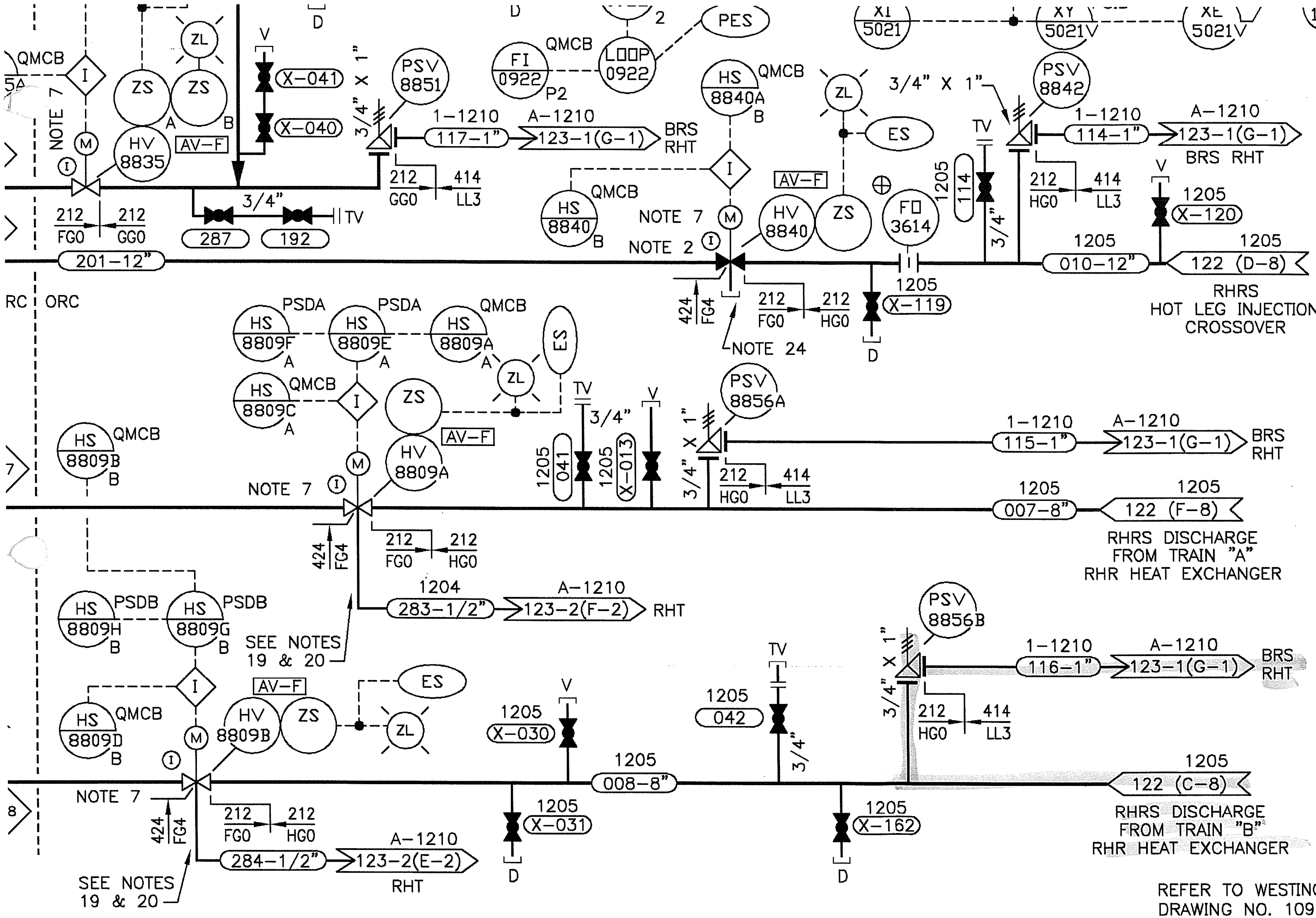
3.3 Reactor Make-Up Water is available to provide a cooling spray for the PRT. ________

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

Date: 2/1/2013 Time : 12:56:21 PM



Title: C:\DATA\HL-18 License Class\HL-18 References\Procedures\Prints, Logics\P&ID\1X4DB122.cal



18. THE PIPING FROM THE OUTLET OF THE RELIEF VALVES TO THE RWST IS SPECIFIED AS 414. HOWEVER, THIS PIPING SHALL BE SUPPORTED AND PROTECTED FROM EXTERNAL HAZARDS/EVENTS SO THAT THE PRESSURE BOUNDARY REMAINS INTACT & THAT THE LINE ALLOWS FLOW FOLLOWING A SAFE SHUTDOWN EARTHQUAKE.
19. REMOVABLE SPOOL.
20. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
21. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK REQUIREMENTS.
22. DRILL VENT HOLE ON THE CONTAINMENT SIDE OF THE VALVE DISC.
23. FOR ISI TESTING USE ONLY, ROOT VALVES TO BE NORMALLY CLOSED.
24. THE VALVE STEM LEAKOFF LINE HAS BEEN REMOVED AND THE VALVE LEAKOFF CONNECTION HAS BEEN CAPPED.
25. PRESSURE RELIEF VALVE WILL VENT TO CONTAINMENT ATMOSPHERE PREVENTING THERMAL OVERPRESSURIZATION OF PIPE DURING DBA.
26. ORIFICE ASSEMBLY; SEE VENDOR DRAWING 1X6AG01–10001 & 10002.

REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E71 SHEET 3 OF 3.

CAD 1X4DB121

ACAD2004 | KB – 02

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
42.0	REVISED PER ABN–1071073501M004, VER. 1.0	10/20/09	KB	CFC	JMR	X

SIZE

5 4 3 2

WINDOW E02

ORIGIN	SETPOINT	
1-PT-0469	8 psig	PRZR REL TANK HI PRESS

1.0 PROBABLE CAUSE

1. One or more of the following valves has lifted or is leaking to the Pressurizer Relief Tank:

   a. Pressurizer Safety Valves,

   b. Pressurizer (PRZR) Power Operated Relief Valves (PORV)s,

   c. Chemical and Volume Control System (CVCS) Letdown Relief Valve 1-PSV-8117,

   d. CVCS Seal Return Relief Valve 1-PSV-8121,

   e. Residual Heat Removal (RHR) Relief Valves 1-PSV-8708A and B during shutdown conditions.

2. Nitrogen Regulator malfunction.

3. Safety grade letdown in use and aligned to the Pressurizer Relief Tank.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

WINDOW E02
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

If PRT pressure increases the PRT rupture disk will fail at 86 to 100 psig, opening the PRT to containment.

1. **Determine** actual Pressurizer Relief Tank pressure using 1-PI-0469 on the QMCB.

2. **Monitor** Pressurizer Relief Tank temperature, level, and pressure.

3. **Check** tailpipe temperatures for the Pressurizer Safety Valves, Power Operated Relief Valves, and CVCS Letdown Relief Valve.

4. <u>IF</u> a PRZR PORV <u>OR</u> Safety Valve has actuated, **check** valve closure when pressure is lowered in the Reactor Coolant System.

5. <u>IF</u> a nitrogen supply malfunction has occurred, **isolate** the supply by shutting valves 1-HV-8033 and 1-HV-8047.

6. <u>IF</u> a Pressurizer Safety Valve is open <u>OR</u> fails to close following an actuation, **Go To** 18004-C, "Reactor Coolant System Leakage."

7. <u>IF</u> a PRZR PORV 455A/456A is open <u>OR</u> fails to close following an actuation:

   a. **Place** the Control Switch for the affected valve to the closed position,

   b. <u>IF</u> the affected valve will <u>NOT</u> close, **close** the associated Block Valve,

   c. **Refer** to Technical Specification LCO 3.4.11.

8. <u>IF</u> the pressure rise is due to the CVCS Letdown Relief Valve being open **isolate** letdown, and **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

9. IF the pressure rise is due to a failed RHR Relief Valve, **isolate** the affected Train of RHR and **initiate** 18019-C, "Loss Of Residual Heat Removal."

10. IF the pressure rise is due to a failed Seal Return Relief Valve, attempt to **isolate** the leak.

11. IF pressure rise is due to a hard bubble, i.e., no temperature or level change, **notify** Chemistry and **place** Pressurizer Steam Space Sample in service and **control** RCS pressure using 12004-C.

12. **Restore** pressure in the Pressurizer Relief Tank to normal per 13004-1, "Pressurizer Relief Tank Operation."

13. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, PLS

9. 007EK1.02 001/1/1/REACTOR TRIP -SDM/MEM - 3.4/3.8/NEW/HL-18 NRC/RO/SRO/TNT

Given the following plant conditions:

- The reactor trips from 100% power.
- The crew is performing 19001-C, "Reactor Trip Response."
- CBD rod H-8 is stuck at 36 steps.

Which ONE of the following completes the following statement?

Per 19001-C, for the given plant conditions, the crew ____(1)____ required to verify adequate Shutdown Margin,

and

Emergency Boration of the RCS ____(2)____ required.

	__(1)__	__(2)__
A.	is	is
B.✓	is	is NOT
C.	is NOT	is
D.	is NOT	is NOT

007EK1.02 Reactor Trip

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR: 41.8 / 41.10 / 45.3)

Shutdown Margin

<u>K/A MATCH ANALYSIS:</u>

The question presents a plausible scenario where a reactor trip has occurred and one control rod is stuck partially out. The candidate has to determine whether a shutdown margin is required to be performed and an emergency boration. The question meets the KA due to a shutdown margin performance is an operational implication due to the stuck rod post reactor trip.

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect - A shutdown margin is required to be performed per step 7 RNO of 19001-C. An emergency boration is NOT required to be performed unless 2 or more rods are not fully inserted. It is plausible the candidate

may think both a shutdown margin and emergency boration are required due to a stuck control rod.

B. Correct - A shutdown margin is required to be performed per step 7 RNO of 19001-C. An emergency boration is NOT required to be performed unless 2 or more rods are not fully inserted.

C. Incorrect - It is plausible a candidate may think a shutdown margin is NOT required since the definition of shutdown margin assumes the highest worth rod is stuck fully out and there is only one rod stuck. An emergency boration is NOT required to be performed unless 2 or more rods are not fully inserted. It is plausible the candidate may think an emergency boration is still required due to a stuck rod.

D. Incorrect - It is plausible a candidate may think a shutdown margin is NOT required since the definition of shutdown margin assumes the highest worth rod is stuck fully out and there is only one rod stuck. An emergency boration is NOT required to be performed unless 2 or more rods are not fully inserted. It is plausible the candidate may think an emergency boration is still required due to a stuck rod.

REFERENCES:

19001-C, "Reactor Trip Response", step # 7

VEGP learning objectives:

LO-LP-33800-27, Evaluate changes in shutdown margin due to changes in plant parameters.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	__4) Open MFP discharge valve.
	__5) Slowly raise MFP speed using GE pot as necessary.
	__b. Verify MFRVs and BFRVs controllers at 0% demand.
	__c. Reset FW Isolation.
	__d. Open BFIV(s) as necessary.
	__e. Open BFRV(s) as necessary.
__7. Check all Rods - FULLY INSERTED.	__7. IF two or more Rods NOT fully inserted, THEN EMERGENCY BORATE 154 ppm for each Rod not fully inserted by initiating 13009, CVCS REACTOR MAKUP CONTROL SYSTEM.
	__ Verify adequate shutdown margin as required by Technical Specification SR 3.1.1.1.

10. 008AG2.4.02 001/1/1/EP - PZR VAPOR SPACE/MEM - 4.5/4.6/BANK-HL16 NRC/HL-18 NRC/RO/SRO/TNT

The following conditions exist:

- The unit is at 5% power following a start up.
- A Pressurizer Safety valve fails open.

With no operator action taken, which ONE of the following will first require the crew to perform 19000-C, "Reactor Trip or Safety Injection"?

A. OP Delta T Reactor Trip

B. OT Delta T Reactor Trip

C. Low Pressurizer Pressure Reactor Trip

D.✓ Pressurizer Pressure Safety Injection Reactor Trip

008AG2.4.02 Pressurizer Vapor Space Accident

Knowledge of system set points, interlock and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)

K/A MATCH ANALYSIS

The question test the ability of the student to determine from a list of Reactor Trips the one that is applicable for the plant conditions.

DISTRACTOR ANALYSIS

A. Incorrect: OPDeltaT is a variable setpoint with no input from Pressurizer Pressure. This trip will not occur due to pressurizer pressure but the inputs are often confused with the OTDeltaT setpoint.

B. Incorrect: OTDeltaT is a variable setpoint with input from Pressurizer Pressure, Tavg and Neutron Flux. With Delta Tat a very low power level it would take a very long time and pressure would have to decrease significantly to reach the setpoint.

C. Incorrect: Low Pressurizer Pressure Rx trip is only applicable with power above the P-7 setpoint (2/4 Power Range NI channels at 10% or 1/2 Turbine Impulse pressure channels at 10%).

D. Correct: Pressurizer Low Pressure Safety Injection is always applicable and would be the first event to trip the reactor in these conditions.

REFERENCES

17009-1 "Annunciator Response for ALB09 Panel Window A04
Functional Diagrams 1X6AA02-00229 and 1X6AA02-00230
Tech Spec LCO 3.3.1, pages 14 through 19

VEGP learning objectives:

LO-PP-28103-02 List all permissives with applicable set points, coincidences and functions.

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

WINDOW A04

PRZR PRESS SI RX TRIP

ORIGIN	SETPOINT
Any two of the following:	1870 psig

a. 1-PT-0455
b. 1-PT-0456
c. 1-PT-0457
d. 1-PT-0458

1.0 **PROBABLE CAUSE**

1. Loss of coolant accident or Steam Generator tube rupture
2. Pressurizer Pressure Control System malfunctions

2.0 **AUTOMATIC ACTIONS**

> NOTE
> This function is manually blocked below P-11 permissive.

1. Reactor Trip
2. Turbine Trip
3. Safety Injection

3.0 **INITIAL OPERATOR ACTIONS**

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

4.0 **SUBSEQUENT OPERATOR ACTIONS**

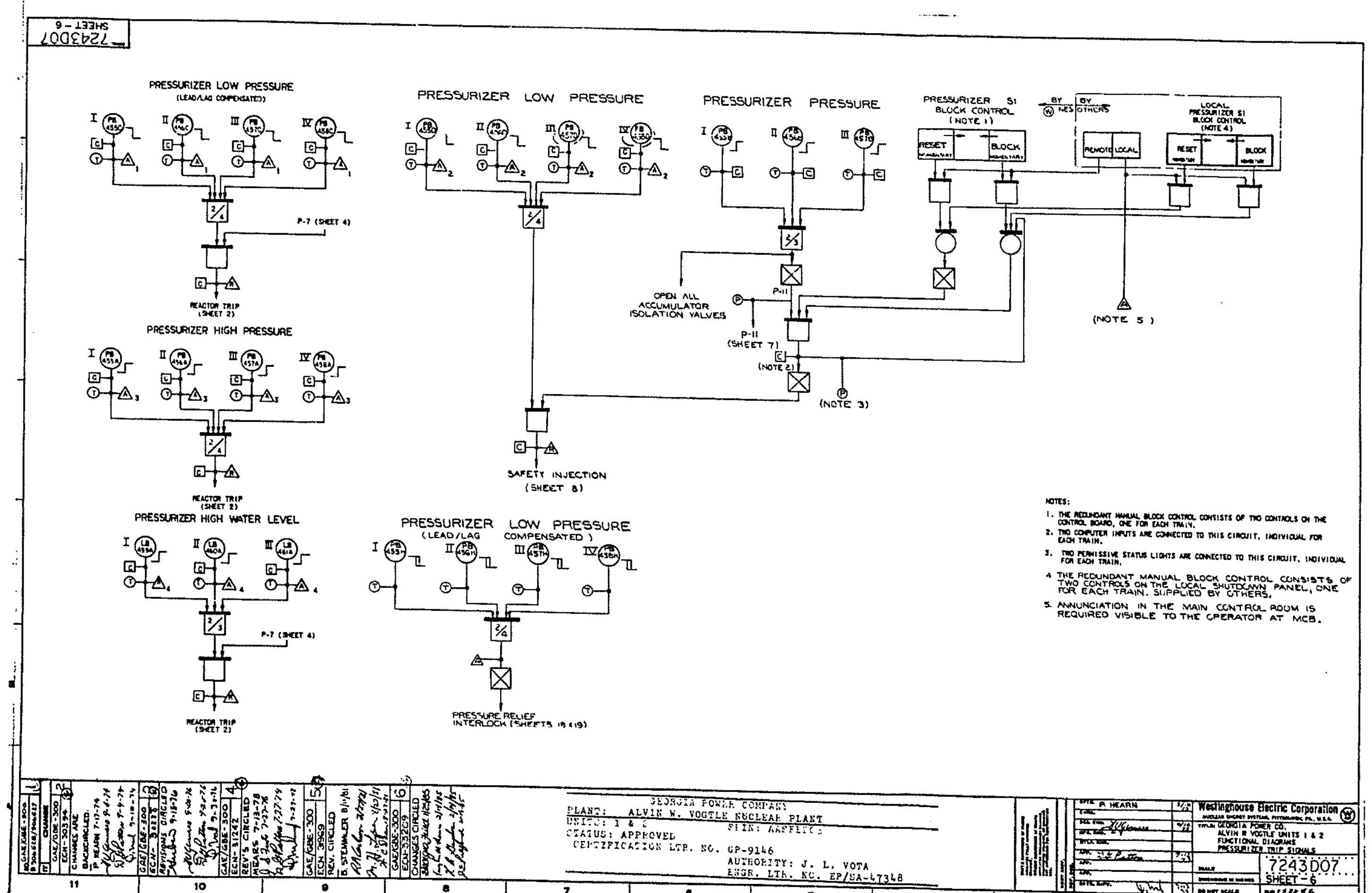
NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: FSAR Section 7.2, 1X6AA02-230, PLS, Technical Specifications LCO 3.3.2, and 3.3.1



VOGTLE ELECTRIC GENERATING PLANT — JOB NO. 9510

EQUIPMENT TAG NO. various

STARTUP DESIGNATION NO. various
ACTIVITY NO. various
SYSTEM NO. various
CATEGORY NO. N/A

RETROFITTING REQUIRED ☐ YES ☒ NO

DISTRIBUTION TO:	FOR: REVIEW	INFO.
• MECHANICAL		
• HVAC		
NSSS		
BOP		
• CONTROL SYSTEMS	✓	
• ELECTRICAL	✓	
WIRING		
CONDUIT		
• HAZARDS		
• CIVIL/STRUCTURAL		
• NUCLEAR		
• STRESS/PLANT DESIGN		
CODES AND STANDARDS		
• ARCHITECTURAL		
• STARTUP		
• CONSTRUCTION		
• NOT REQ'D BY ENGRG		
• CLIENT		
• EQUIP. QUALIFICATION		
• M & QS		
• WESTINGHOUSE		

IDENTIFYING TITLE OF THIS DOCUMENT: functional diag.

Bechtel Log No. 1X6AA02-230-8

IMPORTANT
Permission to proceed does not constitute acceptance or approval of design details, calculations, analyses, test methods or materials developed or selected by the supplier and does not relieve supplier from full compliance with contractual obligations.

DATE RECEIVED 3-14-85 SIGNED [signature]

DOCUMENT STATUS
1 ☒ WORK MAY PROCEED
2 ☐ REVISE AND RESUBMIT. WORK MAY PROCEED SUBJECT TO INCORPORATION OF CHANGES INDICATED
3 ☐ REVISE AND RESUBMIT. WORK MAY NOT PROCEED
4 ☐ INFORMATION ONLY DISTRIBUTION REQ'D? ☐ YES
M ☐ RESUBMIT – NOT ACCEPTABLE FOR MICROFILM. WORK MAY PROCEED

DATE 5/3/85

PF-8957 (9310) 9/84

I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL

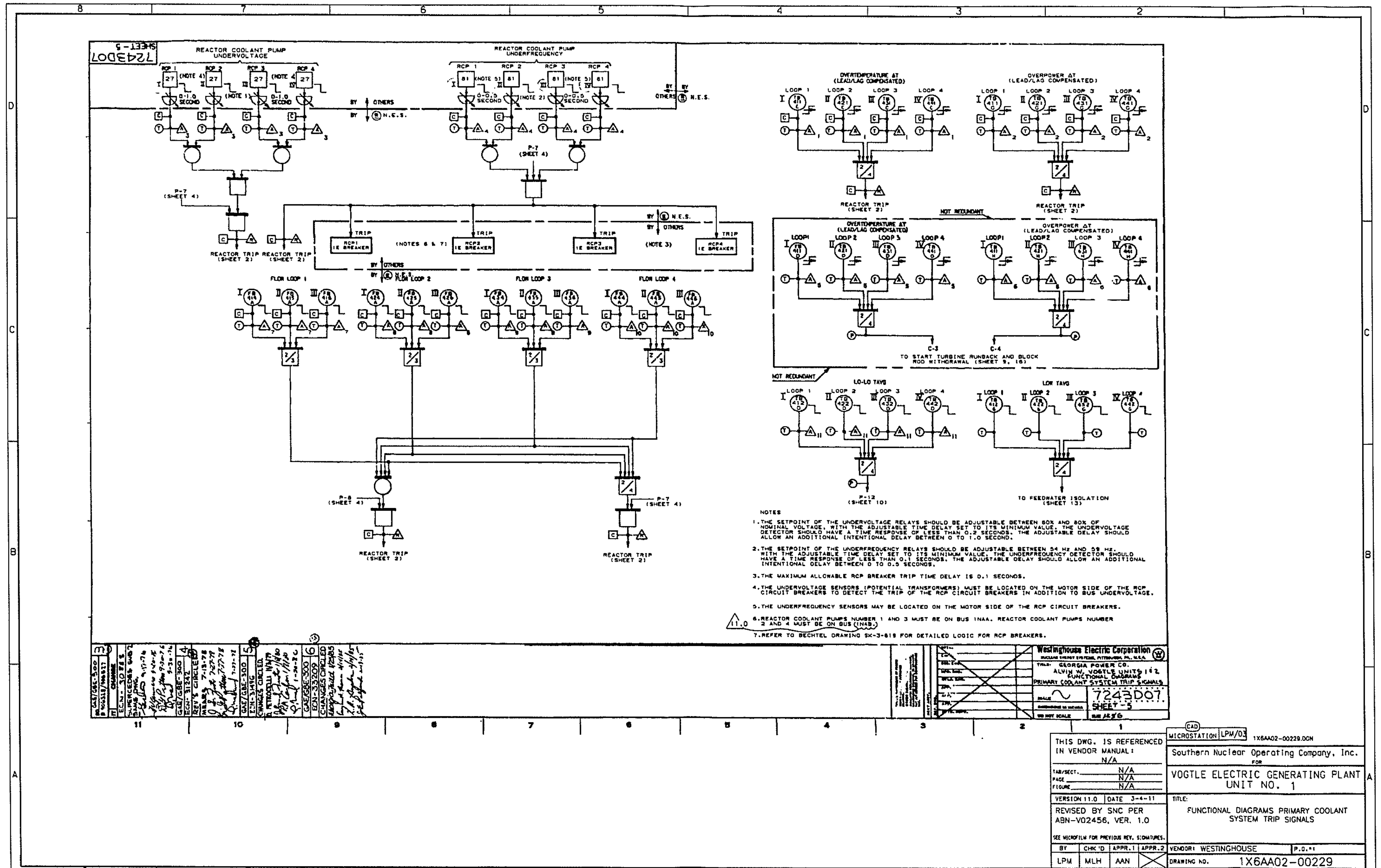

REACTOR COOLANT PUMP UNDERVOLTAGE
REACTOR COOLANT PUMP UNDERFREQUENCY
REACTOR TRIP (SHEET 2)
TO START TURBINE RUNBACK AND BLOCK ROD WITHDRAWAL (SHEET 9, 16)
TO FEEDWATER ISOLATION (SHEET 13)
Westinghouse Electric Corporation
THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
Southern Nuclear Operating Company, Inc.
VOGTLE ELECTRIC GENERATING PLANT UNIT NO. 1
VERSION 11.0
DATE 3-4-11
REVISED BY SNC PER ABN-V02456, VER. 1.0
FUNCTIONAL DIAGRAMS PRIMARY COOLANT SYSTEM TRIP SIGNALS
VENDOR: WESTINGHOUSE
LPM
MLH
AAN
1X6AA02-00229

Table 3.3.1-1 (page 1 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA	NA
		3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA	NA
2.	Power Range Neutron Flux						
	a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7(n)(o) SR 3.3.1.11(n)(o) SR 3.3.1.15	≤ 111.3% RTP	109% RTP
	b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8(n)(o) SR 3.3.1.11(n)(o) SR 3.3.1.15	≤ 27.3% RTP	25% RTP
3.	Power Range Neutron Flux High Positive Rate	1,2	4	E	SR 3.3.1.7(n)(o) SR 3.3.1.11(n)(o) SR 3.3.1.15	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8(n)(o) SR 3.3.1.11(n)(o)	≤ 41.9% RTP	25% RTP
		2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8(n)(o) SR 3.3.1.11(n)(o)	≤ 41.9% RTP	25% RTP

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(n) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(o) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.1-1 (page 2 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5. Source Range Neutron Flux	$2^{(d)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8(n)(o) SR 3.3.1.11(n)(o)	≤ 1.7 E5 cps	1.0 E5 cps
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.11(n)(o)	≤ 1.7 E5 cps	1.0 E5 cps
	$3^{(e)}, 4^{(e)}, 5^{(e)}$	1	L	SR 3.3.1.1 SR 3.3.1.11(n)(o)	NA	NA
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-20)	Refer to Note 1 (Page 3.3.1-20)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-21)	Refer to Note 2 (Page 3.3.1-21)

(continued)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide input to the High Flux at Shutdown Alarm System (LCO 3.3.8) and indication.

(n) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(o) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
8.	Pressurizer Pressure						
	a. Low	1(f)	4	M	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 1950 psig	1960(g) psig
	b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	≤ 2395 psig	2385 psig
9.	Pressurizer Water Level - High	1(f)	3	M	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o)	≤ 93.9%	92%
10.	Reactor Coolant Flow - Low						
	a. Single Loop	1(h)	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 89.4%	90%
	b. Two Loops	1(i)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 89.4%	90%

(continued)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(g) Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag.

(h) Above the P-8 (Power Range Neutron Flux) interlock.

(i) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(n) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(o) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
11.	Undervoltage RCPs	1(f)	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 9481 V	9600 V
12.	Underfrequency RCPs	1(f)	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 57.1 Hz	57.3 Hz
13.	Steam Generator (SG) Water Level - Low Low	1,2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7(n)(o) SR 3.3.1.10(n)(o) SR 3.3.1.15	≥ 35.9%	37.8%

(continued)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(n) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(o) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
14. Turbine Trip						
a. Low Fluid Oil Pressure	1(j)	3	O	SR 3.3.1.10(n)(o) SR 3.3.1.16	≥ 500 psig	580 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.14	≥ 90% open	96.7% open
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.13	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 1.2E-5% RTP	2.0E-5% RTP
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50.3% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 40.6% RTP	40% RTP
e. Power Range Neutron Flux, P-10 and input to P-7	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	(l,m)	(l,m)
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.10 SR 3.3.1.12	≤ 12.3% Impulse Pressure Equivalent turbine	10% Impulse Pressure Equivalent turbine

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(j) Above the P-9 (Power Range Neutron Flux) interlock.
(l) For the P-10 input to P-7, the Allowable Value is ≤ 12.3% RTP and the Nominal Trip Setpoint is 10% RTP.
(m) For the Power Range Neutron Flux, P-10, the Allowable Value is ≥ 7.7% RTP and the Nominal Trip Setpoint is 10% RTP.
(n) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(o) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.1-1 (page 6 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
17. Reactor Trip Breakers[k]	1,2	2 trains	T,V	SR 3.3.1.4	NA	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.4	NA	NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	U,V	SR 3.3.1.4	NA	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA	NA
19. Automatic Trip Logic	1,2	2 trains	Q,V	SR 3.3.1.5	NA	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA	NA

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

11. 008K4.09 001/2/1/CCW - STBY FEATURE/C/A - 2.7/2.9/LOIT BANK/HL-18 NRC/RO/SRO/AML

Initial conditions:

- Unit 1 is at 100% power.
- CCW pumps #1 and #5 are running.
- CCW pump #3 is in AUTO standby.

Current sequence of events:

- Safety Injection actuates.
- Two minutes later, a loss of RAT 'A' occurs.
- SI has NOT been reset.

Which ONE of the following correctly describes the status of the 'A' Train CCW system after the completion of the LOSP sequence?

A. No 'A' Train CCW pump is running.

B✓ CCW pumps #1 and #3 are running.

C. CCW pumps #1 and #5 are running.

D. CCW pumps #1, #3, and #5 are running.

008K4.09 Component Cooling Water System (CCWS)

Knowledge of CCWS design feature(s) and/or interlock(s) which provide the following: (CFR: 41.7)

The standby feature for the CCW pumps.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where the CCW system is in service with pump # 3 in standby. Following an SI and LOSP on 1 bus event, the candidate must determine the running CCW pumps. The candidate has to determine the system response with a particular pump in standby, the final system alignment.

DISTRACTOR ANALYSIS:

A. Incorrect - Plausible if the candidate confuses CCW system response with ACCW, no ACCW pumps would be running during this sequence of events.

B. Correct - After SI actuates, all 3 CCW pumps will be running since pump # 3 will get a start signal. However, following the load shed and re-sequence, the SI sequence has priority. Pumps 1 and 3 will be started on the SI sequence.

C. Incorrect - Plausible that the students may believe # 5 starts on the load shed and re-sequence since it was previously running.

D. Incorrect - After SI actuates, all 3 CCW pumps will be running since pump # 3 will get a start signal. However, following the load shed and re-sequence, the SI sequence has priority. Pumps 1 and 3 will be started on the SI sequence so this choice is incorrect.

REFERENCES:

1X3D-AA-K02A, Sequencer Loading Table

VEGP learning objectives:

LO-PP-10101-04 From memory, describe the expected system response and operator corrective actions for each of the following:

a. SI
b. LOSP
c. SI with LOSP

EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION	PROJECT CLASSIFICATION (ELECTRICAL SYSTEM)	PROCUREMENT SPECIFICATION	BUS NUMBER	BUS VOLTAGE	NAMEPLATE RATING HP (OR KW)	REFERENCE DWG. 1X3D-	LOSS OF COOLANT ACCIDENT SI PREFERRED OFFSITE SOURCE AVAILABLE					LOSS OF OFFSITE POWER & SUBSEQUENT SI					LOSS OF OFFSITE POWER					
										MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SECONDS) TO START (NOTE 7)	RUNNING TIME AFTER START (NOTE 4)			MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.b)	RUNNING TIME AFTER START (NOTE 4)			MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.c)	RUNNING TIME AFTER START (NOTE 4)	
11202P4001M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04A			S	25.5	C			S	25.5	C			S	25.5	C	115
11202P4003M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04C			S	25.5	C			S	25.5	C			S	25.5	C	1HV
11202P4005M01	NUC SERV CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	700	BD-K04E			NOTE. 5	30.5	—			NOTE 5	30.5	—			NOTE 5	30.5	—	1HV
11203P4001M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01A			S	20.5	C			S	20.5	C			S	20.5	C	1HV
11203P4003M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01C			S	20.5	C			S	20.5	C			S	20.5	C	1HV
11203P4005M01	COMP CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	300	BD-L01E			NOTE 5	25.5	—			NOTE 5	25.5	—			NOTE 5	25.5	—	1HV
11204P6003M01	SAFETY INJ PP	11E	X6	1AA02	4160V	450	BD-D01C			S	5.5	C			S	5.5	C			M	>30.5	C	1HV
11205P6001M01	RESID HT REMVL PP	11E	X6	1AA02	4160V	400	BD-E01A			S	10.5	C			S	10.5	C			M	>30.5	C	1HV
11206P6001M01	CONTAINMENT SPRAY PP	11E	X6	1AA02	4160V	400	BD-J01A			NOTE 6	15.5	C			NOTE 6	15.5	C			M	>30.5	C	1HV
11208P6002M01	CENTRIFUGAL CHARG PP	11E	X6	1AA02	4160V	600 **	BD-C01A			S	0.5	C			S	0.5	C			S	0.5	C	1LV
11217P4001M01	AUX COMP CLG WTR PP	11E	X4	1AA02	4160V	600	BD-L03A			M	>30.5	C			M	>30.5	C			S	15.5	C	1PV
11302P4003M01	AUX FDWTR PP-MOTOR DRIVEN	11E	X4	1AA02	4160V	900	BC-F04A			S	20.5	C			S	20.5	C			S	20.5	C	1HV
11592C7001M01	ESF CHILLER (COMPRESSOR)	11E	X4	1AA02	4160V	400	BG-G02A			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV
																							DIS
11805S3B04	LOAD CENTER TRANS. 1AB04	11E	X3	1AB04	4160 -480V	1000KVA	BA-D02F			—	—	C			—	0	C			—	0	C	1PV
	SECONDARY BREAKER 1AB04	11E	X3	1AB04	480V	—	BA-E02A			—	—	C			S	0.5	C			S	0.5	C	1PV
11501A7001M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01A			NOTE 10	NA	NA			NOTE 10	NA	NA			S NOTE 15	50.5	C	1HV
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03F			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	1H
11501A7002M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01B			NOTE 10	NA	NA			NOTE 10	NA	NA			S NOTE 15	50.5	C	1HV
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03G			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	115
11501A7005M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01E			NOTE 10	NA	NA			NOTE 10	NA	NA			S	30.5	C	PNL
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03K			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	115
11501A7006M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01F			NOTE 10	NA	NA			NOTE 10	NA	NA			S	30.5	C	115
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03L			S	30.5	C			S	30.5	C			NOTE 10	NA	NA	115
																							1153
11805S3B05	LOAD CENTER TRANS. 1AB05	11E	X3	1AB05	4.16 -480V	1000KVA	BA-D02L			—	—	C			—	0	C			—	0	C	118
	SECONDARY BREAKER 1AB05	11E	X3	1AB05	480V	—	BA-E02P			—	—	C			S	0.5	C			S	0.5	C	
11566B7001M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01B			I	>30.5	C			I	>30.5	C			I	>30.5	C	
11566B7003M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01C			P	>30.5	C			P	>30.5	C			P	>30.5	C	
11531N7001H01	CB CONTROL ROOM HTR	11E	X4	1AB05	480V	118KW	BG-C05R			N NOTE 11	>30.5	C			N NOTE 11	>30.5	C			N NOTE 11	>30.5	C	118
11531N7001M01	CB CONTROL ROOM FIL UNIT	11E	X4	1AB05	480V	125	BG-C01E			P	>30.5	C			P	>30.5	C			P	>30.5	C	
11513H7001000	CTB HYDROGEN RECOMBINER	11E	X6	1AB05	480V	75KW	BG-B02U			M	>30.5	C			M	>30.5	C			M	>30.5	C	
11805S3B15	LOAD CENTER TRANS. 1AB15	11E	X3	1AB15	4.16 -480V	1000KVA	BA-D02K			—	—	C			—	0	C			—	0	C	118
	SECONDARY BREAKER 1AB15	11E	X3	1AB15	480V	—	BA-E02R			—	—	C			S	0.5	C			S	0.5	C	1HV
11202W4001M01	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03A			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV
11202W4001M02	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03B			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV
11202W4001M03	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03C			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV
11202W4001M04	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03D			P	>30.5	C			P	>30.5	C			P	>30.5	C	1HV

12. 009EA1.04 001/1/1/SMALL LOCA - CVCS/C/A - 3.7/3.5/NEW/HL-18 NRC/RO/SRO/KAJ

Valve list as follows:

- LV-112B, VCT Outlet Isolation
- LV-112C, VCT Outlet Isolation
- LV-112D, RWST to CCP A & B Suction

Initial conditions:

- A small break LOCA has occurred.
- SI has been actuated.
- 19010-C, "Loss of Reactor or Secondary Coolant," has been entered.
- LV-112D will NOT open.

- All other ECCS components have operated as expected.

Which one of the following completes the following statement?

Based on the given conditions, the OATC will observe that __(1)__ and stopping the CCPs is __(2)__.

A. (1) LV-112B is shut

(2) NOT required

B. (1) LV-112B is shut

(2) required

C✓ (1) LV-112C is shut

(2) NOT required

D. (1) LV-112C is shut

(2) required

009EA1.04 Small Break LOCA

Ability to operate and monitor the following as they apply to a small break LOCA: (CFR: 41.7 / 45.5 / 45.6)

CVCS

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where LV-112D fails to open on an SI signal. The candidate has to determine the effect of LV-112D on the 2 VCT Outlet Isolation Valves LV-112B and LV-112C. The candidate has to determine whether or not to allow the CCPs to continue to run (operate and monitor). Since LV-112E functions as designed (not explicitly stated in the stem other than all other ECCS components function as designed), the candidate has to know LV-112D and LV-112E are parallel paths and the CCPs will have adequate suction.

DISTRACTOR ANALYSIS:

A. Incorrect. With a failure of LV-112D to open, LV-112B will not close. Stopping the CCPs is not required as LV-112E will open to provide suction to the CCPs.

B. Incorrect. With a failure of LV-112D to open, LV-112B will not close. Stopping the CCPs is not required as LV-112E will open to provide suction to the CCPs.

C. Correct. LV-112C will shut since it is not interlocked with LV-112B. Stopping the CCPs is not required as LV-112E will open to provide suction to the CCPs.

D. Incorrect. LV-112C will shut since it is not interlocked with LV-112B. Stopping the CCPs is not required as LV-112E will open to provide suction to the CCPs.

REFERENCES:

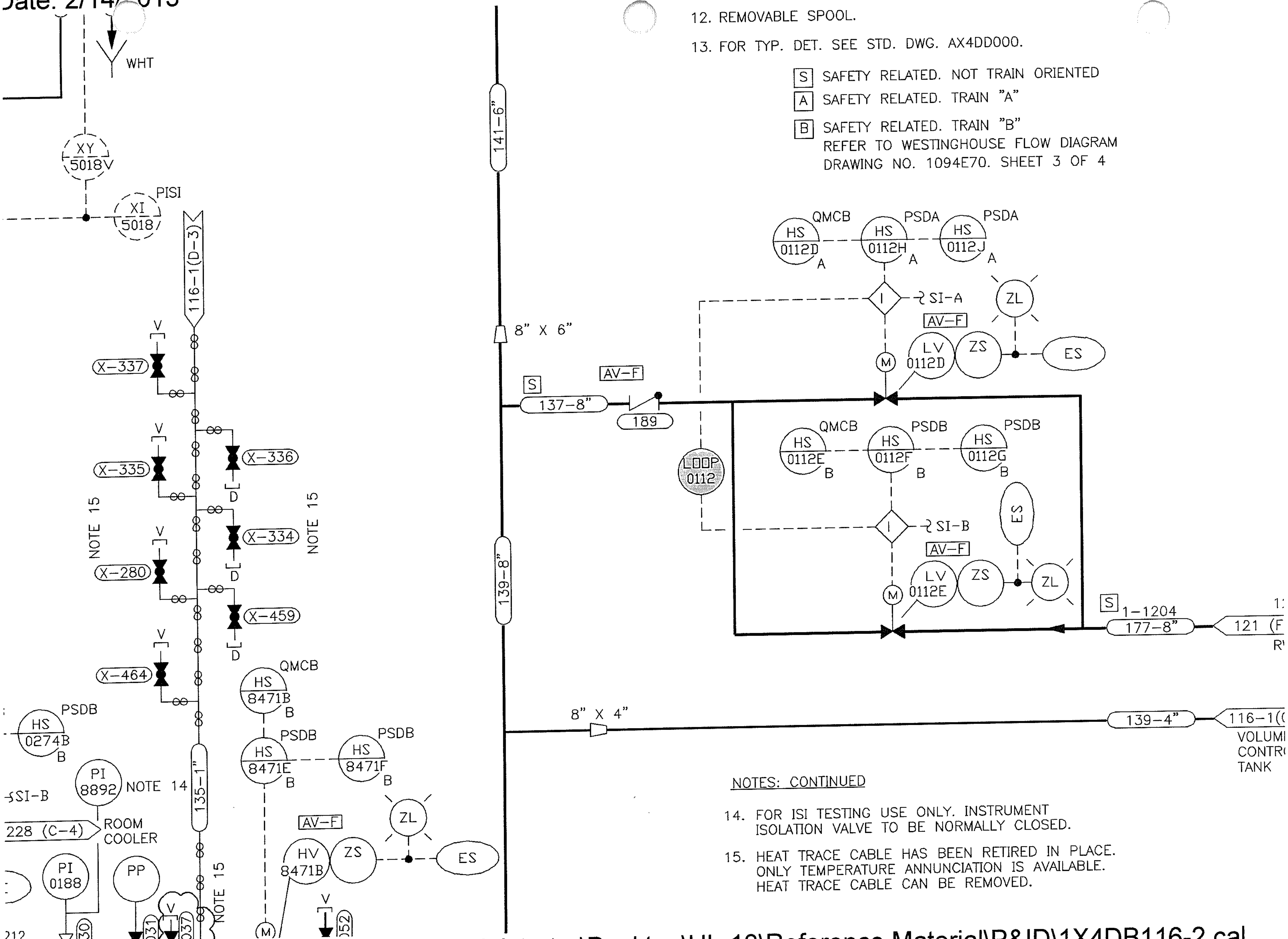
1X4-DB-116-1, CVCS Prints
1X4-DB-116-2, CVCS Prints

VEGP learning objectives:

LO-PP-09200-01 State the purpose and describe the control signals, setpoints, and any interlocks for the following:

a. VCT outlet valves, LV-112B, LV-112C
b. RWST suction supply valves to charging pumps, HV-112D, HV-112E

Date: 2/14/2013 Time : 03:42:18 PM



12. REMOVABLE SPOOL.

13. FOR TYP. DET. SEE STD. DWG. AX4DD000.

- S SAFETY RELATED. NOT TRAIN ORIENTED
- A SAFETY RELATED. TRAIN "A"
- B SAFETY RELATED. TRAIN "B"

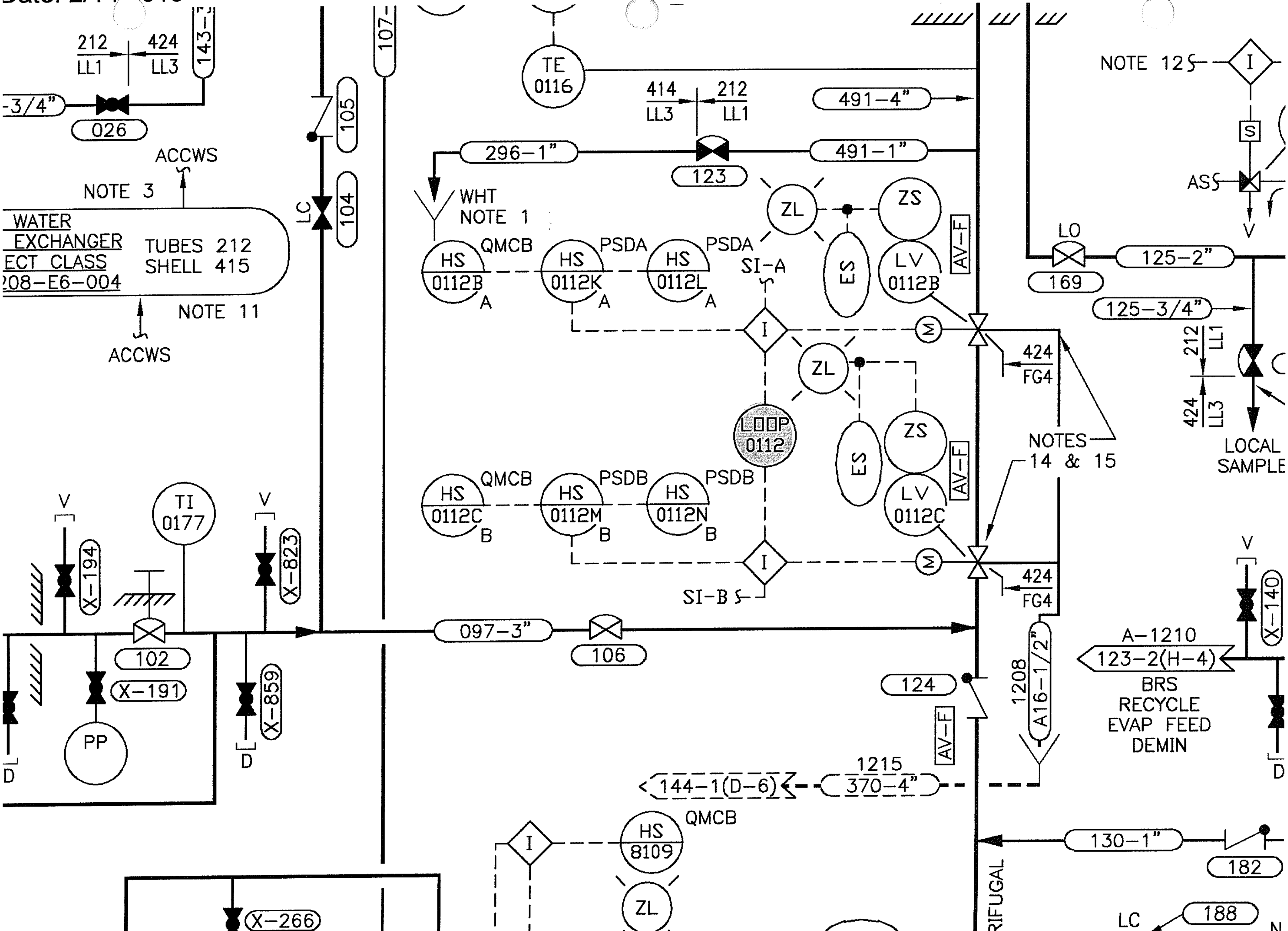
REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70. SHEET 3 OF 4

NOTES: CONTINUED

14. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.

15. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.

Title: C:\Documents and Settings\Administrator\Desktop\HL-18\Reference Material\P&ID\1X4DB116-2.cal



212 LL1
424 LL3
143-
-3/4"
026
ACCWS
NOTE 3
WATER
EXCHANGER
ECT CLASS
208-E6-004
TUBES 212
SHELL 415
NOTE 11
ACCWS
105
LC
104
107-
TE 0116
414 LL3
212 LL1
491-4"
296-1"
123
491-1"
WHT
NOTE 1
HS 0112B A
QMCB
HS 0112K A
PSDA
HS 0112L A
PSDA
SI-A
ZL
ZS
ES
LV 0112B
AV-F
LOOP 0112
ZL
ES
ZS
LV 0112C
AV-F
HS 0112C B
QMCB
HS 0112M B
PSDB
HS 0112N B
PSDB
SI-B
424 FG4
424 FG4
NOTES 14 & 15
NOTE 12
S
AS
V
LO
169
125-2"
125-3/4"
212 LL1
424 LL3
LOCAL SAMPLE
V
X-194
TI 0177
V
X-823
102
X-191
PP
D
X-859
D
097-3"
106
124
AV-F
1208
A16-1/2"
A-1210
123-2(H-4)
BRS
RECYCLE
EVAP FEED
DEMIN
V
X-140
D
1215
370-4"
144-1(D-6)
HS 8109
QMCB
ZL
130-1"
182
RIFUGAL
LC
188
N
X-266

13. 010K4.02 001/2/1/PZR PCS - HTRS/C/A - 3.0/3.4/BANK - FARLEY2004/HL-18 NRC/RO/SRO/KAJ

Initial conditions:

- Unit 2 is at 25% power.
- All pressurizer backup heaters are in AUTO.
- Variable pressurizer heaters are in the ON position.
- Due to a pressurizer level control malfunction, pressurizer level dropped to 15% and pressurizer pressure dropped to 2100 psig.

Current conditions:

- Pressurizer level has returned to 20%.
- Pressurizer pressure has returned to 2215 psig.

Based on the current conditions and assuming no operator actions, which ONE of the following correctly describes the current status of the pressurizer heaters?

A.✓ Variable and backup heaters are off.

B. Variable and backup heaters automatically energized.

C. Backup heaters automatically energized; variable heaters are off.

D. Variable heaters automatically energized; backup heaters are off.

010K4.02 Pressurizer Pressure Control System (PZR PCS)

Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Prevention of uncovering the heaters.

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario where PRZR level drops below 17% and then rises back to 20%. PRZR pressure is also low which will require the heaters to be on. The candidate has to determine if the heaters are on or off given the plant conditions. Once PRZR level drops below 17%, the PRZR heaters will remain off until reset by the operator, this is to prevent uncovering and damaging the PRZR heaters.

DISTRACTOR ANALYSIS:

A. Correct. All PRZR heaters will remain off until reset by the operator.

B. Incorrect. All PRZR heaters will remain off until reset by the operator. It is plausible the candidate may think once level is above 17% and PRZR pressure is low, the heaters will automatically turn on.

C. Incorrect. All PRZR heaters will remain off until reset by the operator. With PRZR pressure at 2215 psig, it is plausible the candidate may think the backup heaters will be energized and the variables off if they don't recall the heaters need to be reset.

D. Incorrect. All PRZR heaters will remain off until reset by the operator. With PRZR pressure at 2215 psig, it is plausible the candidate may think the variable heaters will be energized and the backups off if they invert the function of the heaters and don't recall the heaters need to be reset.

REFERENCES:

1X6AA02-00230, 236, and 496 Pressurizer Heaters.
18000-C, Pressurizer Spray, Safety, or Relief Valve Malfunction, Figure 1.

VEGP learning objectives:

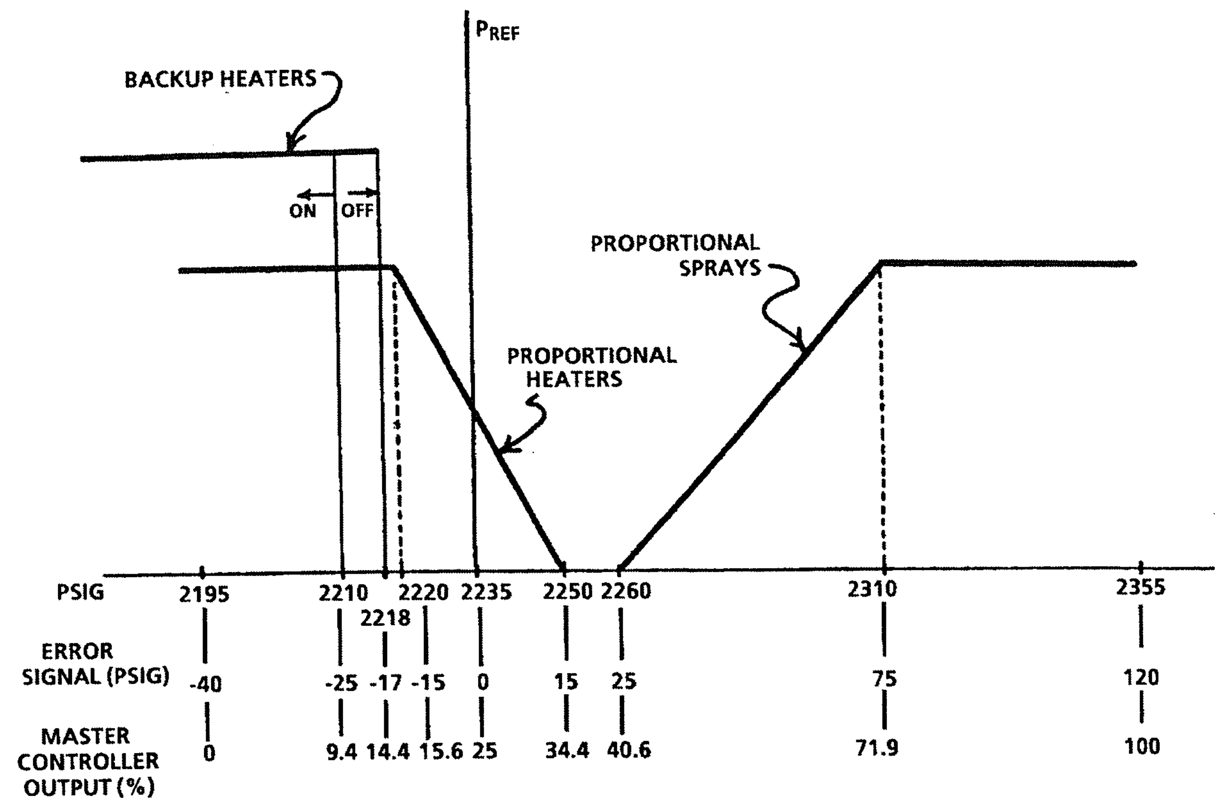
LO-PP-16302-02 Describe how the response of pressurizer level control to the following failures:

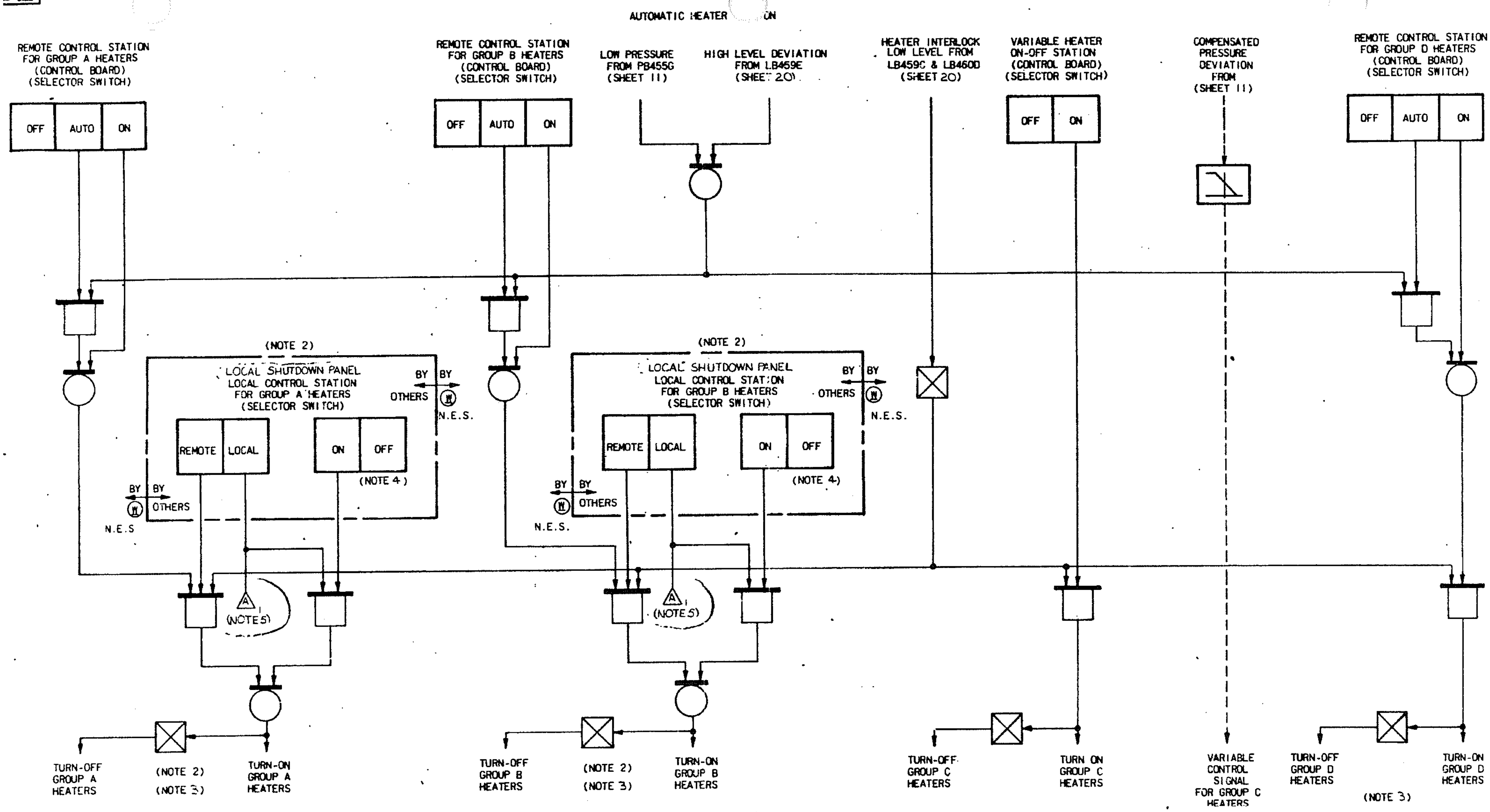
a. controlling (primary & secondary) channel fails low

LO-PP-16302-04 Describe the Hi and Low Pressurizer level protection features including the set points, coincidence, and reason for each.

FIGURE 1 Sheet 1 of 1

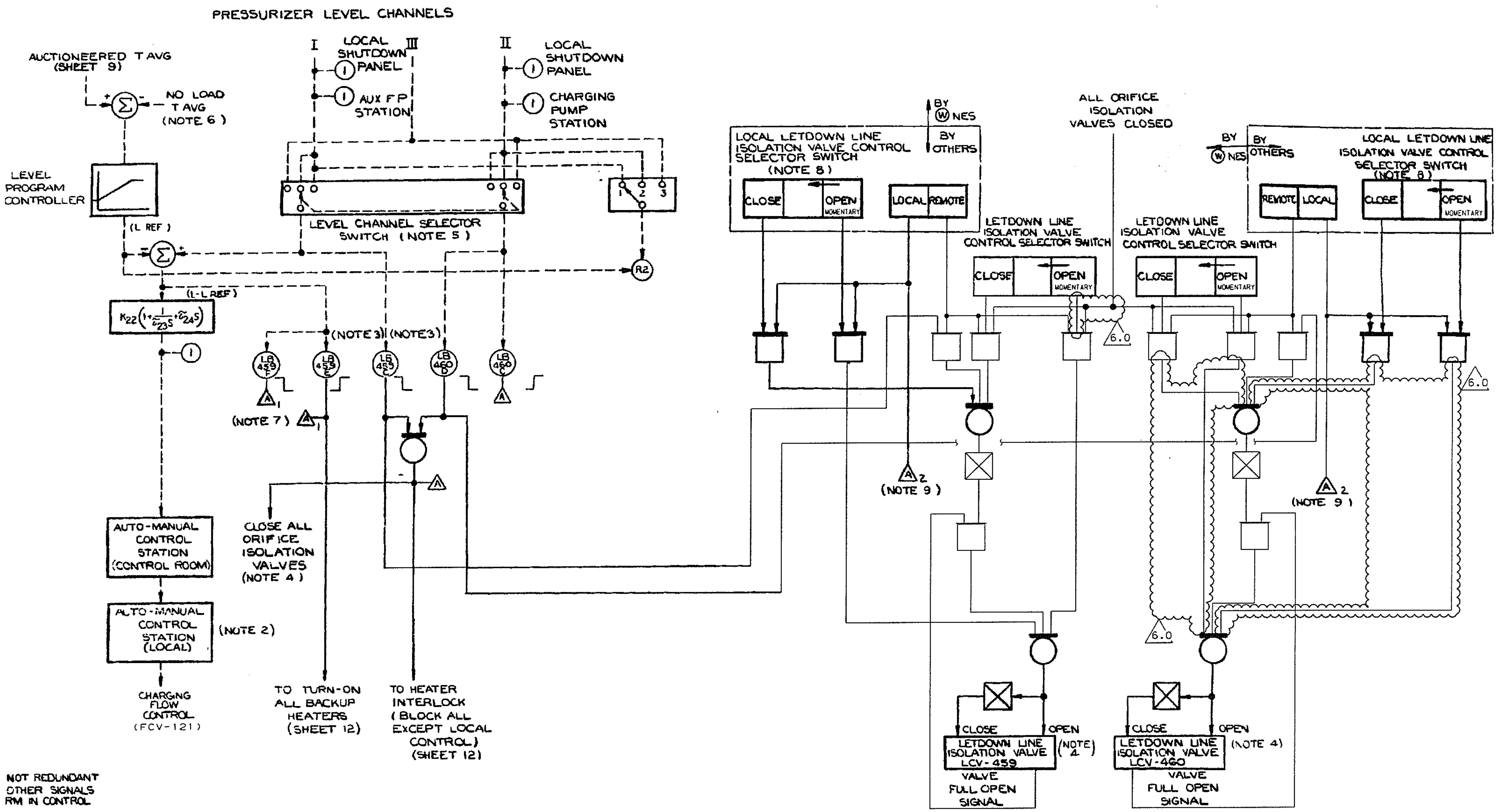
PRESSURIZER PRESSURE CONTROLLER BAND





CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
JP A AND GROUP B HEATERS MUST BE ON SEPARATE VITAL POWER SUPPLIES WITH THE LOCAL CONTROL
ARATED SO THAT ANY SINGLE FAILURE DOES NOT DEFEAT BOTH.
CK UP HEATER STATUS INDICATION IN CONTROL ROOM.
ECAUTIONS SHOULD BE TAKEN TO AVOID MANUAL HEATER OPERATION,
ICH WOULD CAUSE HEATER DAMAGE, IF THE WATER LEVEL UNCOVERS
E HEATERS.
NUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED TO BE VISIBLE TO THE
ERATOR AT THE MAIN CONTROL BOARD.

GEORGIA POWER COMP
PLANT: ALVIN W. VOGTLE NUCLEA
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-5705
AUTHORITY:

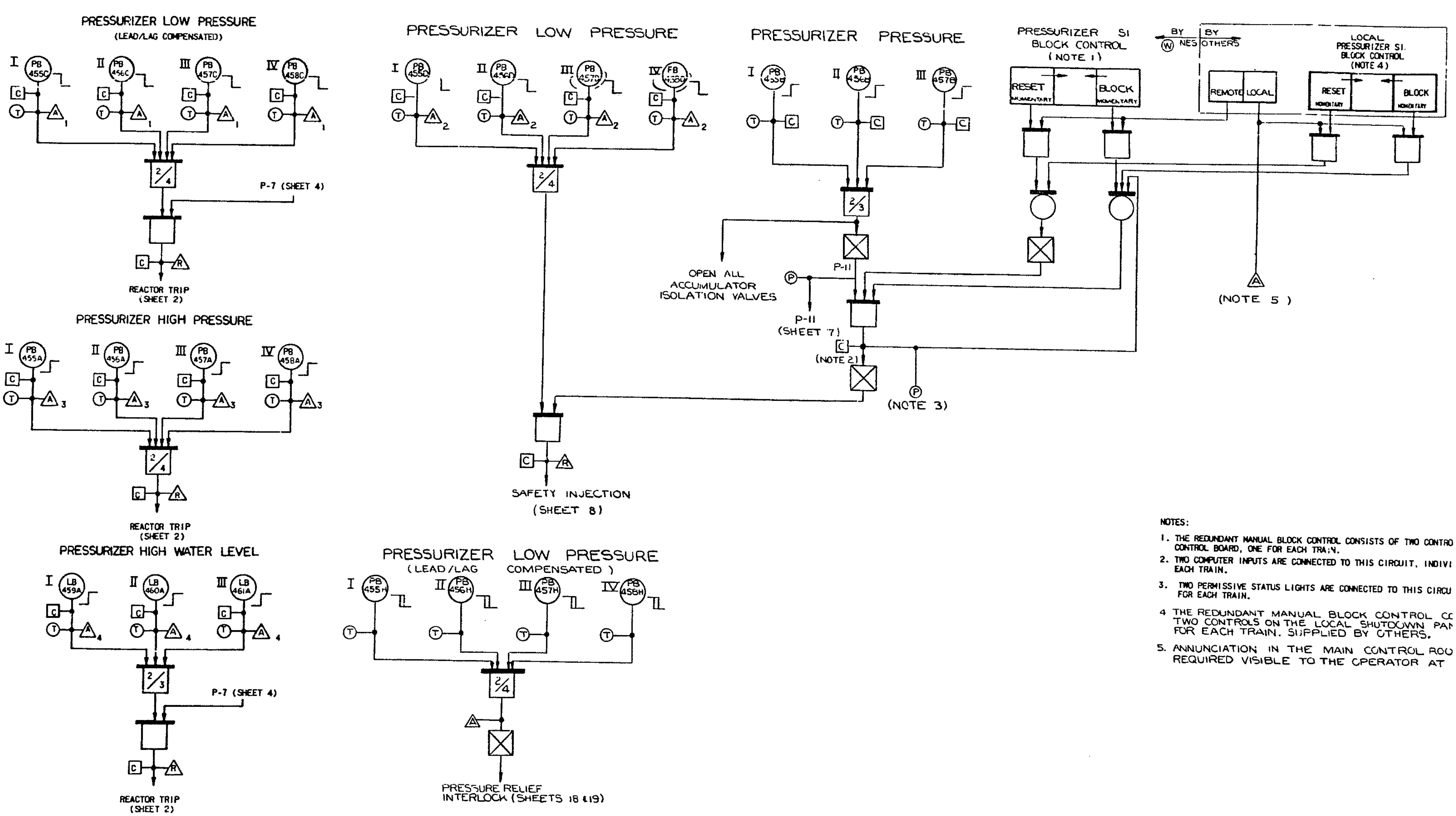


NOT REDUNDANT
OTHER SIGNALS
RM IN CONTROL

LB-459E, & LB-460D

CONTROL ROOM.
SELECTED.
N CONTROLLER.
T CAPABILITY.
VALVES CONSISTS OF
& THE LOCAL SHUTDOWN
IE. SUPPLIED BY OTHERS.
CONTROL ROOM IS REQUIRED
T THE MCB

DRWN B STEMMLER	8 98	Westinghouse Electric Corporation NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U S A
DES ENG		TITLE GEORGIA POWER CO ALVIN W VOGTLE UNITS 1 & 2 FUNCTIONAL DIAGRAMS PRESSURIZER LEVEL CONTR
		7243D07



NOTES:

1. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTRO CONTROL BOARD, ONE FOR EACH TRAIN.
2. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT, INDIVI EACH TRAIN.
3. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCU FOR EACH TRAIN.
4. THE REDUNDANT MANUAL BLOCK CONTROL CC TWO CONTROLS ON THE LOCAL SHUTDOWN PAN FOR EACH TRAIN. SUPPLIED BY OTHERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROO REQUIRED VISIBLE TO THE OPERATOR AT

P HEARN 7-17-74
GAE/GBE-300
ECN-30838
REVISIONS CIRCLED
GAE/GBE-300
ECN-31242
REV'S CIRCLED
MEARS 7-13-78
GAE/GBE-300
ECN 31969
REV. CIRCLED
B. STEMMLER 8/11/81
GAE/GBE-300
ECN-33209
CHANGES CIRCLED

GEORGIA POWER COMPANY
PLANT: ALVIN W. VOGTLE NUCLEAR PLANT
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-9146
AUTHORITY: J. L. VOTA
ENGR. LTR. NO. EP/SA-47348

DFTR. P. HEARN
Westinghouse Electric
NUCLEAR ENERGY SYSTEMS, PITTSBU
TITLE: GEORGIA POWER CO.
ALVIN W VOGTLE UNI
FUNCTIONAL DIAGRAM
PRESSURIZER TRIP S
SCALE
DIMENSIONS IN INCHES
DO NOT SCALE
72
SHEE

10 9 8 7 6 5 4 3 2

14. 011EK2.02 001/1/1/LARGE LOCA - PUMPS/C/A - 2.6/2.7/BANK-VGT 2009 NRC/HL-18 NRC/RO/SRO/TNT

Given the following:

- A DBA LOCA is in progress on Unit 2.
- ECCS has been aligned for "Cold Leg Recirculation."
- 'B' RHR pump tripped 5 minutes after completion of the recirculation alignment.
- All other components are functioning properly.

Which one of the following completes the following statement.

For the given conditions, __________ have an adequate suction source and may continue to run in the recirculation alignment.

A. ONLY the SIPs

B. ONLY the CCPs

C. ONLY 'A' CCP and 'A' SIP

D.✓ both CCPs and both SIPs

011EK2.02 Large Break LOCA

Knowledge of the interrelations between the large break LOCA and the following: (CFR: 41.7 / 45.7)

Pumps

K/A MATCH ANALYSIS:

This question gives a plausible scenario during cold leg recirculation where an RHR pump trips after cold leg recirculation has been established. The candidate must choose how this affects suction to his SIPs and CCPs and determine they should be left running. In addition to this, he may also consider possibly runout of the remaining RHR pump.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible since candidate may think RHR pump 2B supplies the SIP suction header. Candidate could also consider RHR pump runout too for a reason to shut off some ECCS pumps. During swaps from Cold Leg to Hot Leg recirculation, some alignments are performed to keep from running out a single RHR pump.

B. Incorrect. Plausible since candidate may think RHR pump 2B supplies the CCP suction header. Candidate could also consider RHR pump runout too for a reason to shut off some ECCS pumps. During swaps from Cold Leg to Hot Leg recirculation, some alignments are performed to keep from running out a single RHR pump.

C. Incorrect. Plausible since candidate may think RHR pump 2B supplies CCP "B" and SIP "B" suctions. Candidate could also consider RHR pump runout too for a reason to shut off some ECCS pumps. During swaps from Cold Leg to Hot Leg recirculation, some alignments are performed to keep from running out a single RHR pump.

D. Correct.

REFERENCES

19013-C, "Cold Leg Recirculation"
1X4DB-121/122 P&IDs

VEGP learning objectives:

LO-LP-37113-03, Describe the basic lineup of ECCS during Cold Leg Recirculation.

ATTACHMENT A

COLD LEG RECIRCULATION VALVE ALIGNMENT

2. Align RHR Pump A flow path:	2.
__a. Check RHR Pump A – RUNNING.	__a. Start RHR Pump A. __ IF RHR Pump A can NOT be started, THEN go to Step 3.
__b. Check CNMT SUMP TO RHR PMP-A SUCTION HV-8811A - OPEN.	b. IF HV-8811A is NOT open, THEN perform the following: __1) Stop RHR Pump A. __2) Close RWST TO RHR PMP-A SUCTION HV-8812A. __3) Open HV-8811A. __4) Start RHR Pump A. __5) Go to Step 2.d.
__c. Close RWST TO RHR PMP-A SUCTION HV-8812A.	__c. IF HV8812A will not close, THEN stop RHR Pump A. __ Go to Step 3.

Step 2 continued on next page

ATTACHMENT A

Sheet 3 of 7

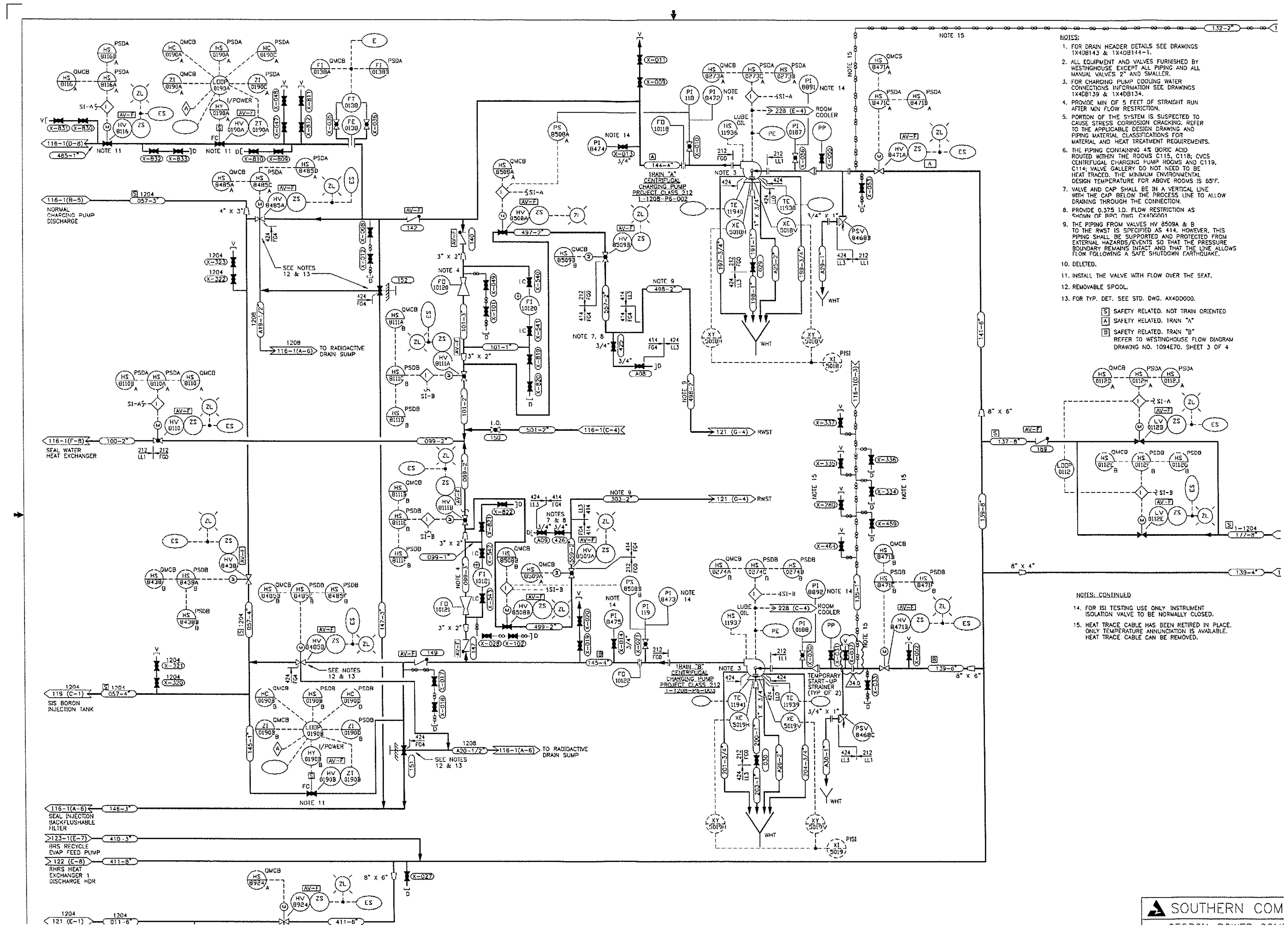
COLD LEG RECIRCULATION VALVE ALIGNMENT

__d. Check RHR PMP-A TO COLD LEG 1&2 ISO VLV HV-8809A – OPEN.	__d. Open HV-8809A. __ IF HV-8809A can NOT be opened from QMCB or locally, THEN go to Step 3. 1-HV-8809A (AB-A09) 2-HV-8809A (AB-A103)
__e. Check RHR Heat Exchanger A flow indicator FI-618A – GREATER THAN 500 GPM.	__e. Recheck valve and pump status. __ Go to Step 3.
3. Align RHR Pump B flow path:	3.
__a. Check RHR Pump B – RUNNING.	__a. Start RHR Pump B. __ IF no RHR Pumps can be started, THEN go to 19111-C, ECA-1.1 LOSS OF EMERGENCY COOLANT RECIRCULATION.

Step 3 continued on next page

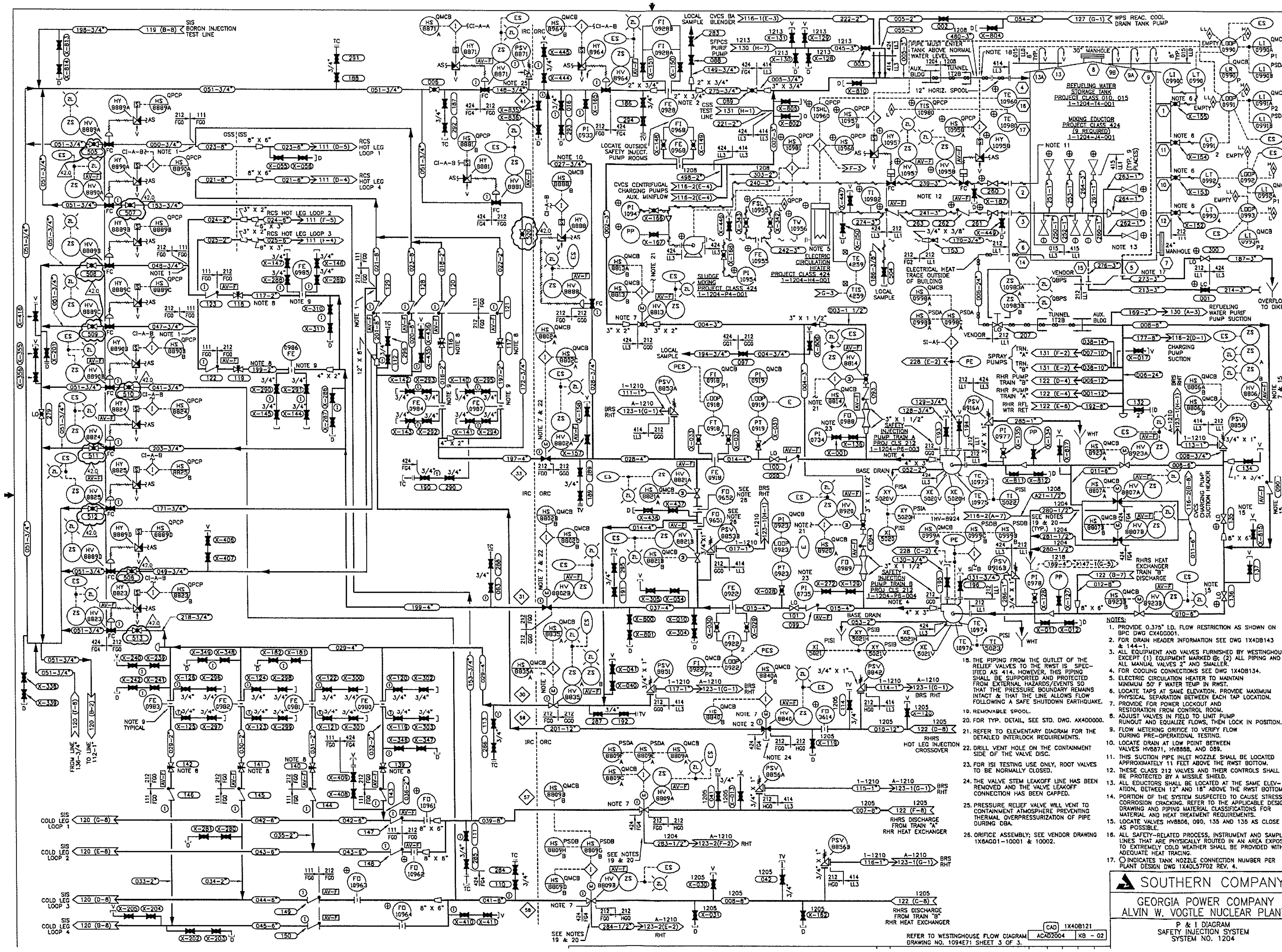
Date: 2/14/2013

Time : 03:57:06 PM



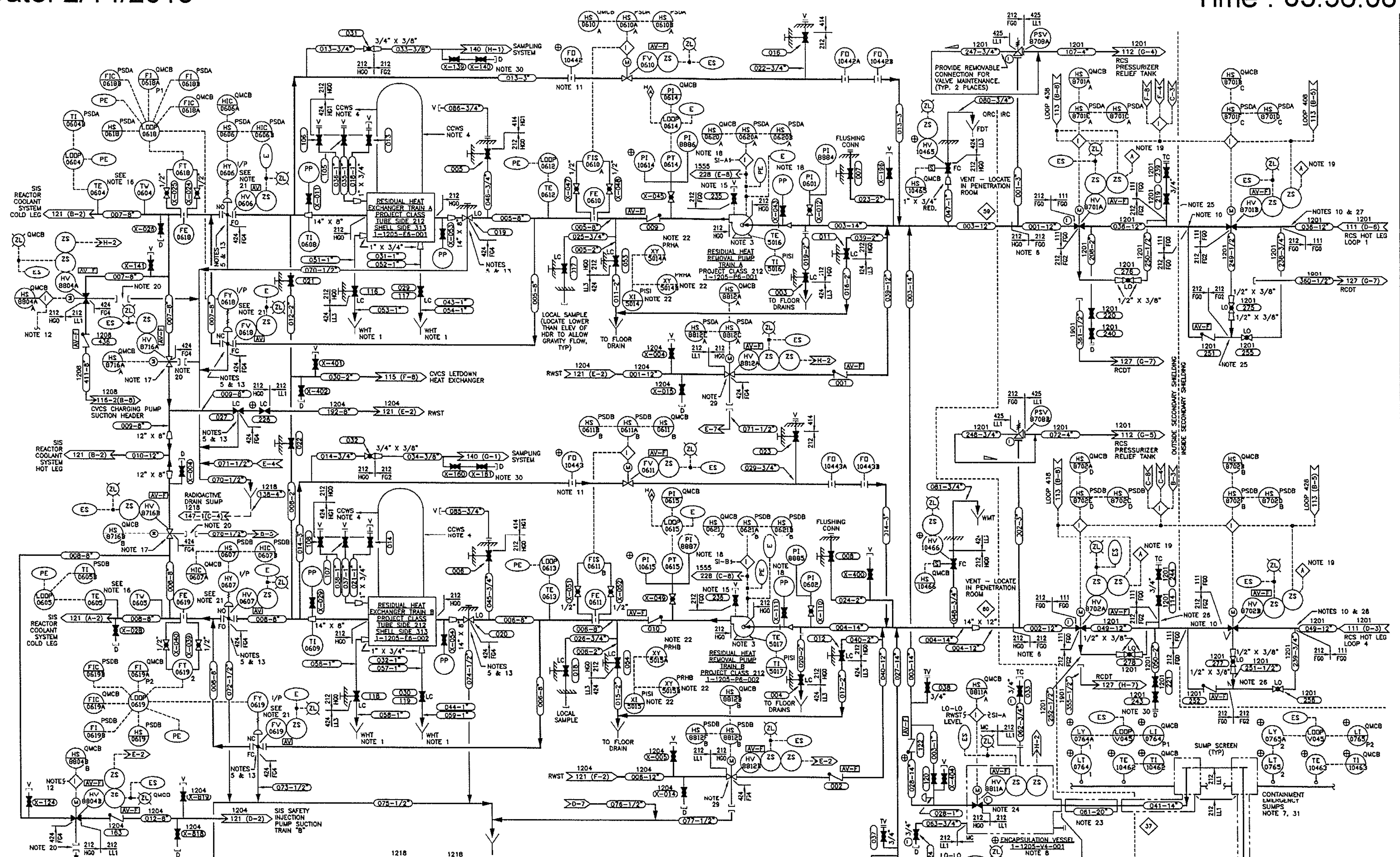
Title: C:\Documents and Settings\Administrator\Desktop\HL-18\Reference Material\P&ID\1X4DB116-2.cal

Date: 2/14/2013 Time : 03:54:26 PM



Title: C:\Documents and Settings\Administrator\Desktop\HL-18\Reference Material\P&ID\1X4DB121.cal

Date: 2/14/2013 Time : 03:56:08 PM



NOTES:

1. FOR DRAIN HEADER INFORMATION SEE DWGS 1X4DB143, 144-1, 1X4DB145-5 AND 1X4DB183.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT: (1) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER, AND (2) EQUIPMENT MARKED ⊕.
3. FOR PUMP AND MOTOR COOLING WATER CONNECTIONS, SEE DWGS 1X4DB134 & 1X4DB137.
4. FOR HEAT EXCHANGER SHELL SIDE CONNECTIONS SEE DWG 1X4DB137.
5. PROVIDE 12" REMOVABLE SPOOL.
6. MAKE PIPING MATERIAL CLASS CHANGE AT FIRST WELD OF PENETRATION SLEEVE INSIDE THE CONTAINMENT.
7. INSTALL 150 #R.F. FLANGE ON INLET OF EACH 14"# PIPE FOR PERIODIC TESTING. PROVIDE ONE MATING BLIND FLANGE TO BE DRILLED AND TAPPED FOR TEST CONNECTION. MATING FLANGE MUST BE REMOVED FOR NORMAL OPERATION.
8. VESSEL IS CONSTRUCTED TO SEISMIC CATEGORY 1 AND ASME III CLASS MC.
9. PORTIONS OF THIS SYSTEM ARE SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON DWG CX4DC001. 51.0
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. FOR TYPICAL DETAIL SEE STD. DWG AX4DD000.
14. DELETED.
15. HIGH POINT VENT VALVE (BECHTEL-FURNISHED) FOR SEAL PIPING - SEE 1X6AF02-25.
16. TE IS A SURFACE TYPE CLAMP ON RTD, TW IS INSTALLED BUT NOT USED.
17. VENT HOLE IS DRILLED ON RESPECTIVE RHR PUMP SIDE OF DISC.
18. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.
19. ALARM ON OPEN HOT LEG VALVE AND HIGH RCS PRESSURE.
20. DELETED.
21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.
23. 4" VIEW PORT TO BE USED TO VERIFY MOTOR OPERATED VALVE STROKE TEST WITHOUT REMOVING THE VESSEL HEAD.
24. VENT HOLE IS DRILLED ON CONTAINMENT SIDE OF DISC.
25. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-249-01.
26. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-251-01.
27. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-036-01.
28. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-049-02.
29. A SECTION OF THE LEAKOFF LINE HAS BEEN REMOVED AND BOTH ENDS CAPPED.
30. THESE ARE KEROTEST VALVES AND MAY NOT ALLOW REVERSE FLOW.
31. SUMP SCREEN TAG NO'S. 1-1205-02-001 (TRAIN "A") AND 1-1205-02-002 (TRAIN "B")

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. NO. 1094E72

CAD 1X4DB122
ACAD2007 LPM - 07

SOUTHERN COMPANY
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
P & I DIAGRAM
RESIDUAL HEAT REMOVAL
SYSTEM NO. 1205

SCALE: NONE | DRAWING NO. 1X4DB122 | VER. 51.0

Title: C:\Documents and Settings\Administrator\Desktop\HL-18\Reference Material\P&ID\1X4DB122.cal

15. 011K2.02 001/2/2/PZR LVL - HEATERS/C/A - 3.1/3.2/MOD-HL15 AUDIT/HL-18 NRC/RO/SRO/AML

Given the following:

- The Reactor is tripped after a loss of both RATs occurs.
- DG1A fails to start.
- All other equipment sequences on as expected.

Which ONE of the following correctly identifies which Pressurizer heater bank(s) is/are available for RCS pressure control?

A. All Back-up Heater Banks.

B✓ Back-up Heater Bank 'B' only.

C. Proportional Heater Banks only.

D. Back-up Heater Banks 'B' and 'D' only.

011K2.02 Pressurizer Level Control System (PZR LCS)

Knowledge of bus power supplies to the following: (CFR 41.7)

PRZR Heaters

K/A MATCH ANALYSIS:

Question gives a plausible scenario where a dual train LOSP has occurred. Candidate must identify the available PRZR Heaters for the plant condition. The candidate has to understand which power supplies remain and the effect on the PRZR.

DISTRACTOR ANALYSIS:

A. Incorrect. Backup heater bank B will be energized from the stub bus. Backup heater A will be de-energized with the A stub bus de-energized. C backup heater will also be de-energized since the non-1E bus power will be lost on fast and residual bus transfer.

B. Correct. With Train B DG powering its stub bus, BU Heater Bank B will be energized after the LOSP sequence.

C. Incorrect. The proportional heaters will also be de-energized since the non-1E bus power will be lost on fast and residual bus transfer.

D. Incorrect. A bus is de-energized, however it is plausible the candidate may think that B and D buses are powered from DG1B since many train B and D components are powered from Train B power.

REFERENCES:

HL-15 Audit Exam question #16
1X3D-AA-F13A 480V Pressurizer Heater Panels
Tech Spec 3.4.9, Pressurizer, and Bases

VEGP learning objectives:

LO-LP-39208-01 For any given item in section 3.4 of Tech Specs, be able to:
a. State the LCO
b. State any one hour or less required actions.

The surge line connects the hot leg of RCS Loop 4 to the bottom head of the pressurizer. The surge line has a diameter of 16 inches in which it is designed for a maximum flow of 20,000 gpm. This, in combination with maximum flow from the three pressurizer safety valves, will limit the over pressurization of the RCS to 110% of the design pressure. The pressurizer is the hottest point in the RCS. When water surges out of the pressurizer, it is approximately 650°F. When RCS hot leg water surges into the pressurizer, it is approximately 35°F cooler. The surge nozzle is protected by a thermal sleeve, which minimizes thermal stresses due to the rapid temperature changes that accompany water surges into or out of the Pressurizer. A retaining screen above the nozzle prevents any foreign matter in the Pressurizer from entering the RCS piping.

16-29 Electric Immersion Heaters

The pressurizer electric heaters are used to increase RCS pressure. Each heater is a resistive heating element with a rated capacity of approximately 23 kW at 480 volts. With a total heater capacity of 1800 kW is divided into four groups with separate controls for the proportional group (Group C) and the backup groups (A, B, and D). Groups A and B can be controlled from the remote shutdown panel A and B, respectively. Group A and B heaters are tech. spec. related and are powered from 1NB01 and 1NB10. These two buses are energized by 1AA02 and 1BA03, which are safety related buses that can be powered from the emergency diesel generators on a LOSP.

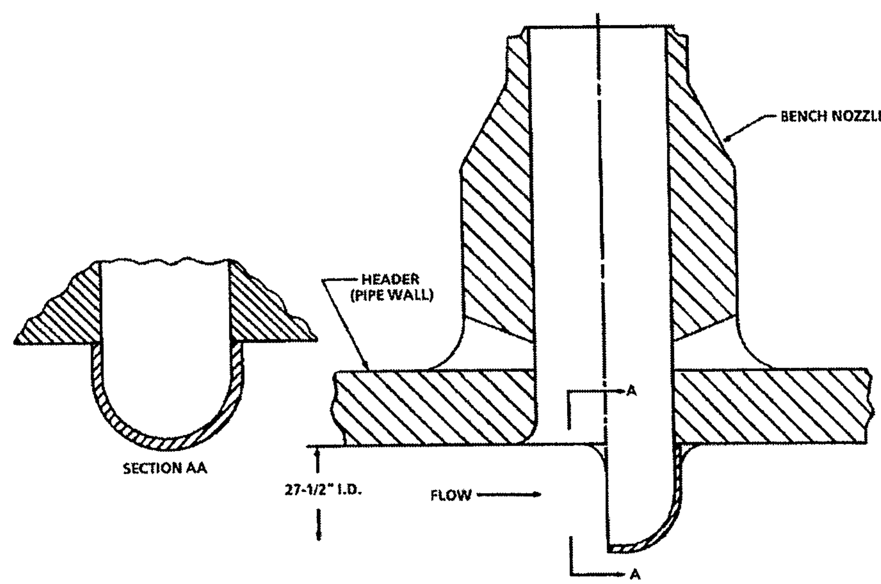
The proportional group heaters are rated at approximately 400 kW and have a variable output depending on the pressure control demand. The backup heaters are rated at approximately 1400 kW, and receive either an ON or OFF control signal. Their primary purpose is to heat the pressurizer to normal operating temperatures from a cold, solid, non-pressurized condition and to react to pressure transients.

The heaters are designed to raise the temperature of the pressurizer at a maximum rate of 50-55°F/hr when the pressurizer is water solid at startup, and 70°F/hr with the plant in a hot standby condition.

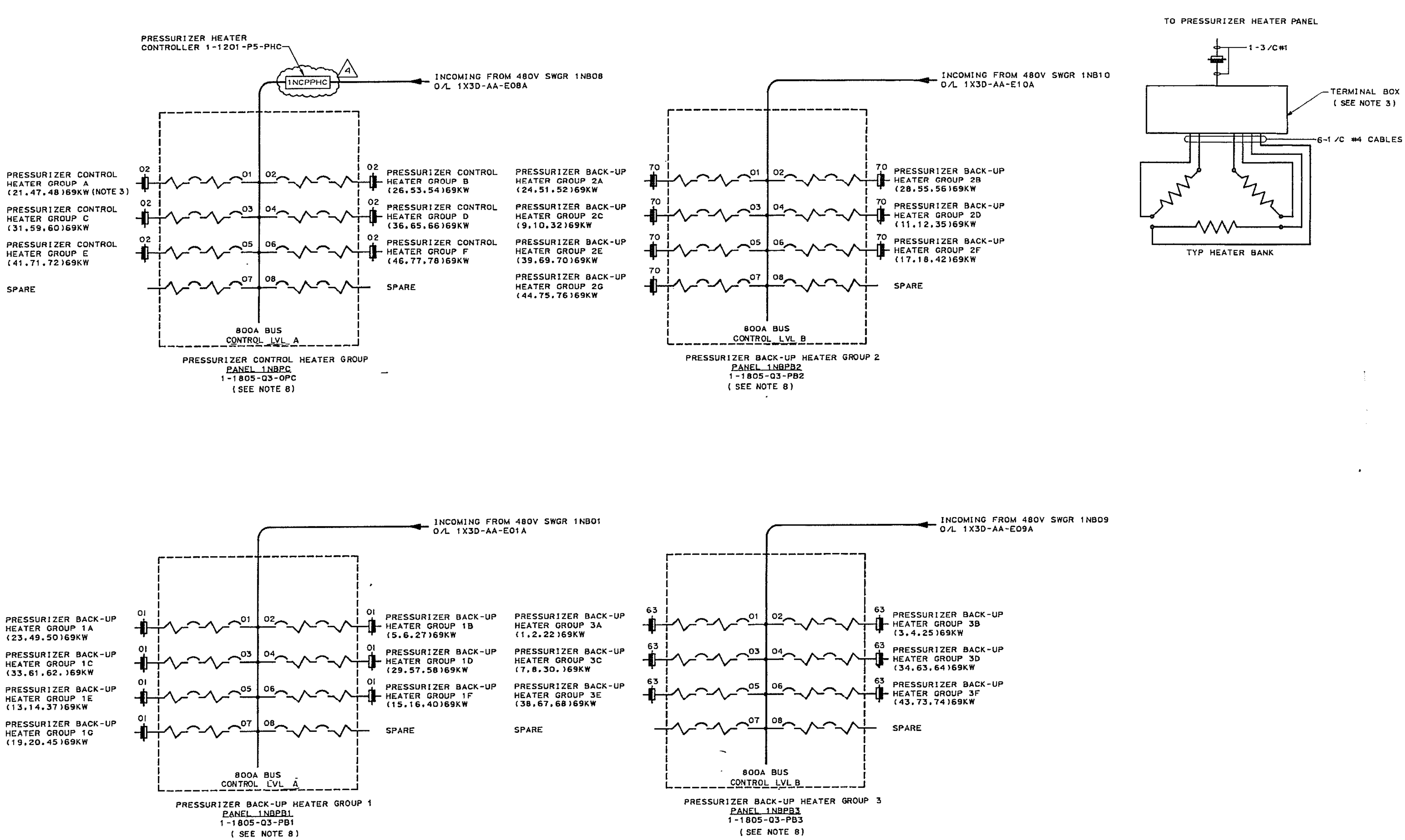
Pressurizer heaters are controlled automatically by the Pressurizer Pressure Control System to maintain RCS pressure at 2235 psig. This ensures an adequate degree of subcooling of the RCS and prevents any boiling which would adversely affect the heat transfer in the core. The pressurizer heaters can also be manually controlled.

16-30 Spray System

Pressurizer spray is used to reduce pressure in the pressurizer and RCS. When spray flow is established, relatively cool water from the RCS is sprayed into the steam space of the pressurizer. The spray condenses some of the steam. When the steam is condensed, the liquid occupies approximately one-sixth the volume that it did when it was steam. The volumetric reduction reduces the vapor space pressure and the force exerted on the liquid in the pressurizer. This reduces pressurizer and RCS pressure.



The pressurizer spray comes from the cold legs of RCS Loops 1 and 4. The driving force for the spray water is the reactor coolant pumps in loops 1 and 4. It is aided by use of a scoop that protrudes into the RCS piping and directs flow through the spray piping. The two 4-inch spray lines from the cold legs tie together after passing through the spray control valves, and supply water to a single spray nozzle in the top of the pressurizer vessel. The common spray line connects via a thermal sleeve to the pressurizer


PRESSURIZER HEATER CONTROLLER 1-1201-P5-PHC
1NCPPHC
INCOMING FROM 480V SWGR 1NB08 O/L 1X3D-AA-E08A
PRESSURIZER CONTROL HEATER GROUP A (21,47,48)69KW (NOTE 3)
PRESSURIZER CONTROL HEATER GROUP C (31,59,60)69KW
PRESSURIZER CONTROL HEATER GROUP E (41,71,72)69KW
SPARE
PRESSURIZER CONTROL HEATER GROUP B (26,53,54)69KW
PRESSURIZER CONTROL HEATER GROUP D (36,65,66)69KW
PRESSURIZER CONTROL HEATER GROUP F (46,77,78)69KW
SPARE
800A BUS CONTROL LVL A
PRESSURIZER CONTROL HEATER GROUP
PANEL 1NBPC
1-1805-Q3-OPC
(SEE NOTE 8)
INCOMING FROM 480V SWGR 1NB10 O/L 1X3D-AA-E10A
PRESSURIZER BACK-UP HEATER GROUP 2A (24,51,52)69KW
PRESSURIZER BACK-UP HEATER GROUP 2C (9,10,32)69KW
PRESSURIZER BACK-UP HEATER GROUP 2E (39,69,70)69KW
PRESSURIZER BACK-UP HEATER GROUP 2G (44,75,76)69KW
PRESSURIZER BACK-UP HEATER GROUP 2B (28,55,56)69KW
PRESSURIZER BACK-UP HEATER GROUP 2D (11,12,35)69KW
PRESSURIZER BACK-UP HEATER GROUP 2F (17,18,42)69KW
SPARE
800A BUS CONTROL LVL B
PRESSURIZER BACK-UP HEATER GROUP 2
PANEL 1NBPB2
1-1805-Q3-PB2
(SEE NOTE 8)
TO PRESSURIZER HEATER PANEL
1-3/C#1
TERMINAL BOX (SEE NOTE 3)
6-1/C #4 CABLES
TYP HEATER BANK
INCOMING FROM 480V SWGR 1NB01 O/L 1X3D-AA-E01A
PRESSURIZER BACK-UP HEATER GROUP 1A (23,49,50)69KW
PRESSURIZER BACK-UP HEATER GROUP 1C (33,61,62,)69KW
PRESSURIZER BACK-UP HEATER GROUP 1E (13,14,37)69KW
PRESSURIZER BACK-UP HEATER GROUP 1G (19,20,45)69KW
PRESSURIZER BACK-UP HEATER GROUP 1B (5,6,27)69KW
PRESSURIZER BACK-UP HEATER GROUP 1D (29,57,58)69KW
PRESSURIZER BACK-UP HEATER GROUP 1F (15,16,40)69KW
SPARE
800A BUS CONTROL LVL A
PRESSURIZER BACK-UP HEATER GROUP 1
PANEL 1NBPB1
1-1805-Q3-PB1
(SEE NOTE 8)
INCOMING FROM 480V SWGR 1NB09 O/L 1X3D-AA-E09A
PRESSURIZER BACK-UP HEATER GROUP 3A (1,2,22)69KW
PRESSURIZER BACK-UP HEATER GROUP 3C (7,8,30,)69KW
PRESSURIZER BACK-UP HEATER GROUP 3E (38,67,68)69KW
SPARE
PRESSURIZER BACK-UP HEATER GROUP 3B (3,4,25)69KW
PRESSURIZER BACK-UP HEATER GROUP 3D (34,63,64)69KW
PRESSURIZER BACK-UP HEATER GROUP 3F (43,73,74)69KW
SPARE
800A BUS CONTROL LVL B
PRESSURIZER BACK-UP HEATER GROUP 3
PANEL 1NBPB3
1-1805-Q3-PB3
(SEE NOTE 8)

BASES

BACKGROUND (continued)	Two groups of pressurizer heaters can be administratively loaded onto the non-Class 1E emergency buses. The Class 1E 4160-V breakers supplying the non-Class 1E buses are automatically opened upon a safety injection signal, but they can be closed under administrative procedure.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.
LCO	The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1656 cubic feet, which is equivalent to 92% (LI-0459A, LI-0460A, LI-0461A), ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from an emergency power supply. This means that the two required groups of pressurizer heaters must be capable of being powered from a Class 1E 4160-V power supply. This is accomplished by administratively loading the two required

(continued)

1. 011K2.02 001/2/2/PRZR LVL-HEATERS/C/A -3.1/M-ANO 2005/RO/HL-15 AUDIT/TNT/DS

Given the following:

- The plant is at full power when a loss of offsite power causes a plant trip.
- Both EDGs start and tie onto their respective ESF buses. All equipment sequences on as expected.

Which **ONE** of the following is **CORRECT** for PRZR heater banks available for RCS pressure control?

A. All Backup Heater Banks only.

B. Proportional Heater Banks only.

C✓ Backup Heater Banks A and B only.

D. All Proportional and Backup Heater Banks.

16. 012A1.01 001/2/1/RPS - TRIP SETPT/MEM - 2.9/3.4/MOD-WC09/HL-18 NRC/RO/SRO/KAJ

The plant is at 100% power.

- An inadvertent dilution causes Tavg to rise.

Based on the given conditions, what will be the effect on the Over-Temperature (OT) and Over-Power (OP) Differential Temperature (DT) Reactor Protection setpoints?

	OTDT setpoint	OPDT setpoint
A.	no change	no change
B.	no change	decrease
C.✓	decrease	decrease
D.	decrease	no change

012A1.01 Reactor Protection System

Ability to predict and/or monitor changes to parameters (to prevent exceeding design limits) associated with operating the RPS controls including: (CFR: 41.5 / 45.5)

Trip setpoint adjustment.

K/A MATCH ANALYSIS:

The question straight forward asks the candidates how the OT Delta T and OP Delta T setpoints adjust as power changes. The setpoints adjust as power / Tavg adjust to prevent exceeding design setpoint. The candidate has to determine the direction of setpoint changes due to the temperature change.

DISTRACTOR ANALYSIS:

A. Incorrect. With a rise in Tavg, Tavg and rate of change Tave will cause both the OT and OP Delta T setpoints to lower. Lowering the trip setpoints will initiate a reactor trip if the setpoint is reached to prevent exceeding design limits. The OPDT setpoint never increases from its nominal value (no change) therefore it is plausible the candidate may confuse the OT and OP setpoints and think the setpoint does not change even with a temperature change.

B. Incorrect. With a rise in Tavg, Tavg and rate of change Tave will cause both the OT and OP Delta T setpoints to lower. Lowering the trip setpoints will initiate a reactor trip if the setpoint is reached to prevent exceeding design limits. The OPDT setpoint

never increases from its nominal value (no change) therefore it is plausible the candidate may confuse the OT and OP setpoints and think the setpoint does not change even with a temperature change.

C. Correct. With a rise in Tavg, Tavg and rate of change Tave will cause both the OT and OP Delta T setpoints to lower. Lowering the trip setpoints will initiate a reactor trip if the setpoint is reached to prevent exceeding design limits.

D. Incorrect. With a rise in Tavg, Tavg and rate of change Tave will cause both the OT and OP Delta T setpoints to lower. Lowering the trip setpoints will initiate a reactor trip if the setpoint is reached to prevent exceeding design limits. The OPDT setpoint never increases from its nominal value (no change) therefore it is plausible the candidate may think the setpoint does not change even with a temperature change.

REFERENCES:

Tech Spec 3.3.1, Reactor Trip System Instrumentation, Table 3.3.1-1

VEGP learning objectives:

LO-PP-16101-11: Describe how set points for OP and OT delta T vary with changes in Tavg, pressurizer pressure, and delta flux.

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T

The Allowable Value of each input to the Overtemperature Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1) 0.5% ΔT span for the ΔT channel
(2) 0.5% ΔT span for the T_{avg} channel
(3) 0.5% ΔT span for the pressurizer pressure channel
(4) 0.5% ΔT span for the $f_1(AFD)$ channel

$$\left[100\frac{\Delta T}{\Delta T_0}\frac{\{1+\tau_1 s\}}{\{1+\tau_2 s\}}\frac{1}{\{1+\tau_3 s\}}\right] \le \left[K_1 - K_2\frac{\{1+\tau_4 s\}}{\{1+\tau_5 s\}}\left[T\frac{1}{\{1+\tau_6 s\}} - T'\right] \overset{(p)}{} - K_3\{P' - P\} - f_1(AFD)\right]$$

Where:

ΔT	measured loop specific RCS differential temperature, degrees F
ΔT_0	indicated loop specific RCS differential at RTP, degrees F
$\frac{1+\tau_1 s}{1+\tau_2 s}$	lead-lag compensator on measured differential temperature
τ_1, τ_2	time constants utilized in lead-lag compensator for differential temperature: τ_1 = 0 seconds, τ_2 = 0 seconds
$\frac{1}{1+\tau_3 s}$	lag compensator on measured differential temperature
τ_3	time constant utilized in lag compensator for differential temperature, ≤ 6 seconds
K_1	fundamental setpoint, ≤ 114.9% RTP
K_2	modifier for temperature, = 2.24% RTP per degree F
$\frac{1+\tau_4 s}{1+\tau_5 s}$	lead-lag compensator on dynamic temperature compensation
τ_4, τ_5	time constants utilized in lead-lag compensator for temperature compensation: τ_4 ≥ 28 seconds, τ_5 ≤ 4 seconds
T	measured loop specific RCS average temperature, degrees F
$\frac{1}{1+\tau_6 s}$	lag compensator on measured average temperature
τ_6	time constant utilized in lag compensator for average temperature, ≤ 6 seconds
T′	indicated loop specific RCS average temperature at RTP, ≤ 588.4 degrees F
K_3	modifier for pressure, = 0.177% RTP per psig
P	measured RCS pressurizer pressure, psig
P′	reference pressure, ≥ 2235 psig
s	Laplace transform variable, inverse seconds

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T (continued)

$f_1(AFD)$ modifier for Axial Flux Difference (AFD):

1. for AFD between -23% and +10%, = 0% RTP
2. for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP
3. for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP

(p) The compensated temperature difference $\frac{\{1 + \tau_4 s\}}{\{1 + \tau_5 s\}}\left[T\frac{1}{\{1 + \tau_6 s\}} - T'\right]$ shall be no more negative than 3 degrees F.

Note 2: Overpower Delta-T

The Allowable Value of each input to the Overpower Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1) 0.5% ΔT span for the ΔT channel
(2) 0.5% ΔT span for the T_{avg} channel

$$\left[100\frac{\Delta T}{\Delta T_0}\frac{\{1 + \tau_1 s\}}{\{1 + \tau_2 s\}}\frac{1}{\{1+\tau_3 s\}}\right] \leq \left[K_4 - \left[K_5\frac{\{\tau_7 s\}}{\{1 + \tau_7 s\}}\frac{1}{\{1+\tau_6 s\}}T\right] - K_6\left[T\frac{1}{\{1 + \tau_6\}} - T'\right] - f_2(AFD)\right]$$

Where:

ΔT	measured loop specific RCS differential temperature, degrees F
ΔT_0	indicated loop specific RCS differential at RTP, degrees F
$\frac{1+\tau_1 s}{1+\tau_2 s}$	lead-lag compensator on measured differential temperature
τ_1, τ_2	time constants utilized in lead-lag compensator for differential temperature: τ_1 = 0 seconds, τ_2 = 0 seconds
$\frac{1}{1+\tau_3 s}$	lag compensator on measured differential temperature
τ_3	time constant utilized in lag compensator for differential temperature, ≤ 6 seconds
K_4	fundamental setpoint, ≤ 110% RTP
K_5	modifier for temperature change: ≥ 2% RTP per degree F for increasing temperature, ≥ 0% RTP per degree F for decreasing temperature
$\frac{\tau_7 s}{1+\tau_7 s}$	rate-lag compensator on dynamic temperature compensation
τ_7	time constant utilized in rate-lag compensator for temperature compensation, ≥ 10 seconds
T	measured loop specific RCS average temperature, degrees F
$\frac{1}{1+\tau_6 s}$	lag compensator on measured average temperature

Table 3.3.1-1 (page 9 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower Delta-T (continued)

τ_6	time constant utilized in lag compensator for average temperature, $\leq$ 6 seconds
K_6	modifier for temperature: $\geq$ 0.244% RTP per degree F for T > T″, = 0% RTP for T $\leq$ T″
T″	indicated loop specific RCS average temperature at RTP, $\leq$ 588.4 degrees F
s	Laplace transform variable, inverse seconds
f_2(AFD)	modifier for Axial Flux Difference (AFD), = 0% RTP for all AFD

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

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

- pressurizer pressure — the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution — f(AFD)x, the f(AFD) Function is used in the calculation of the Overtemperature ΔT trip. It is a function of the indicated difference between the upper and lower NIS power range detectors. This Function measures the axial power distribution. The Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for RTD response time delays.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. A trip occurs if Overtemperature ΔT is indicated in two loops. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

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

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature — including dynamic compensation for RTD response time delays.

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

(continued)

17. 012K6.02 001/2/1/RPS - RED CHANNEL/C/A 2.9/3.1/BANK-LOIT/HL-18 NRC/RO/SRO/AML

Given the following conditions on Unit 2:

- Reactor power is at 74%.

- A loss of power to vital instrument bus 2AY1A has occurred.

- A Safety Injection has initiated on Low Steam Line Pressure.

Which ONE of the following will be the result of the instrument power loss?

A. ONLY Reactor Trip Breaker 'B' opens, and both Train 'A' and 'B' ESFAS equipment realign.

B. Both Reactor Trip Breaker 'A' and Reactor Trip Breaker 'B' open, and both Train 'A' and 'B' ESFAS equipment realign.

C. ONLY Reactor Trip Breaker 'B' opens, and ONLY Train 'B' ESFAS equipment realigns.

D.✓ Both Reactor Trip Breaker 'A' and Reactor Trip Breaker 'B' open, and ONLY Train 'B' ESFAS equipment realigns.

012K6.02 Reactor Protection System

Knowledge of the effect of a loss or malfunction of the following will have on the RPS: (CFR: 41.7 / 45.7)

Redundant Channels.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where AY1A is lost while the reactor is at power. The candidate has to determine the effects on the reactor trip breakers which have redundant instrumentation and the effects on the ESFAS alignment which is affected by the loss of the Train A slave relays.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. 1st half is incorrect as redundant instrumentation will cause both RTBs to open. 2nd half is incorrect as only Train B ESFAS will realign due to the loss of slave relays to the Train A ESFAS equipment on loss of 2AY1A.

B. Incorrect. 1st half is correct as both RTBs open due to redundant instrumentation. 2nd half is incorrect as only Train B ESFAS will realign due to the loss of slave relays to the Train A ESFAS equipment on loss of 2AY1A.

C. Incorrect. 1st half is incorrect as both RTBs open due to redundant instrumentation. 2nd half is correct as only Train B ESFAS will realign due to the loss of slave relays to the Train A ESFAS equipment on loss of 2AY1A.

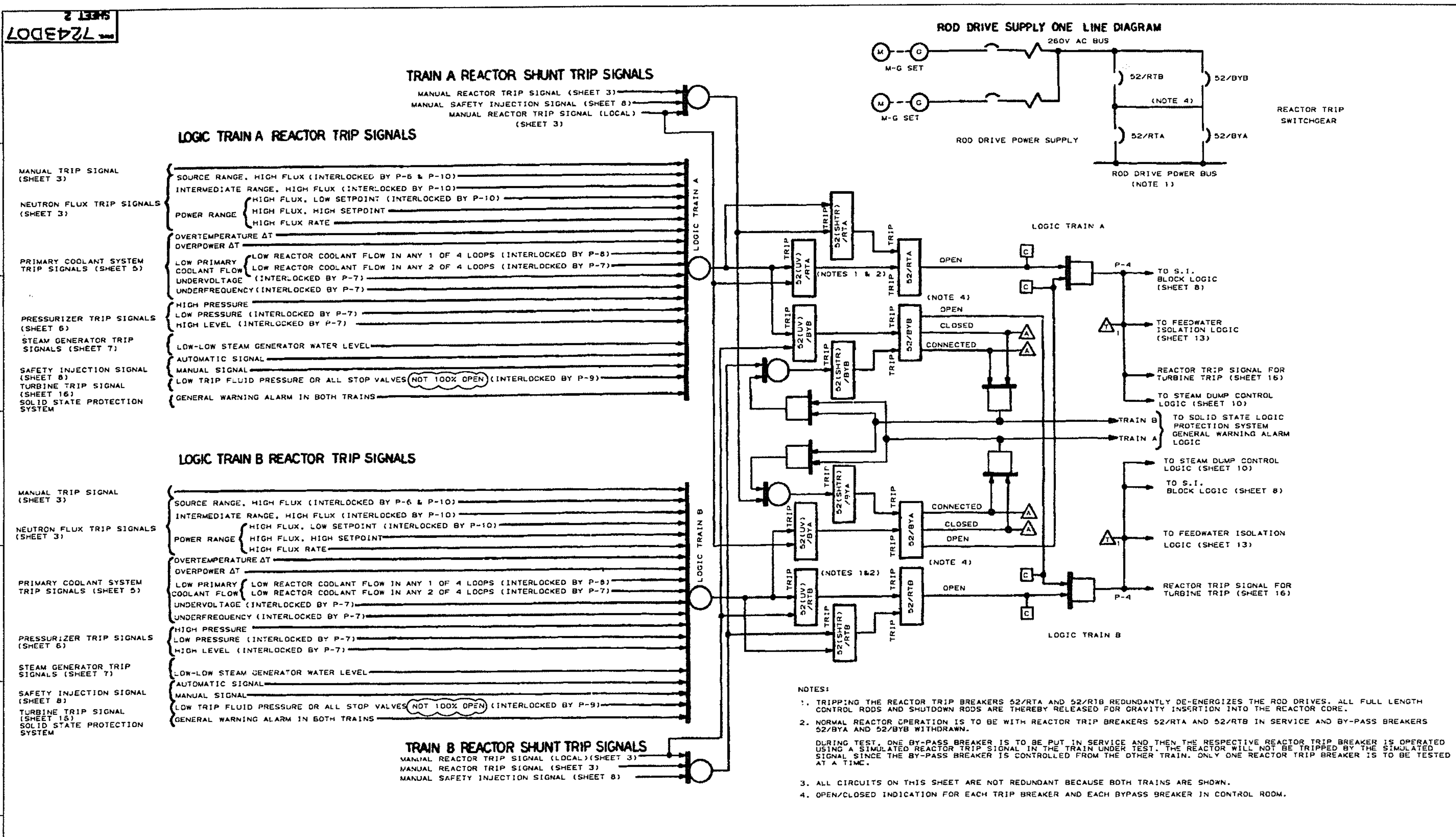
D. Correct. 1st half is correct as both RTBs open due to redundant instrumentation. 2nd half is correct as only Train B ESFAS will realign due to the loss of slave relays to the Train A ESFAS equipment on loss of 2AY1A.

REFERENCES:

2X3D-AA-G02A, One line for 2AY1A
1X6AA02-00232, Logic ESFAS
1X6AA02-00226, Reactor Trip Logic

VEGP learning objectives:

LO-PP-28101-02 Determine how the loss of a power supply will affect RPS.



NOTES:

1. TRIPPING THE REACTOR TRIP BREAKERS 52/RTA AND 52/RTB REDUNDANTLY DE-ENERGIZES THE ROD DRIVES. ALL FULL LENGTH CONTROL RODS AND SHUTDOWN RODS ARE THEREBY RELEASED FOR GRAVITY INSERTION INTO THE REACTOR CORE.
2. 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.

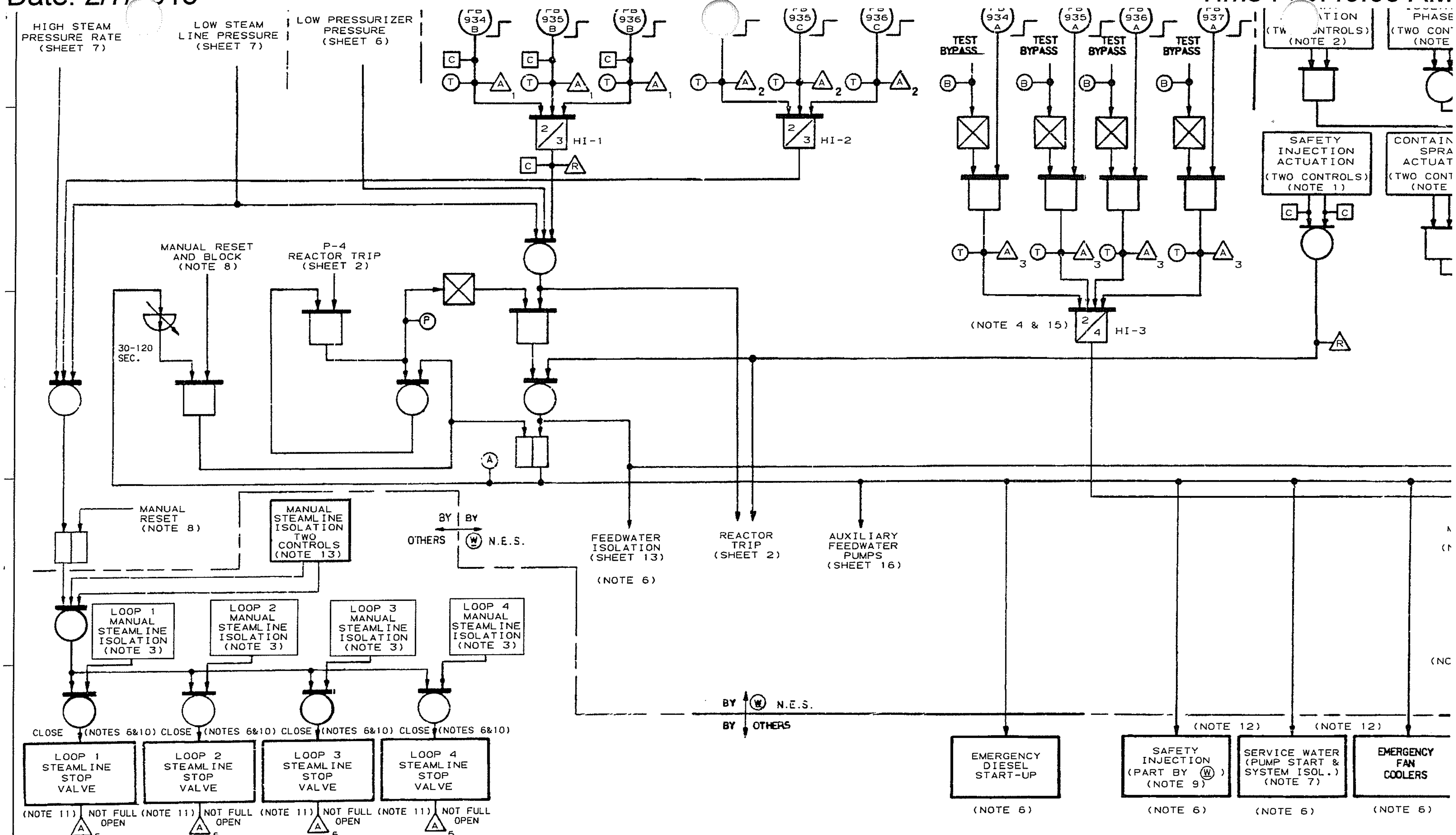
   DURING TEST, ONE BY-PASS BREAKER IS TO BE PUT IN SERVICE AND THEN THE RESPECTIVE REACTOR TRIP BREAKER IS OPERATED USING A SIMULATED REACTOR TRIP SIGNAL IN THE TRAIN UNDER TEST. THE REACTOR WILL NOT BE TRIPPED BY THE SIMULATED SIGNAL SINCE THE BY-PASS BREAKER IS CONTROLLED FROM THE OTHER TRAIN. ONLY ONE REACTOR TRIP BREAKER IS TO BE TESTED AT A TIME.
3. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT BECAUSE BOTH TRAINS ARE SHOWN.
4. OPEN/CLOSED INDICATION FOR EACH TRIP BREAKER AND EACH BYPASS BREAKER IN CONTROL ROOM.

ABN-05114 7 REDRAWN BY SCS AND ISSUED TO INCORP. ABN-05114

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U.S.A.
GEORGIA POWER CO
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
REACTOR TRIP SIGNALS
7243D07
SHEET 2
DO NOT SCALE
1X6AA02-226-7

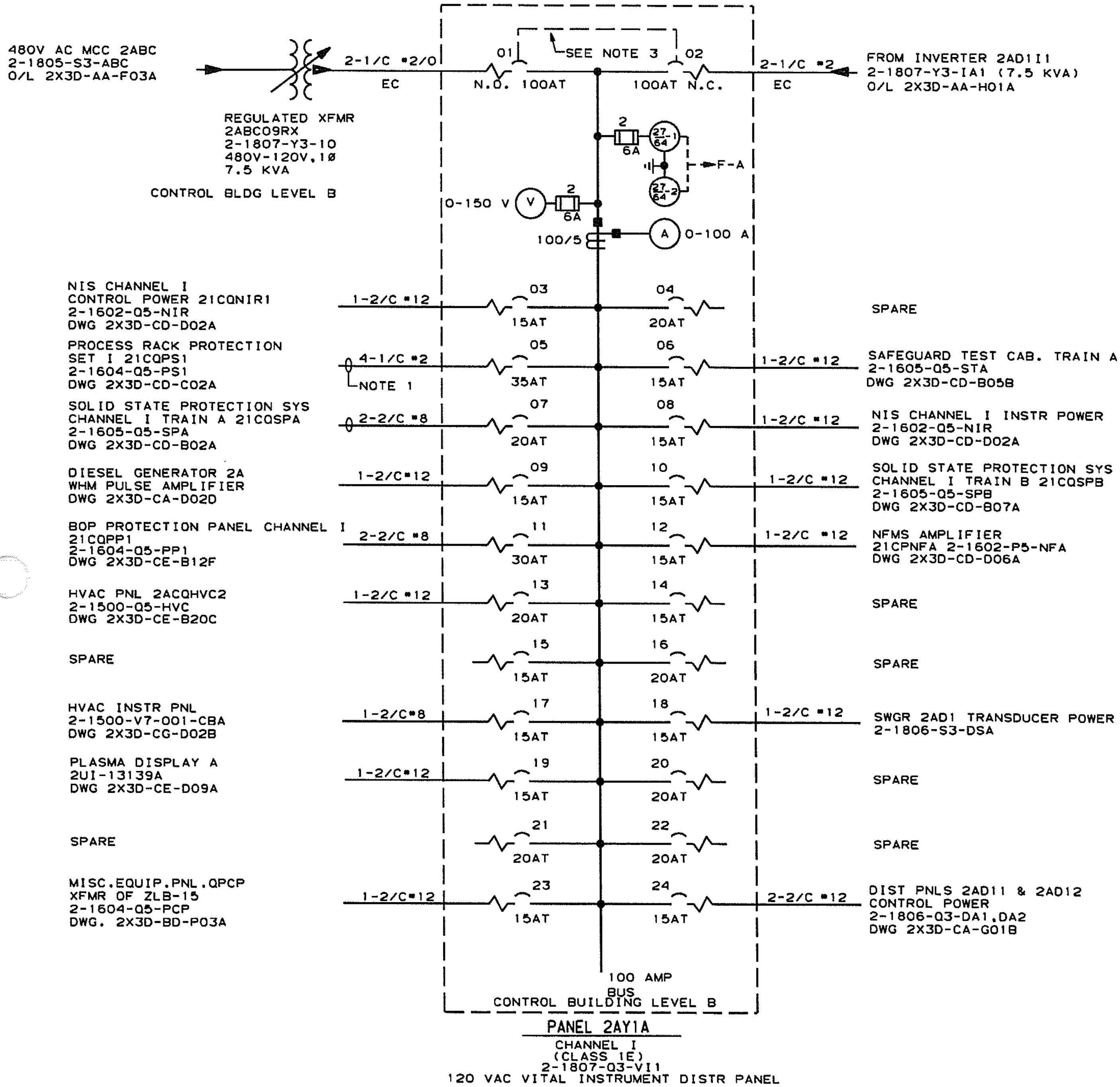
Southern Company Services, Inc.
DATE RECEIVED 12/03/93

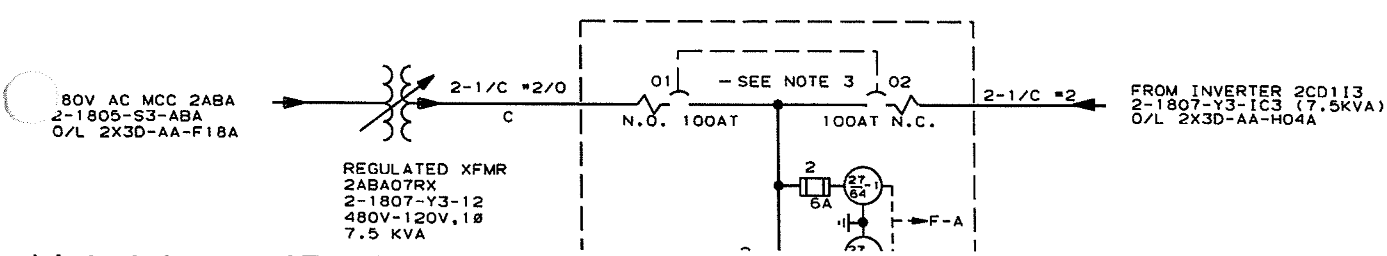
PLANT VOGTLE
1X6AA02-226-9
FUNCTIONAL DIAGRAMS REACTOR



NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, 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 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. CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM, AND IS NOT SAFETY RELATED, HOWEVER, CIRCUITRY IS INPUT TO THE SAFEGUARS SYSTEM THERFORE ISOLATION MUST BE PROVIDED.
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. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
8. 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.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF W NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE W SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE SEQUENCE TO LOAD THE CONTAINMENT SPRAY SYSTEM ON THE EMERGENCY DIESEL POWER SHOULD BE INDEPENDENT OF THE SEQUENCE TO LOAD THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY THE SEQUENCE
17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.


480V AC MCC 2ABC
2-1805-S3-ABC
O/L 2X3D-AA-F03A
2-1/C #2/0
EC
REGULATED XFMR
2ABC09RX
2-1807-Y3-10
480V-120V,1Ø
7.5 KVA
CONTROL BLDG LEVEL B
01
SEE NOTE 3
02
N.O. 100AT
100AT N.C.
2-1/C #2
EC
FROM INVERTER 2AD111
2-1807-Y3-IA1 (7.5 KVA)
O/L 2X3D-AA-H01A
2
6A
27-1
64
27-2
64
F-A
0-150 V
V
2
6A
100/5
A
0-100 A
NIS CHANNEL I
CONTROL POWER 21CQNIR1
2-1602-Q5-NIR
DWG 2X3D-CD-D02A
1-2/C #12
03
15AT
04
20AT
SPARE
PROCESS RACK PROTECTION
SET I 21CQPS1
2-1604-Q5-PS1
DWG 2X3D-CD-C02A
4-1/C #2
NOTE 1
05
35AT
06
15AT
1-2/C #12
SAFEGUARD TEST CAB. TRAIN A
2-1605-Q5-STA
DWG 2X3D-CD-B05B
SOLID STATE PROTECTION SYS
CHANNEL I TRAIN A 21CQSPA
2-1605-Q5-SPA
DWG 2X3D-CD-B02A
2-2/C #8
07
20AT
08
15AT
1-2/C #12
NIS CHANNEL I INSTR POWER
2-1602-Q5-NIR
DWG 2X3D-CD-D02A
DIESEL GENERATOR 2A
WHM PULSE AMPLIFIER
DWG 2X3D-CA-D02D
1-2/C#12
09
15AT
10
15AT
1-2/C #12
SOLID STATE PROTECTION SYS
CHANNEL I TRAIN B 21CQSPB
2-1605-Q5-SPB
DWG 2X3D-CD-B07A
BOP PROTECTION PANEL CHANNEL I
21CQPP1
2-1604-Q5-PP1
DWG 2X3D-CE-B12F
2-2/C #8
11
30AT
12
15AT
1-2/C #12
NFMS AMPLIFIER
21CPNFA 2-1602-P5-NFA
DWG 2X3D-CD-D06A
HVAC PNL 2ACQHVC2
2-1500-Q5-HVC
DWG 2X3D-CE-B20C
1-2/C #12
13
20AT
14
15AT
SPARE
SPARE
15
15AT
16
20AT
SPARE
HVAC INSTR PNL
2-1500-V7-001-CBA
DWG 2X3D-CG-D02B
1-2/C#8
17
15AT
18
15AT
1-2/C #12
SWGR 2AD1 TRANSDUCER POWER
2-1806-S3-DSA
PLASMA DISPLAY A
2UI-13139A
DWG 2X3D-CE-D09A
1-2/C#12
19
15AT
20
20AT
SPARE
SPARE
21
20AT
22
20AT
SPARE
MISC.EQUIP.PNL.QPCP
XFMR OF ZLB-15
2-1604-Q5-PCP
DWG. 2X3D-BD-P03A
1-2/C#12
23
15AT
24
15AT
2-2/C #12
DIST PNLS 2AD11 & 2AD12
CONTROL POWER
2-1806-Q3-DA1,DA2
DWG 2X3D-CA-G01B
100 AMP
BUS
CONTROL BUILDING LEVEL B
PANEL 2AY1A
CHANNEL I
(CLASS 1E)
2-1807-Q3-VI1
120 VAC VITAL INSTRUMENT DISTR PANEL


80V AC MCC 2ABA
2-1805-S3-ABA
O/L 2X3D-AA-F18A
2-1/C #2/0
C
REGULATED XFMR
2ABA07RX
2-1807-Y3-12
480V-120V,1Ø
7.5 KVA
01
SEE NOTE 3
02
N.O. 100AT
100AT N.C.
2-1/C #2
FROM INVERTER 2CD113
2-1807-Y3-IC3 (7.5KVA)
O/L 2X3D-AA-H04A
2
6A
27-1
64
F-A

18. 013G2.1.19 001/2/1/ESFA - PLNT COMPUTER/MEM - 3.9/3.8/NEW/HL-18 NRC/RO/SRO/TNT

Given the following:

- A Reactor trip and Safety Injection have occurred.

- The MLB status lights for CIA valves are NOT available.

- The CNMT ISO PHASE A ACTUATION annunciator is not available.

Which one of the following completes the following statement?

There ___(1)___ a group on the IPC computer where the OATC can determine the status of the CIA valves,

and

when CIA is reset, the OATC ___(2)___ determine whether CIA has reset using the IPC computer.

	__(1)__	__(2)__
A.✓	is	can
B.	is	can NOT
C.	is NOT	can
D.	is NOT	can NOT

013G2.1.19 Engineered Safety Features Actuation

Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12)

<u>K/A MATCH ANALYSIS:</u>

The question presents a plausible scenario where a reactor trip and SI have occurred with complications. The candidate must determine if he has the ability to determine proper CIA alignment using an IPC computer group and can the IPC be used to determine if CIA is reset without the CIA annunciator available.

<u>DISTRACTOR ANALYSIS:</u>

A. Correct. There is a group that has all the CIA and CVI valve positions. On top level digital of the IPC, a CI light illuminates when CIA is present and extinguishes when

CIA is reset.

B. Incorrect. The OATC can determine if CIA is reset on the IPC computer, it is plausible the candidates may not know this indication is present on the IPC computer since it is not usually what is checked when CIA is reset.

C. Incorrect. The OATC can use a group to determine if CIA valves are shut. It is plausible the candidate may not know this group is present since the MLBs and / or handswitches are usually checked. This group is normally checked in the 19100 Loss of All AC Power procedure.

D. Incorrect. The OATC can use a group to determine if CIA valves are shut. It is plausible the candidate may not know this group is present since the MLBs and / or handswitches are usually checked. This group is normally checked in the 19100 Loss of All AC Power procedure.

The OATC can determine if CIA is reset on the IPC computer, it is plausible the candidates may not know this indication is present on the IPC computer since it is not usually what is checked when CIA is reset.

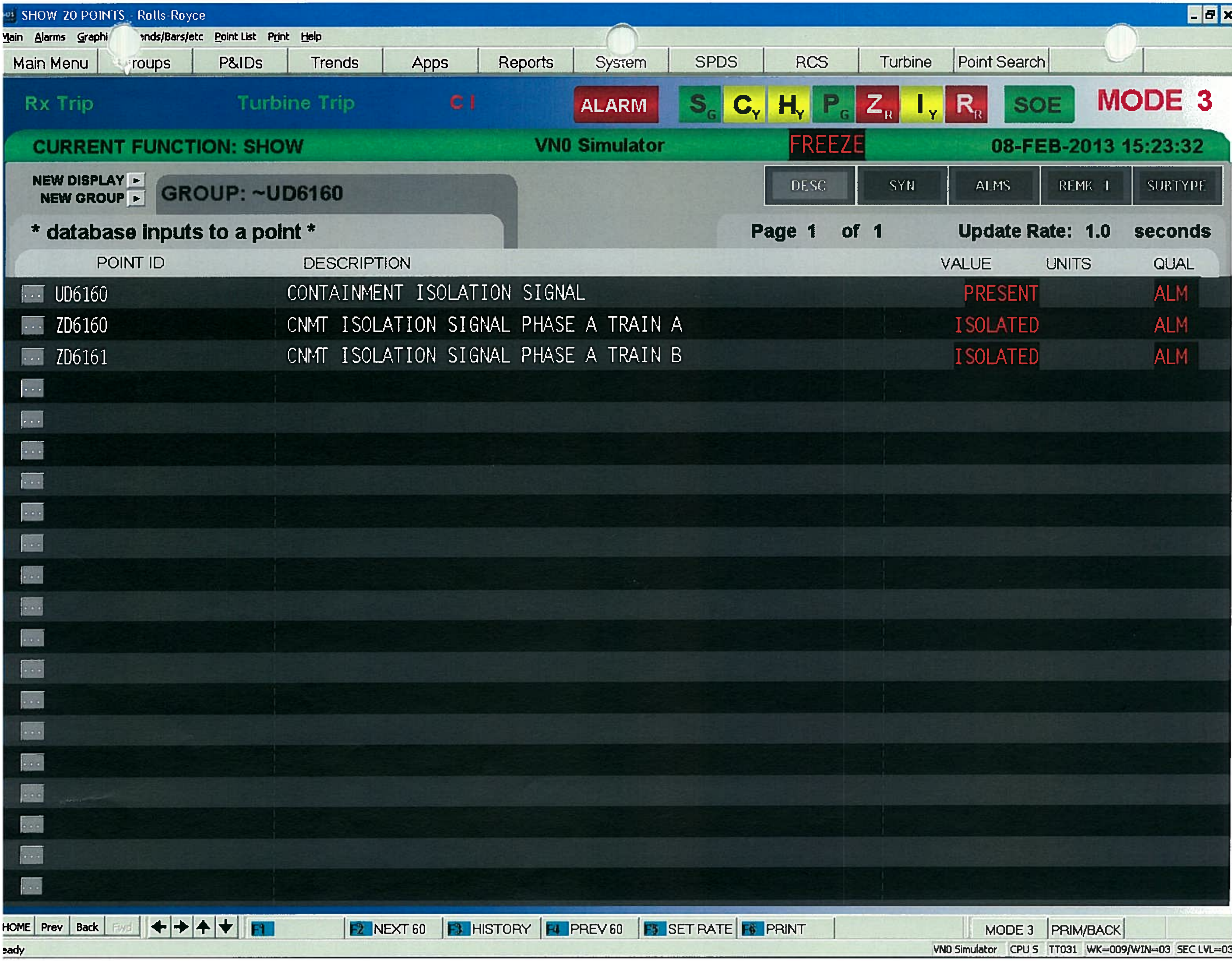
REFERENCES:

IPC Computer Point for CIA actuation signal.
IPC Computer Printout for Containment Isolation Valves screen.
Simulator print outs of GP 4 MON LTS
Simulator print outs of ALB06-E06 CNMT ISO PHASE A ISOLATION

VEGP learning objectives:

LO-PP-05210-04 Describe the IPC screen layout, including the types of information found in each menu:

a. GROUPS
b. SYSTEM
c. TRENDS
d. P&IDs.



SHOW 20 POINTS - Rolls-Royce
Main Alarms Graphi ends/Bars/etc Point List Print Help
Main Menu roups P&IDs Trends Apps Reports System SPDS RCS Turbine Point Search
Rx Trip Turbine Trip C I ALARM S C H P Z I R SOE MODE 3
CURRENT FUNCTION: SHOW VN0 Simulator FREEZE 08-FEB-2013 15:23:32
NEW DISPLAY
NEW GROUP
GROUP: ~UD6160
DESC SYN ALMS REMK I SUBTYPE
* database inputs to a point *
Page 1 of 1 Update Rate: 1.0 seconds
POINT ID DESCRIPTION VALUE UNITS QUAL
UD6160 CONTAINMENT ISOLATION SIGNAL PRESENT ALM
ZD6160 CNMT ISOLATION SIGNAL PHASE A TRAIN A ISOLATED ALM
ZD6161 CNMT ISOLATION SIGNAL PHASE A TRAIN B ISOLATED ALM
HOME Prev Back Fwd F1 F2 NEXT 60 F3 HISTORY F4 PREV 60 F5 SET RATE F6 PRINT MODE 3 PRIM/BACK
eady VN0 Simulator CPU S TT031 WK=009/WIN=03 SEC LVL=03

SHOW 20 POINTS - Rolls-Royce

Main | Alarms | Graphi[...]ends/Bars/etc | Point List | Print | Help

Main Menu | Groups | P&IDs | Trends | Apps | Reports | System | SPDS | RCS | Turbine | Point Search

Rx Trip | Turbine Trip | ALARM | S_G | C_Y | H_Y | P_G | Z_R | I_Y | R_R | SOE | MODE 3

CURRENT FUNCTION: SHOW | VN0 Simulator | FREEZE | 08-FEB-2013 15:23:54

NEW DISPLAY
NEW GROUP

GROUP: ~UD6160

DESC | SYN | ALMS | REMK 1 | SUBTYPE

* database Inputs to a point *

Page 1 of 1 | Update Rate: 1.0 seconds

POINT ID	DESCRIPTION	VALUE	UNITS	QUAL
UD6160	CONTAINMENT ISOLATION SIGNAL	NOT PRESENT		GOOD
ZD6160	CNMT ISOLATION SIGNAL PHASE A TRAIN A	NORMAL		GOOD
ZD6161	CNMT ISOLATION SIGNAL PHASE A TRAIN B	NORMAL		GOOD

HOME | Prev | Back | Fwd | F1 | F2 NEXT 60 | F3 HISTORY | F4 PREV 60 | F5 SET RATE | F6 PRINT | MODE 3 | PRIM/BACK

eady | VN0 Simulator | CPU 5 | TT031 | WK=009/WIN=03 | SEC LVL=03

TOP LEVEL SPDS - Rolls-Royce

Main Alarms Graphics Trends/Bars/etc Point List Print Help

Main Menu | Groups | P&IDs | Trends | Apps | Reports | System | SPDS | RCS | Turbine | Point Search

Rx Trip Turbine Trip S I C I ALARM S_G C_Y H_Y P_G Z_M I_Y R_R SOE **MODE 0**

CURRENT FUNCTION: SPDSTOP **VN0 Simulator** **08-FEB-2013 13:30:57**

SR FLUX (CPS)	1.099E+02	CORE EXIT (F)	524
SR SU RATE (DPM)	-2.857E-01	CNMT ISOL	YES
HEATUP RATE (F/HR)	0.0	CNMT PRES (PSIG)	5.5
SUBCOOLING (F)	7	CNMT HYDROGEN (%)	BAD
RCS PRESS (PSIG)	876.3	CNMT H2O LVL (IN)	0.0
PRZR LEVEL (%)	0.0	CST 1 LEVEL (%)	96.3
RVLIS FULL RANGE (%)	93.7	CST 2 LEVEL (%)	97.5
RVLIS UPPER HEAD (%)	95.9	RWST LEVEL (%)	93.3

	LOOP 1	LOOP 2	LOOP 3	LOOP 4
RCP STATUS	STOPPED	STOPPED	STOPPED	STOPPED
T - HOT (F)	532.1	532.3	532.9	532.8
T - COLD (F)	529.4	529.4	528.0	528.8
SG PRESS (PSIG)	856	856	867	864
SG NR LEVEL (%)	34.6	43.1	43.4	30.8
SG WR LEVEL (%)	55.9	58.7	58.7	54.8
AUX FW FLOW (GPM)	31.6	31.4	15.7	20.6
MS RAD (UCI/CC)	0.000E+00	0.000E+00	0.000E+00	0.000E+00

CNMT	STM GENS	GASEOUS	LIQUID	AREA
RED	GREEN	MAGENTA	MAGENTA	MAGENTA

HOME Prev Back Fwd F1 F2 F3 HISTORY F4 F5 F6 MODE 0 PRIM/BACK

Ready VN0 Simulator CPU S TT031 WK=009/WIN=03 SEC LVL=03

TOP LEVEL SPDS - Rolls-Royce

Main Alarms Graphi… …ends/Bars/etc Point List Print Help

Main Menu | Groups | P&IDs | Trends | Apps | Reports | System | SPDS | RCS | Turbine | Point Search

Rx Trip Turbine Trip ALARM S_G C_Y H_G P_G Z_M I_Y R_R SOE **MODE 0**

CURRENT FUNCTION: SPDSTOP VN0 Simulator 08-FEB-2013 13:40:55

SR FLUX (CPS)	1.666E+01	CORE EXIT (F)	496
SR SU RATE (DPM)	-6.130E-03	CNMT ISOL	YES
HEATUP RATE (F/HR)	0.0	CNMT PRES (PSIG)	7.2
SUBCOOLING (F)	26	CNMT HYDROGEN (%)	BAD
RCS PRESS (PSIG)	813.2	CNMT H2O LVL (IN)	0.0
PRZR LEVEL (%)	0.0	CST 1 LEVEL (%)	96.6
RVLIS FULL RANGE (%)	102.1	CST 2 LEVEL (%)	97.6
RVLIS UPPER HEAD (%)	104.7	RWST LEVEL (%)	91.4

	LOOP 1	LOOP 2	LOOP 3	LOOP 4
RCP STATUS	STOPPED	STOPPED	STOPPED	STOPPED
T - HOT (F)	505.7	506.2	528.0	528.2
T - COLD (F)	520.6	521.1	511.0	509.8
SG PRESS (PSIG)	767	769	844	807
SG NR LEVEL (%)	37.7	45.9	44.6	33.0
SG WR LEVEL (%)	57.5	59.9	59.2	55.8
AUX FW FLOW (GPM)	116.3	114.0	13.0	63.6
MS RAD (UCI/CC)	0.000E+00	0.000E+00	0.000E+00	0.000E+00

CNMT RED | STM GENS GREEN | GASEOUS MAGENTA | LIQUID MAGENTA | AREA MAGENTA

HOME Prev Back Fwd F1 F2 F3 HISTORY F4 F5 F6 MODE 0 PRIM/BACK

eady VN0 Simulator CPU 5 TT031 WK=009/WIN=03 SEC LVL=03

Main Alarms Graphics Trends/Bars/etc Point List Print Help

Main Menu | Groups | P&IDs | Trends | Apps | Reports | System | SPDS | RCS | Turbine | Point Search

ALARM S C H P Z I R SOE **MODE 1**

CURRENT FUNCTION: CNMTISOL **VN1 IS** **DG** **22-FEB-2013 13:21:20**

CONTAINMENT ISOLATION VALVES

SYS#	VALVE	VLV POSN
1201	HV-8028	NOT CLOSED
	HV-8033	CLOSED
	HV-8047	CLOSED
1204	HV-8823	CLOSED
	HV-8824	CLOSED
	HV-8825	CLOSED
	HV-8843	CLOSED
	HV-8871	CLOSED
	HV-8881	CLOSED
	HV-8888	CLOSED
	HV-8890A	CLOSED
	HV-8890B	CLOSED
	HV-10950	CLOSED
	HV-10951	CLOSED
	HV-10952	CLOSED
	HV-10953	CLOSED
	HV-8964	CLOSED
1208	HV-8100	NOT CLOSED
	HV-8112	NOT CLOSED
	HV-8152	NOT CLOSED
	HV-8160	NOT CLOSED

SYS#	VALVE	VLV POSN
1212	HV-3502	NOT CLOSED
	HV-3507	CLOSED
	HV-3508	CLOSED
	HV-3513	CLOSED
	HV-3514	CLOSED
	HV-3548	NOT CLOSED
1214	HV-0780	NOT CLOSED
	HV-0781	NOT CLOSED
1901	HV-7699	NOT CLOSED
	HV-7126	NOT CLOSED
	HV-7136	NOT CLOSED
	HV-7150	NOT CLOSED
2301	HV-27901	CLOSED
2401	HV-9385	CLOSED
2402	HV-8880	CLOSED
2420	HV-9378	NOT CLOSED
2702	HV-8211	CLOSED
	HV-8212	CLOSED

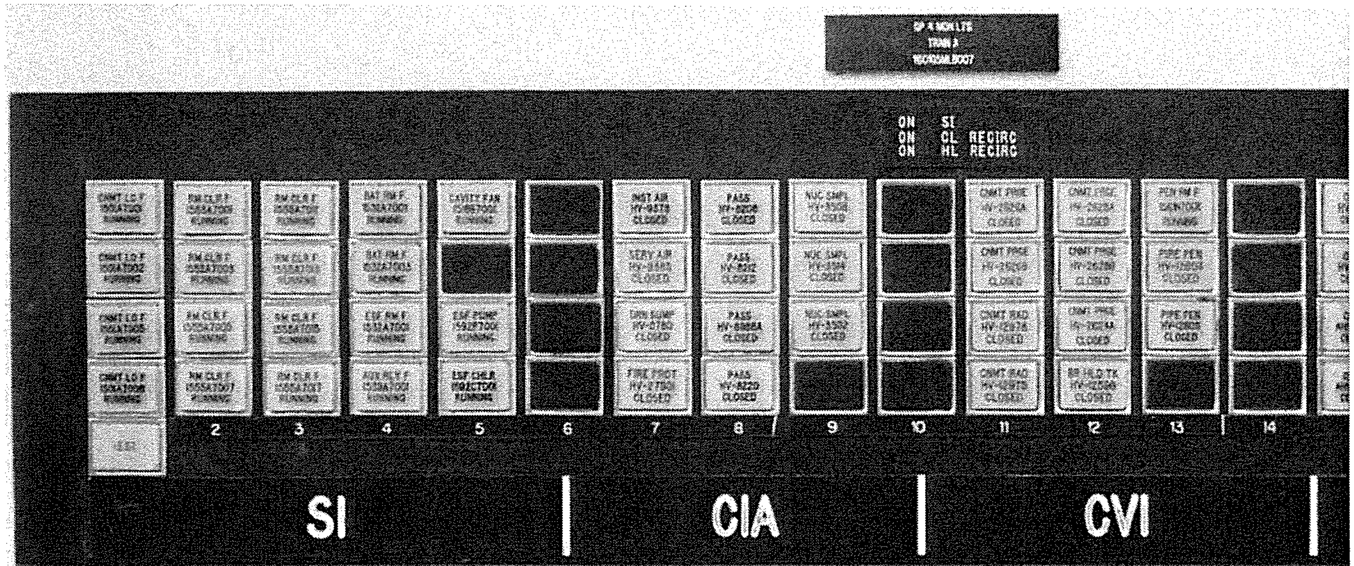
**** CONTAINMENT CVI ****

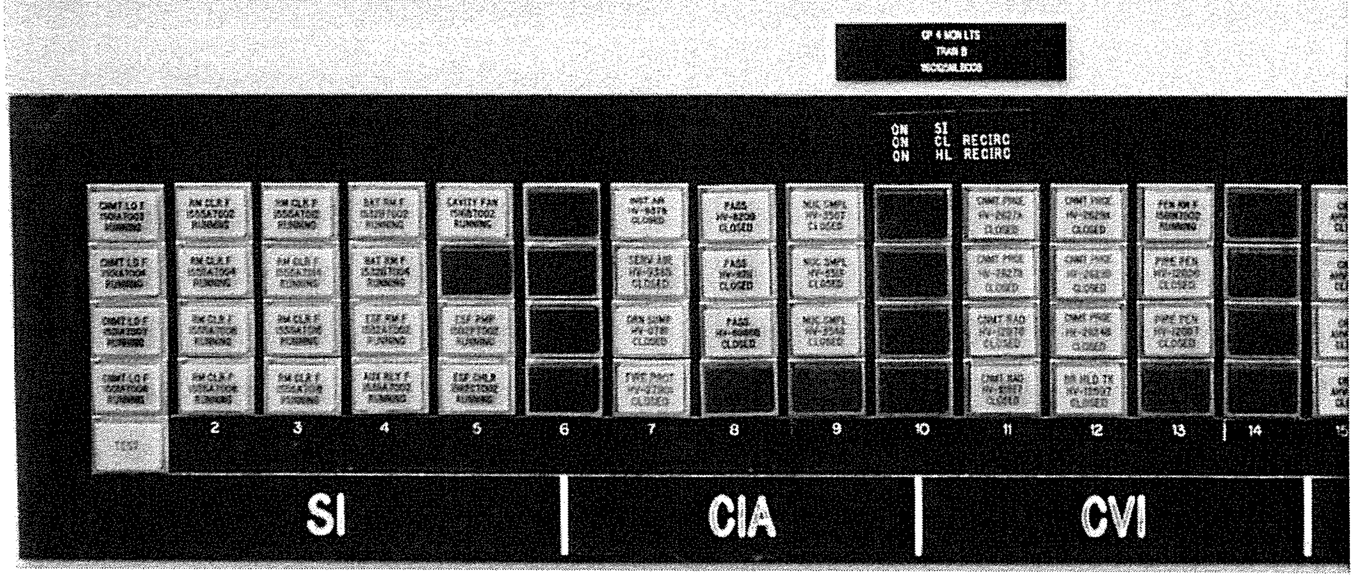
SYS#	VALVE	VLV POSN
1609	HV-12975	NOT CLOSED
	HV-12976	NOT CLOSED
	HV-12977	NOT CLOSED
	HV-12978	NOT CLOSED
1505	HV-2626A	CLOSED
	HV-2626B	CLOSED
	HV-2627A	CLOSED
	HV-2627B	CLOSED
1506	HV-2628A	CLOSED
	HV-2628B	CLOSED
	HV-2629A	CLOSED
	HV-2629B	CLOSED
1508	HV-2624A	CLOSED
	HV-2624B	CLOSED

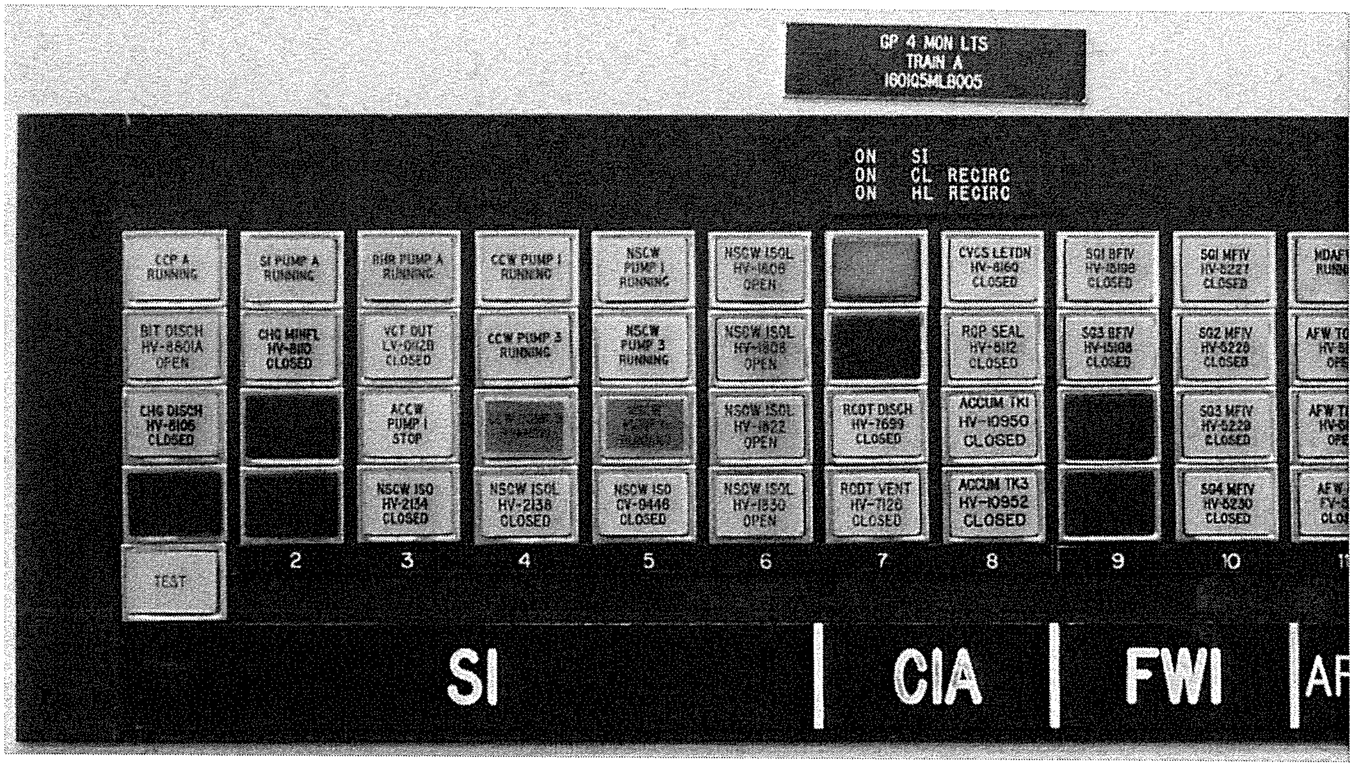
* * * * AUX. BLDG. CVI * * * *
NOT USED IN CVI LOGIC

SYS#	VALVE	VLV POSN
1561	HV-12604	NOT CLOSED
	HV-12605	NOT CLOSED
	HV-12606	NOT CLOSED
	HV-12607	NOT CLOSED

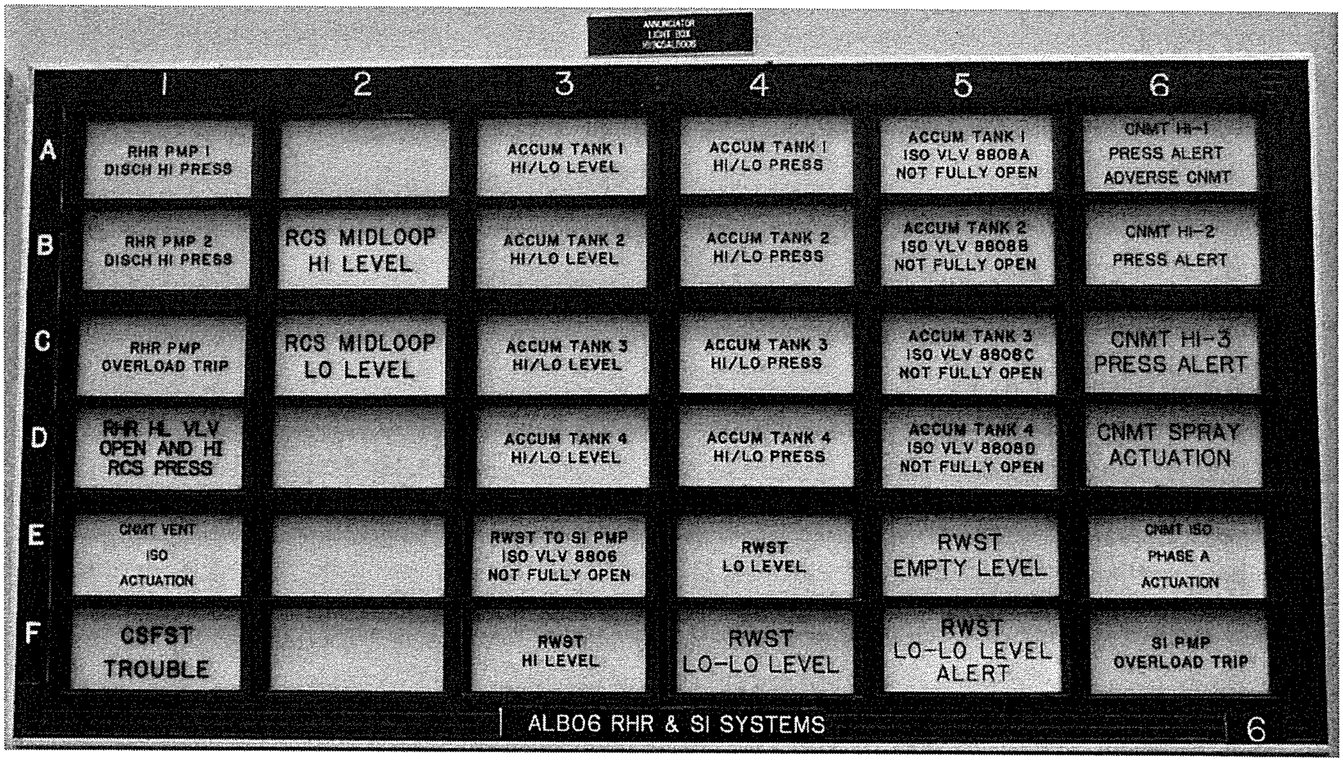
HOME Prev Back Fwd MODE 1 PRIM/BACK


ON SI
ON CL RECIRC
ON HL RECIRC
2
3
4
5
6
7
8
9
10
11
12
13
14
SI
CIA
CVI


ON SI
ON CL RECIRC
ON HL RECIRC
2
3
4
5
6
7
8
9
10
11
12
13
14
SI
CIA
CVI


GP 4 MON LTS
TRAIN A
1601Q5ML8005
ON SI
ON CL RECIRC
ON HL RECIRC
CCP A RUNNING
BIT DISCH HV-8801A OPEN
CHG DISCH HV-8106 CLOSED
TEST
SI PUMP A RUNNING
CHG MINFL HV-8110 CLOSED
RHR PUMP A RUNNING
VCT OUT LV-0112B CLOSED
ACCW PUMP 1 STOP
NSCW ISO HV-2134 CLOSED
CCW PUMP 1 RUNNING
CCW PUMP 3 RUNNING
NSCW ISOL HV-2138 CLOSED
NSCW PUMP 1 RUNNING
NSCW PUMP 3 RUNNING
NSCW ISO CV-9446 CLOSED
NSCW ISOL HV-1806 OPEN
NSCW ISOL HV-1808 OPEN
NSCW ISOL HV-1822 OPEN
NSCW ISOL HV-1830 OPEN
RCDT DISCH HV-7699 CLOSED
RCDT VENT HV-7126 CLOSED
CVCS LETDN HV-8160 CLOSED
RCP SEAL HV-8112 CLOSED
ACCUM TK1 HV-10950 CLOSED
ACCUM TK3 HV-10952 CLOSED
SG1 BFIV HV-15196 CLOSED
SG3 BFIV HV-15198 CLOSED
SG1 MFIV HV-5227 CLOSED
SG2 MFIV HV-5228 CLOSED
SG3 MFIV HV-5229 CLOSED
SG4 MFIV HV-5230 CLOSED
2
3
4
5
6
7
8
9
10
SI
CIA
FWI

ALB05		ALB07
"A" Panel - Right Side	ARP 17006	Full Size Image of ALB06



If you find any problems or have suggestions for improvement of this site, contact Mick Youmans at 8-695-3906 or click here to E-mail.

19. 013K3.03 001/2/1/ESFAS - CONTAINMENT/MEM - 4.3/4.7/NEW/HL-18 NRC/RO/SRO/AML

Given the following conditions:

- A LOCA has occurred.

The OATC is at the Step in 19000-C, "Reactor Trip or Safety Injection," OATC Initial Actions, to "Verify Containment Cooling Units."

Which ONE of the following describes the required condition of the Containment Cooling Units

and

the effect on Containment if the Cooling Units are NOT in the required condition?

A. All running in LOW speed.

Results in higher overall containment temperatures due to exceeding the cooling capacity of the NSCW system.

B✓ All running in LOW speed.

The Cooling units will overload under higher density atmospheric conditions resulting in higher Containment pressure.

C. All running in HIGH speed.

Results in higher localized temperatures due to inadequate mixing of the Containment atmosphere.

D. All running in HIGH speed.

The Cooling Units will overload under higher density atmospheric conditions resulting in higher Containment pressure.

013K3.03 Engineered Safety Features Actuation System (ESFAS)

Knowledge of the effect that a loss or malfunction of ESFAS will have on the following: (CFR: 41.7 / 45.6)

Containment

K/A MATCH ANALYSIS:

The question asks the candidates what a malfunction of the Containment Cooling

temperature will have on Containment conditions.

DISTRACTOR ANALYSIS

A Incorrect. 1st half is correct, coolers are required to be in LOW speed. 2nd half is incorrect, Containment temperature will not be high due to inadequate NSCW flow if coolers are in the wrong speed.

B Correct. 1st half is correct, coolers are required to be in LOW speed. 2nd half is correct, if coolers are in the wrong speed the higher density atmosphere will overload the coolers resulting in a higher containment temperature.

C Incorrect. 1st half is incorrect, LOW speed is required. Plausible if the candidate thinks higher speed is required for adequate mixing and prevention of hot spots. This is a purpose of the lower level circulators.

D Incorrect. 1st half is incorrect, LOW speed is required. 2nd half is correct, if coolers are in the wrong speed the higher density atmosphere will overload the coolers resulting in a higher containment temperature. However this will be if they do not shift to LOW speed.

REFERENCES

19000-C, "Reactor Trip or Safety Injection", OATC Initial Actions
Tech Spec 3.6.6 Bases for Containment Cooling
FSAR, Section 6

VEGP learning objectives:

LO-PP-29101-13 State why two speeds are provided for the Containment Coolers and when each speed is used.

LO-PP-29101-14 State all auto start signals for the Containment Cooling including set points and coincidence where applicable.

LO-PP-29101-15 State the starting interlocks associated with the Containment Cooling fans. Include set points and coincidence where applicable.

BASES

BACKGROUND

Containment Cooling System (continued)

During normal operation, four fan units are operating. The fans are normally operated at high speed with NSCW supplied to the cooling coils. The Containment Cooling System, operating in conjunction with the Containment Ventilation and Air Conditioning systems, is designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the NSCW is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 36.5 psig (experienced during a LOCA). The analysis shows that the peak containment temperature is 303.1°F (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4A, "Containment Pressure," and

(continued)

6.2.2.1.2.3 Component Description. Each 25%-capacity fan cooling unit consists of a vane axial fan, a fan motor, copper-nickel cooling coils with copper fins, a carbon steel housing, round metal ducting, and a concrete discharge duct. The 8 units are located at el 238 ft in the containment building and discharge to the bottom of the containment building through four concrete ducts. Each fan unit has two speeds of operation, high speed for normal operation and low speed for post-accident operation. For purposes of heat removal from the containment atmosphere, any combination of four fans in slow speed is sufficient.

The containment cooling system is separated into two trains consisting of four fan cooler units each. The two trains are supplied cooling water and electrical power from the corresponding train of the NSCW system and the Class 1E electrical power system.

Each concrete air supply duct contains three large outlets located above the maximum calculated containment flood elevation to supply cool air to the containment. The air cooling unit enclosures and concrete air supply duct remain intact following a design basis accident (DBA). There is a backdraft damper located at the inlet of each concrete air supply duct to prevent overpressurizing the ducting and coolers. The capability to remain intact is discussed in relation to the hydrogen mixing function of the containment cooling system in paragraph 6.2.5.2.1. Plan and elevation drawings of the containment showing the routing of airflow guidance ductwork are given in drawings 1X4DJ4103, 1X4DJ4113, 1X4DJ4123 and 1X4DJ4133. The fans and motors are designed to operate in the containment post-accident environment. The heat removal capability of the containment fan coolers versus containment temperature for the maximum NSCW inlet temperature is provided in table 6.2.2-2.

6.2.2.1.2.4 System Operation. The containment cooling system is an ESF system that is in use during normal plant operation with four fans operating at high speed. The containment isolation valves on the NSCW line to the containment air cooling units are normally open. System ESF operation is initiated automatically upon receipt of an SI signal. Upon receipt of an SI signal, all fans are restarted at low speed. The basis for the setpoint for the automatic initiation of the cooling system is the diesel start time (12 + 0.5 s) and the load sequencer for the fan coolers (30.5 s). The containment cooling system is fully operational within 40.5 s following receipt of an SI signal. The containment air cooling units can be stopped and started from the control room and from the shutdown panels.

6.2.2.1.3 Safety Evaluation

A. The safety-related portions of the containment cooling system are located in the containment building. This building is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods.

B. Operation of the containment cooling system is initiated automatically following the receipt of an SI signal. Use of this signal provides a reliable indication that a LOCA or a MSLB accident has occurred inside containment. Operation of the containment air coolers may also be manually initiated from the control room and from the shutdown panels. A detailed description of the actuation system is contained in section 7.3.

C. Two trains, each containing four air cooling units, are supplied from redundant emergency power sources and redundant NSCW trains. Failure of any component in one train will not affect the operability of the other train. The containment analyses of LOCA and MSLB accidents were performed in

subsection 6.2.1, assuming the availability of one of two containment air cooling trains. A failure modes and effects analysis of the containment cooling system is presented in table 6.2.2-3.

D. Capability is provided to periodically test the entire startup sequence of the containment air cooling system. Active components can be tested periodically during plant operation to verify operability. The entire system can be inspected during unit shutdown. Additional information is contained in section 3.1, paragraph 6.2.2.2.4, and the Technical Specifications.

E. The containment air cooling units are tested and demonstrated to perform in a simulated MSLB and LOCA environment. The units are located in a manner to minimize the effects of jet impingement and pipe whip in case of a high-energy line break.

F. The analyses show that the containment cooling system in conjunction with the containment spray system is capable of removing sufficient heat energy and subsequent decay heat from the containment atmosphere to ensure the accident peak pressure is below the containment design pressure. Accident analyses assume the occurrence of a single failure that results in the loss of one air cooling train and one containment spray train. Containment cooler heat removal capacity is provided in table 6.2.2-2.

   Containment accident analysis assumes a constant NSCW temperature equal to the highest anticipated system temperature (95°F) to maximize the calculated containment peak pressure. The assumptions used in calculating this temperature are discussed in subsections 9.2.1 and 9.2.5.

   Curves showing heat removal rates of the containment air cooling system, and containment total pressure, and temperatures as a function of time for minimum ESF performance are given in the figures of subsection 6.2.1.

   Location of the containment air cooling units in different quadrants of the containment, the difference in elevation between suction and discharge points, and the significant flowrate developed (table 6.2.2-1) ensure adequate circulation in the containment following a LOCA which prevents the formation of localized high temperature air pockets or areas of high combustible gas concentration. The mixing capability of this system is supplemented by the containment spray system and natural convection. Additional information is contained in subsection 6.2.5 and section 9.4.

G. The containment cooling system is designed to Seismic Category 1 requirements as specified in section 3.2.

6.2.2.1.4 Testing and Inspection

Fans are tested and rated by the manufacturer in accordance with the standards of the Air Moving and Conditioning Association.

The containment air cooling units are tested and/or analyzed by the manufacturer to ensure operation in the post-accident environmental conditions indicated in section 3.11 and following a safe shutdown earthquake (SSE).

OATC INITIAL ACTIONS

Sheet 2 of 4

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. Verify proper NSCW system operation:	
__a. NSCW Pumps - ONLY TWO RUNNING EACH TRAIN.	
b. NSCW TOWER RTN HDR BYPASS BASIN handswitches - IN AUTO: __• HS-1668A __• HS-1669A	
6. Verify Containment Cooling Units:	
a. ALL RUNNING IN LOW SPEED: __• MLB indication	
b. NSCW Cooler isolation valves - OPEN: __• MLB indication	
7. Check Containment Ventilation Isolation:	
a. Dampers and Valves - CLOSED: __• CVI MLB indication	a. Perform the following: __1) Close Dampers and Valves. __2) Start Piping Pen Units.

20. 013K6.01 001/2/1/ESFAS - DETECTOR/MEM - 2.7/3.1/MOD - HL17NRC/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- Unit 1 is at 100% power.

- The bistables for Containment Pressure Channel I (1PT-937) are de-energized.

- No operator action has been taken.

Based on the given conditions, which ONE of the following indicates the MINIMUM number of ADDITIONAL channels required to initiate an actuation signal on HIGH-1 and HIGH-3 Containment Pressure?

	HIGH-1	HIGH-3
A.	1	1
B.	1	2
C.	2	1
D.✓	2	2

013K6.01 Engineered Safety Features Actuation System (ESFAS)

Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: (CFR: 41.7 / 45.5 to 45.8)

Sensors and detectors.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a Containment pressure channel fails high and no operator action has been taken. The candidate must determine how many of the remaining OPERABLE channels are required for a High-1 and High-3 actuation to occur. Channel 1 does NOT have a bistable for High-1 or High-2 so require 2 more channels to actuate. High-3 is a 2 of 4 coincidence, however, with the bistable de-energized, High-3 is an energize to actuate bistable so it still takes 2 more channels for the actuation.

DISTRACTOR ANALYSIS

A. Incorrect. There is no High 1 bistable for channel 1. The High-3 bistable is bypassed per Tech Spec actions so 2 are still required for the actuation on High 1 and High 3.

B. Incorrect. There is no High 1 bistable for channel 1. The High-3 bistable is bypassed per Tech Spec actions so 2 are still required for the actuation on High 1 and High 3.

C. Incorrect. There is no High 1 bistable for channel 1. The High-3 bistable is bypassed per Tech Spec actions so 2 are still required for the actuation on High 1 and High 3.

D. Correct. There is no High 1 bistable for channel 1. The High-3 bistable is bypassed per Tech Spec actions so 2 are still required for the actuation on High 1 and High 3.

REFERENCES

Tech Spec 3.3.2, Engineered Safety Features Actuation System (ESFAS), FU 1c and FU 2c and Bases.
1X6AA02-00232, ESFAS Logic
HL-15R audit 013K6.01

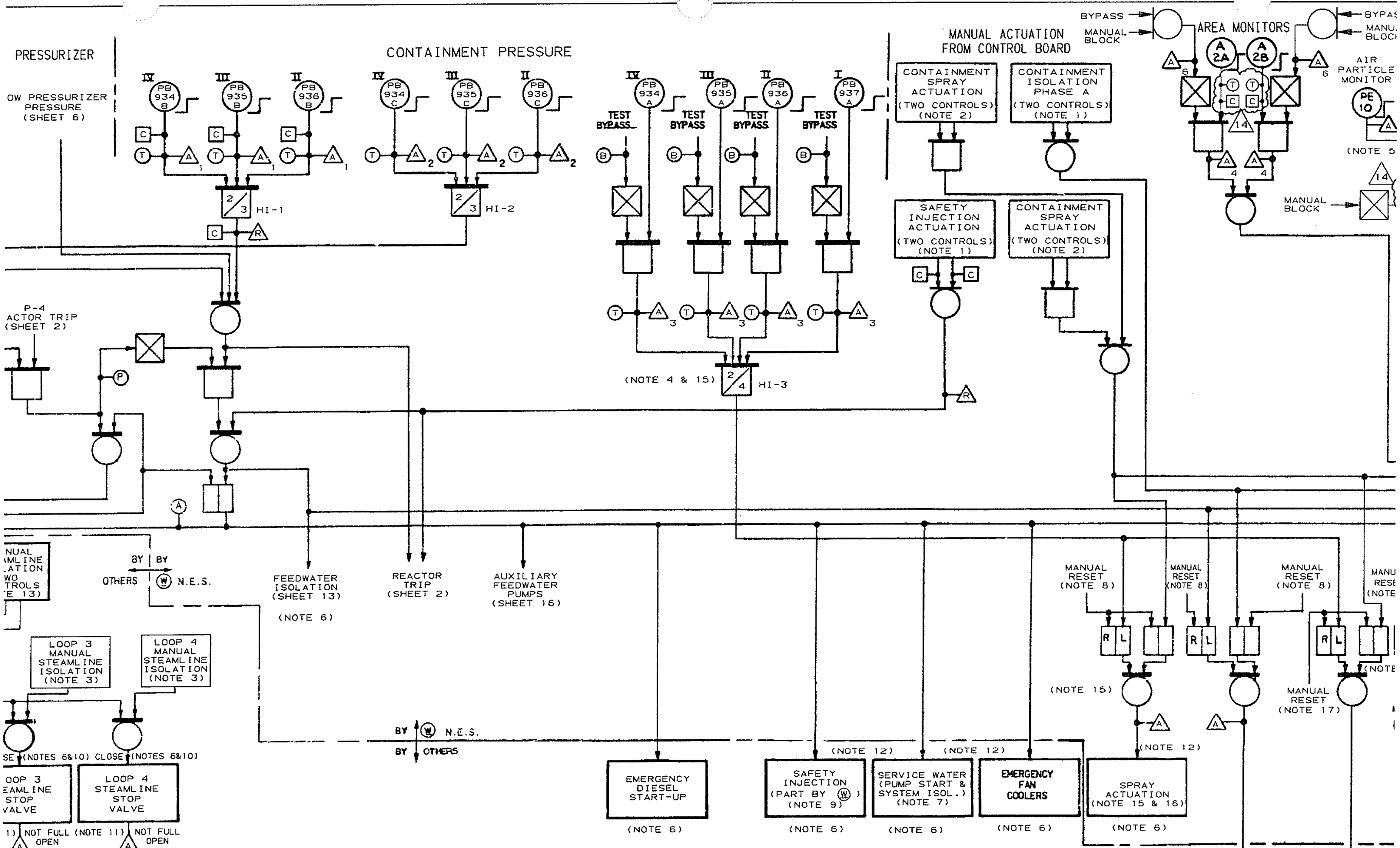
VEGP learning objectives:

LO-LP-39207-02, Given a set of Technical Specification and the Bases, determine for a specific set of plant conditions, equipment availability, and operational mode.

a. Whether any Tech Spec LCOs of section 3.3 are exceeded.

b. The required actions for any section 3.3 LCOs.

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



), OPERATING EITHER CONTROL

R MOMENTARY CONTROLS,
) CONTROLS ARE OPERATED

CTUATION ARE
E-ENERGIZE-TO-ACTUATE).

STEM, AND IS NOT SAFETY
E SAFEGUARS SYSTEM THERFORE

LATCHED), SO THAT LOSS OF

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.

12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.

13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.

14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF W NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE W SUPPLIED EQUIPMENT.

15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH

17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.

Table 3.3.2-1 (page 1 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≤ 4.4 psig	3.8 psig
d. Pressurizer Pressure - Low	1,2,3(a)	4	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≥ 1856 psig	1870 psig
e. Steam Line Pressure - Low	1,2,3(a)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≥ 570(b) psig	585(b) psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.2-1 (page 2 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure						
High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4[(i)(j)] SR 3.3.2.7[(i)(j)] SR 3.3.2.8	≤ 22.4 psig	21.5 psig

(continued)

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Safety Injection - Automatic Actuation Logic and Actuation Relays (continued)

consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure — High 1

(PI-0934, PI-0935, PI-0936)

NOTE: Containment pressure channels are also required OPERABLE by the Post Accident Monitoring Technical Specification.

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure — High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

Thus, the high pressure Function will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties. Containment Pressure — High 1 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Containment Spray - Automatic Actuation Logic and Actuation Relays (continued)

this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation handswitches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure High — 3

(PI-0934, PI-0935, PI-0936, PI-0937)

NOTE: Containment Pressure Channels are also required OPERABLE by the Post Accident Monitoring Technical Specification.

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells and electronics) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Thus, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

This Function requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

(continued)

1. 013K6.01 001/2/1/ESFAS-SENSOR/DETECT/C/A-2.7/B-VC SUMMER 2009/HL-15R AUDIT/RO/TNT / DS

Given the following plant conditions:

- 100% power.
- 1PT-936, Containment Pressure Channel II, has failed HIGH.
- 1PT-936 the proper Technical Specification actions have been taken.

Which ONE of the following indicates the number of OPERABLE channels required to **initiate** a HIGH-1 and HIGH-3 Containment Pressure actuation signal?

REFERENCE PROVIDED

	HIGH-1	HIGH-3
A.	1	1
B.✓	1	2
C.	2	1
D.	2	2

21. 015AK3.01 001/1/1/RCP - HIGH TEMP/MEM 2.5/3.1/BANK-FARLEY 2007/HL-18 NRC/RO/SRO/TNT

Given the following:

- Unit 2 is in Mode 4 with two RCPs running.

The crew has been directed by the UOP to start a third RCP.

Which ONE of the following correctly describes an RCP failure mechanism that will allow the RCP to start, the potential damage that can occur, and the reason?

A. The anti-reverse rotation device pawls are NOT engaged in the ratchet plate.

RCP radial bearing damage due to reverse flow through the RCP.

B✓ The anti-reverse rotation device pawls are NOT engaged in the ratchet plate.

RCP motor winding damage due to high starting currents.

C. The oil lift pump develops 585 psig oil lift pressure when started.

RCP motor winding damage due to high starting torque.

D. The oil lift pump develops 585 psig oil lift pressure when started.

RCP thrust bearing damage due to high temperatures.

015AK3.01 Reactor Coolant Pump (RCP) Malfunctions

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):
(CFR: 41.5 / 41.10 / 45.6 / 45.13)

Potential damage from high winding and / or bearing temperatures.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where an RCP is to be started. The candidate has to determine which condition will allow an RCP start and cause damage to the pump or motor and the type of damage.

DISTRACTOR ANALYSIS:

A. Incorrect. The ratchet and pawl devices will not prevent the pump from starting, but if not engaged, the pump will be rotating in the reverse direction when started. This will cause motor winding damage due to high starting currents.

B. Correct. The ratchet and pawl devices will not prevent the pump from starting but if not engaged the pump will be rotating in the reverse direction when started. This will cause motor winding damage due to high starting currents.

C. Incorrect. The oil lift pump pressure at 585 psig will NOT allow the pump to start. The oil lift pressure must be 600 psig to allow the RCP to start. Radial bearing damage due to high starting torque is plausible.

D. Incorrect. The oil lift pump pressure at 585 psig will NOT allow the pump to start. The oil lift pressure must be 600 psig to allow the RCP to start. Thrust bearing damage due to high temperature is plausible.

REFERENCES:

V-LO-PP-16401, Reactor Coolant Pumps
17011-1/2, window A04 for RCP 1 OIL LIFT PMP PRESS
Farley 2007 NRC RO exam question

VEGP learning objectives:

LO-PP-16401-01 Explain the function of the following RCP components:
e. Anti Rotation Device
f. Oil Lift Pump

WINDOW A04

ORIGIN	SETPOINT	
2-PSL-0491	600 psig	RCP 1 OIL LIFT PMP LO PRESS

1.0 **PROBABLE CAUSE**

1. Oil Lift Pump not running.
2. Oil Lift System malfunction.

2.0 **AUTOMATIC ACTIONS**

Prevents start of Reactor Coolant Pump 1.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

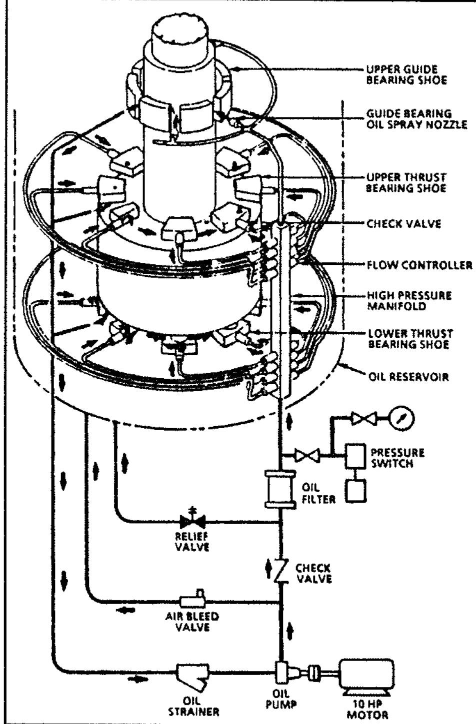
1. **Check** lights above Main Control Board Reactor Coolant Pump 1 Oil Lift Pump Control Switch 2-HS-0555 to verify:
   a. Pump running (red light illuminated),
   b. Sufficient Oil Lift System pressure (blue light illuminated).
2. IF Oil Lift Pump is running with low system pressure, **stop** the Oil Lift Pump and **investigate** for cause of alarm.
3. IF Oil Lift Pump is not running, **check** power available from Breakers 1 and 2 at 2NBE08.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 2X4DB113, 2X3D-BD-B01E, PLS



RCP Lift Oil System

RCP 1
OIL LIFT PMP
LO PRESS

23

Objective 1c,f & 7

A) Lower Motor radial bearing

•Bearing submersed in oil reservoir (20 gallon capacity)

•Oil reservoir has integral cooling coils supplied by ACCW that cools the oil (approx. 6 gpm)

B) **Upper Motor radial / thrust bearing**

•Upper radial and thrust bearing both cooled by the upper oil cooler

•Both bearings are submersed in its oil reservoir (220 gallon capacity)

•Oil circulated through upper lube oil cooler by a viscosity pump built into the outer periphery of the thrust bearing runner.

C) **Oil Lift System**

•Provides lubrication to the upper bearings during motor startup and shutdown.

•10 HP motor and piston type P.D. pump delivers oil at 600 psig discharge pressure through oil filter to :

a) Upper and lower thrust shoes which take axial thrust from rotor weight (lower) and hydraulic forces (upper)

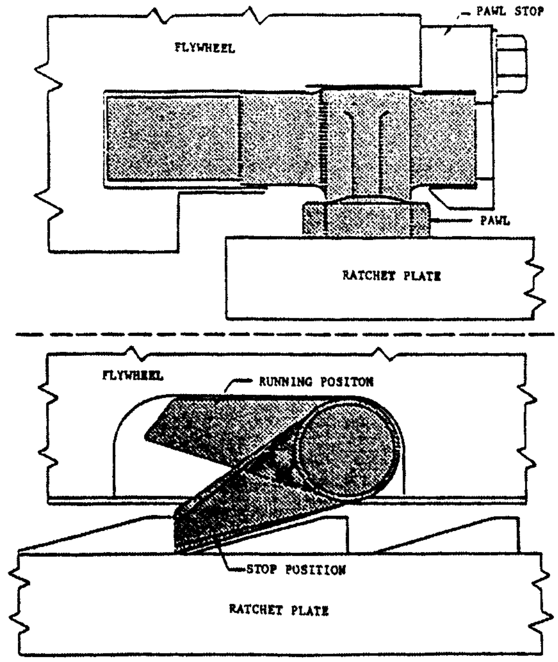
b) Upper guide bearing

•Oil lift system lifts the weight of the rotor off the thrust bearing (also pushes the shaft down off top collar when starting RCP at RCS pressure)

•A .001 inch thick film of oil between the thrust bearing runner and bearing shoes reduces friction during starting of the RCP.

•Not needed when pump is operating.

RCP oil lift pump lo pressure alarm clears at > 600 psig. Alarm normal in at power (colored green) Interlocked with Non 1E breaker (will not close if lift pressure < 600 psig)



V-LO-PP-16401

26

Objective 1e

Anti-reverse rotation device

1) Ratchet and pawl arrangement (5 per pump)

2) No lubrication required

3) Several Pawls attached to flywheel

4) Drop and bounce on ratchet plate at approximately 70 rpm

5) At zero speed, one pawl will engage the ratchet plate and prevent reverse rotation

6) On startup, Pawls kicked up by ratchet plate and held in place by centrifugal force (70 rpm)

7) Prevents reverse rotation of motor which would cause excessive starting currents.

1. 015 AK3.01 002/MODIFIED/FARLEY/HIGHER//RO/FARLEY/GO/NO

Given the following:

- Unit 2 is in Mode 4 with two RCPs running.

The crew is at a step in UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby, to start a third RCP.

Which ONE of the following correctly describes a RCP failure mechanism that will still allow the remaining RCP to be started, the damage that would occur and the reason?

A✓ • The anti-reverse rotation device pawls are not engaged in the ratchet plate.
• RCP motor winding damage due to high starting currents.

B. • The anti-reverse rotation device pawls are not engaged in the ratchet plate.
• RCP radial bearing damage due to reverse flow through the RCP.

C. • The oil lift pump does not develop 600 psig oil lift pressure.
• RCP radial bearing damage due to high starting torque.

D. • The oil lift pump does not develop 600 psig oil lift pressure.
• RCP motor winding damage due high starting torque.

22. 015K6.01 001/2/2/NI - DETECTORS/MEM - 2.9/3.2/BANK - LORQ/HL-18 NRC/RO/SRO/TNT

Given the following:

- Unit 1 is at 50% power with a power descent in progress.
- N43 power range channel fails LOW.

Per 12004-C, "Power Operation (Mode 1)," which ONE of the following is correct regarding when the TURB TRIP/RX-TRIP BLOCKED P-9 light will FIRST illuminate on the BPLB as power is lowered?

A. When any ONE of the remaining PR NIS are ≤ 40% power.

B. When any TWO of the remaining PR NIS are ≤ 40% power.

C. When any ONE of the remaining PR NIS are ≤ 38% power.

D✓ When any TWO of the remaining PR NIS are ≤ 38% power.

015K6.01 Nuclear Instrumentation System (NIS)

Knowledge of the effect of a loss or malfunction of the following will have on the NIS:
(CFR: 41.7 / 45.7)

Sensors, detectors, and indicators.

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario where a power descent is in progress with one PR NIS failed low. The candidate has to determine the correct coincidence and power level where P-9 will reset. With one channel failed low it will take two additional channels since the NIS resets are 3 of 4 coincidence.

<u>ANSWER / DISTRACTOR ANALYSIS:</u>

A. Incorrect. It takes 3 of 4 channels less than 38% power to reset P-9. This choice is plausible since the it takes 2 of 4 channels > 40% for P-9 they may think 2 of 4 < 40% is the reset value. One failed channel and one more is a plausible reset.

B. Incorrect. It takes 3 of 4 channels less than 38% power to reset P-9. This choice is plausible since as it takes 2 of 4 channels > 40% for P-9 they may think 3 of 4 < 40% is the reset value. One failed channel and two more is a plausible reset.

C. Incorrect. It takes 3 of 4 channels less than 38% power to reset P-9. This choice is

plausible since the it takes 2 of 4 channels > 40% for P-9 they may think 2 of 4 < 38% is the reset value.

D. Correct. It takes 3 of 4 channels < 38% power to reset P-9. With one channel failed, it will take two more channels < 38% to reset P-9.

REFERENCES:

12004-C, "Power Operation (Mode 1)", Step 4.2.16
PLS section for NIS
1X6AA02-00228, Logic for P-9

VEGP learning objectives:

LO-PP-17101-04 Discuss all associated Power Range Nuclear Instrument:
c. Permissives

LO-PP-28103-02 List all permissives with applicable set points, coincidences, and functions.

INITIALS

NOTES

- When below 25% power, the Condensate Filter Demineralizer Flow may be erratic. ☐
- Vessels may be removed from service based on Total Flow requirements, System ΔP and Figure 1 of 13616 "CONDENSATE DEMINERALIZER SYSTEM". ☐

4.2.14 Have operator periodically **Monitor** and **control** the Condensate Demineralizer System ΔP per 13616, "Condensate Filter Demineralizer System." ______

4.2.15 At approximately 46% Reactor Power, **verify** the following BPLP illuminates:

1.8 1 LP LO FL TRIP BLKD **P8** ______

4.2.16 At approximately 38% Reactor Power, **verify** the following BPLP illuminates:

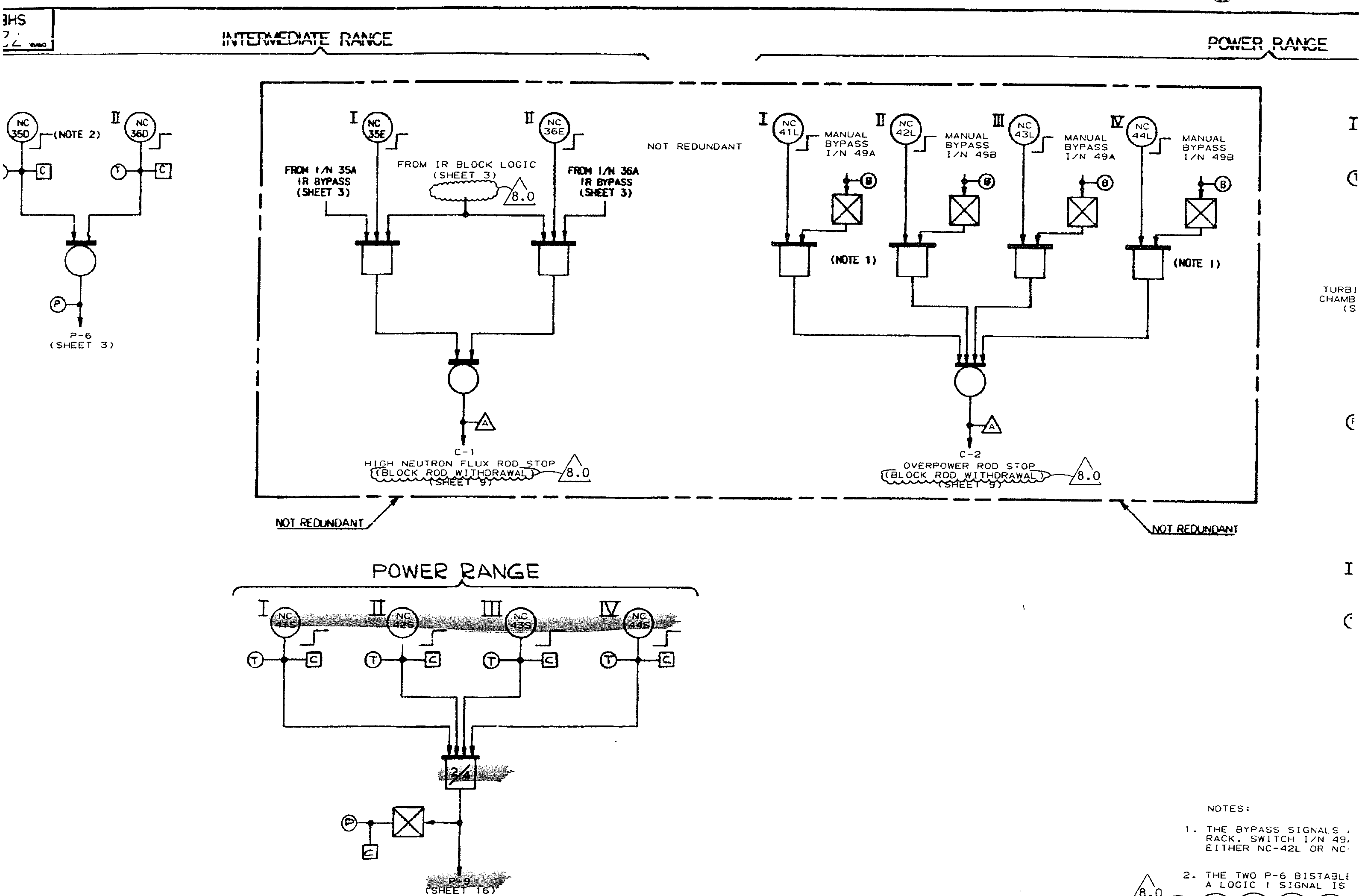
1.9 TURB TRIP/RX-TRIP BLOCKED **P-9** ______

NOTE

C20 automatically blocks AMSAC-ATWAS Mitigation System Actuation Circuitry below 40% power. ☐

4.2.17 At approximately 37% Turbine Power, **verify** C20 clears as indicated by the illuminated BPLP status light,

4.8 AMSAC BYPASSED LO TURBINE LOAD ______



NOTES:

1. THE BYPASS SIGNALS ,
RACK. SWITCH I/N 49,
EITHER NC-42L OR NC-

2. THE TWO P-6 BISTABLE
A LOGIC 1 SIGNAL IS

8.0

C. P-7 (automatically blocks various "at power" trips at low power) — 10% of full power

1. Low neutron flux (See P-10)

2. Low turbine load (See P-13)

D. P-8 (allows one loop loss of flow below setpoint)

(NC-41N, NC-42N, NC-43N, NC-44N) — 48% of full power

(The P-8 setpoint should be set 48% at all times when the 4 loop over-temperature ΔT trip setpoints are used.)

E. P-9 (automatically blocks reactor trip turbine trip below setpoint power level) — 40% of full power

(NC-41S, NC-42S, NC-43S. NC-44S)

F. P-10 (allows manual block of power range (low setpoint) trip and intermediate range trip, and C-1; blocks source range trip and provides a portion of P-7 signal)

(NC-41M, NC-42M, NC-43M. NC-44M) — 10% of full power

G. P-11 (allows manual block of safety injection actuation on low pressurizer pressure and low steamline pressure; automatically blocks steam line isolation on the low steam line pressure signal and unblocks the high steam line pressure rate signal; above P-11 the high steam line pressure rate signal is automatically blocked)

(See I.1.A.3 above)

7. The source range high flux level reactor trip must be blocked when the power ABOVE P6 light on the permissive status panel is illuminated. Blocking is done manually when P6 becomes available and can be verified by the illumination of SOURCE RANGE TRIP BLOCKED on the permissive status panel.

8. Do not reset the source range Reset-Block switches, located on the control console, above 1×10^{-4}% full power indication on either intermediate range channel, as this could cause an undesirable reactor trip.

9. Power range bistables should have loop width adjustment set to give deadband of approximately 2 percent of full power.

10. Source range bistables should have loop width adjustment set to give deadband of approximately 50 percent of the set point. (Example - if the set point of the bistable is at 1000 counts per second then reset of the bistable should occur at 500 counts per second giving a deadband of 500 cps or 50 percent of the set point).

11. Intermediate range bistables should have loop width adjustment set to give deadband of approximately 50 percent of the set point. (Example - if the bistable set point is 10^{-10} amperes then reset should occur at 5×10^{-11} amperes).

12. All bistables which alarm failure of channel power supplies shall have the loop width adjustment set to maximum deadband (full clockwise position).

AX6AA04-30

23. 016K3.12 001/2/2/NNI- S/G/MEM - 3.4/3.6/NEW/HL-18 NRC/RO/SRO/TNT

Unit 1 is at 100% power with all systems in normal alignment.

Which ONE of the following Secondary pressure transmitters failing HIGH will result in a physical change to Steam Generator(s) parameter?

A. 1PT-514 (SG #1 Channel 1 Steam Pressure)

B. 1PT-508 (Main Feed Pump Header Pressure)

C. 1PT-507 (Main Steam Header Pressure)

D✓ 1PT-3010 (Loop 2 ARV Pressure)

016K3.12 Non-Nuclear Instrumentation System (NNIS)

Knowledge of the effect that a loss or malfunction of the NNIS will have on the following:
(CFR: 41.7 / 45.7)

S/G

K/A MATCH ANALYSIS:

The question straight forward asks which failure from a list of SG pressure instruments will result in SG-2 operating parameters changing.

DISTRACTOR ANALYSIS:

A. Incorrect. With installation of Digital Feedwater System and all systems normal, a single steam pressure instrument will not result in parameters changing.

B. Incorrect. With installation of Digital Feedwater System and all systems normal, Feed Header pressure instrument will not result in parameters changing.

C. Incorrect. With installation of Digital Feedwater System and all systems normal, Steam Header pressure instrument will not result in parameters changing.

D. Correct. With PT-3010 failing HIGH will result in ARV-3010 opening venting SG-2 to atmosphere and causing SG-2 pressure to lower.

REFERENCES:

13506-C, "Digital Feedwater Control System"
1X4DB159-1, 2, and 3 Main Steam System

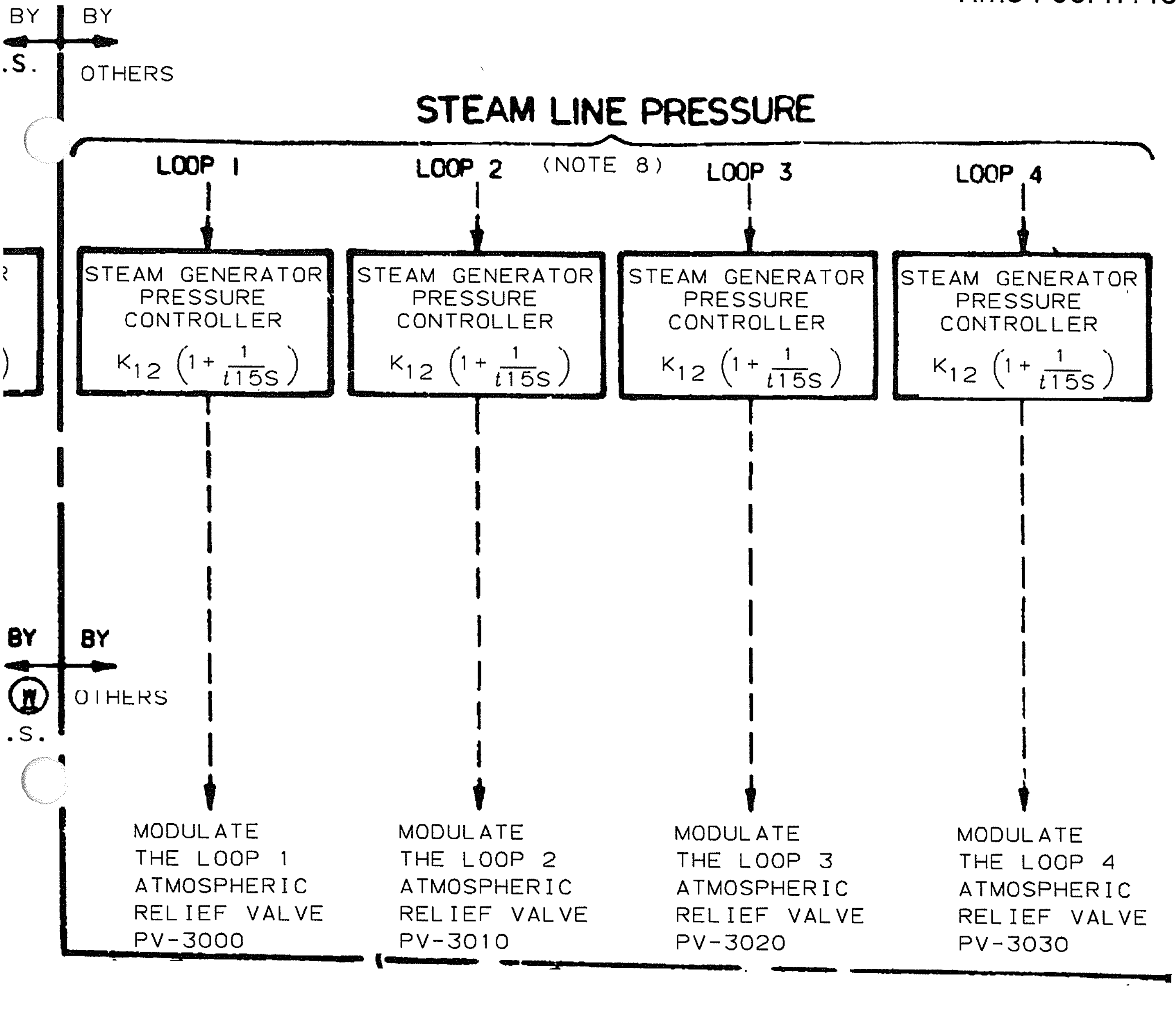
<u>VEGP learning objectives:</u>

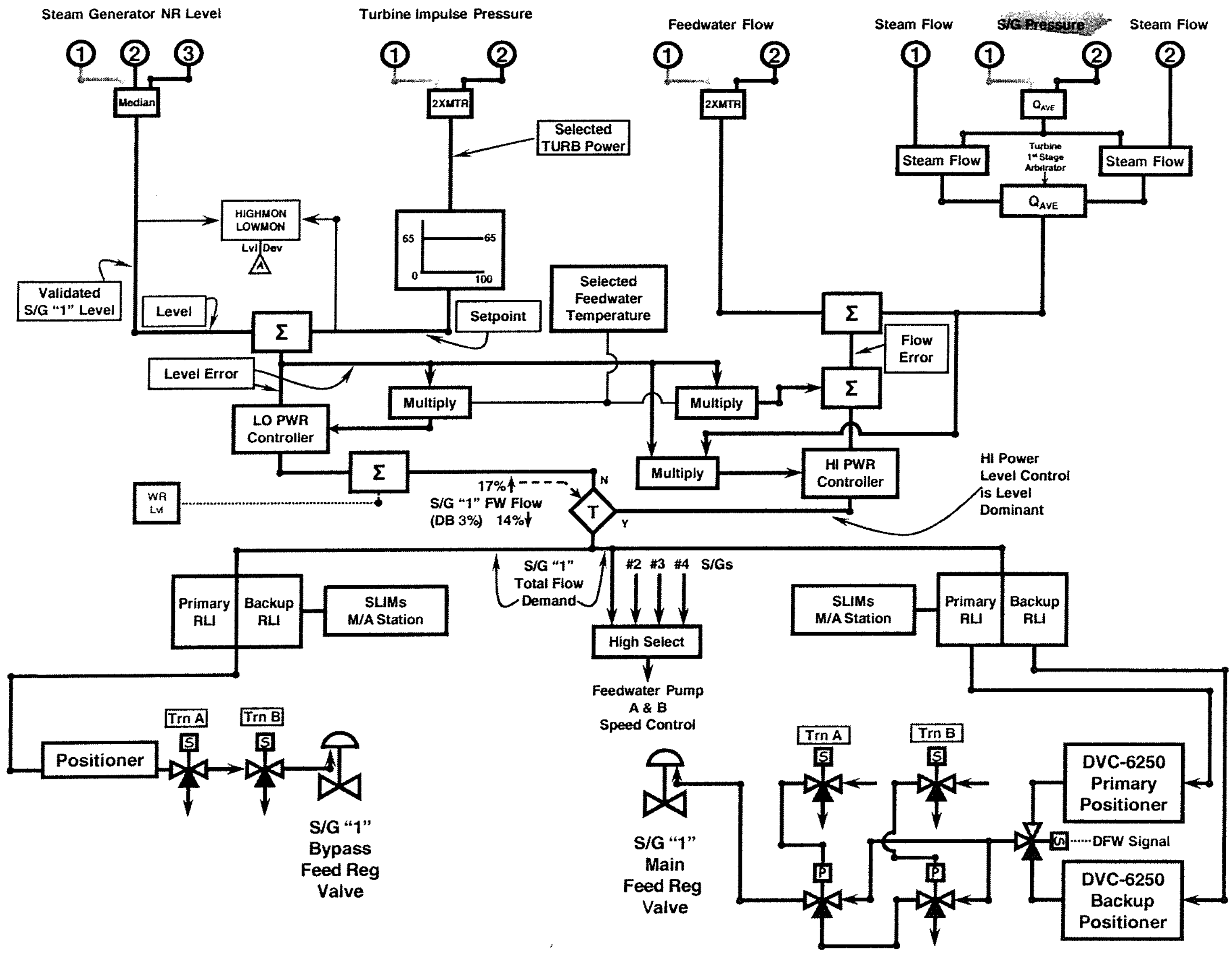
LO-PP-21101-06 Describe the operation of the Main Steamline Pressure instruments to include:

a. Where they can be read (Main Control Room, Shutdown Panels)
b. The number of channels per Main Steamline used by the Reactor Protection System
c. How they function as part of the Reactor Protection System above and below P-11.

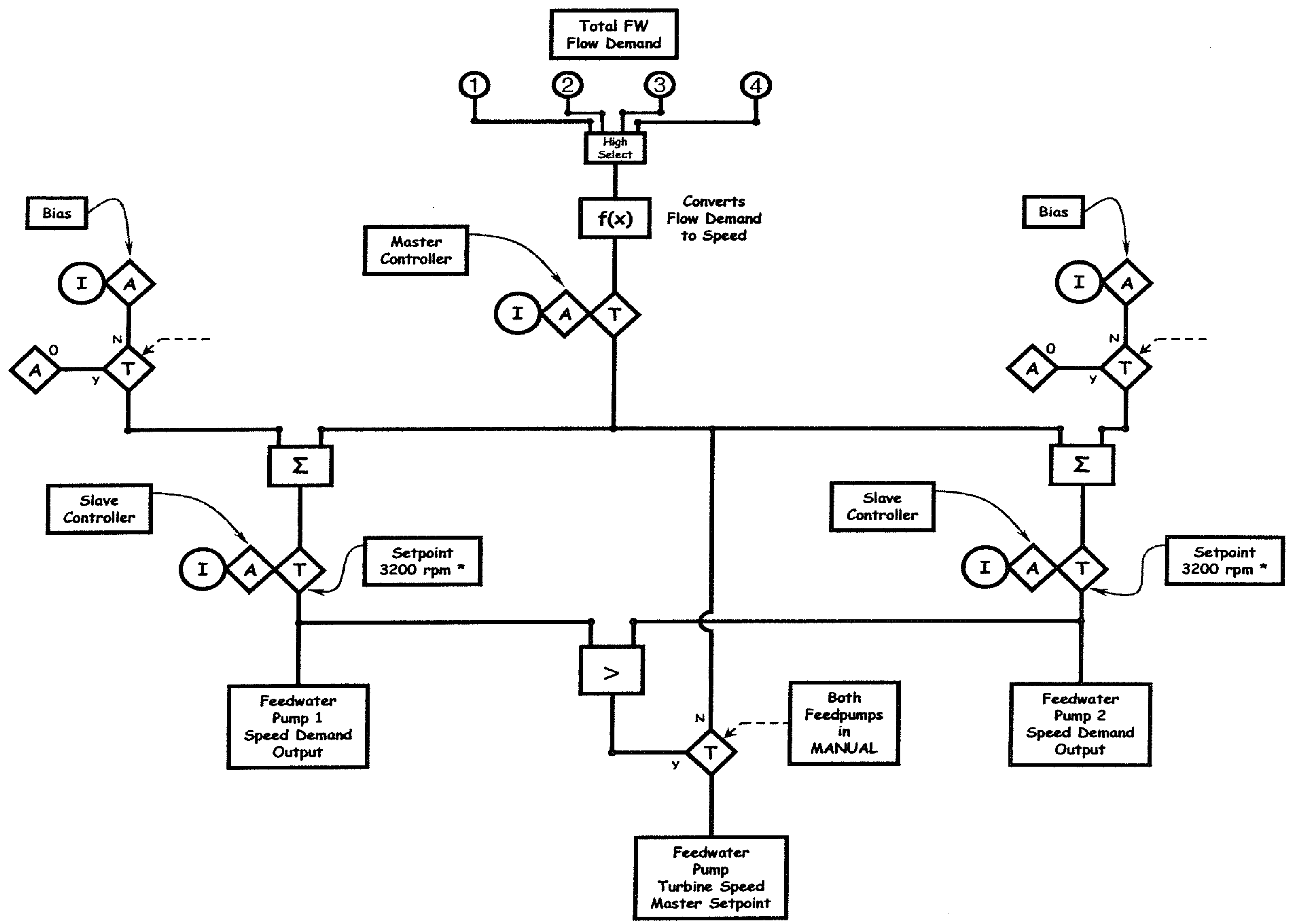
LO-PP-21101-10 Discuss the following concerning the "Atmospheric Relief valves" (ARV):

b. Basic description of how they operate (automatic and manual)
e. Impact on plant if they fail open (include how the operators can determine which valve has failed and operator response to failure)


BY
BY
.S.
OTHERS
STEAM LINE PRESSURE
LOOP 1
LOOP 2
(NOTE 8)
LOOP 3
LOOP 4
STEAM GENERATOR PRESSURE CONTROLLER
$K_{12}\left(1+\frac{1}{t15s}\right)$
STEAM GENERATOR PRESSURE CONTROLLER
$K_{12}\left(1+\frac{1}{t15s}\right)$
STEAM GENERATOR PRESSURE CONTROLLER
$K_{12}\left(1+\frac{1}{t15s}\right)$
STEAM GENERATOR PRESSURE CONTROLLER
$K_{12}\left(1+\frac{1}{t15s}\right)$
BY
BY
OTHERS
.S.
MODULATE THE LOOP 1 ATMOSPHERIC RELIEF VALVE PV-3000
MODULATE THE LOOP 2 ATMOSPHERIC RELIEF VALVE PV-3010
MODULATE THE LOOP 3 ATMOSPHERIC RELIEF VALVE PV-3020
MODULATE THE LOOP 4 ATMOSPHERIC RELIEF VALVE PV-3030



Simplified Drawing of the Digital Feedwater Control System
This is not a Design Drawing and shall not be used for Plant operations; for Information purposes only.
Figure 4



Total Steam Flow Inputs to a High Select Algorithm for Main Feed Pump Speed Control Process
This is not a Design Drawing and shall not be used for Plant operations; for Information purposes only.
Figure 5

1.1 Basic System Description

The Ovation Digital Control has the capability of automatically controlling Feedflow and SG Levels from 3% to 100% power with the addition of fault tolerance to reduce spurious trips. The system calculates Steam Generator Main Feed Reg Valve and Bypass Feed Reg Valve position and feed pump speed required to maintain Steam generator Level. It has an anticipatory circuit which provides the expected response for transient conditions to minimize adverse impact.
Any single system component or controlling input failure automatically swaps to a redundant system or input and maintains the required feedflow to keep SG levels on program. The DFWCS provides effective feedwater control over a wide range of operating conditions without operator intervention.
There are two modes of operation: Low Power and High Power.

1.2 Power Modes of Operation

The Ovation system has two modes of SG level control based on power as determined by steam flow. Low power Mode and High Power Mode.

Low Power Mode
The Low Power Mode uses a NR Level Regulator and a Flow Regulator to control SG Level. The Flow Regulator uses SGWR Level as a feed forward signal for disturbance compensation in the low power control mode because it responds quickly and in proportion to changes in total steam flow. WR SG level is generally within the normal operating range when Tavg. Is ≈ 400°F. Level control in Low Power mode is "sluggish". Both BFRV and MFRV must be in Manual on the affected loop prior to Resetting the quality for associated WR SG level. As power increases the WR Level **indication** will decrease as power increases due to density change of water in the SGs. As the WR Level signal decreases the feedwater flow demand signal will be increased. If the WR Level signal increases, the feedwater flow demand signal will be decreased. This modification of the WR level is referred to as a feed forward signal. The feed forward signal is added/subtracted downstream of the low power PID which controls are based on NR level error.

The purpose of the NR Level Regulator is to maintain the steam generator water level at its setpoint. The validated SG NR level is compensated with a lead/lag and compared to the level setpoint to generate a NR level error signal. The level error is then passed through a PI controller whose gain and integral time constant are a function of feedwater temperature to account for the shrink/swell phenomenon.

24. 022A1.03 001/2/1/CCS - HUMIDITY/C/A - 3.1/3.4/MOD - HL14 NRC/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- Unit 1 at 100% power.
- ALB01-E06 CNMT HI TEMP illuminates.

Current conditions:

- A check of Containment parameters reveals the following:

Temperature	Pressure	Relative Humidity
T2501 = 115°F	PI-0935 = 0.6 psig	MTSH-2564 = 54.5%
T2502 = 123°F	PI-0937 = 0.8 psig	MTSH-2614 = 50.8%
T2503 = 117°F	PI-0934 = 0.7 psig	MTSH-2615 = 52.7%
UT2501(AVG) = 118.3°F	PI-0936 = 0.5 psig	

- Containment Temperatures are rising very slowly.
- Containment pressures are stable.
- Containment relative humidity is stable.
- PRZR level, pressure, and RCS Tavg have remained stable.

Which one of the following completes the following statement?

Containment Temperature is currently ___(1)___ Tech Spec LCO limits

and

the crew is required to ___(2)___.

24. 022A1.03 001/2/1/CCS - HUMIDITY/C/A - 3.1/3.4/MOD - HL14 NRC/HL-18 NRC/RO/SRO/TNT

A. (1) within

(2) enter 18008-C, "Secondary Leakage", due to symptoms of a steam leak

B✓ (1) within

(2) start an additional pair of Containment Cooling Units to maintain temperature under the Tech Spec limit

C. (1) exceeding

(2) enter 18008-C, "Secondary Leakage", due to symptoms of a feedwater leak

D. (1) exceeding

(2) start an additional pair of Containment Cooling Units to return temperature within Tech Spec limits

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:
(CFR: 41.5 / 45.5)

Containment humidity.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where Containment Temperature is high due to inadequate coolers running. The candidate has to determine if the Containment Temperature LCO is exceeded or within limits and any appropriate corrective actions.

DISTRACTOR ANALYSIS:

A. Incorrect. The Containment Temperature is within LCO limits. With the other Containment parameters not trending up, there is no reason to suspect a steam leak so entering 18008-C is not an appropriate action. It is plausible with all the parameters present and temperature rising, the candidate may suspect a steam leak.

B. Correct. The Containment Temperature is within LCO limits and per the ARP 17001 window F05, starting either an additional pair of Containment Coolers or an additional Auxiliary Cooler is the action to take.

C. Incorrect. The Containment Temperature is within LCO limits but it is plausible the

candidate may think with one channel above 120ºF and the alarm in, the Containment Temperature limit is exceeded. The Tech Spec clearly states Containment Average Air Temperature as the measuring point for whether or not the LCO is exceeded. It is plausible the candidate may think with the RCS parameters stable and Containment Temperature rising a feedline leak exists. However, with the other Containment parameters stable, this will rule that out.

D. Incorrect. The Containment Temperature is within LCO limits but it is plausible the candidate may think with one channel above 120ºF and the alarm in, the Containment Temperature limit is exceeded. The Tech Spec clearly states Containment Average Air Temperature as the measuring point for whether or not the LCO is exceeded. The 2nd part for starting an additional pair of Containment Coolers in HIGH speed is a correct action per the ARP.

REFERENCES:

17001-1, E06 CNMT HI TEMP
17001-1, F06 CNMT HI MSTR
Tech Spec LCO 3.6.5 for Containment Average Air Temperature
Tech Spec LCO 3.6.4 for Containment Pressure

VEGP learning objectives:

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

a. Whether any Tech Spec LCOs of Section 3.6 are exceeded.
b. The required actions for all Section 3.6 LCOs.

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. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.3 psig and ≤ +1.8 psig.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	In accordance with the Surveillance Frequency Control Program

WINDOW E06

ORIGIN	SETPOINT	
1-TSH-2563 1-TSH-2612 1-TSH-2613	120°F	CNMT HI TEMP

1.0 **PROBABLE CAUSE**

Insufficient number of Containment Building Cooling Units operating.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Start** an additional pair of Containment Cooling Units or a Containment Auxiliary Cooling Unit per 13120-1, "Containment Building Cooling Systems".

2. **Verify** Nuclear Service Cooling Water flow to coolers, and IF necessary, **dispatch** an operator to inspect the Containment Heat Removal System.

3. **Refer to** Technical Specification LCO 3.6.5 and 3.6.6.

4. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB212, CX5DT101-66, CX5DT101-71

WINDOW F06

ORIGIN	SETPOINT	
1-MTSH-2564 1-MTSH-2614 1-MTSH-2615	85% RH	CNMT HI MSTR

1.0 **PROBABLE CAUSE**

Reactor Coolant System, Steam System, Feedwater System, or other moisture source leaking in containment.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Check** Radiation Monitors for possible increase indicating Reactor Coolant System leakage.

2. **Monitor** Containment pressure and temperature for trends indicating system leakage.

3. **Check** other Reactor Coolant, Steam, and Feedwater System parameters for indications of source of leakage.

4. Refer to the appropriate AOP:

   a. 18004-C, "Reactor Coolant System Leakage",

   b. 18008-C, "Secondary Coolant Leakage".

5. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF PROCEDURE TEXT

REFERENCES: 1X4DB212, CX5DT101-37C

25. 022AA2.01 001/1/1/LOSS RC M/U - CHG LK/C/A - 3.2/3.8/MOD-HL16/HL-18 NRC/RO/SRO/AML

With Unit 1 at 100% power, the OATC observes the following:

- ALB07-A05 REGEN HX LTDN HI TEMP is in alarm.
- ALB07-B06 CHARGING LINE HI/LO FLOW is in alarm.
- RCP seal injection flow rates are rising.

Which one of the following completes the following statement?

Using the above indications, the correct diagnosis is that ____________.

A. a charging line break has occurred

B. a seal injection line break has occurred

C.✓ Seal Flow Control Valve, HV-182, has failed closed

D. Charging Flow Control Valve, FV-121, has failed closed

022AA2.01 Loss of Rx Coolant Makeup

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: (CFR: 43.5 / 45.13)

Whether charging line leak exists.

K/A MATCH ANALYSIS:

The question presents a plausible scenario with various CVCS annunciators and conditions stated. The candidate must determine from a list of choices the cause of the listed parameters and determine whether or not a charging line leak exists.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. With the stated conditions, HV-182 Seal Flow Control Valve has closed. The presented alarms individually are symptoms of a charging line break. The candidate must combine all indications to rule out a charging line break. A charging line break will not cause seal injection flow rates to rise with the 2 annunciators.

B. Incorrect. A seal injection line break will not cause all 4 seal injection flow rates to rise due to the location of the seal injection flow instruments. A seal injection line break downstream of the flow instruments would cause flow to be starved to the other 3 seal injection lines and their flow will actually lower. This requires the candidate to have good knowledge of the location of the seal flow instruments.

C. Correct. With the parameters presented, HV-182 failing closed is correct as flow is forced through the seal injection lines and starved from the Regen Hx. Charging line lo flow will also be received if HV-182 is failed closed.

D. Incorrect. A charging line break will starve flow to the seal injection line causing seal flows to lower versus rise.

REFERENCES:

1X4DB116-1, CVCS
1X4DB114, CVCS Letdown
17007-1/2, windows A05, B06

VEGP learning objectives:

LO-PP-09200-09, State how seal injection flow is controlled.

LO-PP-09200-10, State how changes in charging flow affects seal injection flow and the corrective actions to restore flow to normal range.

WINDOW A05

ORIGIN	SETPOINT	
1-TE-0127	400°F	REGEN HX LTDN HI TEMP

1.0 **PROBABLE CAUSE**

1. Low charging flow.
2. High letdown flow.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** Regenerative Heat Exchanger letdown outlet temperature using 1-TI-0127 on the QMCB.
2. IF a malfunction of the charging system is determined, **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Monitor** letdown and charging flows.
2. IF operating at a minimum charging flow, **adjust** charging or letdown flows, as necessary, to lower the temperature below 380°F.
3. **Verify** compliance with 13006-1, "CVCS Startup And Normal Operation."
4. **Return** the system to normal operation as soon as possible.
5. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DBI14, PLS

WINDOW B06

ORIGIN	SETPOINT	CHARGING LINE HI/LO FLOW
1-FT-0121	Hi: 150 gpm Low: 25 gpm	

1.0 **PROBABLE CAUSE**

1. High flow:
   a. I-HV-182 failed open,
   b. Charging line rupture,
   c. Pressurizer Level Control malfunction.
   d. 1-FIC-0121 malfunction.
2. Low flow:
   a. Operating Charging Pump tripped,
   b. Charging Flow Valve I-HV-182 malfunction,
   c. Pressurizer Level Control malfunction.
   d. 1-FIC-0121 malfunction.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** total charging flow using 1-FI-0121A on the QMCB.
2. IF charging is lost, **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

WINDOW B06
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF charging flow was NOT lost, **place** 1-FIC-0121 in manual and control charging flow as required.

2. **Adjust** RCP seal flow using 1-HC-182, to maintain seal flow of 8 to 13 gpm per pump.

2. IF equipment failure is indicated, **initiate** maintenance as required.

3. IF an operating charging pump fails due to suspected gas binding (fluctuating discharge pressure and flow), THEN the standby pump shall not be started until the cause of the gas binding is understood and all effected piping and components have been vented.

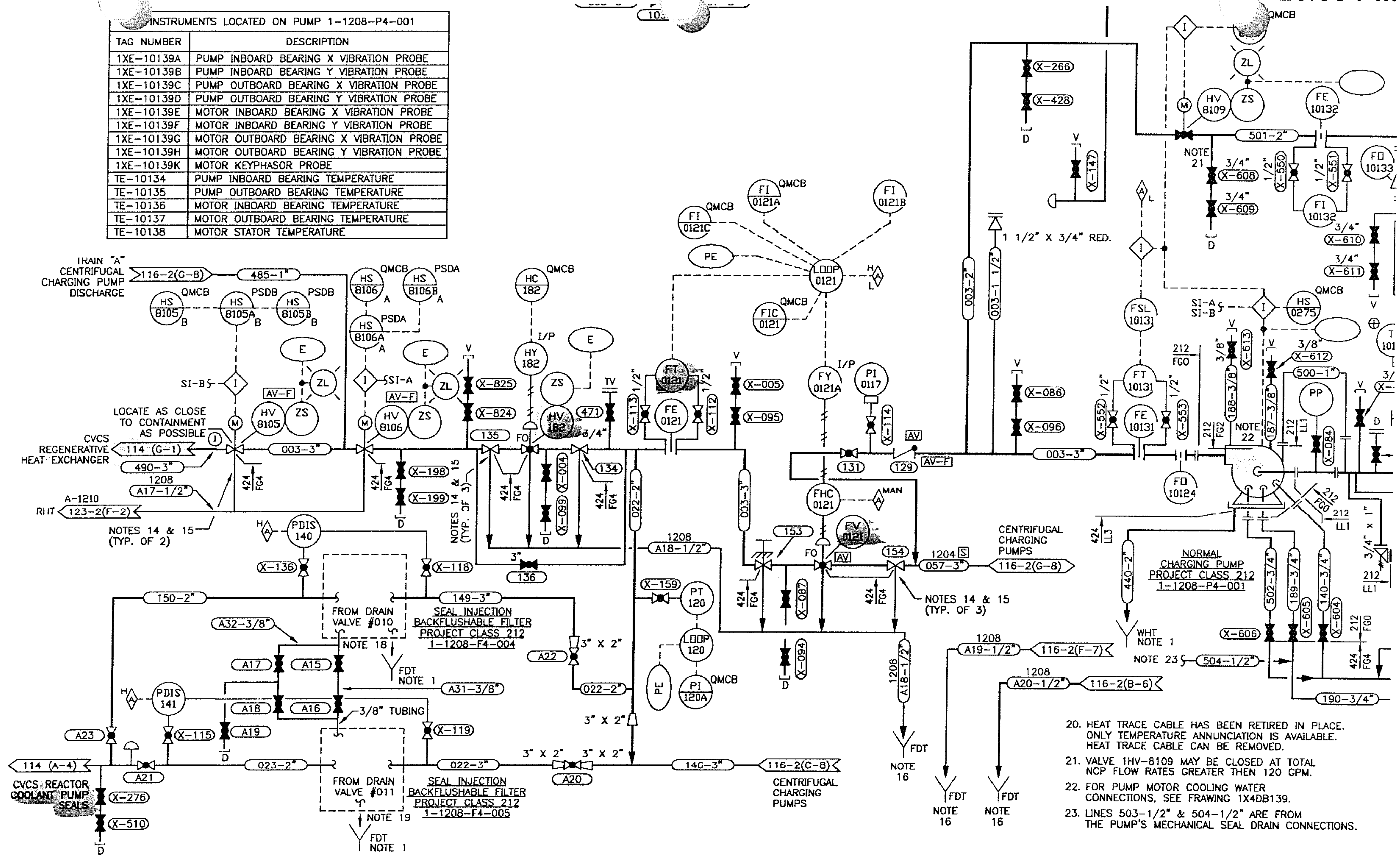
5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

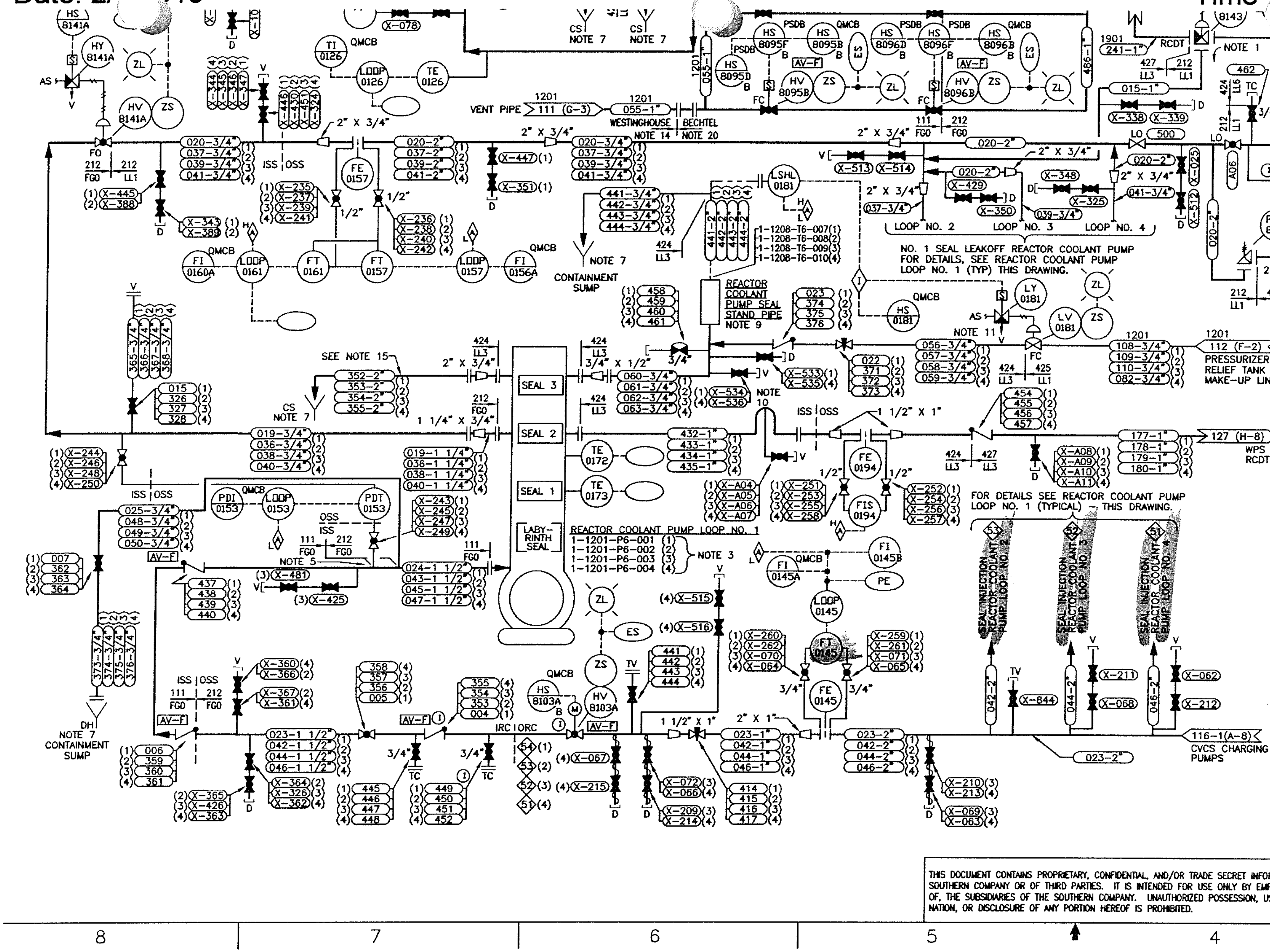
REFERENCES: 1X4DB116-1, PLS

INSTRUMENTS LOCATED ON PUMP 1-1208-P4-001	
TAG NUMBER	DESCRIPTION
1XE-10139A	PUMP INBOARD BEARING X VIBRATION PROBE
1XE-10139B	PUMP INBOARD BEARING Y VIBRATION PROBE
1XE-10139C	PUMP OUTBOARD BEARING X VIBRATION PROBE
1XE-10139D	PUMP OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139E	MOTOR INBOARD BEARING X VIBRATION PROBE
1XE-10139F	MOTOR INBOARD BEARING Y VIBRATION PROBE
1XE-10139G	MOTOR OUTBOARD BEARING X VIBRATION PROBE
1XE-10139H	MOTOR OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139K	MOTOR KEYPHASOR PROBE
TE-10134	PUMP INBOARD BEARING TEMPERATURE
TE-10135	PUMP OUTBOARD BEARING TEMPERATURE
TE-10136	MOTOR INBOARD BEARING TEMPERATURE
TE-10137	MOTOR OUTBOARD BEARING TEMPERATURE
TE-10138	MOTOR STATOR TEMPERATURE

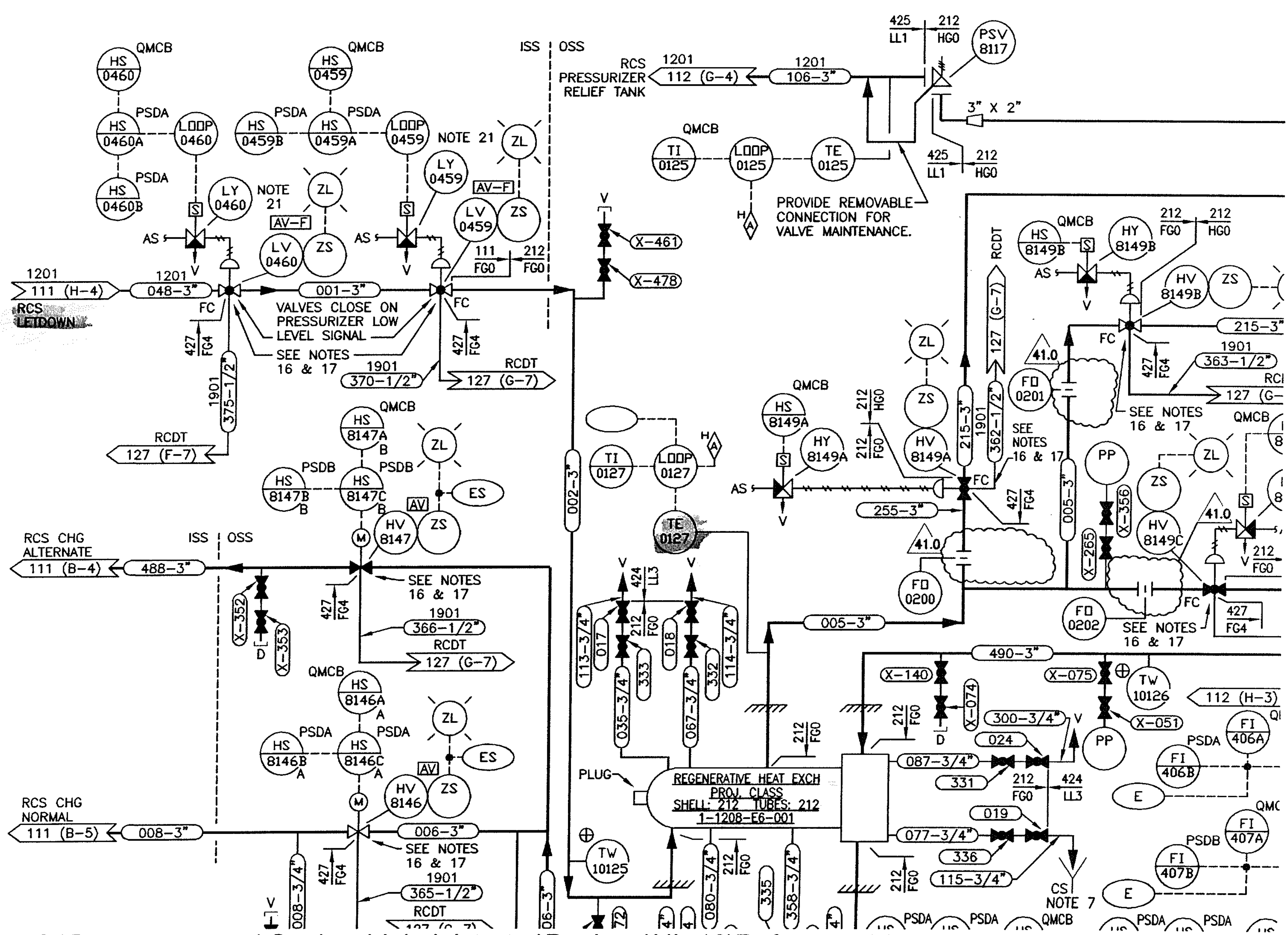


20. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.
21. VALVE 1HV-8109 MAY BE CLOSED AT TOTAL NCP FLOW RATES GREATER THEN 120 GPM.
22. FOR PUMP MOTOR COOLING WATER CONNECTIONS, SEE FRAWING 1X4DB139.
23. LINES 503-1/2" & 504-1/2" ARE FROM THE PUMP'S MECHANICAL SEAL DRAIN CONNECTIONS.

8 7 6 5 4



INSTRUMENT
SEAL WATER FLOW
FLOW ELEMENT
FLOW TRANSMITTER
FLOW INDICATOR (CONTROL
FLOW INDICATOR (LOCAL)
LOWER BEARING WATER & NO.
WATER INLET TEMPERATURE
TEMPERATURE ELEMENT
LOWER BEARING WATER & NO.
WATER INLET TEMPERATURE
TEMPERATURE ELEMENT
NO. 1 SEAL ΔP
DIFFERENTIAL PRESS TRANSM
DIFFERENTIAL PRESS INDICAT
CONTROLLED LEAKAGE LO RAN
FLOW ELEMENT
FLOW TRANSMITTER
FLOW INDICATOR (DUAL MET
CONTROLLED LEAKAGE HI RANG
FLOW TRANSMITTER
FLOW INDICATOR (DUAL MET
RCP SEAL STAND PIPE LEVEL
LEVEL SWITCH (SUPPLY)
LEVEL VALVE
SOLENOID VALVE
HAND SWITCH
NO. 2 SEAL LEAKOFF
LEAKOFF FLOW ELEMENT
LEAKOFF FLOW SWITCH
NO. 1 RCP SEAL WATER INLET
NO. 1 SEAL WATER LEAKOFF


RCS LETDOWN
1201
111 (H-4)
048-3"
001-3"
VALVES CLOSE ON PRESSURIZER LOW LEVEL SIGNAL
SEE NOTES 16 & 17
1901
370-1/2"
375-1/2"
RCDT
127 (F-7)
127 (G-7)
NOTE 21
ISS
OSS
RCS PRESSURIZER RELIEF TANK
112 (G-4)
106-3"
PSV 8117
3" X 2"
PROVIDE REMOVABLE CONNECTION FOR VALVE MAINTENANCE.
X-461
X-478
002-3"
RCS CHG ALTERNATE
111 (B-4)
488-3"
RCS CHG NORMAL
111 (B-5)
008-3"
006-3"
366-1/2"
365-1/2"
005-3"
215-3"
362-1/2"
363-1/2"
255-3"
490-3"
REGENERATIVE HEAT EXCH
PROJ. CLASS
SHELL: 212 TUBES: 212
1-1208-E6-001
PLUG
087-3/4"
077-3/4"
300-3/4"
115-3/4"
035-3/4"
067-3/4"
113-3/4"
114-3/4"
080-3/4"
358-3/4"
112 (H-3)
CS NOTE 7

26. 025AG2.4.21 001/1/1/LOSS RHR - EP/C/A - 4.0/4.6/NEW/HL-18 NRC/RO/SRO/TNT

While performing RCS drain down to mid-loop conditions, the following parameters are observed by the OATC:

- Pressurizer level is at 0%.
- RCS temperature is 97°F and stable.
- RHR flow is fluctuating.
- RHR pump amps are fluctuating.
- Core Cooling CSFST is indicating YELLOW.
- RCS vessel level is 189 feet.

Per 18019-C, "Loss of Residual Heat Removal," which ONE of the following is the correct action for the operating crew to take?

A. Stop the RCS drain down and wait for the RCS level to stabilize and the Core Cooling CSFST to indicate GREEN.

B. Ensure local RCS sightglass reading is within 10% of the control room indications, then continue drain down.

C. Continue the RCS drain down, but at a reduced rate, since indicated level is still above 188 feet.

D✓ Stop the RCS drain down and reduce flow through the operating RHR pump to stop the cavitation.

025AG2.4.21 Loss of RHR System

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
(CFR: 41.7 / 43.5 / 45.12)

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario with indications from various plant parameters that indicate RHR pump cavitation while at midloop. The candidate must choose the correct action per guidance of 18019-C, "Loss of RHR". The core cooling CSFST yellow indicates there is a problem with RHR cooling from RHR inputs.

DISTRACTOR ANALYSIS:

A. Incorrect. Stopping the drain down is a correct action, however, 18019-C, does not direct the stabilization of level while waiting on core cooling to turn green.

B. Incorrect. This is a plausible action from some of the midloop procedures while performing an RCS draindown, however, this action is not directed per 18019-C. The draindown will not be continued with RHR pump cavitation indications present.

C. Incorrect. With RHR pump cavitation indications present, the draindown will not be continued even at a reduced rate.

D. Correct. This is the correct actions as specified in 18019-C.

REFERENCES:

18019-C, Loss of Residual Heat Removal

ARG-1, Background for Loss of RHR

VEGP learning objectives:

LO-LP-37002-06 State the layout of the Critical Safety Function status trees (CSFSTs)

LO-LP-37002-07 State how the Critical Safety Functions are prioritized. Be able to demonstrate how the highest priority CSF is selected and addressed.

LO-LP-37002-08 State the rules of usage for functional restoration procedures in terms of:
a. when they must be used
b. when they may be terminated
c. transition criteria for higher priority CSF challenge.

LO-LP-37002-09 Using EOP 19200, as a guide, briefly describe how the steps are accomplished.

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

System status information on loss of RHR:
- Figure 1 - Heatup Rate
- Figure 2 - Core Flow to Maintain 195 Degrees F
- Figure 3 & 4 - Time to Boiling
- Figure 5 - Time For Core Uncovery

CAUTION

Changes in RCS pressure could result in inaccuracies in RCS level readings.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B1. Check if RHR Pumps should be stopped:	
__a. RHR Pumps - ANY RUNNING. yes	a. Perform the following: __1) IF RCS level is less than 188 feet, THEN Go to Step B2. __2) IF RCS is greater than 188 feet, THEN start an available RHR pump by initiating 13011, RESIDUAL HEAT REMOVAL SYSTEM. __3) Go to Step B1.c.
__b. RCS level - GREATER THAN 188 FEET. yes	__b. Stop RHR pumps and Go to Step B2.

Step 1 continued on next page

B. LOSS OF RHR - MODE 5 OR 6 BELOW PRZR IR OR SG NOZZLE DAMS INSTALLED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__c. RHR Pumps - NOT CAVITATING. —> NO —>	c. Perform the following: 1) Throttle RHR Heat Exchanger Bypass valve to reduce flow through affected pump to stop cavitation: __ Train A: FV-0618 __ Train B: FV-0619 __2) IF flow must be reduced to less than 1500 gpm to stop cavitation, THEN stop RHR Pump and Go to Step B2.
__d. Verify at least one RHR train in service with flow - ADEQUATE TO MAINTAIN RCS TEMPERATURE.	__d. Go to Step B2.
__e. Check RCS level - STABLE OR RISING.	__e. Go to Step B2.
__f. Return to procedure and step in effect.	
__B2. Initiate the Continuous Actions Page.	

STEP: Check If RHR Pumps Should Be Stopped

PURPOSE: To stop the RHR pumps if RCS level is not adequate to support pump operation or if the pumps are cavitating.

BASIS:

If air is ingested into the suction of the RHR pumps, the pumps are likely to start cavitating. The symptoms of cavitation are typically erratic behavior of pump current (oscillations between high amperage values and low amperage values), erratic behavior of pump flow and excessive pump noise. Pump suction pressure (typically a local measurement) will also oscillate. If any plant personnel are in the immediate vicinity of the pumps, loud audible noises may be heard when air ingestion starts. If no actions are taken, damage to the pumps may occur due to vibration and the pumps may eventually seize and stop running. As soon as the operator is aware that air ingestion or cavitation is occurring, actions should be initiated to protect the RHR pumps. If the air ingestion started during an RHR flow increase evolution, then a reduction of flow to below the pre-air ingestion value may stop the air ingestion and the pump cavitation. Inadequate level in the RCS hot leg will also result in pump cavitation and air ingestion. If level is greater than (1), the operator should reduce flow to (2) which is an acceptable level vs. flow combination. If the RHR pumps continue to cavitate or RCS level is not adequate to allow continued pump operation, then the operator should stop the RHR pumps until the proper operating conditions are reestablished.

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

- o (1) Enter plant-specific RCS level corresponding to the lowest allowable RHR flow from RCS level vs. flow curves including allowances for normal channel accuracies.
- o (2) Enter plant-specific lowest allowable RHR flow for mid-loop operations including allowances for normal channel accuracies.

27. 026A4.05 001/2/1/CSS - RESET SWITCH/MEM 3.5/3.5/NEW/HL-18 NRC/RO/SRO/AML

Initial conditions:

- DBA LOCA has occurred.

- Containment Spray has actuated.

Current conditions:

- Containment pressure is 23.8 psig.

Which one of the following completes the following statement?

Based on the current conditions, if the OATC takes the Containment Spray reset switches to the "Reset" position, the Containment Spray actuation signal __(1)__ reset

and

per 19010-C, "Loss of Reactor or Secondary Coolant," the Containment Spray pumps must operate in the recirculation mode for a MINIMUM of _(2)_ hours and containment pressure must be less than 15 psig before they can be stopped.

	_(1)_	_(2)_
A✓	will	1.5
B.	will	2.0
C.	will NOT	1.5
D.	will NOT	2.0

026A4.05 Containment Spray System (CSS)

Ability to manually operate and / or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

Containment spray reset switches.

K/A MATCH ANALYSIS:

This question meets the KA by questioning the students knowledge of the containment spray reset logic and when spray can be procedurally stopped.

DISTRACTOR ANALYSIS:

A. Correct.

B. Incorrect - 19010 states they should be run for 2 hours and 1.5 hours in recirc mode.

C. Incorrect - Plausible if the students believe that an actuation signal that is present will prevent you from resetting the system.

D. Incorrect - 19010 states they should be run for 2 hours and 1.5 hours in recirc mode.

REFERENCES:

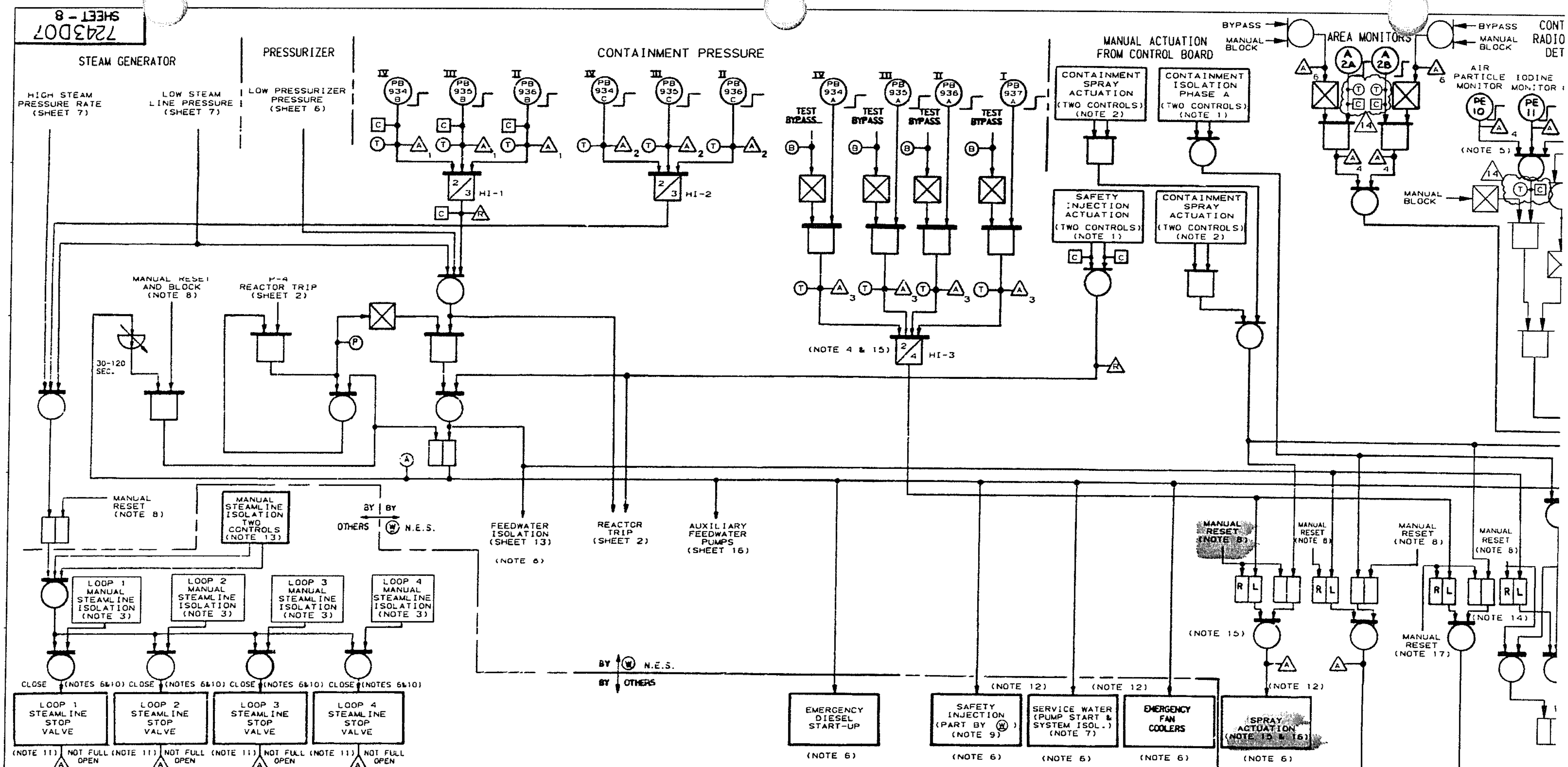
19010-C, "Loss of Reactor or Secondary Coolant", Step 12d
1X6AA02-00232, Spray Actuation ESFAS Logic

VEGP learning objectives:

LO-LP-15101-06 State the reason for a minimum required time the Containment Spray system is left on recirculation following a LOCA.

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

LO-LP-28103-06 Determine when ESF actuation signal can be reset and describe actions to reset the signal.



NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, 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 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. CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM, AND IS NOT SAFETY RELATED, HOWEVER, CIRCUITRY IS INPUT TO THE SAFEGUARS SYSTEM THERFORE ISOLATION MUST BE PROVIDED.
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. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
8. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
9. SAFETY INJECTION SEQUENCE REQUIREMENTS (IF SEQUENCING IS NECESSARY) ARE SPECIFIED BY Ⓦ NUCLEAR ENERGY SYSTEMS.
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.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF Ⓦ NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE Ⓦ SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE SEQUENCE TO LOAD THE CONTAINMENT SPRAY SYSTEM ON THE EMERGENCY DIESEL POWER SHOULD BE INDEPENDENT OF THE SEQUENCE TO LOAD THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY THE SEQUENCE INTERLOCK SHOULD BE SUCH THAT SPRAY WILL START WITHIN THE REQUIRED TIME INDEPENDENT OF THE SAFETY INJECTION SIGNAL STATUS.
17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.

14	INCORPORATED PER OCP 97-VIN0067	11-16-00	ELC	TSL	GLB
NO.	REVISIONS	DATE	DR	CHK	APPV

SCS REVISIONS

DOCUMENT STATUS CODE 1

1X6AA02-00232-17

WESTINGHOUSE PROPRIETARY DATA

THIS DOCUMENT CONTAINS PROPRIETARY INFORMATION OF THE WESTINGHOUSE ELECTRIC CORPORATION WATER REACTOR DIVISIONS. IT IS TRANSMITTED TO YOU IN CONFIDENCE AND TRUST, AND IS TO BE RETURNED UPON REQUEST. ITS CONTENTS MAY NOT BE DISCLOSED IN WHOLE OR IN PART TO OTHERS OR USED FOR OTHER THAN THE PURPOSE FOR WHICH TRANSMITTED WITHOUT THE PRIOR WRITTEN PERMISSION OF THE WESTINGHOUSE WATER REACTOR DIVISIONS.

DFTM. R HEARN 5/10/73
CHKD.
DES. ENG. [signature] 7/14/73
APPD. ENG. CV
APP. [signature] 7-72-73
APP.
APP.
[signature] 7-13-73

Westinghouse NUCLEAR ENERGY
TITLE: GEOR... ALV... FUNC... SAFE...
SCALE
DO NOT SCALE

11 10 9 8 7 6 5 4 3 2

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***12. Check if Containment Spray should be stopped:**	
__a. CS Pumps - RUNNING.	__a. Go to Step 13.
b. Containment pressure - LESS THAN 15 PSIG.	__b. <u>WHEN</u> Containment pressure is less than 15 psig, <u>THEN</u> go to Step 12.c. __Go to Step 13.
c. Any Containment radiation levels - INDICATE HIGH DUE TO PRIMARY LOCA: __ RE-002 __ RE-003 __ RE-005 __ RE-006	c. Perform the following: __1) Reset Containment Spray signal. __2) Stop Containment Spray Pumps. 3) Close CNMT SPRAY ISO VLV: __• HV-9001A __• HV-9001B __Go to Step 13.
d. Operate CS Pumps: __• Minimum of 2 hours. __• At least 1.5 hours in recirculation mode.	__d. <u>WHEN</u> CS Pumps have operated for at least 2 hours <u>AND</u> in the recirculation mode for at least 1.5 hours, <u>THEN</u> perform Step 12.c RNO.

28. 026AA2.02 001/1/1/LOSS CCW - CAUSE/C/A - 2.9/3.6/BANK-HARRIS07/HL-18 NRC/RO/SRO/AML

Given the following:

- The unit is at 95% power.
- ACCW surge tank level is cycling between makeup start and stop levels.
- ACCW Surge Tank level lowers slowly each time the ACCW Surge Tank makeup valve shuts.

Which one of the following completes the following statement?

The cause of the surge tank level cycling is a leak in the _____ Heat Exchanger.

A. RCP Thermal Barrier

B✓ Seal Return

C. Letdown

D. ACCW

026AA2.02 Loss of Component Cooling Water (CCW)

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water (CCW):
(CFR: 43.5 / 45.13)

The cause of possible CCW loss.

K/A MATCH ANALYSIS:

Question meets the KA by testing students understanding of various system pressures and using that information to determine where a leak may be coming from.

DISTRACTOR ANALYSIS:

A. Incorrect - ACCW surge tank level would be rising if the leak were in the thermal barrier heat exchanger

B. Correct - Seal return HX is at a lower pressure than ACCW, and a leak will go to the VCT.

C. Incorrect - Letdown HX would result in the ACCW surge tank level rising.

D. Incorrect - An ACCW Hx. leak will result in rising ACCW surge tank level as NSCW is designed to be at a higher pressure leaking into ACCW versus to the environment.

REFERENCES:

11874-1, Control Room Rounds Sheets
V-LO-TX-16001, Reactor Coolant System, page # 28
V-LO-PP-04101, Auxiliary Component Cooling Water, Figure 04-1
IPC photos of VCT and ACCW Surge Tank

VEGP learning objectives:

LO-PP-04101-03 Describe how ACCW surge tank level and RE-1950 are used to determine source of in-leakage and when the in-leakage is isolated.

FIGURE 1 - UNIT 1
DATA SHEET

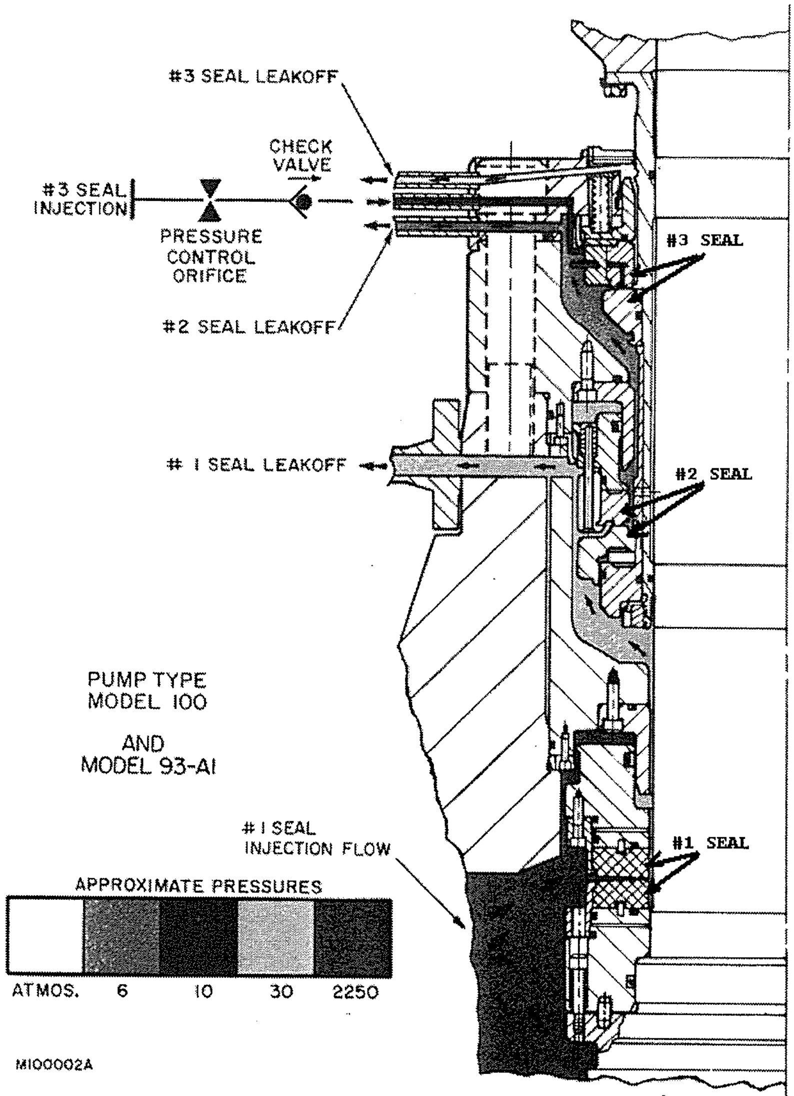
Sheet 1 of 9

EQUIPMENT/COMPONENT	PARAMETER	INSTRUMENT	MAXIMUM	MINIMUM	SHIFT DAY	SHIFT NIGHT	COMMENT NUMBER
CIRC WATER	SUPPLY TEMP (°F)	1-TI-7138		40			
	SUPPLY PRESS (PSIG)	1-PI-7170		22			
NSCW TRAIN A	RETURN TEMP (°F)	1-TI-1676A					
	HDR PRESS (PSIG)	1-PI-1636	120	70			
RMW STOR TANK	LEVEL (%)	1-LI-7761B	98	74			
NSCW TRAIN B	RETURN TEMP (°F)	1-TI-1677A					
	HDR PRESS (PSIG)	1-PI-1637	120	70			
CCW TRAIN A SUPPLY HEADER	HDR PRESS (PSIG	1-PI-1874	100	87			
	FLOW (GPM)	1-FI-1876	10,200	8500			
ACCW SUPPLY HDR	PRESS (PSIG)	1-PI-1977		135			
CCW TRAIN B SUPPLY HEADER	HDR PRESS (PSIG)	1-PI-1875	100	87			
	FLOW (GPM)	1-FI-1877	10,200	8500			
TPCW SUPPLY HEADER	TEMP (°F)	1-TI-6721		40			
	PRESS (PSIG)	1-PI-7235	115	80			
	FLOW (GPM)	1-FI-7093	25,000	5000			
INSTRUMENT AIR HDR	PRESS (PSIG)	1-PI-9361		100			
SERVICEAIR HDR	PRESS (PSIG)	1-PI-9377		100			
AIR COMPRESSOR(S) RUNNING	#1, #2, #3 AND/OR #4						
NSCW TRAIN A FLOW	SUPPLY (GPM)	1-FI-1640B	20,000				
	RETURN (GPM)	1-FI-1640A	20,000				
NSCW TRAIN B FLOW	SUPPLY (GPM)	1-FI-1641B	20,000				
	RETURN (GPM)	1-FI-1641A	20,000				
DEMIN WTR TANK	LEVEL (FT)	1-LI-7526A	27	15			
ACCW RETURN FRM RCP	FLOW (GPM)	1-FI-11829		2800			
	FLOW (GPM)	1-FI-11830		2800			
TPCCW	SUPPLY TEMP (°F)	1-TI-7318	110				
	SURGE TANK LEVEL (IN.)	1-LI-7312	+16	-16			
	SUPPLY PRESS (PSIG)	1-PI-7315	110	90			
SI Pump A	DISCHARGE PRESS (PSIG)	1-PI-919	(12)				
SI Pump B	DISCHARGE PRESS (PSIG)	1-PI-923	(12)				
RHR Pump A	DISCHARGE PRESS (PSIG)	1-PI-614	100 (11)				
RHR Pump B	DISCHARGE PRESS (PSIG)	1-PI-615	100 (11)				

(11) DEPRESS RHR SYSTEM PER GUIDANCE OF 13011-1 RESIDUAL HEAT REMOVAL SYSTEM

(12) SI DISCHARGE HEADER PRESSURE SHOULD BE LESS THAN OR EQUAL TO APPROXIMATELY SI ACCUMULATOR PRESSURE. INCREASING PRESSURE MAY INDICATE RCS CHECK VALVE LEAKAGE.

COMMENTS: ______________________________



seal, consisting of a carbon-graphite insert assembled with a shrink fit into a stainless steel seal ring. The carbon-graphite insert has two sealing faces called "dams". These dams rub on a chrome-carbide coated stainless steel runner, which rotates with the shaft. Water from the RCP standpipe is injected between the two dams of the seal ring at a pressure greater than in the number 2 seal leak-off connection. Two leakage paths are thus provided for this injected water (thus the term "double dam"). Part of the injected water flows past the outer dam where it joins the leakage from the number 2 seal and passes out of the pump through the number 2 seal leak-off connection. The remainder of the injected water flows past the inner dam and is diverted to the splashguard to flow out through the number 3 seal leak-off connection to the normal containment sump. A pressure control orifice is installed in the number 3 injection connection to compensate for dam profile variations. A check valve is also provided in this location to allow the inner dam pressure to support the seal faces during number 2 leak-off high-pressure conditions.

Seal injection provided by the CVCS system is approximately 8 gallons per minute per RCP number 1 seal. 5 gpm of the 8 gpm total go directly to the lower pump bearing providing lubrication and cooling. 3 gallons is directed through the number 1 seal were a pressure drop of approximately 2220 psid occurs. ALL but 3 gph of the number 1 seal leak off is returned to the volume control tank (VCT). 3 gph of number 1 seal leak off enters the number 2 seal. 3 gph passes through the number 2 seal and its leak off is directed to the reactor coolant drain tank (RCDT). The RCP standpipe provides 800 cc/hr of seal injection flow to the number 3 seal. Each RCP has its own standpipe that is located at a higher elevation to provide gravity flow. The Standpipe make up is provided by reactor makeup water system tank (RMWST). Number 3 seal injection is injected between the two dams and the sealing surface. The number 3 seal injection is at a slightly higher pressure than number 2 seal injection leak off. This prevents RCS liquids or gases from escaping to the containment environment. The number 3 seal off has two paths: (1) the outer dam leak off (400 cc/hr) combines with number 2 seal leak off and is routed to the RCDT, (2) the inner dam is directed to the containment sump.

ACCW SYSTEM

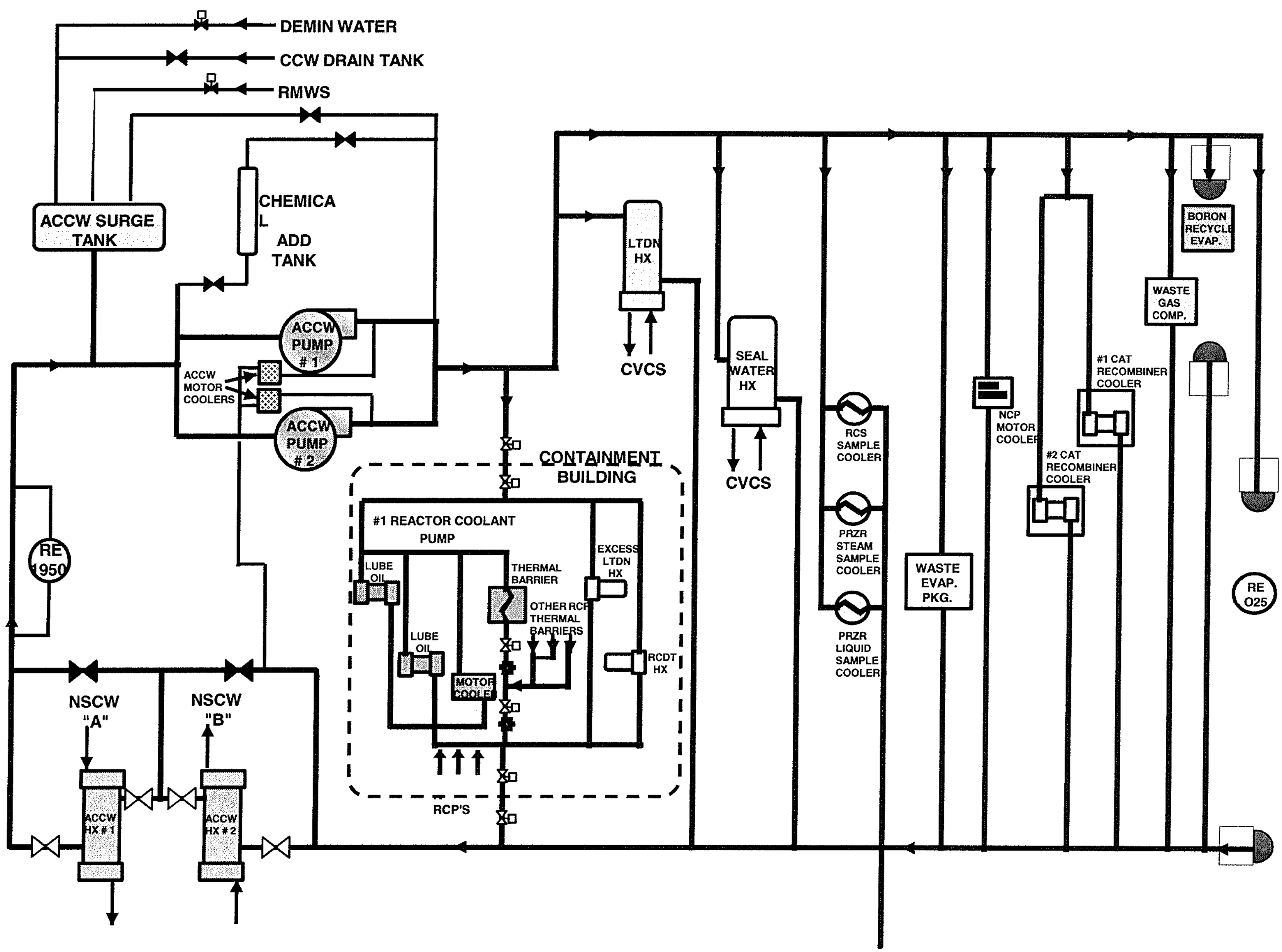
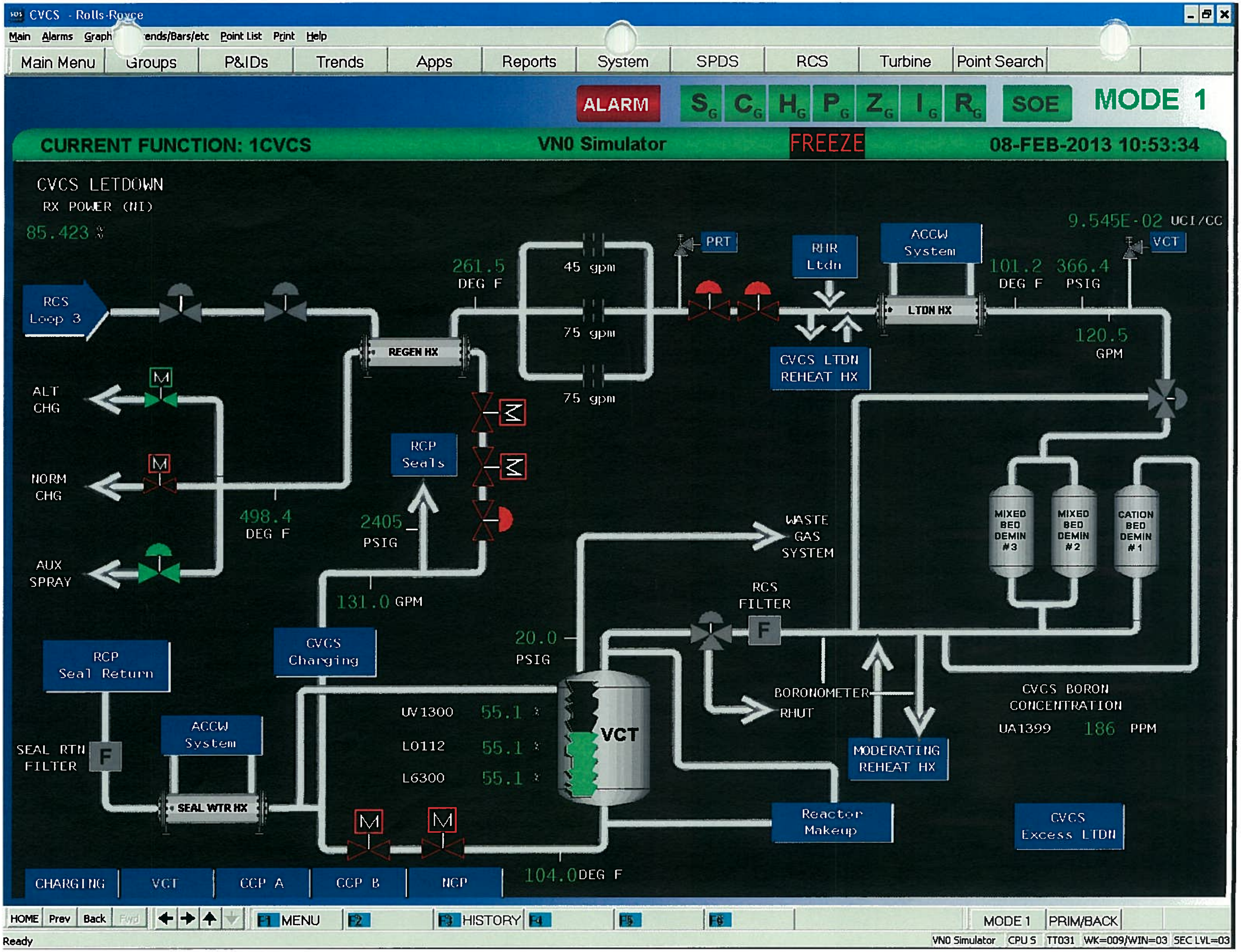
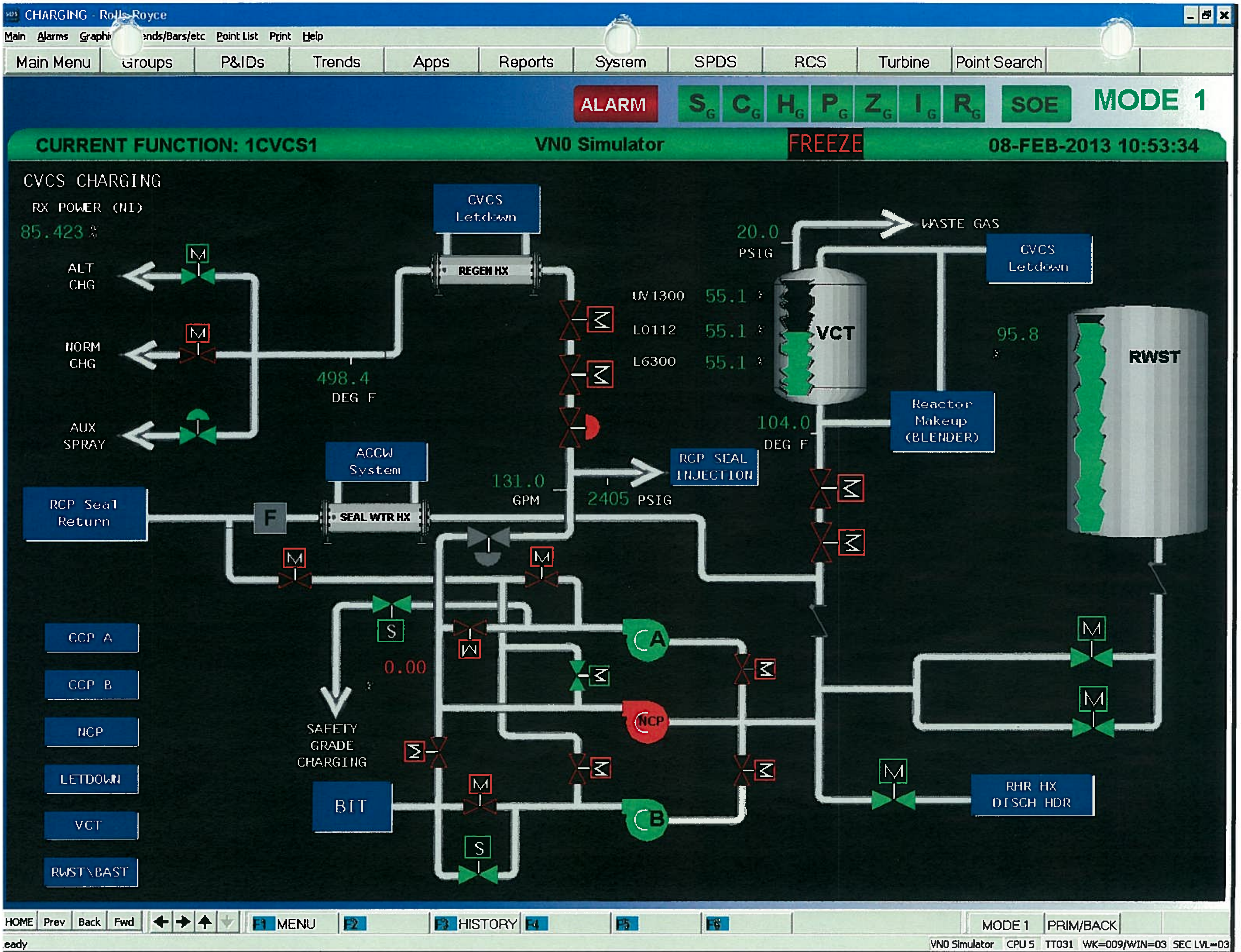
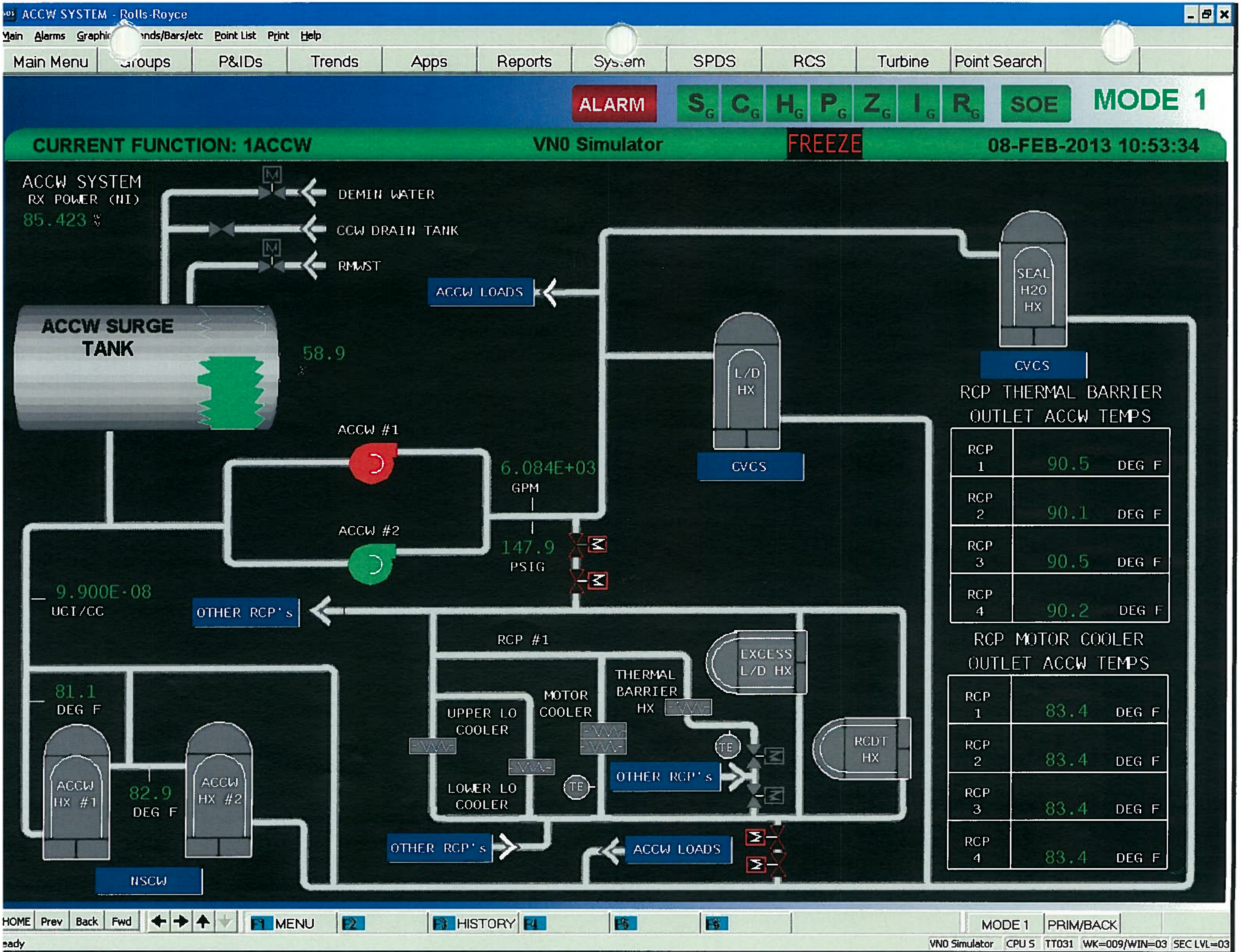


FIGURE 04-1


CVCS - Rolls-Royce
Main Alarms Graph ends/Bars/etc Point List Print Help
Main Menu Groups P&IDs Trends Apps Reports System SPDS RCS Turbine Point Search
ALARM
SOE
MODE 1
CURRENT FUNCTION: 1CVCS
VN0 Simulator
FREEZE
08-FEB-2013 10:53:34
CVCS LETDOWN
RX POWER (NI)
85.423 %
RCS Loop 3
ALT CHG
NORM CHG
AUX SPRAY
REGEN HX
261.5 DEG F
45 gpm
75 gpm
75 gpm
PRT
RHR Ltdn
ACCW System
LTDN HX
101.2 DEG F
366.4 PSIG
9.545E-02 UCI/CC
VCT
120.5 GPM
CVCS LTDN REHEAT HX
RCP Seals
498.4 DEG F
2405 PSIG
131.0 GPM
WASTE GAS SYSTEM
MIXED BED DEMIN #3
MIXED BED DEMIN #2
CATION BED DEMIN #1
RCS FILTER
20.0 PSIG
CVCS Charging
RCP Seal Return
SEAL RTN FILTER
ACCW System
SEAL WTR HX
UV1300 55.1 %
L0112 55.1 %
L6300 55.1 %
VCT
BORONOMETER
RHUT
MODERATING REHEAT HX
CVCS BORON CONCENTRATION
UA1399 186 PPM
Reactor Makeup
CVCS Excess LTDN
104.0 DEG F
CHARGING VCT CCP A CCP B NCP
HOME Prev Back Fwd
F1 MENU F2 F3 HISTORY F4 F5 F6
MODE 1 PRIM/BACK
Ready
VN0 Simulator CPU 5 TT031 WK=009/WIN=03 SEC LVL=03


CHARGING - Rolls-Royce
Main Alarms Graphi… …nds/Bars/etc Point List Print Help
Main Menu Groups P&IDs Trends Apps Reports System SPDS RCS Turbine Point Search
ALARM
SOE
MODE 1
CURRENT FUNCTION: 1CVCS1
VN0 Simulator
FREEZE
08-FEB-2013 10:53:34
CVCS CHARGING
RX POWER (NI)
85.423 %
ALT CHG
NORM CHG
AUX SPRAY
CVCS Letdown
REGEN HX
498.4
DEG F
ACCW System
SEAL WTR HX
RCP Seal Return
131.0
GPM
2405 PSIG
RCP SEAL INJECTION
20.0
PSIG
WASTE GAS
UV1300 55.1 %
L0112 55.1 %
L6300 55.1 %
VCT
104.0
DEG F
Reactor Makeup (BLENDER)
95.8
%
RWST
0.00
%
SAFETY GRADE CHARGING
BIT
NCP
RHR HX DISCH HDR
CCP A
CCP B
NCP
LETDOWN
VCT
RWST\BAST
HOME Prev Back Fwd
F1 MENU
F3 HISTORY
MODE 1
PRIM/BACK
eady
VN0 Simulator CPU 5 TT031 WK=009/WIN=03 SEC LVL=03


ACCW SYSTEM - Rolls-Royce
Main Alarms Graphic ...nds/Bars/etc Point List Print Help
Main Menu
...roups
P&IDs
Trends
Apps
Reports
Sys..em
SPDS
RCS
Turbine
Point Search
ALARM
S G C G H G P G Z G I G R G
SOE
MODE 1
CURRENT FUNCTION: 1ACCW
VN0 Simulator
FREEZE
08-FEB-2013 10:53:34
ACCW SYSTEM
RX POWER (NI)
85.423 %
DEMIN WATER
CCW DRAIN TANK
RMWST
ACCW LOADS
ACCW SURGE TANK
58.9
ACCW #1
ACCW #2
6.084E+03
GPM
147.9
PSIG
9.900E-08
UCI/CC
OTHER RCP's
RCP #1
81.1
DEG F
82.9
DEG F
ACCW HX #1
ACCW HX #2
NSCW
UPPER LO COOLER
LOWER LO COOLER
MOTOR COOLER
THERMAL BARRIER HX
TE
TE
OTHER RCP's
OTHER RCP's
ACCW LOADS
EXCESS L/D HX
RCDT HX
L/D HX
CVCS
SEAL H20 HX
CVCS
RCP THERMAL BARRIER OUTLET ACCW TEMPS
RCP 1 90.5 DEG F
RCP 2 90.1 DEG F
RCP 3 90.5 DEG F
RCP 4 90.2 DEG F
RCP MOTOR COOLER OUTLET ACCW TEMPS
RCP 1 83.4 DEG F
RCP 2 83.4 DEG F
RCP 3 83.4 DEG F
RCP 4 83.4 DEG F
HOME Prev Back Fwd
F1 MENU
F2
F3 HISTORY
F4
F5
F6
MODE 1
PRIM/BACK
...eady
VN0 Simulator CPU S TT031 WK=009/WIN=03 SEC LVL=03

29. 029EK3.03 001/1/1/ATWT -BIT VLV/MEM - 3.7/3.6/MOD - BANK/HL-18 NRC/RO/SRO/TNT

During an ATWT event, the following conditions exist:

- Emergency Boration is in progress through HV-8801A and HV-8801B, BIT Outlet Isolation Valves.

Which one of the following completes the following statement?

For an adequate Emergency Boration flow path to exist, flow as read on BIT Flow Indicator (FI-917A), at a MINIMUM, must be greater than ___(1)___,

and

the reason for the minimum flow rate is to account for ___(2)___.

A. (1) 87.5 gpm

(2) the amount of CVCS Letdown that is in service

B. (1) 100 gpm

(2) the amount of CVCS Letdown that is in service

C. (1) 100 gpm

(2) RCP seal injection flow minus total seal return flow

D✓ (1) 87.5 gpm

(2) RCP seal injection flow minus total seal return flow

029EK3.03 Anticipated Transient Without Scram (ATWS)

Knowledge of the reasons for the following responses as they apply to the ATWS:
(CFR: 41.5 / 41.10 / 45.6 / 45.13)

Opening the BIT inlet and outlet valves.

K/A MATCH ANALYSIS:

The question asks a plausible scenario where an ATWT is in progress and the BIT Valve flowpath from the RWST is used for the Emergency Boration. BIT path is an approved Emergency Boration flowpath requiring a higher flowrate than would be if the BAST were used due to the Cb concentration.The candidate must know the flowrate and the reason behind it.

DISTRACTOR ANALYSIS:

A. Incorrect. 87.5 gpm is correct, but the compensation for the amount of letdown that may be in service is not related to the flowrate.

B. Incorrect. 100 gpm is not the minimum flow required. Plausible because this amount of flow is required if RWST was being used through the normal charging flowpath. Compensation for the amount of letdown flow is not related.

C. Incorrect. 100 gpm is not the minimum flow required. Plausible because this amount of flow is required if RWST was being used through the normal charging flowpath. The compensation for seal injection versus seal leakoff is correct in order to inject the proper amount of borated water necessary coming from the RWST.

D. Correct. 87.5 gpm is correct. The compensation for seal injection versus seal leakoff is correct in order to inject the proper amount of borated water necessary coming from the RWST.

REFERENCES:

19211-C, "Response to Nuclear Power Generation/ATWT"
13009-1, "CVCS Reactor Makeup Control System", Section 4.9 for Emergency Boration, page # 47 and # 69
13009-1, "CVCS Reactor Makeup Control System", Table 1 Emergency Boration Flow Path Alternatives

VEGP learning objectives:

LO-PP-09300-06 Describe for all emergency flow paths:
a. borated water source and discharge flow path
b. minimum flow requirements

LO-PP-09300-14 Describe the alignment, and state the concerns when borating the RCS using the RWST with the BAST out of service.

INITIALS

4.9.4 Emergency Boration From The RWST Through The BIT Isolation Valves

4.9.4.1 **Verify** one (1) Charging Pump is running and supplied with cooling water. ________

4.9.4.2 **Open** the following Charging Pump Suctions from the RWST:

- 1-LV-0112D ________
- 1-LV-0112E ________

4.9.4.3 **Close** the following VCT Outlet Isolations:

- 1-LV-0112B ________
- 1-LV-0112C ________

4.9.4.4 **Place** 1-LV-0112A to the HUT position. ________

4.9.4.5 **Open** the following BIT DISCH ISOLATION valves:

- 1-HV-8801A ________
- 1-HV-8801B ________

4.9.4.6 **Verify** BIT Flow (1-FI-0917A), plus total seal injection flow, minus total seal return flow is greater than 87.5 gpm. ________

4.9.4.7 **Adjust** Charging Line Flow Controller 1-FIC-0121 as necessary to maintain RCP seal injection flow at maximum flow less than 13 gpm per pump. ________

4.9.4.8 <u>IF</u> required for RCS inventory control, **place** an additional letdown orifice in service per 13006-1. ________

4.9.4.9 **Operate** the Pressurizer Backup Heaters as necessary to equalize boron concentrations between the RCS and the Pressurizer. ________

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
SUBSEQUENT OPERATOR ACTIONS	
3. Check AFW Pumps - RUNNING:	
__• MDAFW Pumps	__Start Pumps.
__• TDAFW Pump, if required	__Open Steam Supply valve HV-5106.
4. Emergency borate the RCS:	
__a. Start at least one Boric Acid Transfer Pump.	
__b. Verify a Charging Pump is running.	
__c. Open EMERGENCY BORATE valve HV-8104.	c. IF HV-8104 will NOT open, THEN open the following: __• FV-110A, BA TO BLENDER. __• FV-110B, BLENDER OUTLET TO CHARGING PUMPS SUCT.
__d. Verify charging flow - GREATER THAN 42 GPM.	
__e. Verify boric acid flow - GREATER THAN 30 GPM.	__e. Initiate 13009, CVCS REACTOR MAKEUP CONTROL SYSTEM as necessary to establish an alternate emergency boration flowpath.

Step 4 continued on next page

TABLE 1

EMERGENCY BORATION FLOW PATH ALTERNATIVES

Flow path	BATP	Valve Alignments	Other Pump Required	Flows	Flow	Note
HV8104	At least one	**OPEN** 1HV-8104	Any charging pump	**>42 GPM** 1FI-0121C	**>30 GPM** 1FI-0183A	Operate heaters
Charging Flow path	At least one	**OPEN** 1FV-0110A 1FV-0110B	Any charging pump	**>42 GPM** 1FI-0121C	**>30 GPM** 1FI-0110A	Operate heaters
RWST to Regen Hx	NA	**OPEN** 1LV-0112D 1LV-0112E **CLOSE** 1LV-0112B 1LV-0112C **HUT** 1LV-0112A	Any charging pump	**>100 GPM** 1FI-0121C	**8 to 13 GPM** seal injection flow 1HV-0182	Operate heaters
RWST to BIT	NA	**OPEN** 1LV-0112D 1LV-0112E 1HV-8801A 1HV-8801B **CLOSE** 1LV-0112B 1LV-0112C **HUT** 1LV-0112A	Any charging pump	BIT flow (1FI-0917A) + total seal flow - seal return flow **>87.5 GPM**	Adjust 1FIC-0121C to **<13 GPM** per RCP	Operate heaters
RHR (Mode 6)	NA	**OPEN** HV-8812A/B HV-8809A/B	RHR other than S/D Cooling	>100 gpm	See Proc.	Establish water removal path to prevent vessel overflow
SI (Mode 6)	NA	**OPEN** HV-8923A/B HV-8821A/B HV-8835	SI	>100 gpm	See Proc.	Establish water removal path to prevent vessel or cavity overflow

30. 034A2.03 001/2/2/FUEL HAND - MISPO/MEM - 3.3/4.0/NEW/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- Unit 2 is in Mode 6, Refueling outage in progress.

Current conditions:

- A fuel assembly has been lowered into a wrong core location.
- SR counts are unexpectedly rising on NR-45.
- The Fuel Handling Supervisor (FHS) requests permission to unlatch the assembly just lowered.

Which one of the following completes the following statement?

The FHS ___(1)___ be directed to disengage the mispositioned fuel assembly at the current location,

and

per 93300-C, "Conduct of Refueling Operations," the OATC (Reactor Operator) ___(2)___ have the authority to suspend refueling operations.

	__(1)__	__(2)__
A.	will	does
B.	will	does NOT
C.✓	will NOT	does
D.	will NOT	does NOT

034A2.03 Fuel Handling Equipment System (FHES)

Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR 41.5 / 43.5 / 45.3 / 45.13):

Mispositioned fuel element.

<u>K/A MATCH ANALYSIS:</u>

The question presents a plausible scenario where a fuel assembly has been lowered into the wrong core location resulting in a unexpected increase in SR counts. The candidate must determine per 93300-C the person responsible for directing whether or not to disengage the assembly and whether or not the OATC has the authority to suspend core alterations.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect. Per 93300-C, the OATC is the person to direct release of fuel bundles in the core once counts have stabilized. In this case, the OATC will tell the FHS to NOT release the assembly. The OATC (Reactor Operator) does have the authority to suspend fuel handling operations on his own authority if he feels personnel safety or safe fuel movement is threatened.

B. Incorrect. Per 93300-C, the OATC is the person to direct release of fuel bundles in the core once counts have stabilized. In this case, the OATC will tell the FHS to NOT release the assembly. The OATC (Reactor Operator) does have the authority to suspend fuel handling operations on his own authority if he feels personnel safety or safe fuel movement is threatened.

It is plausible the candidate may think the SS, Rx Engineer, or FHS or some other higher authority is required to suspend refueling operations.

C. Correct. Per 93300-C, the OATC is the person to direct release of fuel bundles in the core once counts have stabilized. In this case, the OATC will tell the FHS to NOT release the assembly. The OATC (Reactor Operator) does have the authority to suspend fuel handling operations on his own authority if he feels personnel safety or safe fuel movement is threatened.

D. Incorrect. Per 93300-C, the OATC is the person to direct release of fuel bundles in the core once counts have stabilized. In this case, the OATC will tell the FHS to NOT release the assembly. The OATC (Reactor Operator) does have the authority to suspend fuel handling operations on his own authority if he feels personnel safety or safe fuel movement is threatened.

It is plausible since the assembly is mispositioned the Reactor Engineer could be the one to authorize whether to disengage or not.

It is plausible the candidate may think the SS, Rx Engineer, or FHS or some other higher authority is required to suspend refueling operations.

REFERENCES:

93300-C, Conduct of Refueling Operations

VEGP learning objectives:

LO-PP-25101-17	Explain which operations are covered by the Conduct of Refueling Operations Procedure (93300-C)
LO-PP-25101-19	State which members of the refueling team have the authority to suspend refueling operations.
LO-PP-25101-24	Describe the responsibilities that each of the following positions have during refueling operations: a. Fuel Handling Coordinator b. Fuel Handling Supervisor c. Unit Shift Superintendent d. OATC e. Shift Superintendent f. Reactor Engineer

3.0 PRECAUTIONS AND LIMITATIONS

3.1 The Fuel Handling Coordinator, Shift Manager, Fuel Handling Supervisor, Reactor Engineer, Shift Supervisor, Health Physics Technician, or Reactor Operator shall have the authority and responsibility to suspend refueling operations if, in his judgment, any conditions exist which threaten personnel safety or safe handling of fuel.

3.2 Suspension of core alterations shall not preclude any individual assigned to the refueling crew from completion of movement of a component to a safe conservative position.

3.3 All core alterations shall be observed and directly supervised by either a licensed SRO or SRO Limited to Fuel Handling who is in the Containment Building, of the affected unit and has no other concurrent responsibility during this operation.

3.4 The Shift Supervisor shall ensure that the applicable Core off load or Core reload section of Checklist 1 has been verified complete prior to the activity commencing, and an entry stating so has made in the Electronic Log.

3.5 No more than one fuel assembly shall be out of the storage racks, fuel cleaning canisters, new fuel elevator, fuel transfer system upender, or fuel assembly leak test canister at any given time in the Fuel Building. A minimum distance of 12 inches, edge-to-edge, shall be maintained between the assembly being manipulated and any assembly not located in a storage rack. No more than two fuel assemblies shall be out of the Reactor Vessel at any given time in the Containment and then the minimum edge-to-edge distance between those two assemblies must be at least 12 inches.

3.6 On-Shift operations will be handled in accordance with Procedure NMP-OS-007, "Conduct of Operations," except as supplemented by this procedure.

3.7 Prior to any movement of irradiated fuel through the transfer tube, the removable access plugs located in containment and the Fuel Building shall be verified closed. To ensure there will be no access during fuel movement, notify Health Physics to establish a locked or posted access on the concrete plugs for the Fuel Transfer Tube Bellows in the Fuel Handling Building (Unit 1 rooms 104 and A09, Unit 2 rooms 101 and A02) and Containment Building.(Unit 1 or 2 197" elevation between col 22 and 23)

3.8 After commencement of 1/M plotting, a fuel bundle will not be lowered into the vessel until data gathered from previous bundle has been plotted and a determination made that it is safe to proceed with the reload sequence.

4.3 SHIFT SUPERVISOR

The responsibilities of this individual during refueling operations are as follows:

4.3.1 As part of his normal duties, the Shift Supervisor will be responsible for maintaining the required status of all support systems needed for the refueling operation, such as the Residual Heat Removal System, electrical and air systems, Spent Fuel Pool Water Systems, etc.

4.3.2 Monitors the plant and refueling operation for plant safety and compliance with Technical Specifications.

4.3.3 Ensures that direct communications are maintained between the refueling area in Containment, the Fuel Handling Building, and the Control Room when core alterations are in progress.

4.3.4 Resolves any anomalous plant conditions that may arise.

4.3.5 Initiates cavity water level surveillance whenever the reactor cavity is flooded.

4.4 OPERATOR AT THE CONTROLS

The duties and responsibilities of this individual during refueling activities include:

4.4.1 An OATC shall be stationed in the Control Room during all fuel movement and control rod latching and unlatching..

4.4.2 Maintains cognizance of overall plant conditions during all periods of fuel movement.

4.4.3 Monitors nuclear instrumentation during core alterations and upon determination that the count rate is stable directs disengagement of fuel bundles in core.

4.4.4 Monitors nuclear instrumentation during Control Rod latching and unlatching activities. Ensures count rate stability during the ten foot rod withdrawal portion of the latching process.

31. 035A4.05 001/2/2/S/G - LVL CTRL/MEM - 3.8/4.0/NEW/HL-18 NRC/RO/SRO/AML

Given the following:

- Unit 1 tripped due to a loss of offsite power.

- Natural circulation flow is currently developing in the Reactor Coolant System (RCS).

Which ONE of the following describes operator actions that will enhance RCS natural circulation flow per 19002-C, "Natural Circulation Cooldown"?

1. Maintain PRZR pressure ≥ 1950 psig to keep voids from forming in vessel head.
2. Maintain PRZR level ≥ 25% to maintain water inventory for pressure control.
3. Maintain SG NR levels ~65% to maintain heat sink.
4. Maintain steam flow as necessary to minimize loop delta T.

A.✓ 1, 2, and 3 ONLY

B. 2 and 3 ONLY

C. 2, 3, and 4 ONLY

D. 1 and 3 ONLY

035A4.05 Steam Generator System (S/Gs)

Ability to manually operate and/or monitor in the control room: (CFR 41.7 / 45.5 to 45.8)

Level control to enhance natural circulation.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a natural circulation condition is developing in the RCS. The question straight forward asks what actions the candidate may take to enhance natural circulation conditions.

DISTRACTOR ANALYSIS:

A. Correct. Actions 1, 2, 3 are used to enhance natural circulation cooldown, action # 4 is incorrect.

B. Incorrect. Actions 1, 2, 3 are used to enhance natural circulation cooldown, action # 4 is incorrect.

C. Incorrect. Actions 1, 2, 3 are used to enhance natural circulation cooldown, action # 4 is incorrect.

D. Incorrect. Actions 1, 2, 3 are used to enhance natural circulation cooldown, action # 4 is incorrect.

REFERENCES:

19002-C, ES-0.2 Natural Circulation Cooldown.

VEGP learning objectives:

LO-LP-37012-14 State the initial steps that must be taken to prepare the plant for natural circulation cooldown. Include the reason for each.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__b. As RCS cooldown is initiated, hold HS-0500A and HS-0500B in the BYPASS INTERLOCK position until RCS temperature is less than 550°F.	
***10. Initiate RCS cooldown to cold shutdown:**	
__a. Check RCS boron concentration greater than required boron concentration for xenon free cold shutdown.	__a. Return to Step 6.
__b. Maintain cooldown rate in RCS Cold Legs - LESS THAN 50°F/Hr.	
__c. Dump steam to Condenser using Steam Dumps.	__c. Use SG ARVs.
__d. Maintain SG NR levels - AT APPROXIMATELY 65%.	
__e. Check RCS cooldown at 15 minute intervals.	
f. Maintain RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR): __ Use 60°F/HR curve and RCS Cold Leg temperature.	
__g. Perform other appropriate actions required to take the unit to cold shutdown by initiating 12006-C, RCS COOLDOWN TO COLD SHUTDOWN.	

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

CAUTION

SI actuation circuits will automatically unblock if PRZR pressure rises to greater than 2000 psig.

17. Block SI actuation:

- Low steamline pressure SI
- Low PRZR pressure SI

*18. **Maintain following RCS conditions:**

__• RCS pressure - AT 1950 psig.

__• PRZR level - AT 25%.

__• Cooldown rate in RCS Cold Legs - LESS THAN 50°F/Hr.

• RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR):

__ Use 60°F/HR curve and RCS Cold Leg temperature.

*19. **Monitor RCS cooldown:**

__• Core Exit TCs - LOWERING.

__• RCS WR Hot Leg temperatures - LOWERING.

__• RCS subcooling - RISING.

32. 036AA1.04 001/1/2/FUEL HAND - INCIDENT/MEM - 3.1/3.7/NEW/HL-18 NRC/RO/SRO/TNT

Given the following conditions:

- Unit 1 is in Mode 6 with Core off-load in progress.
- A fuel assembly is to be withdrawn from the core.
- The assembly is twisted and cannot be latched.

Which ONE of the following completes the following statement?

Per 93270-C, "Refueling Machine Operation," and with Fuel Handling Supervisor permission, rotating of the Refueling Machine Mast __(1)__ allowed to latch the fuel assembly,

and

if necessary, the Refueling Machine Hoist __(2)__ be manually operated.

A. (1) is NOT

(2) can

B.✓ (1) is

(2) can

C. (1) is NOT

(2) cannot

D. (1) is

(2) cannot

036AA1.04 Fuel Handling Incidents

Ability to operate and / or monitor the following as they apply to Fuel Handling Incidents:
(CFR: 41.7 / 45.5/ 45.6)

Fuel handling equipment during an incident.

K/A MATCH ANALYSIS:

The question presents a plausible scenario requiring the candidate to know whether or not RF Machine mast rotation to latch onto a twisted assembly is allowed and manual operation of the RF hoist are allowed.

DISTRACTOR ANALYSIS:

A. Incorrect. Mast rotation is allowed and the RF Machine Hoist can be manually operated per 93270-C and 93500-C.

B. Correct. Mast rotation is allowed and the RF Machine Hoist can be manually operated per 93270-C and 93500-C.

C. Incorrect. Mast rotation is allowed and the RF Machine Hoist can be manually operated per 93270-C and 93500-C.

D. Incorrect. Mast rotation is allowed and the RF Machine Hoist can be manually operated per 93270-C and 93500-C.

REFERENCES:

93270-C, Refueling Machine Operation, section 4.4.5 Mast Rotation

93500-C, Manual Operation of Fuel Handling Equipment

VEGP learning objectives:

LO-PP-25101-03 Describe the operations of the refueling machine controls to include:

c. Mast control

LO-PP-25101-21 Explain what operations are covered by the "Manual Operation of Fuel Handling Equipment" procedure (93500-C).

INITIALS

4.4.5 Mast Rotation

> NOTE
>
> Mast rotation is allowed when fuel assembly bowing or twisting is so severe that a fuel assembly in the core cannot be latched or an assembly cannot be pulled up into the mast.

4.4.5.1 **Obtain** Fuel Handling Supervisor permission to rotate the mast. ________

4.4.5.2 **Verify** a common alignment mark is present on both the upper rotating portion of the mast and the lower stationary portion of the mast. ________

4.4.5.3 **Loosen** the mast rotation lock device. ________

4.4.5.4 **Rotate** the mast a sufficient amount to allow the gripper to enter the assembly OR to allow the assembly to enter the mast. ________

4.4.5.5 **Tighten** the mast rotation locking device. ________

4.4.5.6 **Verify** the "MAST NOT AT ZERO DEGREE" message on the screen. ________

4.4.5.7 **Latch** the gripper OR **raise** the assembly one foot into the mast. ________

4.4.5.8 **Loosen** the mast rotation locking device. ________

4.4.5.9 **Rotate** the mast back to its initial position. ________

4.4.5.10 **Tighten** the mast rotation locking device. ________

4.4.5.11 **Verify** "MAST AT ZERO DEGREE" Green icon on screen. ________

INITIALS

4.2 REFUELING MACHINE

4.2.1 Bridge Motor Failure

4.2.1.1 **De-energize** power to the motor by turning OFF breaker CB-D located in the back of the RFM console on the right side.

4.2.1.2 **Lift** and **secure** deck plates to gain access to the drive line.

4.2.1.3 Manually **release** the drive motor brake.

4.2.1.4 **Install** the emergency handwheel on the speed reducer shaft.

4.2.1.5 **Hand crank** the handwheel, as required, to move the bridge in the desired direction.

4.2.1.6 **Confirm** position of the bridge by means of the bridge index marks.

4.2.2 Trolley Motor Failure

4.2.2.1 **De-energize** power to the motor by turning OFF breaker CB-D located in the back of the RFM console on the right side.

4.2.2.2 **Lift** and **secure** deck plates to gain access to the drive line.

4.2.2.3 Manually **release** the trolley motor brake.

4.2.2.4 **Install** the emergency handwheel on the speed reducer shaft.

4.2.2.5 **Hand crank** the handwheel as required to move the trolley in the desired direction.

4.2.2.6 **Confirm** position of the trolley by means of the trolley index marks.

4.2.3 Hoist Motor Failure

4.2.3.1 **De-energize** power to the motor by turning OFF breaker CB-D located in the back of the RFM console on the right side.

4.2.3.2 Manually **release** the hoist motor brake using the release (T-handle) on the housing.

4.2.3.3 **Install** the emergency chain wheel (stored in trolley drive compartment) on the hoist winch shaft.

INITIALS

4.2.3.4 **Operate** the chain wheel up or down to move hoist as required.

4.2.3.5 **Confirm** position of the hoist by means of the Z-tape or other positive means (e.g.; underwater TV camera).

4.2.4 Solenoid Air Valve Failure

4.2.4.1 Each solenoid air valve is equipped with a manual operator to be used in the event of electrical failure. A small button located just below the solenoid on the valve body is depressed by using a small round pin approximately 1/8 inch diameter.

4.2.5 Fuel Gripper Cylinder Failure

4.2.5.1 **Raise** the mast to full up.

4.2.5.2 **Locate** the 2 eyebolts on the top of the gripper cylinder and **attach** cables to them.

4.2.5.3 **Lower** the fuel assembly to the desired location.

4.2.5.4 **Select** the gripper unlatch position on the gripper selector switch located on the control console OR, if necessary, manually **operate** the unlatch solenoid as in Step 4.3.4.1 above.

4.2.5.5 **Apply** upward force to the cables to **unlatch** the fuel assembly.

4.2.6 Auxiliary Hoist Motor Failure

4.2.6.1 **De-energize** power to the motor.

4.2.6.2 **Pull out** the brake release knob on the end of the motor.

4.2.6.3 **Install** the emergency handwheel on the motor extension shaft.

4.2.6.4 **Hand crank** the handwheel up or down to the desired location.

33. 037AA2.08 001/1/2/SG LEAK - A/E MON/C/A - 2.8/3.3/MOD-FARLEY 2010 NRC/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- Reactor power is 12%.

Current conditions:

- RE-12839C, SJAE - Wide Range Radiogas (Low Range), has failed.
- A 10 gpm SG tube leak develops on SG # 3.

Which ONE of the following radiation monitors will provide the EARLIEST indication of the SG tube leak?

A. RE-0724, N16

B. RE-13122, MSL Loop 3

C.✓ RE-0810, SJAE Exhaust

D. RE-0019, SG Sample Liquid

037AA2.08 Steam Generator Tube Leak

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak:
(CFR: 43.5/ 45.13)

Failure on Condensate air ejector exhaust monitor.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where the Condensate Air Ejector Exhaust Monitor is failed. The candidate must determine with a SGTL present, which rad monitor will provide the earliest indication of the SGTL.

DISTRACTOR ANALYSIS:

A. Incorrect. Although providing a quick response, RE-0724 for measuring N16 is only accurate at higher power levels.

B. Incorrect. Although located on SG # 3 main steam line, this radiation monitor is a PAMS monitor and will not alarm until radiation levels are very high.

C. Correct. RE-0810 is the quickest radiation monitor to detect a SGTL, this is RE-0810s purpose, early detection of an SGTL.

D. Incorrect. RE-0019 is located on the SG Blowdown lines but is after the SGBD demins and will read lower than other radiation monitors, RE-0019 will detect the SGTL but will be slower than RE-0810.

REFERENCES:

17100-1, "Annunciator Response Procedure for the Process and Effluent Radiation Monitoring System (RMS)"
WOG Background Document for E-3 Steam Generator Tube Rupture
V-LO-LP-37311, Steam Generator Tube Rupture

VEGP learning objectives:

LO-LP-60309-10 Discuss how changes in the following affect radiation monitor response to a steam generator tube leak / rupture.
a. RCS activity
b. Power level
c. Process flow rate (i.e., SG blowdown)
d. Rupture size

LO-LP-60309-11 Describe the guidelines for plant operation with minor steam generator tube leakage including:

a. Selection of appropriate computer trends and radiation alarm setpoints to identify changes in the amount of leakage.

Since the primary system pressure (nominally 2235 psig) is initially much greater than the steam generator pressures (nominally 1000 psig) reactor coolant flows from the primary into the secondary side of the affected steam generator. In response to this loss of reactor coolant, pressurizer level decreases at a rate which is dependent upon the size and number of failed tubes, as shown in Figure 1. RCS pressure, Figure 2, also decreases as the steam bubble in the pressurizer expands. Normally, charging flow will automatically increase and pressurizer heaters will energize in an effort to stabilize pressure and level. However, if leakage exceeds the capacity of the Chemical and Volume Control System (CVCS), reactor coolant inventory will continue to decrease and eventually lead to an automatic reactor trip signal. If turbine load is not reduced, reactor trip will most likely occur on over temperature delta T. For the expected case, however, turbine load will be decreased either automatically or manually so that reactor trip will occur on low pressurizer pressure. Normal letdown flow would isolate and pressurizer heaters would turn off on low pressurizer level.

On the secondary side, leakage of contaminated primary coolant will increase the activity of the secondary coolant resulting in high radiation indications from the air ejector radiation monitor and blowdown line radiation monitors. Although these alarms may lag indications of a loss of reactor coolant, depending on the transport time to the radiation monitors, they have sounded nearly simultaneously with pressurizer low level indications during past tube failure events and generally provide the earliest diagnosis of a steam generator tube rupture. As primary coolant accumulates in the affected steam generator, normal feedwater flow is automatically reduced to compensate for high steam generator level. Consequently, a mismatch between steam flow from and feedwater flow to the affected steam generator may be observed. This potentially provides early confirmation of a tube failure event and also identifies the affected steam generators. However, such a mismatch may not be noticeable for smaller tube failures because of the relatively large normal feedwater/steam flow rates. The water level in the affected steam generators may not be significantly greater than that of the intact steam generators prior to reactor trip as the normal feedwater control system automatically compensates for changes in steam flow rate and

(6) Because of the magnitude of primary-to-secondary leakage during a SGTR, plant power level does not significantly affect radiation monitor response.

(7) Will not respond if isolated as part of SI or another SGBD isolation actuation signal.

(d) SG Tube Leakage Monitors (RE-724 & RE-810)

(1) IPC calculated **sensitive** radiation monitors that are calibrated to detect steam generator tube leakage at low powers and low levels (GPD indication)

(2) Post trip (< 16% power), RE-724 radiation monitor calculation BAD

(3) These radiation monitors can be used as a diagnostic tool when accompanied by SG levels rising in an uncontrolled manner. Transitions should not be made solely on a high reading on these radiation monitors

f. Step 6 Caution: At least one SG should be maintained available for RCS cooldown

Obj. V-LO-LP-37311-11

Bases: To alert the operator that a feed flow and a steam release path must be maintained from at least one steam generator in order to cool the RCS. If no intact steam generator is available, steam release must be maintained from either a ruptured or faulted steam generator to cool the RCS to RHR system operating conditions. If a ruptured steam generator is selected, steam release from that steam generator should not be isolated as directed in the following step.

g. Step 6, 7, 8, and 9: Isolate ruptured SG(s):

Step 27 provides minimum dp of 250 psi between ruptures and intact SG used for cooldown

Bases: Isolation of the ruptured steam generator(s) effectively minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary-to-secondary leakage. In order to remove heat generated in the primary system the ruptured steam generator pressure and RCS pressure must be maintained greater than the non-ruptured steam generator pressures. Secondly, as this pressure differential is increased, so is the subcooling in the primary system.

If sufficient pressure differential CANNOT be maintained, leakage from the RCS will continue since RCS pressure will remain greater than the ruptured steam generator pressure in order to remove decay heat. In that case, the operator is directed to 19131-C, ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED to minimize this leakage.

1) If MSIVs and bypass valves of the ruptured SG can not be closed

a) The ruptured SG is isolated from the intact SGs by isolating closing the MSIVs and bypass valves of the intact SGs. RCS cooldown will then be using the ARVs from the intact SGs.

ORIGIN	**SETPOINT**	
SJAE Rad Monitor	As determined by Chemistry Department	1-RE-0810 (High)

NOTES

- For other than HIGH conditions see Pages 4 and 5.
- The IPC calculates primary-to-secondary leakage using the output of RE-0810 and SJAE exhaust flow, and displays GPD and GPD/HR on the IPC. For this indication to be valid IPC constant K6422 (SJAE in-service flag) must be updated for any change in SJAE status.
- High alarm is only active when the 10 minute average of the signal (UR6810) exceeds 5 GPD.
- The IPC indication of GPD and GPD/HR from RE-0810 will be "BAD" when the mechanical vacuum pumps are in-service.

1.0 **PROBABLE CAUSE**

Steam Generator Tube leakage

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

1-RE-0810
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Evaluate** plant parameters to determine if a Steam Generator Tube leak is indicated:

   a. VCT makeup frequency and/or Charging flow has increased.

   b. Pressurizer level and/or pressure has decreased.

   c. Steam Flow/Feed Flow mismatch and SG level response.

   d. Other secondary system radiation monitors indicate increasing radiation level.

      1-RE-0724

      1-RE-12839C

      1-RE-0019

      1-RE-0021

      Any Steam Line Rad Monitor

2. **Obtain** detector trend data per 13508-1, "Radiation Monitoring Systems".

3. **Request** Chemistry to sample and count Steam Generators and Condenser Off-Gas for activity.

4. **Notify** Health Physics of the alarm.

5. IF plant parameters indicate the presence of a Steam Generator Tube Leak OR indicate that an existing SGTL has increased in size, **initiate** 18009-C, "Steam Generator Tube Leak".

6. IF a SGTL is NOT indicated:

   a. **Monitor** the channel for further changes.

   b. IF sampling and analysis determine the channel has malfunctioned, **request** Chemistry to deactivate the channel.

1-RE-0810
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X3D-AA-F14B

34. 038EK1.03 001/1/1/SGTR - NAT CIRC/C/A - 3.9/4.2/BANK-FARLEY 2004/HL-18 NRC/RO/SRO/AML

Given the following:

- A SGTR has occurred on Unit 1.
- 19030-C, "Steam Generator Tube Rupture," is in progress.
- All RCPs have been tripped.
- Natural circulation has been verified.

Which ONE of the following could be a direct result of the step, "Depressurize RCS using a PRZR PORV to refill PRZR," per 19030-C?

A. A rapid rise in containment pressure due to overpressurization of the PRT and subsequent rupture disc failure.

B. A rapid drop in the cold leg temperature due to the loop being stagnant during the pressure reduction.

C. A rapid drop in the ruptured SG level due to backfill from the SG into the RCS.

D.✓ A rapid rise in pressurizer level due to Reactor Vessel steam voiding.

038EK1.03 Steam Generator Tube Rupture (SGTR)

Knowledge of the operational implications of the following concepts as they apply to the SGTR:
(CFR: 41.8 / 41.10 / 45.3)

Natural Circulation.

K/A MATCH ANALYSIS:

The question requires the knowledge of the implications of what might happen upon performing the depressurization step during SGTR.

DISTRACTOR ANALYSIS

A. Incorrect. If the PRT would rupture due to the use of the PORVs, which is highly unlikely, Ctmt pressure would not rise rapidly. It would be a very small rise in pressure, humidity and temperature.

B. Incorrect. The RCS CL temperature may rapidly decrease due to the SI flow going into the stagnant loop cold leg.

C. Incorrect. The ruptured SG level will rise versus drop due to backfill from the SG into the RCS.

D. Correct. A rapid rise in pressurizer level due to reactor vessel steam voiding. This is a caution prior to step 19 of EEP-3.0.

REFERENCES

19030-C, E-3 Steam Generator Tube Rupture, page # 20 note prior to step # 34.

Farley 2004 NRC Exam question

VEGP learning objectives:

LO-LP-36101-04, State three conditions required to establish natural circulation flow.

LO-LP-36101-14, State, in order of effectiveness, alternate methods of cooling available to the operator during accident conditions.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

The Upper Head region of the vessel may void during RCS depressurization if RCPs are not running. This will result in a rapidly rising PRZR level.

CAUTIONS

- The PRT may rupture if a PRZR PORV is used to depressurize the RCS. This may result in abnormal Containment conditions.
- Cycling of the PRZR PORV should be minimized.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
34. Depressurize RCS using a PRZR PORV to refill PRZR:	
__a. Arm one available train of COPS and check PRZR PORV Block Valve - OPEN.	__a. Open PRZR PORV Block Valve.
__b. Open one PRZR PORV.	__b. Go to Step 35.
__c. Go to Step 37.	
__35. Check at least one SI Pump - RUNNING.	__35. Go to 19133-C, ECA-3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.
__36. Establish Auxiliary Spray by performing the following:	__36. IF Auxiliary Spray can NOT be established, THEN go to 19133-C, ECA-3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.
__a. Verify PRZR Heaters - OFF.	
__b. Verify at least one CCP running.	

Step 36 continued on next page

1. 038EK1.03 001/1/1/SGTR/MEM 3.9/4.2/NEW/FA011005/RO/

Given the following:

- A SG tube rupture has occurred on Unit 1.
- EEP-3, Steam Generator Tube Rupture, has been entered.
- ALL RCPs have been tripped.

Which one of the following could be a result of the step, "Reduce RCS pressure using pressurizer PORV to minimize break flow and refill pressurizer" per EEP-3?

A. A rapid drop in core delta T as natural circulation flow is enhanced.

B✓ A rapid rise in pressurizer level due to reactor vessel steam voiding.

C. A rapid drop in the cold leg temperature due to the loop being stagnant during the pressure reduction.

D. A rapid rise in containment pressure due to overpressurization of the PRT and subsequent rupture disc failure.

35. 039A2.04 001/2/1/MS - SD MALFUNCT/C/A - 3.4/3.7/MOD - CALLAWAY 04/HL-18 NRC/RO/SRO/AML

Given the following conditions:

- Unit 1 is 100% power.
- A Steam Dump valve failed open and cannot be closed.

Which ONE of the following describes the correct plant response and the required operator action in accordance with 18008-C, "Secondary Coolant Leakage"?

A. Turbine power increases. Withdraw control rods to match Tavg and Tref.

B✓ Reactor power increases. Reduce turbine load to stabilize power <100%.

C. Turbine power increases. Reduce turbine load to stabilize power <100%.

D. Reactor power increases. Insert control rods to match Tavg and Tref.

039A2.04 Main and Reheat Steam System (MRSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Malfunctioning steam dump.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a steam dump valve fails open. The candidate must determine the correct effect on reactor or turbine power + the correct operator response.

DISTRACTOR ANALYSIS:

A. Incorrect. Turbine power will lower due to steam bypassing the turbine. The operator action half is correct if turbine power is causing Tavg to lower.

B. Correct. Reactor power will increase due to raised steam demand, per 18008-C, Secondary Steam Leak, turbine load reduction is the proper method to reduce power.

C. Incorrect. Turbine power will lower due to steam bypassing the turbine per 18008-C, Secondary Steam Leak, turbine load reduction is the proper method to

reduce power.

D. Incorrect, reactor power will increase, per 18008-C, Secondary Steam Leak, the proper method to reduce power is lowering turbine load.

REFERENCES:

18008, "Secondary Coolant Leakage"
Callaway 2004 NRC Exam

VEGP learning objectives:

LO-LP-60308-03 Given the entire AOP, describe:

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

LO-LP-60308-04 Discuss the parameters that distinguish primary coolant leakage from secondary coolant leakage.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. Perform the following as necessary:

   - Reduce Turbine load if any of the following indications exceed 100% power:

     UQ1118 (GREATER THAN 100% MWT)

     NIs

     ΔTs

   - Isolate the leak.

   - IF leakage is such that significant hazard to personnel or equipment exists OR leakage rate is unstable and is worsening, THEN:

     1) Trip the reactor.

     2) WHEN reactor trip is verified, THEN close MSIVs and BSIVs.

     3) Go to 19000 - C, E - 0 REACTOR TRIP OR SAFETY INJECTION.

2. Initiate the Continuous Actions Page.

Examination Outline Cross-reference:	**Level**	**RO**	**SRO**
	Tier #	2	2
	Group #	1	1
	K/A #	039A2.04	
	Importance Rating	3.4	3.7

Proposed Question:

The plant is operating at 100% power.

The 'C' Steam Generator Atmospheric Steam Dump fails OPEN and CANNOT be closed.

Which ONE of the following describes the plant response and required operator action?

A. Reactor Power DECREASES. Withdraw Control Rods to match TAVG and TREF.

B. Reactor Power INCREASES. Reduce Turbine load to stabilize power <3565 MW.

C. Reactor Power DECREASES. Raise Turbine load as required to restore full power.

D. Reactor Power INCREASES. Insert Control Rods to match TAVG and TREF.

Proposed Answer: B

Explanation:

Q. Incorrect-Power increases. Reduction in steam flow is required
R. Correct- Power increases. Reduction in steam flow is required
S. Incorrect-Power increases. Reduction in steam flow is required
T. Incorrect-Reduction in steam flow is required

36. 054AA1.04 001/1/1/LOSS MFW - HPI/C/A - 4.4/4.5/MOD - HL17 NRC/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- 19231-C, "Response to Loss of Secondary Heat Sink," is in use.
- Bleed and Feed has been initiated.
- SIPs are NOT available.
- Both CCPs are running.
- PORV-455 is CLOSED.
- PORV-456 is OPEN.

Current conditions:

- ALB07-C06 CHARGING PUMP OVERLOAD TRIP illuminates.
- 'B' CCP handswitch green and amber lights are LIT.

Based on the current conditions, which ONE of the following completes the following statement?

Per 19231-C, the minimum requirement for the RCS Feed path ___(1)___ met,

and

the minimum requirement for the RCS Bleed path ___(2)___ met.

	__(1)__	__(2)__
A.	is	is
B✓	is	is NOT
C.	is NOT	is
D.	is NOT	is NOT

054AA1.04 Loss of Main Feedwater (MFW)

Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW):
(CFR: 41.7 / 45.5 / 45.6)

High Pressure Injection, under total feedwater loss conditions.

<u>K/A MATCH ANALYSIS:</u>

The question meets the KA since the candidate has to take given information to

determine that one CCP is available and have the knowledge if 19231-C feed and bleed is in effect that one CCP or SIP available meets the minimum ECCS pumps required for RCS feed. The candidate also has to determine with only one PORV open that minimum RCS bleed path is not met.

DISTRACTOR ANALYSIS:

A. Incorrect. Feed and bleed criteria ARE met with one CCP available. The minimum amount of high head pumps required for adequate feed path is one CCP or SIP. One PORV is inadequate heat removal.

B. Correct. Feed and bleed criteria ARE met with one CCP available. The minimum amount of high head pumps required for adequate feed path is one CCP or SIP. One PORV is inadequate heat removal.

C. Incorrect. Feed and bleed criteria ARE met with one CCP available. The minimum amount of high head pumps required for adequate feed path is one CCP or SIP. One PORV is inadequate heat removal.

D.Incorrect. Feed and bleed criteria ARE met with one CCP available. The minimum amount of high head pumps required for adequate feed path is one CCP or SIP. One PORV is inadequate heat removal.

REFERENCES:

19231-C, "Response to Loss of Secondary Heat Sink", steps 5 and 36.
HL-17 NRC Exam
WOG Background Document for 19231-C, FR-H.1 Response to Loss of Secondary Heat Sink

VEGP learning objectives:

LO-LP-37051-11 Define loss of secondary heat sink in accordance with 19231-C, Response to Loss of Secondary Heat Sink, requiring immediate initiation of bleed and feed control.

LO-LP-37051-08 Using EOP 19231 as a guide, briefly describe how each major step is accomplished. Describe the bases for each.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RCS WR temperature - GREATER THAN 350°F.	b. Try to place the RHR system in service by initiating 13011, RESIDUAL HEAT REMOVAL SYSTEM. IF adequate cooling with the RHR system is established, THEN return to procedure and step in effect.
***5. Check CCP status - AT LEAST ONE AVAILABLE.**	*5. Stop all RCPs. Go to Step 35.
***6. Check if RCS bleed and feed is required:**	
a. Check the following: WR level in any 3 SGs - LESS THAN 29% [44% ADVERSE]. -OR- RCS pressure - GREATER THAN 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK	a. WHEN criteria for bleed and feed are met, THEN perform Steps 6.b and 6.c. Go to Step 7.
b. Trip all RCPs.	
c. Go to Step 35 and perform bleed and feed actions.	
7. Place Containment Hydrogen Monitors in service by initiating 13130, POST -ACCIDENT HYDROGEN CONTROL.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
33. Check for loss of secondary heat sink: WR level in any 3 SGs - LESS THAN 29% [44% ADVERSE]. -OR- RCS pressure - GREATER THAN 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK.	33. Return to Step 4.

CAUTION

Steps 35 thru 38 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
34. Initiate Continuous Actions Page For After Establishing Bleed And Feed.	
35. Verify SI actuated.	
36. Verify RCS feed path: a. Verify ECCS Pump status: CCPs - AT LEAST ONE RUNNING. -OR- SI Pumps - AT LEAST ONE RUNNING. b. Verify ECCS valve alignment - PROPER INJECTION LINEUP INDICATED ON MLBs.	36. Start pumps and align valves as necessary to establish injection flow using ATTACHMENT A or ATTACHMENT B. <u>IF</u> a feed path can <u>NOT</u> be established, <u>THEN</u> continue attempts to establish feed flow. Return to Step 10.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

During bleed and feed operation the PRT may rupture.

37. Establish RCS bleed path:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Place all PRZR Heaters in OFF/PTL.	
b. Check power to PRZR PORV Block Valves - AVAILABLE.	b. Restore power to block valves.
c. Arm COPS and check PRZR PORV Block Valves - BOTH OPEN.	c. Open both PRZR PORV Block Valves.
d. Open both PRZR PORVs.	

about 40 lbm/sec (290 gpm), with both trains operating, at an RCS pressure of 2300 psig. Since makeup flow from the charging/SI pump system will not keep up with inventory lost out of the pressurizer PORVs, the RCS will eventually dry out enough to cause core uncovery.

In summary, the loss of all feedwater transient from a power condition without operator action will lead to a loss of secondary heat sink followed by a loss of RCS inventory through the pressurizer PORVs. Core uncovery will result at an RCS pressure equal to or greater than the pressurizer PORV setpoint and charging/SI flow, if manually initiated late in the transient, will not be sufficient to prevent core uncovery.

2.2 RCS Bleed and Feed Heat Removal

For a loss of all secondary heat sink, operator action to establish RCS bleed and feed heat removal can prevent or minimize core uncovery. To establish RCS bleed and feed heat removal the operator must initiate and verify high pressure SI flow to feed subcooled fluid to the RCS and then manually open all pressurizer PORVs to bleed hot reactor coolant out of the RCS. To be certain that the bleed and feed heat removal path will be effective, typically at least two PORVs must open.

The effectiveness of RCS bleed and feed heat removal depends on four basic considerations. These are: 1) the timeliness of operator action to initiate bleed and feed following indications of the symptoms of loss of all secondary heat sink (see subsections 2.2.3 and 2.2.4), 2) the core decay heat at the time of RCS bleed and feed initiation, 3) the capacity of the pressurizer PORVs (i.e., number and size of valves), and 4) the capacity of the high pressure SI system (i.e., number, size, and shutoff head of the high pressure SI pumps). These considerations govern the RCS depressurization, repressurization and pressure stabilization after RCS bleed and feed heat removal is established. The fourth consideration also governs the amount of SI flow delivered to the RCS at any RCS pressure. RCS bleed and feed effectiveness is maximized by a combination of these considerations which maximizes the initial RCS depressurization, minimizes the subsequent RCS repressurization and the pressure

1. WE05EK2.1 001/1/1/LOHS - COMPONENTS/H-3.7/3.9/NEW/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

- 19231-C, "FR-H.1 Response to Loss of Secondary Heat Sink" is in progress.

- RCS Bleed and Feed has been initiated.

Current conditions:

- One SIP is running.
- The CCPs and other SIP are NOT available.

- One PRZR PORV is open.
- The other PORV is NOT available.

Per 19231-C, which one of the following correctly completes the following statement?

One SIP running is ___(1)___ for the RCS Feed path

and

one PRZR PORV open is ___(2)___ for the RCS Bleed path.

For the purposes of this question, "adequate" as defined by procedure 19231-C means that 19231-C will NOT direct further adjustments to the RCS Feed or Bleed path.

	__(1)__	__(2)__
A.	adequate	adequate
B✓	adequate	NOT adequate
C.	NOT adequate	adequate
D.	NOT adequate	NOT adequate

37. 056AK3.02 001/1/1/LOSP - EOP ACTIONS/MEM - 4.4/4.7/NEW/HL-18 NRC/RO/SRO/KAJ

Given the following:

- A loss of all AC occurred and 19100-C, "Loss of All AC Power," is entered.

- A depressurization of all SGs at the maximum rate is in progress.

Which ONE of the following completes the following statement?

The reason for stopping the SG depressurization at 300 psig is to prevent ________.

A. a steam bubble from forming in the Reactor Vessel Head

B✓ N_2 injection into the RCS from the ECCS Accumulators

C. challenging the integrity critical safety function

D. a rapid loss of pressurizer level

056AK3.02 Loss of Offsite Power

Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power:
(CFR: 41.5 / 41.10 / 45.6 / 45.13)

Actions contained in EOP for loss of offsite power.

K/A MATCH ANALYSIS:

The question asks the basis for lowering SG pressures to 300 psig and then stopping during a loss of all AC power while in 19100-C, Loss Of All AC Power.

DISTRACTOR ANALYSIS:

A. Incorrect. While it is plausible a steam bubble may occur in the reactor vessel head, the reason is to prevent N_2 injection into the RCS.

B. Correct. Prevention of N_2 injection into the RCS is the reason.

C. Incorrect. It is plausible the candidate may confuse the 300 psig with the 280°F requirement for stopping to prevent a challenge to the integrity safety function.

D. Incorrect. Pressurizer level may rapidly lower during this evolution, however, a note in the procedure says this is NOT a reason to stop the depressurization.

REFERENCES:

19100-C, ECA-0.0 Loss of All AC Power.
WOG Background Document for ECA-0.0 Loss of All AC Power

VEGP learning objectives:

LO-LP-37031-05 State the special concerns regarding the following items should the operator begin the secondary side depressurization:

a. return to critical condition
b. introduction of non-condensible gases

LO-LP-37031-07 State the bases for "Loss of All AC Power" procedure.

LO-LP-37031-09 Given a NOTE or CAUTION statement from the EOP, state the bases for that NOTE or CAUTION statement.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- The SGs should be depressurized at maximum rate to minimize RCS inventory loss.
- PRZR level may be lost and Reactor Vessel Upper Head voiding may occur due to depressurization of the SGs. Depressurization should not be stopped to prevent these occurrences.

*29. **Depressurize intact SGs to 300 psig:**

a. Check SG NR levels - GREATER THAN 10% [32% ADVERSE] IN AT LEAST ONE SG.

a. Perform the following:

1) IF all SG NR levels less than 10% [32% ADVERSE], THEN maintain maximum TDAFW flow.

2) WHEN NR level in at least one SG greater than 10% [32% ADVERSE], THEN go to Step 29.b.

Go to Step 33.

Step 29 continued on next page

CAUTION: SG pressures should not be decreased to less than (0.07) psig to prevent injection of accumulator nitrogen into the RCS.

PURPOSE: To alert the operator that steam generator pressures must be maintained above the specified limit

BASIS:

Steam generators should be depressurized to maximize delivery (into the RCS) of the water contained in the SI accumulators while minimizing delivery of nitrogen. Maintaining steam generator pressures above a value that prevents introduction of a significant volume of nitrogen into the RCS ensures that accumulator nitrogen will not impede natural circulation.

A steam generator pressure limit is set to preclude significant nitrogen injection into the RCS. To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tank pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume. The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^{\gamma} = P_2 V_2^{\gamma}$$

where γ = 1.25 for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta P from P_2. 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.

ACTIONS:

Determine if SG pressures are greater than (0.07) psig

INSTRUMENTATION:

SG pressure indication for each SG

38. 059G2.2.22 001/2/1/MFW - LCO & SL/C/A - 4.0/4.7/NEW/HL-18 NRC/RO/SRO/TNT

Given the following:

- Unit 1 is in Mode 2.

Which ONE of the following conditions in the Main Feedwater System will cause the MFW system to be declared inoperable per Tech Specs?

A. A BFRV will NOT close; the BFRV has been isolated with a closed manual valve.

B. The UO places a MFRV in MANUAL control on the Operator Work Station (OWS).

C.✓ Accumulator Gas pressures for Loop 1 MFIV are reading low (alarm illuminated).

D. During AFW to MFW swapover, one MFPT is feeding forward, the other is tripped.

059G2.2.22 Main Feedwater (MFW)

Knowledge of limiting conditions for operations and safety limits: (CFR: 41.5 / 43.2 / 45.2)

K/A MATCH ANALYSIS:

The candidate is presented with plausible situations regarding the MFW system. The candidate will be required to determine which of the situations will render the MFW system to be inoperable.

DISTRACTOR ANALYSIS:

A. Incorrect. With the manual bypass valve closed, the MFW system is operable. It is plausible the candidate may think MFW is inoperable. Per Tech Spec LCO 3.7.3, if the valve is manually isolated, the LCO is met and the safety function is met.

B. Incorrect. Taking manual control at the OWS does not prevent the MFRV from automatically performing its safety function by closing when designed.

C. Correct. With both accumulator N2 pressures < 4800 psig, the MFIV is considered inoperable. ARP 17016-1, NOTE states a minimum of 4800 psig in both accumulators is necessary to satisfy Tech Spec operability requirements. The alarm comes in when the setpoint of 4800 psig is reached.

D. Incorrect. During the AFW to MFW swapover, with one MFPT feeding forward, the other MFPT has to be tripped to allow an auto AFW actuation on trip of both MFPTs. If the 2nd MFPT was tripped, then a Tech Spec LCO entry is required. per TS 3.3.2, ESFAS, FU 6d. Also, SOP 13615 Condensate and Feedwater for MFPT startup mentions LCO 3.3.2 FU 6d entry is required.

REFERENCES:

TS LCO 3.7.3 and Bases for Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves.
TS LCO 3.3.2, FU 6d, for Auxiliary Feedwater, Trip of all Main Feedwater Pumps
17016-1, window A06 MFW LOOP 1 MFIV ACCUM GAS LO PRESS
11874-1, Control Room Rounds Sheets

VEGP learning objectives:

LO-LP-39211-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

WINDOW A06

ORIGIN	SETPOINT	
1-PSL-5227A 1-PSL-5227B	4800 psig	MFW LOOP 1 MFIV ACCUM GAS LO PRESS

1.0 **PROBABLE CAUSE**

1. Depletion due to use
2. System leakage
3. Malfunction of the Oil Pump or loss of air to the pump
4. Loss of instrument loop power
5. Excessive leak-by through Thermal Relief Valve

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Verify** Main Feedwater Isolation Valve (MFIV), 1-HV-5227, in the desired position.
2. **Determine** MFIV accumulator pressure by selecting "SG-1" on 1-HS-5227G and 1-HS-5227H.
3. **Verify** instrument loop power as follows:
   a. 1NYK1-24 STRAIN GA AMPL/CDTR 1-1305-P5-FW1/FW2 BRKR is ON.
   b. 1NCPFW1, Aux Bldg room A14, ON/OFF Switch is ON.
   c. 1NCPFW2, Aux Bldg Room A17, ON/OFF Switch is ON.

WINDOW A06
(Continued)

CAUTION

MFIV hydraulic fluid (Fyrquel) may decompose at temperatures above 149°F to release toxic materials. If a hydraulic fluid leak is found to be in contact with hot piping and producing vapors, proper protection including respirator and gloves should be utilized. Wash unprotected skin thoroughly after exposure.

4. **Dispatch** an operator to **investigate** and **determine** cause of the alarm. IF QPCP pressure indications are unavailable, **request** the operator to **check** the pressure locally on the accumulator gauges.

5. IF cause of low pressure is NOT determined in Step 4, **request** Maintenance to **inspect** for Thermal Relief Valve leak-by.

6. IF leak-by is detected, **close** the manual isolation valve (1HV-5227-H Valve - A Train, 1HV-5227-H1 Valve - B Train) for the leaking relief valve and **log** in the Unit 1 Out of Position Log.

NOTE

The actuator requires a minimum of 4800 psig N2 pressure in both accumulators to satisfy Tech Spec operability requirements.

7. IF no abnormal conditions or leaks are found, **perform** an MFIV Accumulator Pressure Reduction per 13615-1, "Condensate And Feedwater System," in an attempt to verify the MFIV Air Pump is not bound and the Accumulator Thermal Relief Valve is seating properly.

8. IF pressure CANNOT be restored to the minimum of 4800 psig, **contact** Maintenance and **perform** an MFIV Accumulator Precharge Check per 13615-1, "Condensate And Feedwater System."

9. IF pressure CANNOT be restored, refer to Technical Specification LCO 3.7.3.

10. IF equipment failure is indicated, **initiate** maintenance

WINDOW A06
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

IF annunciator is in solid, **verify** accumulator gas pressures greater than 4800 psig once per shift by reading on 1PI-5227A and 1PI-5227B on QPCP.

END OF SUB-PROCEDURE

REFERENCES: 1X4DB168-3, 1X5DV312, 1X5DV313, CX5DT101-124A, CX5DT101-124A1
AX4AR19-00037-12

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

LCO 3.7.3 Four MFIVs, four MFRVs, and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 except when MFIV, MFRV, or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Close or isolate MFIV. <u>AND</u> A.2 Verify MFIV is closed or isolated.	72 hours Once per 7 days
B. One or more MFRVs inoperable.	B.1 Close or isolate MFRV. <u>AND</u> B.2 Verify MFRV is closed or isolated.	72 hours Once per 7 days

(continued)

BASES

LCO (continued)	This LCO requires that four MFIVs and associated bypass valves and four MFRVs and associated bypass valves be OPERABLE. The MFIVs are provided with dual pneumatic/hydraulic power trains each receiving a feedwater isolation signal from separate ESFAS actuation logic trains. Actuation of either pneumatic/hydraulic power train will cause the MFIVs to close. The MFRVs are equipped with dual solenoids to actuate the valve on a feedwater isolation signal. Each solenoid gets an actuation signal from separate ESFAS actuation logic trains. The solenoid logic for the MFRVs requires both solenoids to actuate for the MFRVs to isolate. The redundancy built into the MFIV closure system prevents any single failure other than a mechanical failure of the valve itself from preventing the MFIV from performing its design function. If a mechanical failure of an MFIV does occur, it becomes the assumed single failure, and the MFRVs would be assumed to perform their isolation function. If an MFRV fails to actuate due to a mechanical failure of the valve itself, or a failure of one train of actuation logic, this becomes the assumed single failure, and the MFIVs would be assumed to perform their isolation function. The MFIVs and MFRVs and the associated bypass valves are considered OPERABLE when isolation times are within limits and capable of closing on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high steam generator level occurs due to an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MFIVs and MFRVs and the associated bypass valves must be OPERABLE whenever there is significant energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1 and 2, the MFIVs and MFRVs and the associated bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function. In MODES 3, 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and the associated bypass valves are normally closed since MFW is not required.

(continued)

Table 3.3.2-1 (page 6 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Trip of all Main Feedwater Pumps	1,2(g)	1 per pump	J	SR 3.3.2.6	NA	NA
7. Semi-automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4(h)	2	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. Refueling Water Storage Tank (RWST) Level-Low Low(j)	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≤ 216.6 in. and ≥ 210.4 in.	213.5 in.
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(g) When the Main Feedwater System is operating to supply the SGs.

(h) In MODE 4, only 1 train is required to be OPERABLE to support semi-automatic switchover for the RHR pump that is required to be OPERABLE in accordance with Specification 3.5.3, ECCS-shutdown.

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

FIGURE 1 - UNIT 1
DATA SHEET

Sheet 5 of 9

EQUIPMENT/COMPONENT	PARAMETER		INSTRUMENT	MAXIMUM	MINIMUM	SHIFT DAY	SHIFT NIGHT	COMMENT NUMBER
H2 UNIT 1	PRESS (PSIG)		A-PI-9708	130	80			
N2 SPLY TURB BLDG	PRESS (PSIG)		A-PI-9718	150	90			
RIVER CANAL	LEVEL (FT) (17)		A-LI-27623A OR A-LI-27623B		73'			
CW INTAKE STRUCTURE	LEVEL (FT)		1-LI-27268	32.5	31			
FIRE PROTECTION	NO STORAGE TANK #2 (FT)		C-LI-7955	30	28			
	SO STORAGE TANK #1 (FT)		C-LI-7956	30	28			
STEAM GENERATOR BLOWDOWN	SG1 (GPM)		1-FI-1171B	90				
	SG2 (GPM)		1-FI-1172B	90				
	SG3 (GPM)		1-FI-1173B	90				
	SG4 (GPM)		1-FI-1174B	90				
	INLET FLTR (GPM)		1-FI-1152B	360				
MFIV ACCUM N2 PRESSURE	PRESS (PSIG)	SW POS						
	(18)	SG-1	1-PI-5227A	5600	4850			
		SG-2	1-PI-5227A	5600	4850			
		SG-3	1-PI-5227A	5600	4850			
		SG-4	1-PI-5227A	5600	4850			
		SG-1	1-PI-5227B	5600	4850			
		SG-2	1-PI-5227B	5600	4850			
		SG-3	1-PI-5227B	5600	4850			
		SG-4	1-PI-5227B	5600	4850			
PLANT FAULT RECORDER 1NCQFRP	COOLING FAN RUNNING							

EQUIPMENT/COMPONENT	PARAMETER	REQUIRED STATUS	SHIFT DAY	SHIFT NIGHT	COMMENT NUMBER
SEISMIC MONITORING SYSTEM (19)	SYSTEM STATUS	AUTOMATIC MONITORING			
	CURRENT TIME	DATE AND TIME CORRECT, SECONDS CHANGING			
	NEXT SENSOR TEST	"NONE"			
	EVENT ALARM	GREEN			
	OBE EXCEEDENCE	GREEN			
	SYSTEM HEALTH	GREEN			
	(blinking indicator)	GREEN			
	R01-R07	GREEN			

(17) IF RIVER CANAL LEVEL DROPS TO LESS THAN 78' AS INDICATED ON A-LI-27623A & A-LI-27623B, INITIATE AN MWO AND REQUEST MAINTENANCE TO RAISE RIVER INTAKE WEIR GATE PILING SECTIONS PER DRAWING AX2D13V003. DRAWING AX2D13V003 PROVIDES DIRECTIONS TO RAISE THREE MOVABLE SECTIONS OF WEIR GATE PILINGS SEQUENTIALLY AT THE MOUTH OF THE INTAKE STRUCTURE IF RIVER LEVEL FALLS TO 77'6", 76', AND 75'.
IF RIVER LEVEL DROPS TO LESS THAN 72'8" SHUTDOWN RIVER WATER PUMPS PER 13727-C

(18) NOTIFY MAINTENANCE IF MFIV ACCUM N2 PRESSURE IS LESS THAN MINIMUM. IF MAXIMUM PRESSURE IS EXCEEDED, NOTIFY MAINTENANCE AND THE SYSTEM ENGINEER TO DETERMINE CAUSE OF HIGH PRESSURE. IF REQUIRED, REDUCE PRESSURE PER 13615-1.

(19) IN THE EVENT OF ANY ANOMALY, CONTACT I&C.

COMMENTS: ____________________

39. 061K6.01 001/2/1/AFW - CONTROLLERS/C/A - 2.6/2.7/NEW/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- The Unit 1 TDAFW pump has received an auto start signal.

Current conditions and sequence of events:

- The UO was attempting to control TDAFW speed by reducing demand on 1PDIC-5180, TDAFW Pump Speed Controller.

- ALB16-F03 AFW TURB OVERSPEED MECH TRIP illuminates.

- The auto start signal is still present.

Which one of the following completes the following statement?

Based on the current conditions, the TDAFW Pump Trip and Throttle Valve (T&T) handswitch amber light ___(1)___ be lit,

and

per 13610-1, "Auxiliary Feedwater System," to reset the governor ramp control circuit requires holding HV-5106, TDAFW Pump Steam Supply Valve, closed and ___(2)___.

A.✓ (1) will

(2) manually opening the Trip and Throttle Valve (T&T)

B. (1) will

(2) manually raising 1PDIC-5180 to 100% demand

C. (1) will NOT

(2) manually opening the Trip and Throttle Valve (T&T)

D. (1) will NOT

(2) manually raising 1PDIC-5180 to 100% demand

061K6.01 Auxiliary / Emergency Feedwater (AFW) System

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components:
(CFR 41.7 / 45.7)

Controllers and positioners.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a TDAFW overspeed trip has occurred. The candidate has to determine the status of the T & TV amber light and the proper way to reset the governor controller.

DISTRACTOR ANALYSIS:

A. Correct. The T & TV amber light will illuminate on a mechanical overspeed trip. The proper way to reset the governor control circuit is to hold HV-5106 shut and cycle the T & TV open.

B. Incorrect. The T & TV amber light will illuminate on a mechanical overspeed trip. Raising PDIC-5180 speed controller to 100% is the normal position but is not required to reset the governor control circuit.

C. Incorrect. The T & TV amber light will illuminate on a mechanical overspeed trip. The proper way to reset the governor control circuit is to hold HV-5106 shut and cycle the T & TV open.

D. Incorrect. The T & TV amber light will illuminate on a mechanical overspeed trip. Raising PDIC-5180 speed controller to 100% is the normal position but is not required to reset the governor control circuit.

REFERENCES:

17016-1, window F03 AFW TURB OVERSPEED MECH TRIP
13610-1, "Auxiliary Feedwater System", section 4.4.7 for Resetting of TDAFW Pump Trip and Throttle Valve Following an Overspeed Trip Actuation (Actual Overspeed or Surveillance).

VEGP learning objectives:

LO-PP-20101-07 Describe the control room indications of a TDAFW pump over speed trip.

WINDOW F03

ORIGIN	SETPOINT	AFW TURB OVERSPEED MECH TRIP
AFW Turbine Overspeed Relay Contact, 4CR	4830 ±50 rpm	

1.0 PROBABLE CAUSE

Overspeed trip

2.0 AUTOMATIC ACTIONS

Trip and Throttle (T&T) Valve 1-PV-15129 closes.

3.0 INITIAL OPERATOR ACTIONS

If required, **verify** adequate flow to maintain Steam Generator levels.

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Verify** Turbine Driven Auxiliary Feedwater (TDAFW) Pump Speed 1-SI-15109A decreasing.

2. IF TDAFW flow is required, **restart** the pump as follows:

   a. **Dispatch** an operator to the TDAFW Pump to **reset** the mechanical overspeed trip linkage per 13610-1.

   b. **Initiate** maintenance, if required, to **correct** the cause of the overspeed.

   c. **Restart** the TDAFW Pump per 13610-1, Auxiliary Feedwater System."

3. Refer to Technical Specification LCO 3.7.5.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB162-3, 1X3D-BC-F02X, 1X4AF03A-0001, 1X4AF03-224

INITIALS

4.4.7 Resetting of TDAFW Pump Trip and Throttle Valve Following an Overspeed Trip Actuation (Actual Overspeed or Surveillance / Maintenance Testing) (SNC11418, 1986309008)

CAUTION

If this reset of the Trip & Throttle valve is following an actual overspeed trip and the potential exists to initiate AFW flow to a "hot dry S/G", closing of the TDAFW discharge valves should be considered prior to reset of the Trip & Throttle valve. ☐

4.4.7.1 **Verify** motor actuator has driven the shaft to approximately 80% closed as indicated by T&T VALVE OPERATOR CLOSED green light lit on the local valve panel (PAFT). ________

NOTE

In the following step, the trip linkage bar may have to be moved. (Figures 3 and 4) ☐

4.4.7.2 On the governor, **lift** and **release** Tappet several times to ensure that there is no binding. ________

4.4.7.3 IF binding is observed, **contact** the Shift Supervisor. ________

4.4.7.4 **Verify** that the flat part of the tappet nut is in contact with the head lever. (See Figure 4 and 4a) ________

4.4.7.5 **Reset** the mechanical linkage by pushing the trip linkage towards the Trip And Throttle Valve and **observe** trip lever moves up (see Figures 3 and 4). ________

4.4.7.6 **Push** the Tappet down to ensure proper seating. ________

4.4.7.7 **Verify** mechanical overspeed trip indicator limit switch roller arm is properly positioned (Figure 4). ________

Critical

4.4.7.8 **Place** TDAFW Pump Steam Admission Valve 1-HV-5106 handswitch 1HS-5106A (QMCB) in CLOSE. ________

____
CV

INITIALS

> NOTE
>
> Holding 1HS-5106A in the CLOSE position allows the speed controller startup logic to reset when the T&T valve is electrically opened. ☐

4.4.7.9 IF AFW Actuation signal is present, **hold** 1HS-5106A in the CLOSE position until completion of Step 4.4.7.10. ________

> NOTE
>
> When Handswitch 1HS-15111 is placed in OPEN, the Trip And Throttle Valve latches and then opens. ☐

4.4.7.10 **Place** Handswitch 1HS-15111 (QMCB) in OPEN, THEN **release**. ________

> **CAUTIONS**
>
> - Steam Admission Valve 1-HV-5106 will open when 1HS-5106A is released if an open signal is present. ☐
> - If pump speed can not be controlled or overspeed trip occurs again when 1HS-5106 is released in the next step, do not attempt to reset again until the speed control problem has been corrected. ☐

4.4.7.11 **Verify** proper latching between the latch up lever and the trip hook lever. (Figure 3 and 3a). ________

4.4.7.12 WHEN the Trip And Throttle Valve is fully open as indicated at MLB13-4.2 OR 1HS-15111 (QMCB), **release** 1HS-5106A IF applicable. ________

40. 062AA1.02 001/1/1/LOSS NSCW - LOADS/C/A - 3.2/3.3/LOIT BANK/HL-18 NRC/RO/SRO/AML

Given the following:

- The plant is at 100% power.
- NSCW Train 'B' tagged out for piping repair.
- NSCW Train 'A' pump # 3 trips.
- NSCW Train 'A' pump # 5 cannot be started.
- 18021-C, "Loss of Nuclear Service Cooling Water," has been entered.
- No other operator actions have been taken.

Which ONE of the following is the required crew action(s) per 18021-C?

A. Place all Train 'A' NSCW pumps in PTL, Emergency Trip DG1A.

Within 7 hours, shutdown to Mode 3 per 12004-C, "Power Operation (Mode 1)."

B✓ Place all Train 'A' NSCW pumps in PTL, Emergency Trip DG1A.

Trip the Reactor, initiate 19000-C, "Reactor Trip or Safety Injection," align NSCW Train 'A' for single pump operation.

C. Allow NSCW Pump #1 to continue running, DG1A should be left in AUTO.

Within 7 hours, shutdown to Mode 3 per 12004-C, "Power Operation (Mode 1)."

D. Allow NSCW Pump #1 to continue running, DG1A should be left in AUTO.

Trip the Reactor, initiate 19000-C, "Reactor Trip or Safety Injection," align NSCW Train 'A' for single pump operation.

062AA1.02 Loss of Nuclear Service Water

Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS):
(CFR: 41.7 / 45.5 / 45.6)

Loads on the SWS in the control room.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where both NSCW trains are lost at power with 1 NSCW pump running on one train. The candidate must determine the correct action to take for the NSCW system and the DG1A. The candidate must also determine when to shutdown the plant.

DISTRACTOR ANALYSIS

A. Incorrect. 1st actions are correct. AOP-18021 has the crew plant the trip, enter E-0, and place one train NSCW in single pump operations. UOP-12004 shutdown is plausible as this would be the 7 hour Tech Spec action for both NSCW trains inoperable.

B. Correct.

C. Incorrect. Plausible to leave one pump in operation for DG1A cooling. This is an incorrect action as the pump would runout. DG1A is also emergency tripped per the AOP action. UOP-12004 shutdown is plausible as this would be the 7 hour Tech Spec action for both NSCW trains inoperable.

D. Incorrect. Plausible to leave one pump in operation for DG1A cooling. This is an incorrect action as the pump would runout. DG1A is also emergency tripped per the AOP action. The actions to trip the reactor and place one train in single pump operation is correct

REFERENCES

18021-C, "Loss of Nuclear Service Cooling Water", steps 1 thru 6.
Technical Specification 3.0.3

VEGP learning objectives:

LO-LP-60317-01: Describe how the loss of NSCW system affects the operation of the Diesel Generators.

LO-LP-60317-02: Describe the operator action(s) required if NSCW is lost and neither train can be placed in operation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check if catastrophic leakage from NSCW system – EXISTS.	1. Go to Step 6.
2. Place affected train NSCW pump handswitches in PULL-TO-LOCK.	
3. Depress both Emergency Stop pushbuttons for the affected DG.	
4. Verify proper operation of <u>UNAFFECTED</u> NSCW train:	4. <u>IF</u> neither NSCW train can be placed in normal, two pump operation, <u>THEN</u> perform the following:
• Two pumps running.	a. Trip the reactor.
• Supply header pressure greater than 70 psig: Train A: PI-1636 Train B: PI-1637	b. Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. c. Trip all reactor coolant pumps.
• Supply header temperature computer indication less than 90°F: Train A: T2601 Train B: T2602	d. Isolate letdown. e. Place one train of NSCW in single pump operation by initiating 13150, NUCLEAR SERVICE COOLING WATER SYSTEM.
• Supply header flow approximately 17,000 gpm: Train A: FI-1640B Train B: FI-1641B	f. Verify train-related CCP or NCP running and seal injection flow established using 13006, CHEMICAL AND VOLUME CONTROL SYSTEM.

Step 4 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	g. Check RCP No. 1 seal temperatures less than 220°F.
	h. <u>IF</u> RCP No. 1 seal temperatures greater than 220°F, <u>THEN</u> do <u>NOT</u> attempt to restart RCPs prior to a status evaluation.
5. Go to Step 13.	
6. Verify NSCW pumps in affected train - TWO <u>OR</u> MORE OPERATING:	6. Perform the following:
• Supply header pressure greater than 70 psig. Train A: PI-1636 Train B: PI-1637	a. Place affected train NSCW pump handswitches in PULL-TO-LOCK. b. Depress both Emergency Stop pushbuttons for the affected DG.
• Supply header flow approximately 17,000 gpm. Train A: FI-1640B Train B: FI-1641B	c. Investigate cause for trip of running pump(s).

Step 6 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

d. Verify proper operation of UNAFFECTED NSCW train:

- Two pumps running.
- Supply header pressure greater than 70 psig:

  Train A: PI-1636
  Train B: PI-1637

- Supply header temperature computer indication less than 90°F:

  Train A: TE-1642
  Train B: TE-1643

- Supply header flow approximately 17,000 gpm:

  Train A: FI-1640B
  Train B: FI-1641B

Go to Step 13.

e. IF neither NSCW train can be placed in normal, two pump operation,
THEN perform the following:

1) Trip the reactor.

2) Initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.

3) Trip all reactor coolant pumps.

4) Isolate letdown.

Step 6 continued on next page

41. 062K1.02 001/2/1/AC - EDG/C/A - 4.1/4.4/BANK-LOIT/HL-18 NRC/RO/SRO/AML

Given the following:

- An SI occurred and has NOT been reset.

- An LOSP then occurs.

- 1AA02 is powered from **DG1A**.

- 1BA03 is powered from **DG1B**.

While the DGs are operating, an electrical perturbation results in the following:

- **DG1A** 186A lockout relay energizes (Generator Differential)

- **DG1B** 186B lockout relay energizes (Phase Overcurrent)

Which ONE of the following is correct with respect to the status of power to the 4160 VAC 1E Emergency Buses at this time?

A. Both 1AA02 and 1BA03 are energized.

B. Both 1AA02 and 1BA03 are de-energized.

C. 1AA02 is energized; 1BA03 is de-energized.

D✓ 1AA02 is de-energized; 1BA03 is energized.

062K1.02 A.C. Electrical Distribution

Knowledge of the physical connections and/or cause-effect relationships between the ac distribution system and the following systems: (CFR 41.2 to 41.9)

EDG

K/A MATCH ANALYSIS:

Question gives a plausible scenario with SI occuring and not yet reset. An LOSP results in the EDG's carrying both 4160 1E emergency buses. Different overcurrent relays energize on each EDG and the candidate must determine the power supply status of both 4160 1E emergency buses.

DISTRACTOR ANALYSIS

A. Incorrect. Generator Differential (186A) lockout will emergency trip DG1A under all conditions resulting in bus AA02 de-energization. Plausible candidate may not recognize this as an emergency trip or that it will trip DG during SI condition.

B. Incorrect. DG1B or breaker would not trip on Phase Overcurrent (186B) as this trip is not active during SI conditions and output breaker would remain closed. Plausible candidate may think both DG and / or output breakers could trip with the listed conditions since the 186A lockout would trip DG1A.

C. Incorrect. As stated above 186A lockout would trip DG1A resulting in AA02 being de-energized. Plausible candidate could invert the malfunctions or not recognize the effect of SI on the DG's and breakers.

D. Correct. AA02 would de-energize on DG1A trip, BA03 would remain energized as 186B lockout does not trip DG or output breaker on SI.

REFERENCES

13145-1/2, "Emergency Diesel Generators", Precautions 2.1.3 and 2.2.18.

VEGP learning objectives:

LO-PP-11101-55, Identify the primary relays which will actuate each of the following lockout relays and how the diesel generator will respond to each normal start from SI, UV, Local Emergency Start, and Normal Start.

a. 186A lockout relay
b. 186B lockout relay

4.4.12 Removing Train A Diesel Generator And Auxiliary Systems From Service

4.4.13 Response To An Air Receiver With Dewpoint Outside Required Limits Or Depressurized

4.4.14 Swapping Compressors To Opposite Air Receivers

4.4.15 Making Adjustments To Jacket Water Level

4.4.16 Inspecting The Control Air System Following Maintenance That Isolates And Restores The Control Air System

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 A Diesel Generator must be taken out of service if any resistance to engine rotation is encountered while operating the Pneumatic Barring Device. ______

2.1.2 The following Diesel Generator Electrical Protection Relays are enabled when the Diesel Generator is paralleled to the offsite power grid (i.e., surveillance testing). These are normally bypassed during a normal start when NOT in Parallel Mode.

a. Reverse Power 132. (trips 186C Lockout Relay)

b. Underfrequency 181.

c. Negative Phase Sequence 146. (trips 186C Lockout Relay)

______

NOTE

Phase Overcurrent and Loss of Field will trip engine and breaker on a normal start. ☐

2.1.3 When operating under Emergency Start conditions, the only active Diesel Generator protective devices are:

a. Generator Differential (trips 186A lockout relay).

b. Low Lube Oil Pressure (2 out of 3 Logic).

c. Engine Overspeed.

d. Loss of Field (trips 186B lockout relay; trips only the breaker during LOSP or Emergency Start Switch starts; NOT active on SI start).

e. Phase Overcurrent (trips 186B lockout relay; trips only the breaker during LOSP or Emergency Start Switch starts; NOT active on SI start). ________

2.1.4 The Lube Oil and Jacket Water Keep-Warm Pumps and Heaters and the Generator Space Heater should be operating whenever a Diesel Generator is aligned for automatic startup. ________

2.1.5 The Maintenance Department should be notified per 00350-C, "Work Request Program" to make any changes or corrections to the governor settings. The governor Load Limit, Speed, or Speed Droop settings should NOT be altered unless:

Required by an approved test procedure.

OR

The Torque Seal has been damaged or broken. ________

2.1.6 If the Diesel Generator is in continuous operation, additional supplies of fuel oil shall be ordered on, or before, the fifth day of continuous operation. ________

2.1.7 Emergency Diesel Generators shall NOT be used for peaking service. ________

2.1.8 If the Diesel Generator is being operated in the Parallel Mode, the LOCAL-REMOTE Switch 1HS-4516 on PDG1 shall NOT be transferred to LOCAL, as this will take governor and voltage regulator out of Droop Mode. ________

2.1.9 When the Diesel Generator is paralleled to the offsite power grid, the kVAR load should be maintained OUT and less than one half of the kilowatt load as shown in Vogtle Administrative Limits in Figure 2. The System Engineer must approve operation outside Vogtle Administrative Limits. ________

2.1.10 The Diesel Generators should NOT be operated in parallel with the offsite grid for prolonged periods of time. This is to keep disturbances in the grid from affecting the Diesel Generators. ________

2.1.11 Only one Diesel Generator should be operated at a time EXCEPT during Emergency Conditions. ________

2.2.12 One DG shall be OPERABLE in Modes 5 and 6. (Technical Specification LCO 3.8.2)

2.2.13 If during a Diesel Engine start, the Fail To Start alarm comes in but the engine keeps running, the support systems will operate as if the engine was shutdown. To reset these systems, the START pushbutton must be pressed. This will stop the Keep Warm Pumps, turn off the Keep Warm Heaters, start the Crankcase Fans, and place the alarms in service that are bypassed when shutdown.

2.2.14 If a Diesel Generator has been operated for at least one hour, the Diesel Fuel Oil Day Tank shall be checked for water.

2.2.15 If a DG is being restarted following a DG failure, Checklist 1 shall be completed prior to restart except for emergency situations.

2.2.16 Due to level instrument inaccuracies, an indicated level of 82% in the Diesel Fuel Storage Tank is required to satisfy the 68,000 gallon requirement of Tech Spec 3.8.3.

2.2.17 To prevent overheating and degradation of internal parts, the DG Air-start compressors should be limited to a continuous run-time of 30 minutes, with an equal amount of shutdown time.

2.2.18 The following table lists the DG Lockout Relays and the related functions:

LOCKOUT RELAYS	PRIMARY RELAYS	BREAKER STATUS	ENGINE STATUS
186A	Generator Differential	**Trips Open** Always	**Shuts Down** Always
186B	Phase Overcurrent or Loss of Field	**Trips Open** on Normal Start, Local Emergency Start, or LOSP	**Shuts Down** on Normal Start
		Remains Closed on SI	**Remains Running** on LOSP, Local Emergency Start, or SI
186C	Reverse Power or Negative Phase Sequence	**Trips Open** in Parallel Mode Only	**Remains Running**

42. 063K1.03 001/2/1/DC - BAT CHARGER/C/A - 2.9/3.5/NEW/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- Unit 1 is at 100% power.
- Offsite power is lost.
- DG1A starts and loads.
- DG1B did NOT start.
- No operator action has been taken.

Which one of the following completes the following statement?

The effects of this failure will be the loss of ___(1)___,

and

per 18031-C, "Loss of Class 1E Electrical Systems," IF the 1E battery voltage drops to 105V DC, then ___(2)___.

A. (1) both battery chargers for 1BD1 and both battery chargers for 1DD1

(2) open the battery breaker only

B. (1) one battery charger each on 1AD1, 1BD1, 1CD1, and 1DD1

(2) open the battery breaker only

C✓ (1) both battery chargers for 1BD1 and both battery chargers for 1DD1

(2) shutdown the associated inverter, then open the battery breaker

D. (1) one battery charger each on 1AD1, 1BD1, 1CD1, and 1DD1

(2) shutdown the associated inverter, then open the battery breaker

063K1.03 D.C. Electrical Distribution

Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems:
(CFR 41.2 to 41.9 / 45.7 to 45.8)

Battery charger and battery.

<u>K/A MATCH ANALYSIS:</u>

This question meets the KA by testing the students knowledge on the DC distribution system, what will be lost in the event of a EDG malfunction,

DISTRACTOR ANALYSIS:

A. Incorrect. 1st half is correct, 2nd part is incorrect as the inverter must be shut down prior to opening the battery breaker.

B. Incorrect - 1st half is incorrect. it is plausible the student may think the "B" charger on each bus is lost, 2nd part is incorrect as the inverter must be shut down prior to opening the battery breaker.

C. Correct - The chargers for 1BD1 and 1DD1 will both be lost, the associated inverter and then the battery breaker must be opened if voltages drop to 105V DC.

D. Incorrect - 1st half is incorrect. it is plausible the student may think the "B" charger on each bus is lost, 2nd half is correct, the associated inverter and then the battery breaker must be opened if voltages drop to 105V DC.

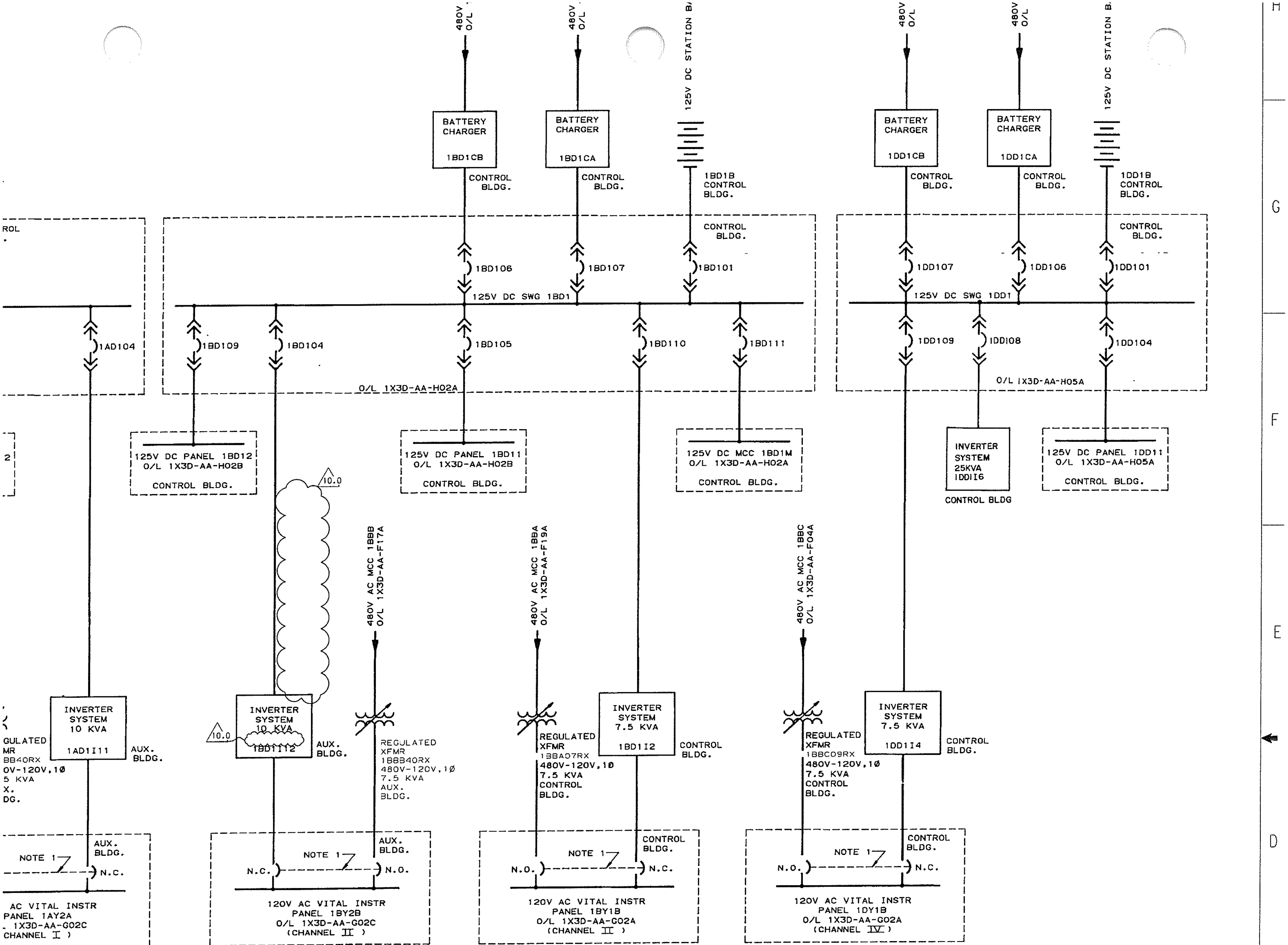
REFERENCES:

LO-TX-01101, pg 27, figure 10
1X3D-AA-G01A
1X3D-AA-D03A
18031-C, Loss of Class 1E Electrical Systems

VEGP learning objectives:

LO-LP-60329-01 Given that a loss of power has occurred to any of the following 125VDC vital buses and given the appropriate plant procedures, describe the operator actions required and why these actions are taken.

a. 1AD1
b. 1BD1
c. 1CD1
d. 1DD1



NOTE:
1. MECHANICAL INTERLOCKS ARE PROVIDED SUCH THAT ONLY ONE BREAKER OF THE TWO

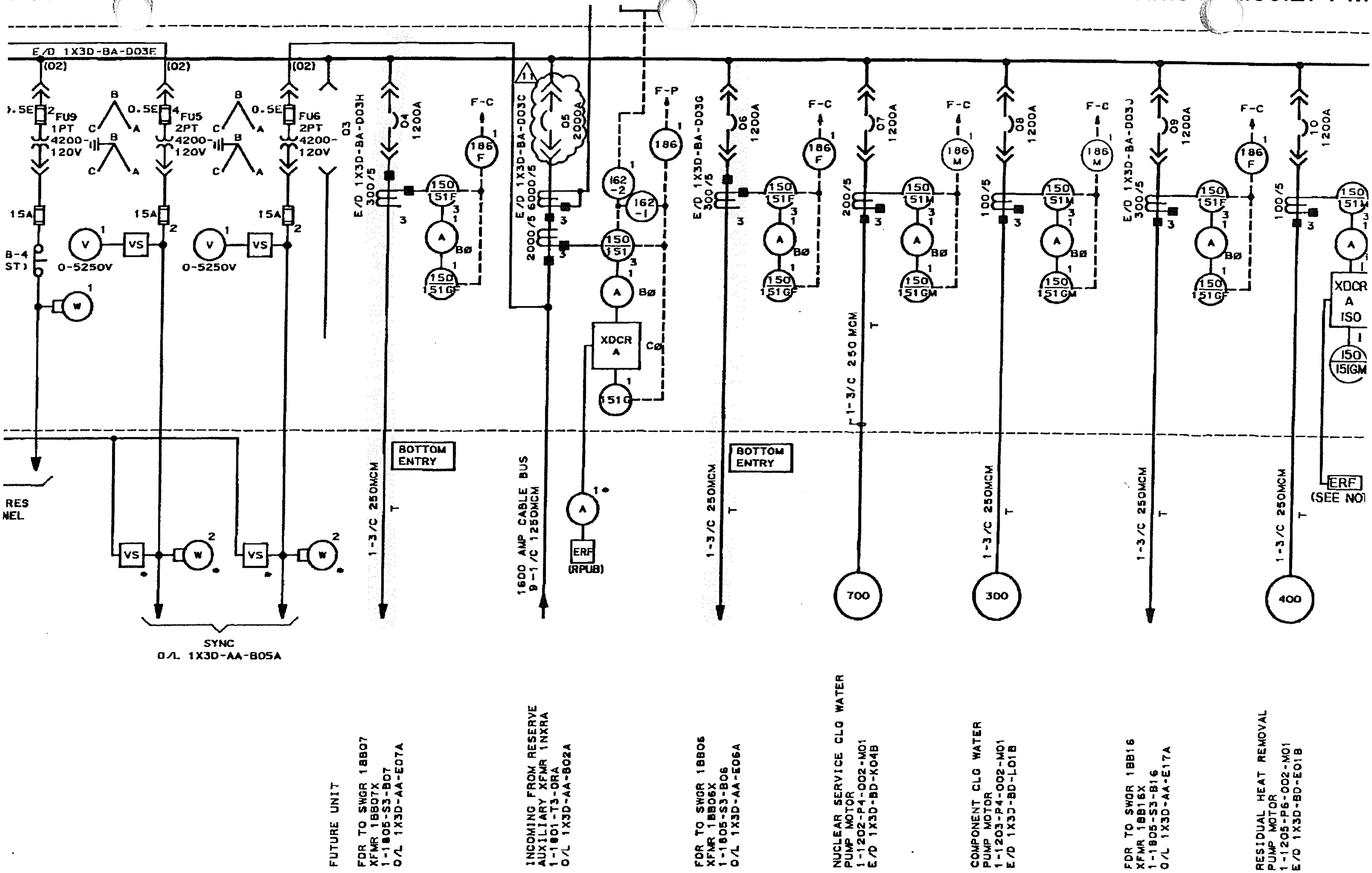


TABLE 1:
1. TRS-LR HS-1BA0319B
2. CS-LR HS-1BA0319A
3. TRS-LR HS-1BA0301B
4. CS-LR HS-1BA0301A

NOTES CONTD:

2. The 480V 1-E System

The ESF Electrical System is divided into two Trains with 4.16kV bus 1AA02 supplying the Train "A" 480V ESF System, and 4.16kV bus 1BA03 supplying the Train "B" 480V ESF System. Redundant ESF loads are assigned to both 480V ESF Trains, thus insuring that no single credible failure of a 480V component will prevent a safe Reactor shut down and decay heat removal following a design basis accident. One non-ESF 480V switchgear bus is powered through a Transformer from each safety-related 4.16kV bus (NB01 and NB10), also known as the "stub buses".

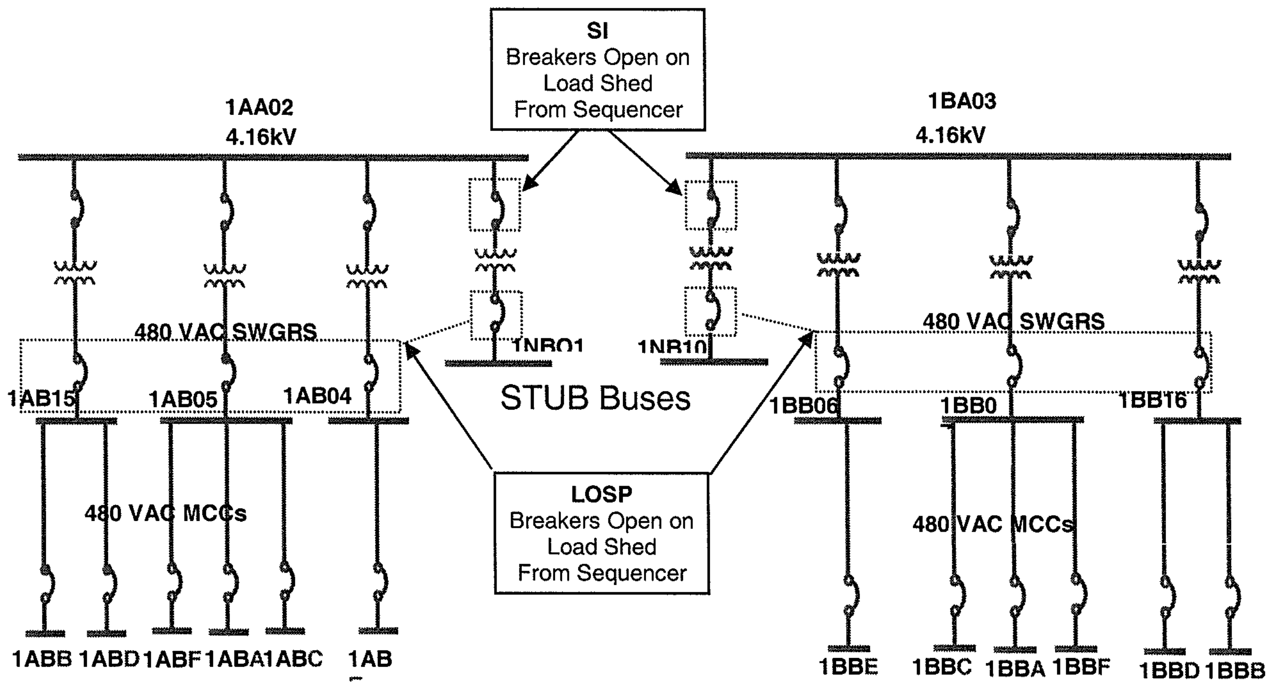


Figure 9

These stub buses are automatically tripped on the 4.16kV side breaker by the SF Sequencer upon the receipt of a Safety Injection Signal, but the breakers can be manually re-closed after resetting the SI Signal or pressing a reset pushbutton on the 4.16kV ESF buses. On an LOSP Signal, the stub buses are automatically tripped on the 480V side breaker by the SF Sequencer and are reloaded 10.5 seconds after the Diesel breaker is closed. The stub buses are not safety-related but are important for Plant operation.

The 480V ESF System also supplies backup power to the 120 volt AC instrument buses. The Inverters are normally in service supplying power from the Vital 125V DC buses to the 120 volt AC instrument buses. The backup supply to the 120 volt bus through a self-regulating Transformer is available if the Inverter is out of service. The two supply breakers are interlocked to prevent simultaneous closure and paralleling of the Inverter with the Transformer.

A. LOSS OF POWER WITH DG FAILING TO TIE TO BUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***A17. Check DC bus loads:**	
a. Verify 125V DC battery loads - LESS THAN THE FOLLOWING LIMITS: __• AD1B 300 AMPS __• BD1B 300 AMPS __• CD1B 100 AMPS __• DD1B 80 AMPS	__a. Evaluate selective load stripping using ATTACHMENT B, DC LOADS TO EVALUATE FOR LOAD STRIPPING DURING LOSS OF 1E BUS.
__b. Monitor all 1E battery bus voltages - REMAIN GREATER THAN 105V DC.	b. Perform the following: __1) Shutdown associated inverter(s) by initiating 13431, 120V AC 1E VITAL INSTRUMENT DISTRIBUTION SYSTEM. __2) Initiate 18032, LOSS OF 120 VOLT AC INSTRUMENT POWER. __3) Open battery breaker after inverter(s) shutdown. __4) Initiate 18034, LOSS OF CLASS 1E 125V DC POWER.

43. 064A4.05 001/2/1/EDG - XFER CONTROL/MEM - 3.1/3.2/NEW/HL-18 NRC/RO/SRO/AML

Given the following:

- Unit 1 is at 100% power.
- DG1B is to be started locally for a post-maintenance run.

Based on the given conditions, which ONE of the following identifies the alarm(s) received in the Control Room indicating diesel control has been transferred from remote to local?

1. ALB38-E01 DG1B GENERATOR TROUBLE
2. ALB38-E05 DG1B DISABLED ENGINE CONTROL IN LOCAL
3. ALB38-E10 DG1B DISABLED MAINTENANCE LOCK OUT

A.✓ 2 only

B. 1, 2, and 3

C. 1 and 2 only

D. 2 and 3 only

064A4.05 Emergency Diesel Generator (EDG) System

Ability to manually operate and/or monitor in the control room: (CFR 41.7 / 45.5 to 45.8)

Transfer of EDG control between manual and automatic.

K/A MATCH ANALYSIS:

The candidate is presented with a list of annunciators and has to determine which annunciator(s) will illuminate when the DG is placed in Local control.

DISTRACTOR ANALYSIS:

A. Correct. ALB38-E05 is the only annunciator that will illuminate.

B. Incorrect. ALB38-E05 is the only annunciator that will illuminate. The other two are plausible distractors.

C. Incorrect. ALB38-E05 is the only annunciator that will illuminate. The other two are plausible distractors.

D. Incorrect. ALB38-E05 is the only annunciator that will illuminate. The other two are plausible distractors.

REFERENCES:

13145B-1, "Emergency Diesel Generators", step 4.1.2.5
ALB38-E01 DG1B GENERATOR TROUBLE
ALB38-E05 DG1B DISABLED ENGINE CONTROL IN LOCAL
ALB38-E10 DG1B DISABLED MAINTENANCE LOCK OUT

VEGP learning objectives:

LO-PP-11101-36 Describe the function and result of operation of the following controls on the engine and generator control panels.

b. Maintenance Mode p/b
g. Local Remote switch

INITIALS

4.1.2 Local Startup Of Train B Diesel Generator

4.1.2.1 IF Diesel Generator is being started following:

A Diesel Generator failure, **complete** Checklist 1. ________

AND/OR

Any maintenance that isolated and restored control air, **perform** Section 4.4.16. ________

> NOTE
>
> Removing a Diesel Generator from standby will require entry into Technical Specification LCO 3.8.1 (Modes 1-4) or LCO 3.8.2 (Modes 5 and 6). ☐

4.1.2.2 **Request** permission to take the Train B Diesel Generator out of Standby. ________

> NOTE
>
> Cylinder-moisture checks are NOT required if the Diesel Generator is started within four (4) hours of a shutdown. ☐

4.1.2.3 IF engine cylinders have NOT been checked for moisture within the last 4 hours, **perform** Section 4.4.1. ________

4.1.2.4 **Initiate** 11885B-1, "Diesel Generator 1B Operating Log." ________

4.1.2.5 At Generator Control Panel PDG3, perform the following:

> ALB38-E05 DG1B DISABLED ENGINE CONTROL IN LOCAL

a. **Place** LOCAL-REMOTE switch 1HS-4517 in LOCAL. ________

b. At Engine Control Panel PDG4, **check** the DG1B DISABLED ENGINE CONTROL IN LOCAL annunciator illuminates. ________

WINDOW E01

DG1B GENERATOR TROUBLE

ORIGIN

One or more of the following:

a. 132X - Reverse Power Relay,
b. 140X - Loss of field,
c. 146X - Negative Phase Sequence,
d. 151V/AX - Voltage Restrained OC - Phase A,
e. 151V/BX - Voltage Restrained OC - Phase B,
f. 151V/CX - Voltage Restrained OC - Phase C,
g. 151NX - Neutral Ground Time Overcurrent,
h. 159X - Overvoltage,
i. 160X - PT Failure,
j. 160/XB, PT failure,
k. 164X - Field Ground
l. 127X - Undervoltage

SETPOINT

Not Applicable

1.0 **PROBABLE CAUSE**

1. Trouble with generator of DG1B.

2. Instrument failure.

3. IF Diesel Generator is supplying power to the grid via the SAT, probable cause of alarm from sensor g is a ground fault on the 13.8kV cable to Plant Wilson.

4. This may be a spurious alarm due to a ground on the input to the optical isolator for this annunciator. Evaluate parameters associated with this annunciator to determine if alarm is valid.

WINDOW E05

ORIGIN	SETPOINT	
1-HS-4517	Not Applicable	DG1B DISABLED ENGINE CONTROL IN LOCAL

1.0 PROBABLE CAUSE

1. The LOCAL/REMOTE 1-HS-4517 Switch, at Panel PDG3 placed in LOCAL position.
2. This may be a spurious alarm due to a ground on the input to the optical isolator for this annunciator. Evaluate parameters associated with this annunciator to determine if alarm is valid.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF alarm NOT expected, **notify** Shift Supervisor and **dispatch** an operator to:
   a. **Verify**, at Panel PDG3, that 1-HS-4517 is in LOCAL.
   b. **Determine** reason for LOCAL position (e.g., the diesel is in the MAINTENANCE Mode).
2. **Refer** to Technical Specifications LCO 3.8.1 or 3.8.2.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X3D-BH-G03T, 1X3D-BH-G03M, 1X3D-BH-G03Q

* WINDOW E10

ORIGIN	SETPOINT	DG1B DISABLED MAINTENANCE LOCK OUT
Maintenance pushbutton on Engine Control Panel	Not Applicable	

1.0 **PROBABLE CAUSE**

1. Diesel engine is in Maintenance mode.
2. Malfunction in Electric-Pneumatic Control System.

2.0 **AUTOMATIC ACTIONS**

Shuts down fuel rack and prevents starting of Diesel Generator.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. IF alarm not expected, **notify** Shift Supervisor and:
   a. **Verify** that maintenance is being performed on diesel.
   b. **Check** for an associated alarm on DG1B BARRING DEVICE ENGAGED.
   c. IF maintenance is NOT being performed on diesel, **determine** reason for maintenance lock-out.
2. **Refer** to Technical Specifications LCO 3.8.1 or 3.8.2.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

* See Note on Page 4.

END OF SUB-PROCEDURE

REFERENCES: 1X4AK01-47, 1X3D-BH-G03N, 1X3D-BH-G03Q

44. 064K1.03 001/2/1/EDG - FUEL OIL SPPLY/MEM - 3.6/4.0/BANK-LOIT/HL-18 NRC/RO/SRO/A,ML

An extended loss of offsite power to 1AA02 has resulted in depletion of the DG1A Fuel Oil Storage Tank (FOST). The Fuel Oil transfer trucks have not been able to reach the site due to severe weather conditions.

Which ONE of the following completes the following statement?

Unit 1, Train 'A' FOST can physically receive Diesel Fuel from ______________.

A. the Aux Boiler FOST ONLY

B. the Unit 1, Train 'B' FOST ONLY

C. the Unit 2, Train 'A' FOST ONLY

D✓ EITHER Train on EITHER Unit FOST

064K1.03 Emergency Diesel Generator (EDG) System

Knowledge of the physical connections and/or cause-effect relationships between the EDG system and the following systems (CFR 41.2 to 41.9 / 45.7 to 45.8):

Diesel fuel oil supply system.

K/A MATCH ANALYSIS:

Question meets the KA by testing the students knowledge on the physical connections when trying to cross tie between FOSTs.

DISTRACTOR ANALYSIS:

A. Incorrect - The Aux Boiler FOST is a possible makeup source to the DG FOSTs, it is plausible the student may think you cannot cross tie trains or units and this will be the only additional supply that can be used for makeup.

B. Incorrect - It is plausible the candidate may think the only possible crosstie source is from the opposite train on the same unit.

C. Incorrect - It is plausible the candidate may think the only possible crosstie source is from the same train on the opposite unit.

D. Correct - with manual valve alignment, any DG FOST can be aligned to any DG.

REFERENCES:

LO-TX-11101, page 10
1X4DB170-1
13146A-1 Train A Diesel Generator Fuel Oil Transfer System

VEGP learning objectives:

LO-PP-11101-05 Describe the Fuel Oil Storage and Transfer System flow paths for each of the following operations:

a. Normal system alignment
b. Recirculation of the Storage Tank
c. Supplying Diesel Fuel Oil Day tank from the opposite train Fuel oil Storage Tank

1.0 PURPOSE

This procedure provides instructions for the operation of the Diesel Generator Fuel Oil (DFO) Storage System. Specific instructions are contained in the following sections:

4.1.1 Filling Diesel Generator A DFO Storage Tank

4.2.1 Aligning Diesel Generator A DFO Day Tank For Automatic Filling

4.4.1 Diesel Generator A DFO Day Tank Supply From Diesel Generator B DFO Storage Tank

4.4.2 Diesel Generator A DFO System Recirculation

4.4.3 Transferring Diesel Generator A DFO Storage Tank To The Auxiliary Boiler Fuel Oil Storage Tank

4.4.4 Filling Diesel Generator A DFO Storage Tank Through The Manway

4.4.5 Fuel Oil Storage Tank Emergency Venting

4.4.6 Transferring Unit 1 Diesel Generator DFO Storage Tanks To Unit 2 Diesel Generator DFO Storage Tanks

4.4.7 Transferring Unit 2 Diesel Generator DFO Storage Tanks To Unit 1 Diesel Generator DFO Storage Tanks

4.4.8 Portable Diesel Generator DFO Storage Tank Filtration

4.4.9 Transferring Diesel Generator B DFO Storage Tank To Train A DFO Storage Tank When Train B Is De-energized

Attachment 1 Diesel Generator Fuel Oil Capacity

Checklist 1 DFO Restoration Valve Alignment

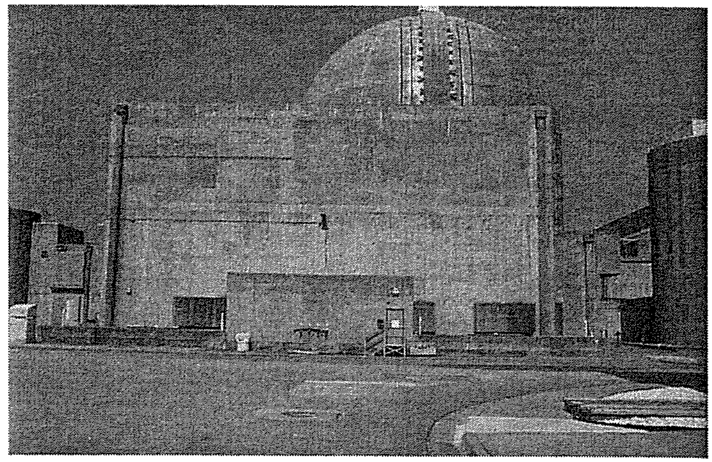
Checklist 2 Temporary Diesel Generator Fuel Oil Filtration Skid Installation/Removal

Checklist 3 Independent Verification Sections 4.1, 4.2, 4.4.1, 4.4.2

Checklist 4 Independent Verification Section 4.4.3

Checklist 5 Independent Verification Sections 4.4.4 and 4.4.5

Checklist 6 Independent Verification Section 4.4.9



There is one water drain line that extends to a low point in the bottom of the tank. Water is detrimental to diesel engine operation because it will cause corrosion, scoring of the injector pump parts, and damage to fuel injector tips. Water can also displace fuel, causing the engine to misfire or stop running. Fuel oil is replenished to the storage tank at the truck quick-fill connection. Diesel fuel oil from the same truck shall not be split between Diesel Generator Fuel Oil Storage tanks on the same unit. This will prevent contaminating both fuel oil storage tanks in the event the fuel oil sample results are found to be unacceptable. Procedure 00261-C describes the process for receipt, sampling and accountability of fuel oil and provides handling safety precautions when transferring fuel oil from a tank truck to permanent tanks. This procedure also covers the fuel oil monthly inventory. Fuel oil is normally capped by way of a normally closed valve, and basket strainer. The design of the diesel fuel oil storage system allows replenishment of fuel without interrupting operation of the diesel generator. The design of the system also prevents turbulence of the sediment in the bottom of the storage tank from degrading overall fuel quality to an unacceptable level.

In the event the diesel fuel oil degenerates during storage, it may be transferred from the diesel fuel oil storage tanks by using the transfer pumps and piping, which are interconnected with locked closed valves, to the plant's auxiliary boiler fuel oil tank. Additional fuel oil can be delivered to VEGP quickly if necessary, from several sources. Also, it should be noted that Georgia Power's Plant Wilson located adjacent to VEGP Units 1 and 2, could be a source of emergency fuel oil if necessary.

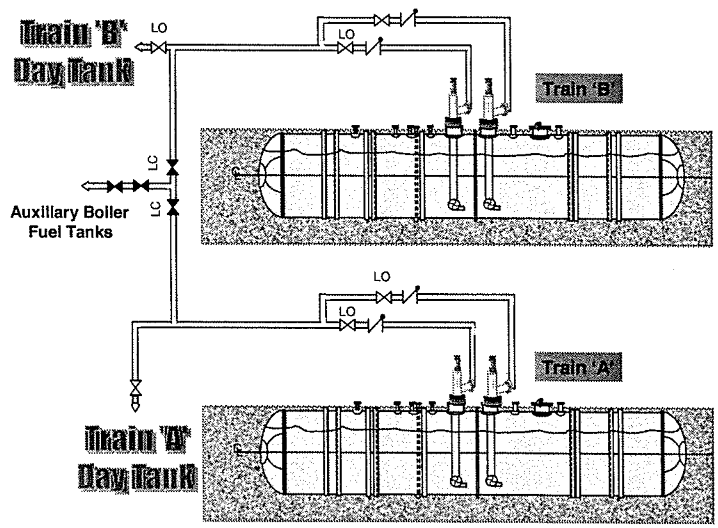


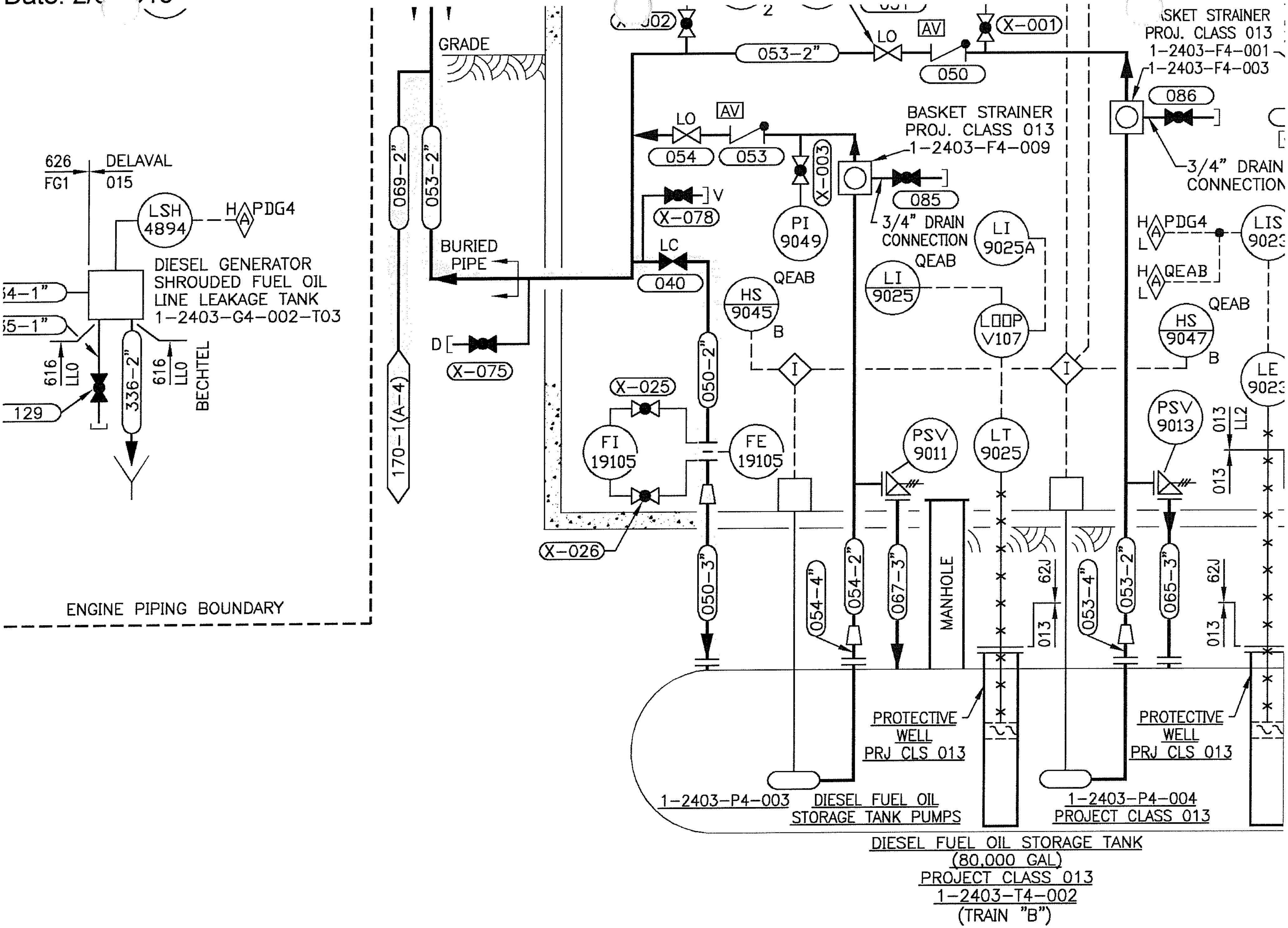
Figure 4

The fuel oil transfer pumps are the submerged, vertical-centrifugal type. Each pump has a capacity of 25 gallons per minute, (approximately 3 times the 8.4 gallons per minute full-load consumption rate of the associated diesel generator). The pump is located in the sump near the bottom of the tank with the pump bearings immersed in the pumped fluid, providing continuous lubrication.

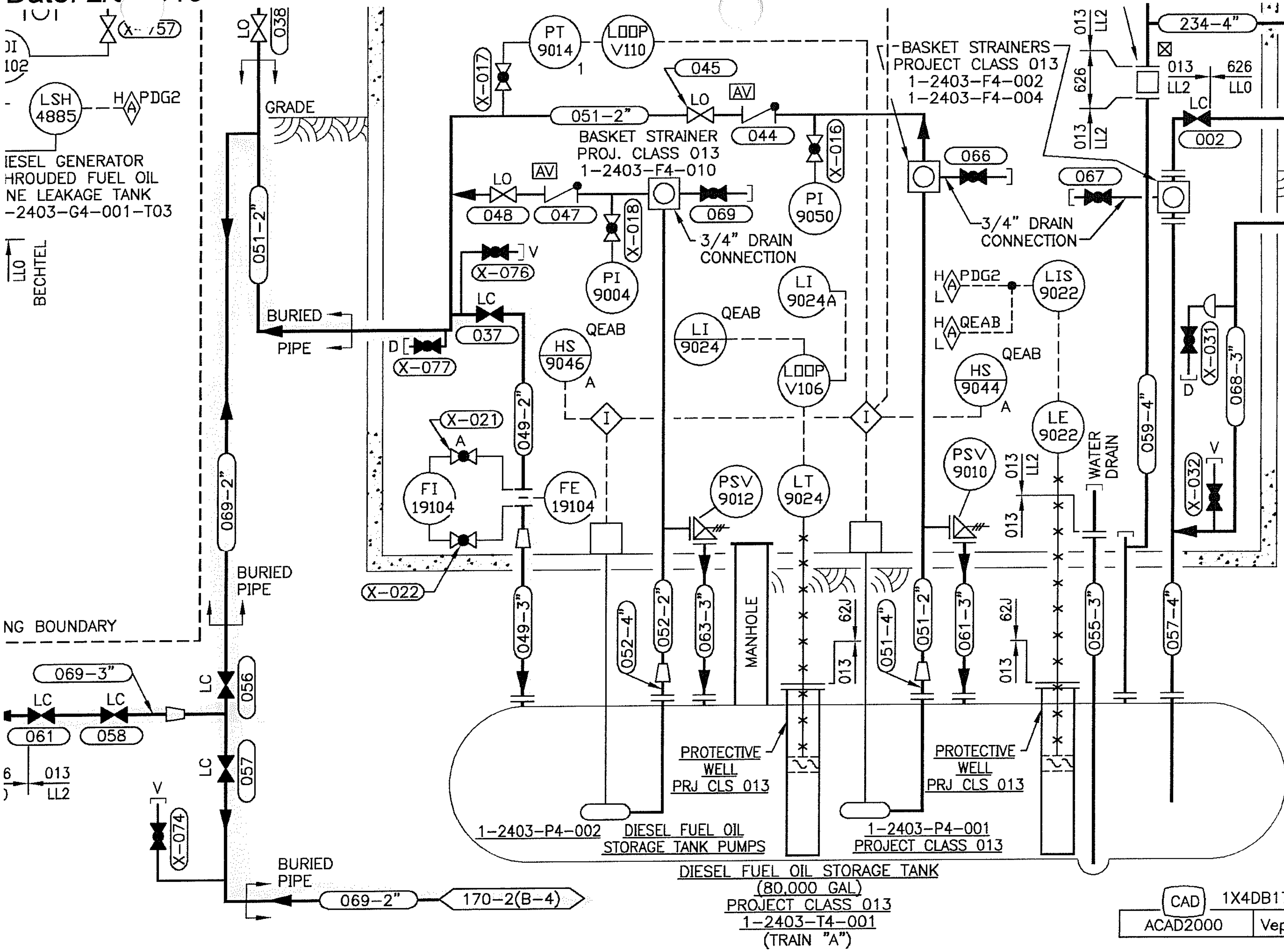
The Fuel Oil Storage System has the following alternate flow path capabilities:

1. Cross train supply of day tank via manual isolation valves
2. Cross unit transfer between storage tanks via manual isolation valves
3. Cross train transfer between storage tanks with one train de-energized via manual power cable quick disconnects.

Each Transfer pump is controlled by one of the two level switches in the DFO Storage Day Tank:


BASKET STRAINER
PROJ. CLASS 013
1-2403-F4-001
1-2403-F4-003
3/4" DRAIN CONNECTION
086
X-001
AV
LO
050
053-2"
GRADE
069-2"
053-2"
170-1(A-4)
BURIED PIPE
D
X-075
LO
AV
054
053
X-003
BASKET STRAINER
PROJ. CLASS 013
1-2403-F4-009
085
3/4" DRAIN CONNECTION
V
X-078
LC
040
PI 9049
HS 9045
QEAB
B
LI 9025
LI 9025A
LOOP V107
H PDG4 L
H QEAB L
LIS 9023
HS 9047
QEAB
B
LE 9023
X-025
FI 19105
FE 19105
X-026
050-2"
050-3"
PSV 9011
LT 9025
PSV 9013
054-4"
054-2"
067-3"
MANHOLE
62J
013
053-4"
053-2"
065-3"
013
LL2
PROTECTIVE WELL PRJ CLS 013
PROTECTIVE WELL PRJ CLS 013
1-2403-P4-003
DIESEL FUEL OIL STORAGE TANK PUMPS
1-2403-P4-004
PROJECT CLASS 013
DIESEL FUEL OIL STORAGE TANK
(80,000 GAL)
PROJECT CLASS 013
1-2403-T4-002
(TRAIN "B")
626
FG1
DELAVAL
015
LSH 4894
H PDG4
DIESEL GENERATOR SHROUDED FUEL OIL LINE LEAKAGE TANK
1-2403-G4-002-T03
616 LLO
336-2"
616 LLO
BECHTEL
129
ENGINE PIPING BOUNDARY

Date: 2/6/2013 Time : 01:25:26 PM



NOTE

1. DA... OI... M... L... IS...
2. ALL PIPING INTERFACE... BECHTEL —|— DELA...
3. A PORTION OF THIS ... OF VENDOR DRAWINGS...
4. ALL PIPING AND COM... PIPING BOUNDARY AR... PIPING LOCATED ON ...
5. PANEL QEAB EQUIPME... PROJECT CLASS 11E.
6. ALL FUEL OIL PIPING ... CLASS LL2 UNLESS C...
7. BECHTEL SUPPLIED D... CLASS IS 013/LL2.
8. BECHTEL SUPPLIED D... CLASS IS 015/LLO.
9. FOR LUBE OIL SYS S...
10. FOR JACKET WATER S...
11. FOR STARTING AIR SY...
12. FOR FUEL OIL SYSTEM...
13. BECHTEL SUPPLIED D...
14. BECHTEL SUPPLIED D... TROUGHS PROJECT/P...
15. BLANK INSTALLED PER...
16. AIR COMPRESSOR #2 ...

CAD 1X4DB170-1

ACAD2000	Vep – 05

SOU...
GEOR...
ALVIN W. ...
DI...

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
45.0	REVISED PER ABN-V00765, VER. 1.0	11-20-07	Vep	RKF	JMR	X

SCALE: NONE

JOB NO.10604

SIZE E 34X44

45. 065AG2.4.46 001/1/1/EP- LOSS IA/MEM - 4.2/4.2/NEW/HL-18 NRC/RO/SRO/AML

Given the following Unit 1 conditions at 100% power:

- 18028-C, "Loss of Instrument Air," has been entered.
- Service Air Dryer Inlet Isolation Valve, 1-PV-9375, has closed.
- Procedural requirements for tripping the Reactor have been met.

Alarm windows are as follows:

- ALB01-B06 INSTR AIR EQUIP LO PRESS
- ALB01-C06 SERVICE AIR HDR LO PRESS

Which ONE of the following is correct concerning:

(1) which alarm(s) on ALB01 should be illuminated,

and

(2) where is the location of pressure switch 1-PSL-9375, which enables the re-opening of 1-PV-9375 after air pressure is recovered?

A✓ (1) B06 and C06

(2) Turbine Building, Level 1, near Powdex Vessels

B. (1) B06 ONLY

(2) Turbine Building, Level 1, near Powdex Vessels

C. (1) B06 and C06

(2) Turbine Building, Level A, on local PMEC panel

D. (1) C06 ONLY

(2) Turbine Building, Level A, on local PMEC panel

065AG2.4.46 Loss of Instrument Air

Ability to verify that the alarms are consistent with the plant conditions:
(CFR: 41.10 / 43.5 / 45.12)

K/A MATCH ANALYSIS:

Question meets the KA by testing the students ability to understand with given conditions which annunciator(s) will be lit. The candidate also has to know the location of the 1-PSL-9375 in order to reset the switch.

DISTRACTOR ANALYSIS:

A. Correct. Both annunciators will be illuminated if a reactor trip is required. PSL-9375 is reset on Turbine Bldg level 1, near the powdex vessels.

B. Incorrect. Both annunciators will be illuminated if a reactor trip is required. The location for resetting PSL-9375 is correct.

C. Incorrect. Both annunciators will be illuminated if a reactor trip is required. PSL-9375 is reset on Turbine Bldg level 1, near the powdex vessels, NOT Turbine Building level A on the PMEC panel. Level A is plausible since other items with instrument air are reset on level A in the vicinity of the air compressors.

D. Both annunciators will be illuminated if a reactor trip is required. PSL-9375 is reset on Turbine Bldg level 1, near the powdex vessels, NOT Turbine Building level A on the PMEC panel. Level A is plausible since other items with instrument air are reset on level A in the vicinity of the air compressors.

REFERENCES:

18028-C, "Loss of Instrument Air", pg 5
ALB01-B06 INSTR AIR EQUIP LO PRESS
ALB01-C06 SERVICE AIR HDR LO PRESS
13710-1, Service Air System, note on page # 22

VEGP learning objectives:

LO-PP-60321-10 Describe why a loss of instrument air precludes plant operation.

LO-PP-02101-09 List the sequence of major events on a decreasing instrument air pressure condition.

ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED	

CAUTION

Loss of Turbine Building instrument air will cause all extraction steam stop valves to close. MFP miniflow valves and feedwater heater and drain tank hi-level dump valves will fail open.

__5.	Check Instrument Air header pressure – LOWERING.	__5.	Go to Step 17.
__*6.	**Check Instrument Air header pressure - REMAINS GREATER THAN 80 PSIG.**	__*6.	Dispatch an operator to verify PV-9375 Service Air Header Isolation Valve is closed: UNIT 1 (TB-A-TD11) UNIT 2 (TB-A-TD10)
__7.	Check UNIT 1 Service Air – AVAILABLE.	7.	Verify seals supplied with bottled nitrogen at greater than or equal to 50 psig: __• Cask loading pit gates. __• Fuel transfer canal gates.

INITIALS

4.4.3 Restoring Service Air System Pressure Following Low Pressure Isolation

4.4.3.1 **Verify** the following valves are closed:

- SERVICE AIR AIR DRYER 1 BYPASS VLV 1-2401-U4-551. ________
- SERVICE AIR AIR DRYER 1 OUTLET ISO VLV 1-2401-U4-554. ________
- SERVICE AIR AIR DRYER 1 PREFILTER 503 ISO VLV 1-2401-U4-548. ________

NOTE

Pressure Switch 1-PSL-9375 is located on Instrument Rack 15 (1-1624-P5-R15) Turbine Building Level 1 near the Powdex Vessels. □

4.4.3.2 At PMEC, When Service Air header pressure is greater than 97 psig on 1-PI-19380, **reset** Pressure Switch 1-PSL-9375 by simultaneously depressing both RESET switches. ________

4.4.3.3 **Verify** 1-PV-9375 opens. ________

CAUTION

When repressurizing the Service Air header, the valve should be opened slowly to prevent reducing pressure to less than 100 psig. □

4.4.3.4 **Re-pressurize** the Service Air header as follows:

a. Slowly **open** SERVICE AIR AIR DRYER 1 BYPASS VLV 1-2401-U4-551. ________

b. If desired, **start up** Service Air Dryer per Section 4.1.5. ________

WINDOW B06

ORIGIN	SETPOINT	
1-PSL-19414	70 psig	INSTR AIR EQUIP LO PRESS

1.0 **PROBABLE CAUSE**

1. Instrument Air Dryer, Prefilter or Afterfilter clogged.
2. System piping leak.
3. System valve misalignment.
4. Loss of all Air Compressors.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Go To 18028-C, "Loss Of Instrument Air."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X3D-BH-R50L, 1X4DB175-2, CX5DT1101-95B

WINDOW C06

ORIGIN	SETPOINT
1-PSL-9375	95 psig

SERVICE AIR
HDR LO PRESS

1.0 **PROBABLE CAUSE**

1. Excessive service air demand.
2. Air Compressor trip.
3. System leak.
4. Standby compressor failed to start.

2.0 **AUTOMATIC ACTIONS**

1. Service Air Dryer Inlet Isolation Valve 1-PV-9375 closes at a service air pressure of 80 psig.
2. Any standby air compressor with its handswitch in AUTO-PTL position will auto start at a discharge pressure of 100 psig decreasing.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

46. 071K4.05 001/2/2/WASTE GAS - RELEASE/C/A -2.7/3.0/MOD - LOIT BANK/HL-18 NRC/RO/SRO/AML

Given the following:

- Waste Gas Decay Tank #10 relief valve is lifting and discharging to the Waste Gas Decay Tank Relief Valve Discharge Header.

- A-RV-0014, Waste Gas Discharge Valve, closes on a high radiation signal.

Which ONE of the following completes the following statement?

A-RV-0014 closing __(1)__ automatically isolate the release path and the Auxiliary Building Normal Ventilation System will __(2)__.

	__(1)__	__(2)__
A.	will	trip
B.	will	remain running
C.	will NOT	trip
D✓	will NOT	remain running

071K4.05 Waste Gas Disposal System (WGDS)

Knowledge of the design feature(s) and/or interlock(s) which provide for the following: (CFR 41.7)

Point of release.

<u>K/A MATCH ANALYSIS:</u>

Question meets the KA by testing the students knowledge on automatic features on a high rad alarm and its effect on the ventilation system.

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect - It is plausible to think that the ventilation will secure on a high rad alarm to prevent release to public but Aux. Building ventilation does NOT trip on a high radiation signal from RV-0014. , the WGDT # 10 discharges downstream of the RV-0014 isolation valve but upstream of the rad monitor. The release will continue.

B. Incorrect - Aux. Building ventilation does NOT trip on a high radiation signal from RV-0014. WGDT # 10 discharges downstream of the RV-0014 isolation valve but upstream of the rad monitor. The release will continue.

C. Incorrect - It is plausible to think that the ventilation will secure on a high rad alarm to prevent release to public but Aux. Building ventilation does NOT trip on a high radiation signal from RV-0014. WGDT # 10 discharges downstream of the RV-0014 isolation valve but upstream of the rad monitor. The release will continue.

D. Correct - WGDT # 10 discharges downstream of the RV-0014 isolation valve but upstream of the rad monitor. The release will continue. Aux. Building ventilation does NOT trip on a high radiation signal from RV-0014.

REFERENCES:

13202-1, Gaseous Releases
1X4DB129, Waste Processing System Gas

VEGP learning objectives:

LO-LP-46101-03 State the purposes of the following Gaseous Radwaste System components:

a. gas decay tanks
b. gas decay shutdown tanks
e. trip valve RV-014
f. rad monitors RE-013 and RE-014
g. gas decay tank relief valves and header

LO-LP-46101-11 State the events that require immediate termination of a gaseous release.

INITIALS

4.0 INSTRUCTIONS

4.1 INITIAL PREPARATIONS FOR RELEASE

NOTES

- Since all Gaseous Waste Processing System Relief Valves relieve to Waste Gas Decay Shutdown Tank A-1902-V6-010, the system MUST be shut down PRIOR to isolating this tank WHEN preparing to release it.
- The Gas Decay Tank is placed in recirculation to ensure all piping in the system is homogeneous to prevent the release path from tripping closed.
- Since pressures may be higher than typical during normal recombiner operation, the flow AND pressure requirements for recirculation per 13201-1 may NOT be achievable. A flow OR greater than 1000 SCFH satisfies this mixing requirement.

CAUTIONS

- The tank, which is to be released, MUST remain isolated except for sampling. This will ensure the validity of the gaseous release permit.
- The Gaseous Waste Processing System can NOT be restarted until the Gas Decay Tank has been released.

4.1.1 **Place** the Gas Decay Tank to be released in the Recirculation Mode by one of the following methods:

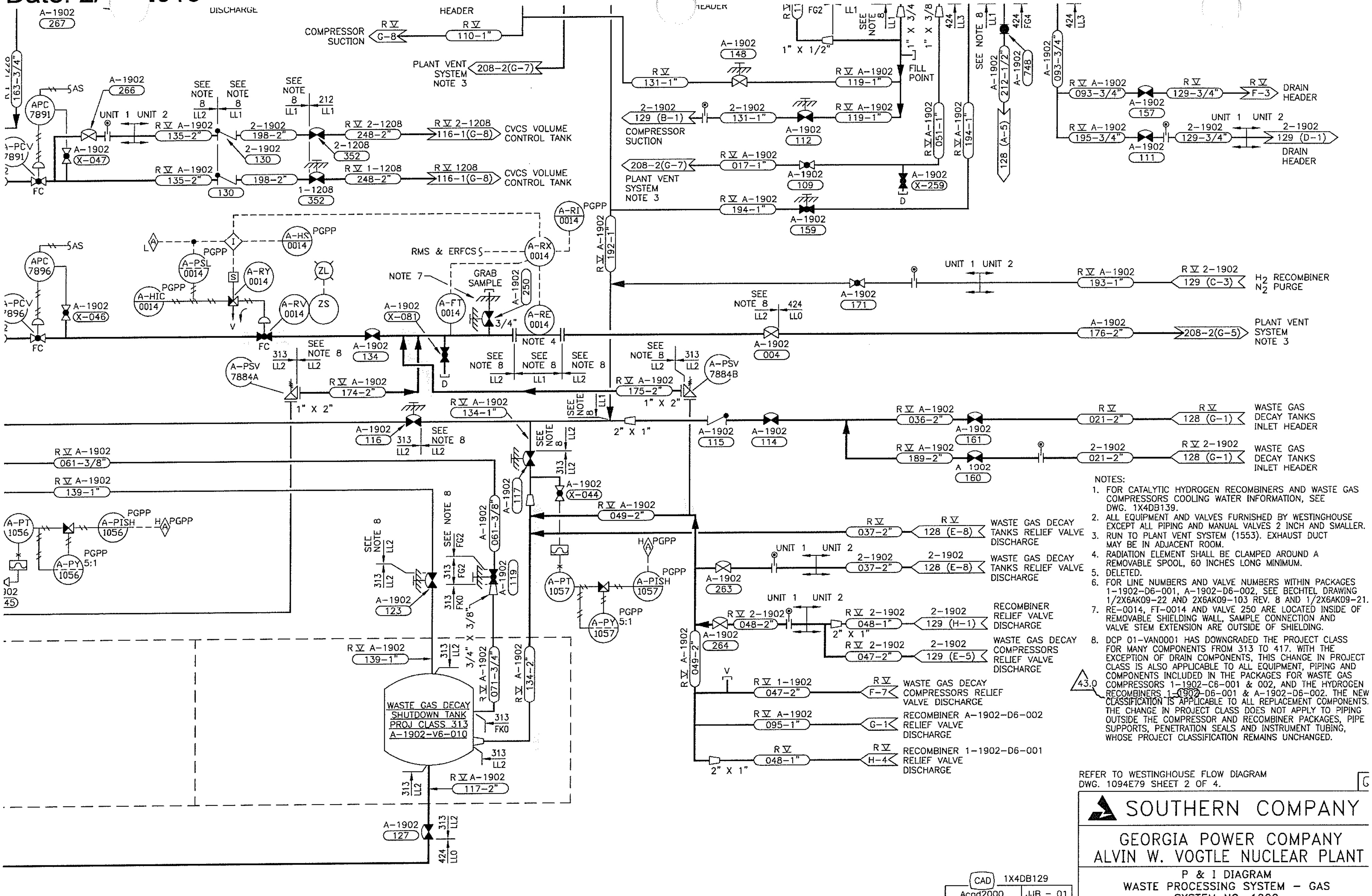
IF the WGS is operating, **swap** the inservice gas decay tank to the GDT to be released per 13201-1, "Gaseous Waste Processing System" ________

OR

IF the WGS is NOT operating, **startup** the WGS with the GDT to be released per 13201-1, "Gaseous Waste Processing System" ________

4.1.2 AFTER 1 hour, **shutdown** the Waste Gas System per 13201-1, "Gaseous Waste Processing System." ________

Date: 2/1?/2013 Time : ?8:35:35 AM



NOTES:

1. FOR CATALYTIC HYDROGEN RECOMBINERS AND WASTE GAS COMPRESSORS COOLING WATER INFORMATION, SEE DWG. 1X4DB139.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND MANUAL VALVES 2 INCH AND SMALLER.
3. RUN TO PLANT VENT SYSTEM (1553). EXHAUST DUCT MAY BE IN ADJACENT ROOM.
4. RADIATION ELEMENT SHALL BE CLAMPED AROUND A REMOVABLE SPOOL, 60 INCHES LONG MINIMUM.
5. DELETED.
6. FOR LINE NUMBERS AND VALVE NUMBERS WITHIN PACKAGES 1-1902-D6-001, A-1902-D6-002, SEE BECHTEL DRAWING 1/2X6AK09-22 AND 2X6AK09-103 REV. 8 AND 1/2X6AK09-21.
7. RE-0014, FT-0014 AND VALVE 250 ARE LOCATED INSIDE OF REMOVABLE SHIELDING WALL, SAMPLE CONNECTION AND VALVE STEM EXTENSION ARE OUTSIDE OF SHIELDING.
8. DCP 01-VAN0001 HAS DOWNGRADED THE PROJECT CLASS FOR MANY COMPONENTS FROM 313 TO 417. WITH THE EXCEPTION OF DRAIN COMPONENTS, THIS CHANGE IN PROJECT CLASS IS ALSO APPLICABLE TO ALL EQUIPMENT, PIPING AND COMPONENTS INCLUDED IN THE PACKAGES FOR WASTE GAS COMPRESSORS 1-1902-C6-001 & 002, AND THE HYDROGEN RECOMBINERS 1-1902-D6-001 & A-1902-D6-002. THE NEW CLASSIFICATION IS APPLICABLE TO ALL REPLACEMENT COMPONENTS. THE CHANGE IN PROJECT CLASS DOES NOT APPLY TO PIPING OUTSIDE THE COMPRESSOR AND RECOMBINER PACKAGES, PIPE SUPPORTS, PENETRATION SEALS AND INSTRUMENT TUBING, WHOSE PROJECT CLASSIFICATION REMAINS UNCHANGED.

43.0

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. 1094E79 SHEET 2 OF 4.

CAD 1X4DB129
Acad2000 JJB - 01

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
WASTE PROCESSING SYSTEM - GAS
SYSTEM NO. 1902

DRAWING NO VER

47. 072K1.03 001/2/2/ARM - FHBI/C/A - 3.6/3.7/MOD - HL-15R AUDIT/HL-18 NRC/RO/SRO/KAJ

A dropped spent fuel assembly in the Unit 1 Spent Fuel Pool has resulted in the following radiation monitor alarms:

- 1-RE-0008, FHB Area Monitor, indicates HIGH.
- A-RE-2532A(B) and A-RE-2533A(B), FHB Effluent Monitors, indicate ALERT.
- The crew is implementing 18006-C, "Fuel Handling Event".

For the given conditions, which ONE of the following completes the following statement?

1-RE-0008 ___(1)___ provide audible and visual indications of the alarm in the Unit 1 SFP area,

and

the FHB Post-Accident Filtration Units ___(2)___ automatically start.

A. (1) will

(2) will

B✓ (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

072K1.03 **Area Radiation Monitoring (ARM) System**

Knowledge of the physical connections and/or cause-effect relationships between the ARM and the following systems: (CFR 41.2 to 41.9 / 45.7 / 45.8)

Fuel Building Isolation.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a fuel assembly has been dropped, the candidate has to know whether or not RE-0008 ARM has both audible and visual indication of an alarm condition on the FHB deck. The candidate also has to determine whether an INTMD alarm on RE-2532/2533 and HIGH on RE-0008 will result in an FHB ventilation actuation.

DISTRACTOR ANALYSIS:

A. Incorrect. 1RE-0008 provides both audible and visual indication on High alarm. RE-2532A/B and ARE-2533A/B do NOT cause an actuation at the Alert level.

B. Correct. 1RE-0008 provides both audible and visual indication on High alarm. RE-2532A/B and ARE-2533A/B do NOT cause an actuation at the Alert level.

C. Incorrect. 1RE-0008 provides both audible and visual indication on High alarm, it is plausible the candidate may confuse this with most other area alarms which do NOT provide both audible and visual alarms. RE-2532A/B and ARE-2533A/B do NOT cause an actuation at the Alert level.

D. Incorrect. 1RE-0008 provides both audible and visual indication on High alarm. it is plausible the candidate may confuse this with most other area alarms which do NOT provide both audible and visual alarms. RE-2532A/B and ARE-2533A/B do NOT cause an actuation at the Alert level.

REFERENCES:

17100-1 1-RE-0008 (High)
17102-1 ARE-2532A & B, ARE-2533A & B Alert

VEGP learning objectives:

LO-LP-32101-08 List all the safety-related radiation monitors by tag number and name. Descirbe those automatic actions that occur for each of the following safety-related monitors when its high alarm setepoint is exceeded.

b. Fuel Handling Building Effluent (ARE-2532A & B and ARE-2533A & B)

ORIGIN

Area Monitor

SETPOINT

As determined by Chemistry Department

1-RE-0008
(High)

NOTE

For other than HIGH conditions see Pages 4 and 5.

1.0 **PROBABLE CAUSE**

Increase in radiation level near Unit 1 Spent Fuel Pool in the Fuel Handling Building.

2.0 **AUTOMATIC ACTIONS**

On the south wall of the Fuel Handling Building Spent Fuel Pool Room near the door:

a. Alarm horn on 1-RA-0008 sounds.

b. Strobe light on 1-RA-0008 blinks.

3.0 **INITIAL OPERATOR ACTIONS**

Evacuate the Fuel Handling Building.

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Check** for elevated radiation levels on A-RE-2533A and A-RE-2533B on the SRDC.

2. **Notify** Health Physics to survey Spent Fuel Pool area to determine cause of the alarm.

3. **Isolate** the source of radioactivity if possible.

4. **Refer** to NMP-EP-110, "Emergency Classification And Implementing Instructions".

5. **Obtain** detector trend data per 13508-1, "Radiation Monitoring Systems".

6. **Monitor** the channel for further changes.

7. IF sampling and analysis determine the channel has malfunctioned, **request** Chemistry to deactivate the channel.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: AX4DB204-2, 1X5DS3A02

ANNUNCIATOR AND DETECTOR LOCATIONS

Radiation Detectors, Data Processing Modules (DPM), Radiation Indicators and Alarms are located as indicated on the individual sub-procedures.

Any RMS alarm will annunciate on ALB05 on the MCB. They will also be indicated on the plant computer and the monochrome CRT of Panel QRM1.

TYPES OF ALARMS

Each channel may display several conditions indicated on the ERF Color CRT by the channel identifier displayed in colors as:

Top of Scale	Red
High Alarm	Red
Alert Alarm	Yellow
Equipment Trouble	Magenta
Test Mode	Magenta
Out of Service	Magenta
Normal	Green

The Top of Scale alarm is a latching alarm. The detector may be damaged, and should be recalibrated by Chemistry prior to continued use.

A High Alarm indication will remain active until reset.

Instructions in this procedure are for a High Alarm on the designated channel. The following "generic" actions should be taken for other alarms on any channel:

Alert Alarm

1. **Notify** the Shift Supervisor.
2. **Notify** Health Physics/Chemistry.
3. **Determine** cause and **fix** if possible.
4. **Obtain** historic trend data for channel.
5. **Monitor** the channel for further changes.
6. IF behavior is erratic, **perform** the ARP actions for a malfunctioning condition.

NOTES

- On loss of power to an RMS instrument, the instrument should be declared inoperable.
- If a Radiation Monitor is inoperable, it should have the current operating parameters entered by Chemistry before being declared operable.

Equipment Trouble

1. **Notify** the Shift Supervisor.
2. **Notify** Chemistry.
3. **Identify** cause of trouble alarm if possible.
4. **Check** Technical Specifications LCO 3.3.3, 3.3.6, 3.3.7, 3.4.15 and Technical Requirements TR 13.3.6 for operability requirements and comply with action statements.

Test Mode

1. **Determine** if an authorized test is in progress.
2. IF NOT undergoing an authorized test:
   a. **Notify** the Shift Supervisor.
   b. **Request** Chemistry investigate.

Out of Service

1. **Determine** if authorized maintenance work is in progress.
2. IF NOT authorized maintenance:
   a. **Notify** the Shift Supervisor.
   b. **Request** Maintenance/I&C to determine cause.

48. 073A2.02 001/2/1/RAD MON - DETECTOR/MEM - 2.7/3.2/BANK - HL-16 NRC/HL-18 NRC/RO/SRO/AML

List of Unit 1 radiation monitors and isolation valves:

- 1RE-0019, SG Sample Liquid
- 1RE-0021, SG Blowdown Liquid

- 1FV-1150, Blowdown Inlet Isolation Valve
- 1HV-7600, Blowdown Recycle Isolation Valve

Initial conditions:

- Steam Generator Blowdown (SGBD) is in service on Unit 1.

Current conditions:

- A SGBD system radiation detector has failed HIGH.

Which ONE of the following completes the following statements?

Radiation monitor ___(1)___ will initiate a SGBD isolation signal,

and

per 17100-C, "Annunciator Response Procedure for the Process and Effluent Radiation Monitoring System (RMS)," the operators will verify that blowdown isolation valve ___(2)___ has automatically closed to isolate the SGBD flow path.

	__(1)__	__(2)__
A.	1RE-0019	1FV-1150
B.	1RE-0019	1HV-7600
C.✓	1RE-0021	1FV-1150
D.	1RE-0021	1HV-7600

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: (CFR 41.5 / 43.5 / 45.3 / 45.13)

Detector failure.

K/A MATCH ANALYSIS:

KA is matched because SGBD Rad Monitor RE-021 is a process Rad Monitor and is physically connected to the system. The scenario is prediction of a detector failure, and the automatic action that should take place to mitigate the malfunction.

DISTRACTOR ANALYSIS

A. Incorrect-Plausible because RE-019 is a SGBD Rad Monitor but provides only alarm indication and causes no isolation of the system. FV-1150 will close on High Rad from RE-021.

B. Incorrect-RE-019 same as "A". Plausible because HV-7600 is an isolation valve but not isolated by High Radiation.

C. Correct-RE-021 will isolate SGBD on High Rad. FV-1150 will close along with FV-021.

D. Incorrect-RE-021 is the Rad Monitor that isolates SGBD on High Rad. Plausible because HV-7600 is an isolation valve but not isolated by High Radiation.

REFERENCES

Vogtle 2011 NRC Exam
ARP 17100-1 for RE-019 Liquid Monitor and RE-021 Process Monitor
P&ID series 1X4DB179

VEGP learning objectives:

LO-PP-32101-09 Describe those automatic actions that occur for each of the following non-safety monitors when its high alarm setpoint is exceeded:

d. steam generator blowdown (RE-0021)

ANNUNCIATOR RESPONSE INDEX

RADIATION MONITOR INDEX

* Safety Related. Go to 17102-1, "ARP For The SRDC QRM2".
** Not Functional - Detectors Removed.
*** Passive collector. No electronic components.

ORIGIN	**SETPOINT**	
Liquid Monitor	As determined by Chemistry Department	1-RE-0019 (High)

NOTE

For other than HIGH or intermediate conditions see Pages 4 and 5.

1.0 **PROBABLE CAUSE**

Increase in radiation in the Steam Generator Blowdown from:

a. Steam Generator tube leak.

b. Blowdown processing system malfunction.

2.0 **AUTOMATIC ACTIONS**

1. At the Steam Generator Blowdown Instrument Rack PSGI
   a. Alarm horn on 1-RA-0019 sounds.
   b. Strobe light on 1-RA-0019 blinks.
2. In the Component Cooling Water Train B Pump Room, 1-RX-1950 indicates the high radiation alarm.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

ORIGIN	SETPOINT	
Liquid Process	As determined by Chemistry Department	1-RE-0021 (High)

NOTE

For other than HIGH conditions see Pages 4 and 5.

1.0 PROBABLE CAUSE

Increase in radiation in the Steam Generator Blowdown from:

a. Steam Generator tube leak,

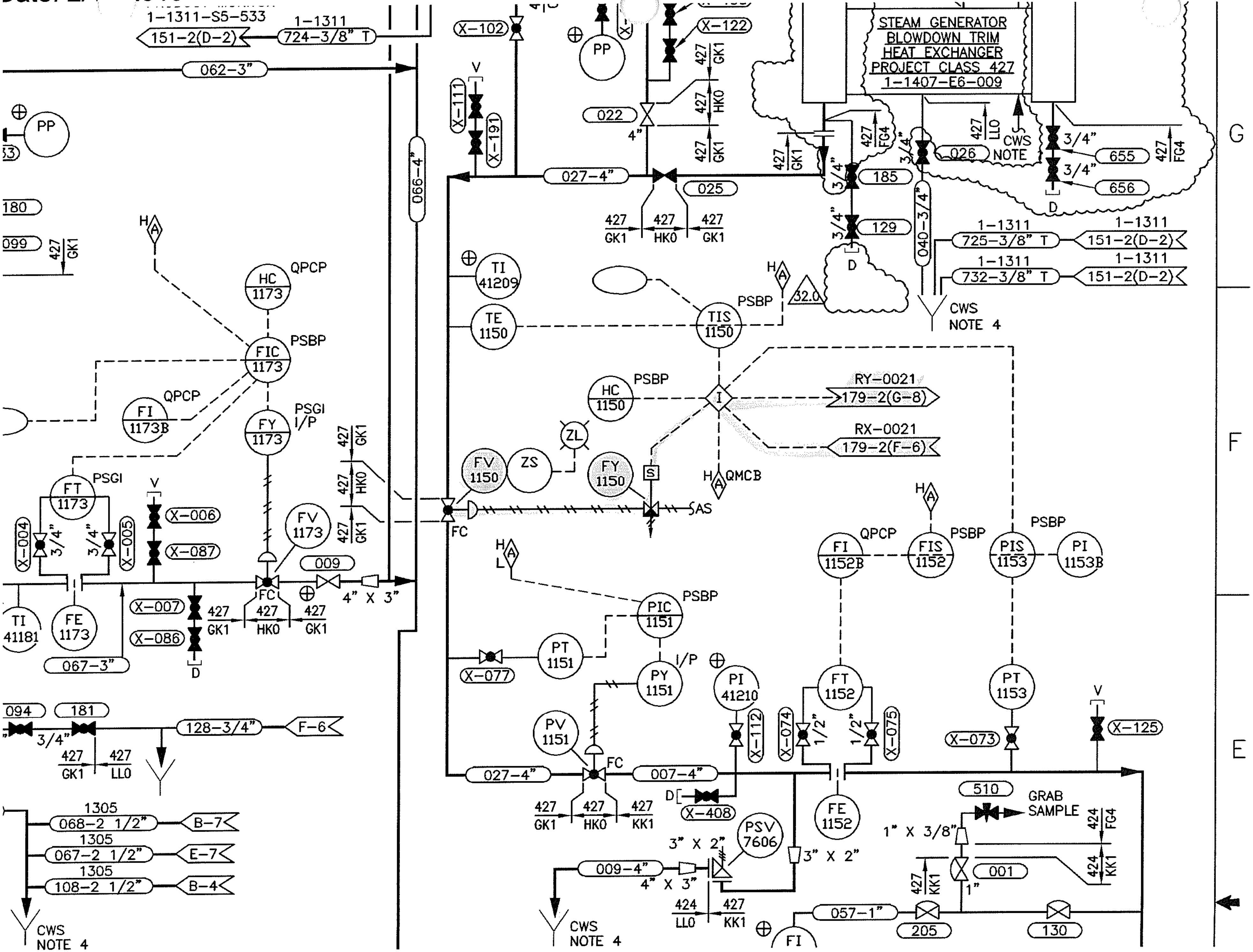
b. Blowdown processing system malfunction.

2.0 AUTOMATIC ACTIONS

1. Isolates Blowdown Heat Exchanger discharge to Blowdown Demineralizers via 1-FV-1150.
2. At the Steam Generator Blowdown Process Panel PSBP: 1-RI-0021 indicates current radiation level.
3. Isolates Blowdown to WWRB via 1-RV-0021.

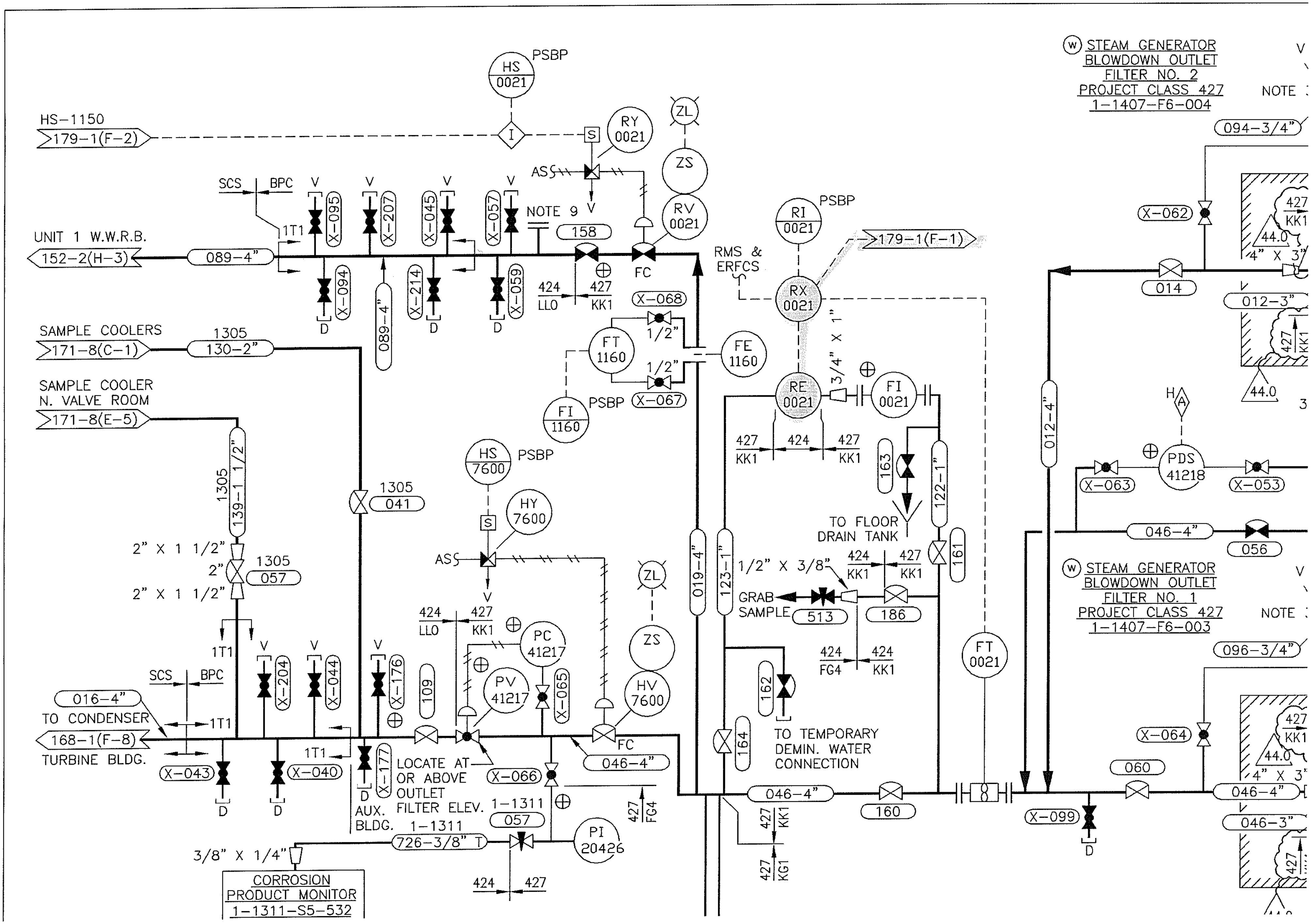
3.0 INITIAL OPERATOR ACTIONS

NONE


STEAM GENERATOR
BLOWDOWN TRIM
HEAT EXCHANGER
PROJECT CLASS 427
1-1407-E6-009
1-1311-S5-533
151-2(D-2)
1-1311
724-3/8" T
062-3"
066-4"
027-4"
025
022
X-102
X-111
X-191
X-122
185
129
026
655
656
040-3/4"
1-1311
725-3/8" T
732-3/8" T
151-2(D-2)
CWS
NOTE 4
HC 1173
QPCP
FIC 1173
PSBP
FI 1173B
FY 1173
PSGI
I/P
FT 1173
FE 1173
FV 1173
TI 41181
X-004
X-005
X-006
X-087
X-007
X-086
067-3"
009
4" X 3"
FC
094
181
128-3/4"
F-6
1305
068-2 1/2"
B-7
067-2 1/2"
E-7
108-2 1/2"
B-4
TI 41209
TE 1150
TIS 1150
HC 1150
FV 1150
ZS
ZL
FY 1150
32.0
RY-0021
179-2(G-8)
RX-0021
179-2(F-6)
QMCB
SAS
PIC 1151
PT 1151
PY 1151
PV 1151
X-077
PI 41210
X-112
X-408
FI 1152B
FIS 1152
FT 1152
FE 1152
X-074
X-075
PIS 1153
PI 1153B
PT 1153
X-073
X-125
007-4"
009-4"
PSV 7606
3" X 2"
510
GRAB SAMPLE
1" X 3/8"
001
057-1"
205
130
FI
427 GK1
427 HK0
427 KK1
427 LLO
427 FG4
424 FG4
424 KK1
424 LLO
G
F
E

Date: 2/1?/2013

Time : 09:52:59 AM



\Documents and Settings\Administrator.SOUTHERN-F2F7D1\Desktop\HL-18\HL-18 References\P&ID\1X4DB17

53. 073K1.01 002/2/1/PRM SYSTEMS/3.6/3.9 MEM/LOIT BANK/RO/SRO/NRC/GCW

HIGH radiation has been detected in the Steam Generator Blowdown System (SGBD).

Regarding the following rad monitors:

RE-0019, "SG Sample Liquid"
RE-0021, "SG Blowdown Liquid"

Which ONE of the following would (1) initiate a SGBD isolation on High Radiation and (2) which valve listed below would automatically close to isolate the system on High Radiation?

A. (1) RE-0019

(2) FV-1150, Blowdown Inlet Isolation Valve

B. (1) RE-0019

(2) HV-7600, Blowdown Recycle Isolation Valve

C✓ (1) RE-0021

(2) FV-1150, Blowdown Inlet Isolation Valve

D. (1) RE-0021

(2) HV-7600, Blowdown Recycle Isolation Valve

49. 076K2.04 001/2/1/SWS - CCW/C/A - 2.5/2.6/NEW/HL-18 NRC/RO/SRO/AML

Initial conditions:

- NSCW pumps # 1 and # 5 are running.

- ACCW pump # 1 is running.

Current conditions:

- SI actuates.

- 2 minutes later an LOSP occurs to 4160 1E bus 1AA02.

- SI has NOT been reset.

Which one of the following correctly describes the pumps that will be supplying cooling water to the Containment Building after the load sequencing is complete?

A. NSCW pumps # 1, # 3, and # 5

Both ACCW pumps

B. NSCW pumps # 1, # 3, and # 5

Neither ACCW pump

C. NSCW pumps # 1 and # 3 only

Both ACCW pumps

D✓ NSCW pumps # 1 and # 3 only

Neither ACCW pump

076K2.04 Service Water System (SWS)

Knowledge of the bus power supplies to the following:
(CFR 41.7)

Reactor building closed cooling water.

<u>K/A MATCH ANALYSIS:</u>

The candidate is presented with a plausible scenario where an SI occurs followed shortly by an LOSP. The candidate has to determine which NSCW and ACCW pumps

are providing cooling water to the containment building.

DISTRACTOR ANALYSIS:

A. Incorrect. On the SI, all 3 pumps will be running, after the LOSP only pumps 1 & 3 will be running. It is plausible that the candidate will forget the LOSP will strip the running pumps and resequence pumps 1 & 3 only.

On an SI, ACCW pump # 1 will remain running. After the LOSP, the ACCW pumps will load shed and the SI signal will prevent the ACCW pumps from restarting. On an LOSP only, both ACCW pumps will be running whether a single or dual train LOSP, it is plausible for the candidate to pick both ACCW pumps running if he forgets that SI prevents the pumps from starting.

B. Incorrect. On the SI, all 3 pumps will be running, after the LOSP only pumps 1 & 3 will be running. It is plausible that the candidate will forget the LOSP will strip the running pumps and resequence pumps 1 & 3 only.

The 2nd half is correct, neither ACCW pump will be running, After the LOSP, the ACCW pumps will load shed and the SI signal will prevent the ACCW pumps from restarting.

C. The first half is correct. On the SI, all 3 pumps will be running, after the LOSP only pumps 1 & 3 will be running. It is plausible that the candidate will forget the LOSP will strip the running pumps and resequence pumps 1 & 3 only.

The 2nd half is incorrect. On an SI, ACCW pump # 1 will remain running. After the LOSP, the ACCW pumps will load shed and the SI signal will prevent the ACCW pumps from restarting. On an LOSP only, both ACCW pumps will be running whether a single or dual train LOSP, it is plausible for the candidate to pick both ACCW pumps running if he forgets that SI prevents the pumps from starting.

D. Correct. See A, B, and C above.

REFERENCES:

1X3D-AA-K02A, DG1A and Train A AC Buses Loading Table

VEGP learning objectives:

LO-LP-28201-03 Describe Sequencer operation, including load shedding, load sequencing and Diesel Generator operation under the following conditions.

a. Undervoltage (UV)
b. Safety Injection (SI)
c. UV followed by SI
d. SI followed by UV

EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION	PROJECT CLASSIFICATION (ELECTRICAL SYSTEM)	PROCUREMENT SPECIFICATION	BUS NUMBER	BUS VOLTAGE	NAMEPLATE RATING HP (OR KW)	REFERENCE DWG. 1X3D-	LOSS OF COOLANT ACCIDENT SI PREFERRED OFFSITE SOURCE AVAILABLE: MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SECONDS) TO START (NOTE 7)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER & SUBSEQUENT SI: MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.b)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER: MANUAL, PROCESS, ACDT, OR SEQUENCER START SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.c)	RUNNING TIME AFTER START (NOTE 4)
11202P4001M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04A	S	25.5	C	S	25.5	C	S	25.5	C
11202P4003M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04C	S	25.5	C	S	25.5	C	S	25.5	C
11202P4005M01	NUC SERV CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	700	BD-K04E	NOTE. 5	30.5	—	NOTE 5	30.5	—	NOTE 5	30.5	—
11203P4001M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01A	S	20.5	C	S	20.5	C	S	20.5	C
11203P4003M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01C	S	20.5	C	S	20.5	C	S	20.5	C
11203P4005M01	COMP CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	300	BD-L01E	NOTE 5	25.5	—	NOTE 5	25.5	—	NOTE 5	25.5	—
11204P6003M01	SAFETY INJ PP	11E	X6	1AA02	4160V	450	BD-D01C	S	5.5	C	S	5.5	C	M	>30.5	C
11205P6001M01	RESID HT REMVL PP	11E	X6	1AA02	4160V	400	BD-E01A	S	10.5	C	S	10.5	C	M	>30.5	C
11206P6001M01	CONTAINMENT SPRAY PP	11E	X6	1AA02	4160V	400	BD-J01A	NOTE 6	15.5	C	NOTE 6	15.5	C	M	>30.5	C
11208P6002M01	CENTRIFUGAL CHARG PP	11E	X6	1AA02	4160V	600 **	BD-C01A	S	0.5	C	S	0.5	C	S	0.5	C
11217P4001M01	AUX COMP CLG WTR PP	11E	X4	1AA02	4160V	600	BD-L03A	M	>30.5	C	M	>30.5	C	S	15.5	C
11302P4003M01	AUX FDWTR PP-MOTOR DRIVEN	11E	X4	1AA02	4160V	900	BC-F04A	S	20.5	C	S	20.5	C	S	20.5	C
11592C7001M01	ESF CHILLER (COMPRESSOR)	11E	X4	1AA02	4160V	400	BG-G02A	P	>30.5	C	P	>30.5	C	P	>30.5	C
11805S3B04	LOAD CENTER TRANS. 1AB04	11E	X3	1AB04	4160 -480V	1000KVA	BA-D02F									
	SECONDARY BREAKER 1AB04	11E	X3	1AB04	480V	—	BA-E02A	—	—	C	—	0	C	—	0	C
11501A7001M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01A	—	—	C	S	0.5	C	S	0.5	C
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03F	NOTE 10	NA	NA	NOTE 10	NA	NA	S NOTE 15	50.5	C
11501A7002M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01B	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03G	NOTE 10	NA	NA	NOTE 10	NA	NA	S NOTE 15	50.5	C
11501A7005M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01E	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03K	NOTE 10	NA	NA	NOTE 10	NA	NA	S	30.5	C
11501A7006M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01F	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03L	NOTE 10	NA	NA	NOTE 10	NA	NA	S	30.5	C
								S	30.5	C	S	30.5	C	NOTE 10	NA	NA
11805S3B05	LOAD CENTER TRANS. 1AB05	11E	X3	1AB05	4.16 -480V	1000KVA	BA-D02L									
	SECONDARY BREAKER 1AB05	11E	X3	1AB05	480V	—	BA-E02P	—	—	C	—	0	C	—	0	C
11566B7001M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01B	—	—	C	S	0.5	C	S	0.5	C
11566B7003M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01C	I	>30.5	C	I	>30.5	C	I	>30.5	C
11531N7001H01	CB CONTROL ROOM HTR	11E	X4	1AB05	480V	118KW	BG-C05R	P	>30.5	C	P	>30.5	C	P	>30.5	C
11531N7001M01	CB CONTROL ROOM FIL UNIT	11E	X4	1AB05	480V	125	BG-C01E	N NOTE 11	>30.5	C	N NOTE 11	>30.5	C	N NOTE 11	>30.5	C
11513H7001000	CTB HYDROGEN RECOMBINER	11E	X6	1AB05	480V	75KW	BG-B02U	P	>30.5	C	P	>30.5	C	P	>30.5	C
								M	>30.5	C	M	>30.5	C	M	>30.5	C
11805S3B15	LOAD CENTER TRANS. 1AB15	11E	X3	1AB15	4.16 -480V	1000KVA	BA-D02K									
	SECONDARY BREAKER 1AB15	11E	X3	1AB15	480V	—	BA-E02R	—	—	C	—	0	C	—	0	C
11202W4001M01	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03A	—	—	C	S	0.5	C	S	0.5	C
11202W4001M02	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K0[illegible]	P	>30.5	C	P	>30.5	C	P	>30.5	C
								P	>30.5	C	P	>30.5	C	P	>30.5	C

C			S	10.5	C
C			K	10.5	C
C			MM	10.5	C
C			MM	10.5	C
C			N	0.5	C
C			N	. :0.5	C
C			N	0.5	C
C			K	0.5	C
C			K	0.5	C
C			I,K	0.5	C
C			M	>30.5	C
C			I,K NOTE 16B,C	>30.5	C
C			K	0.5	C
C			N	0.5	C
10SEC			K	0.5	10SEC
10 SEC			I NOTE 11	>30.5	10SEC
10 SEC			M	>30.5	10SEC
10 SEC			I NOTE 11	>30.5	10SEC
15SEC			I	70.5	15SEC
10 SEC			I	0.5	10 SEC.
10 SEC			M	>30.5	10 SEC.
C			SEE NOTE 14	0.5	C
C			N	0.5	C
4DAYS			M	>30.5	C
C			K	0.5	C
C			K	0.5	C
C			K	0.5	C
15SEC			I	70.5	15SEC

NOTES:

1.) LOSS OF OFFSITE POWER (LOP) IS DEFINED AS A LOSS OF VOLTAGE ON THE 4.16KV BUS.

2.)
a. TIME t=0 SECONDS IS DEFINED AS THE TIME THE DIESEL GENERATOR BREAKER IS CLOSED FOR ALL LOP CONDITIONS AND THE TIME THE SI SIGNAL IS RECEIVED IN THE SAFETY FEATURES (SF) SEQUENCER WITH PREFERRED OFFSITE SOURCE AVAILABLE TO THE 4.16KV BUS.
b. DURING LOP CONDITIONS THE MAXIMUM TIME AFTER THE D.G. RECEIVES THE START SIGNAL TO THE FIRST LOAD STEP IS 10 SECONDS. THIS 10 SECOND INTERVAL INCLUDES A MAXIMUM OF 9.5 SECONDS FOR THE DIESEL GENERATOR TO START AND COME UP TO RATED VOLTAGE AND FREQUENCY PLUS 0.5 SECONDS, AFTER THE DIESEL GENERATOR BREAKER CLOSES, TO ENERGIZE THE UNLOADED 480V LOADCENTER TRANSFORMERS.

3.)
a. THE 4.16KV BREAKERS SUPPLYING 1NB01 AND 1NB10 15 WILL BE TRIPPED DURING SI CONDITIONS BY SSPS. 15 POWER CAN BE RESTORED MANUALLY TO BUS 1NB01 AND 1NB10 UNDER ADMINISTRATIVE PROCEDURE AFTER THE SI SIGNAL HAS BEEN MANUALLY BYPASSED.
b. DURING LOP CONDITIONS AND SUBSEQUENT SI ALL BUSSES (ECEPT 1NB01,1NB10,1NBS,1NBR,1NBO AND 1NBI) WILL HAVE POWER AVALIABLE FOR LOADING 0.5 SECONDS AFTER THE DIESEL GENERATOR BREAKER CLOSES.
c. DURING LOP CONDITIONS ALL BUSSES WILL HAVE POWER AVAILABLE FOR LOADING 0.5 SECONDS AFTER THE DIESEL GENERATOR BREAKER CLOSES.

4.)
a. C MEANS CONTINOUS S MEANS STANDBY
b. UNLESS OTHERWISE INDICATED MOV(S) ARE ASSUMED TO OPERATE FOR 10 SECONDS ONLY.

5.) THIS PUMP WILL BE SEQUENCED ON ONLY IF ONE OR BOTH OF THE PRIMARY PUMPS FAIL TO START ON THE PREVIOUSLY SEQUENCED STEP

6.) THE SF SEQUENCER WILL ONLY PERMIT THIS LOAD TO START FOR 1 SECOND PERIODS OF TIME AT THE BEGINNING OF THE INDICATED SEQUENCE STEP. ALL SUBSEQUENT STEPS AND AT ANYTIME AFTER 30.5 SECONDS

7.) THE FIRST LOAD STEP WILL OCCUR (WITH PREFERRED OFFSITE SOURCE AVAILABLE) 0.5 SECONDS AFTER THE SI SIGNAL IS RECEIVED AT THE SF SEQUENCER

10.) UNDER SI CONDITIONS,THESE LOADS ARE SEQUENCED ON AT LOW SPEED ONLY FOR LOP CONDITIONS THEY ARE SEQUENCED ON AT HIGH SPEED ONLY

11.) THESE LOADS ARE INTERLOCKED WITH OTHER LOADS WHICH ARE BLOCKED BY SEQUENCER UNITL 30.5 SECONDS

12.) TRAIN B SIMILAR EXCEPT WHERE NOTED BY ASTERISK (*) OR IN TRAIN B TABLE (SEE NOTES 18,19 &20)

13.) SI MEANS SAFETY INJECTION SYSTEM SIGNAL ACTUATED.

14 .) THIS LOAD FED FROM BREAKER CLOSED ONLY DURING STARTUP OF THE LOAD

15.) SEQUENCER PROVIDES MOTOR AUTO-START CIRCUIT WITH START SIGNAL AT 30.5 SECOND STEP. HOWEVER MOTOR WILL START AT 50.5 SECONDS DUE TO AN ADDITIONAL TIME DELAY OF 20 SECONDS BY AGASTAT TIME DELAY RELAY IN THE AUTO-START CIRCUIT.

NOTES CONT

16 A) SI PROVIDES CB ESF CHILLER WAT
CIRCUIT WITH START SIGNAL. HOW
WILL BE DELAYED APPROXIMATELY
OF AGASTAT TIME DELAY RELAY IN

B) CB ESF CHILLER PURGE COMP MTR I
ESF CHILLER COMPRESSOR WHICH IS BL
SEQUENCER UNTIL 30.5 SEC

C) CB ESF CHILLER OIL PUMP MOTOR
ESF CHILLER CHILLED WATER FLOW
OPERATES DUE TO FLOW FROM THE
PUMP APPROXIMATELY 120 SECONDS
SI SIGNAL IS INITIATED.

17. FAN 1-1561-N7-001 WILL LOAD SH
AUTO-START UNLESS A CVI SIGNAL
HEATER 1-1561-N7-001-H01 IS IN
WITH FAN 1-1561-N7-001 AND IS
CONTROLLED.

TABLE 1

INITIATING LETTER DESIGNATOR	TIME SEC TO OPERATE AFTER DIESEL GENERATOR BREAKER CLOSES	ST
K	0.5	PRC (TH PRE ETC
L	0.5	ACC OR
M	>30.5	MANU ACTI SEQU IS C
N	0.5	PAN FED WHI
P	>30.5 (BLOCKED BY SEQUENCER)	PRO
S	<30.5	AUTC CAN ONLY
MM	0.5	MNL (SLI LOAD RESTC
I	VARIOUS	INT OTH

NO.	REVISIONS
15	INCORP. PER ABN-36788

5 4 3

50. 076K3.05 001/2/1/SWS - RHR/C/A - 3.0/3.2/MOD - LOIT BANK/HL-18 NRC/RO/SRO/KAJ

Given the following conditions / events:

- RCS is on solid-plant pressure control.

- 'A' RHR in service for RCS temperature control.

- RHR letdown is in service.

- All controls are in AUTOMATIC.

- All 'A' CCW pumps are stopped due to a system rupture.

With no additional operator actions taken, which ONE of the following correctly describes the expected INITIAL system response?

A. PSV-8117 (low pressure letdown relief) opens.

B. PSV-8708A (RHR pump suction relief) opens.

C✓ PV-131 (low pressure letdown control valve) modulates open.

D. FV-619 (RHR heat exchanger bypass valve) modulates closed.

076K3.05 Service Water System (SWS)

Knowledge of the effect that a loss or malfunction of the SWS will have on the following: (CFR 41.7 / 45.6)

RHR components, controls, sensors, indicators and alarms, including rad monitors.

K/A MATCH ANALYSIS:

The candidate is presented with a plausible scenario where CCW has been lost due to a pipe rupture while RHR train A is providing letdown flow during solid plant operations. The candidate has to determine which relief or control valve will initially respond to mitigate a possible RCS overpressure condition as temperature rises due to the loss of CCW cooling to the RHR Hx.

DISTRACTOR ANALYSIS:

A. Incorrect. PSV-8117 lifts at 600 psig, PV-131 will modulate to try to control pressure before this setpoint is reached.

B. Incorrect. RHR pumps suction PSV-8708A lifts at 450 psig, PV-131 will modulate to try to control pressure before this setpoint is reached.

C. Correct. PV-131 will modulate to try to control RCS pressure prior to a relief valve setpoint being reached.

D. Incorrect. FV-619 RHR Heat Exchanger Bypass will be unaffected by the CCW leak. It control at a set flow rate depending on the potentiometer setting. PV-131 will modulate to try to control pressure before this setpoint is reached.

REFERENCES:

17007-1, Window C05 for LP LTDN RELIEF HI TEMP
V-LO-TX-12101, Residual Heat Removal
PLS pages 62, 72, and 73
13011-1, Residual Heat Removal System

VEGP learning objectives:

LO-LP-12101-06 State the lift setpoints for the RHR suction and discharge relief valves and the discharge flow paths for each.

LO-LP-12101-09 Describe how the RHR heat exchanger bypass valve automatically controls flow to the RCS.

LO-LP-12101-10 Describe the RHR flow path for low pressure letdown to the CVCS system.

LO-LP-12101-21 Describe how RCS pressure is controlled while in solid plant operations.

WINDOW C05

ORIGIN	SETPOINT	
1-TE-0125	160°F	LP LTDN RELIEF HI TEMP

1.0 **PROBABLE CAUSE**

Low Pressure Letdown Relief Valve 1-PSV-8117 lifted or leaking (Relief Valve set at 600 psig).

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

1. **Monitor** relief line temperature using 1-TI-0125 and letdown pressure using 1-PI-0131A on the QMCB.

2. IF letdown is lost, **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. IF temperature continues to rise, indicating a lifted or leaking Relief Valve, **evaluate** the affect of continued operation with a malfunctioning Relief Valve.

2. **Monitor** Pressurizer Relief Tank level and pressure using 1-LI-0470 and 1-PI-0469 on the QMCB.

3. IF PRT pressure increases due to CVCS Letdown Relief Valve being open or leaking, THEN **evaluate** the possible need to isolate letdown and **initiate** 18007-C "Chemical And Volume Control System Malfunction."

4. IF equipment failure is indicated, **initiate** maintenance as required.

INITIALS

NOTES

- Design maximum CVCS letdown flow is 120 gpm. ☐
- The RHR Hx outlet Low Pressure Letdown Relief Valve 1-PSV-8856A (1-PSV-8856B) lifts to the BRS RHT at 600 psig. ☐
- During Solid Plant conditions only 1-PIC-0131 should be used for letdown flow control and 1-HV-0128 should remain in the FULL OPEN position. ☐

4.5.4.4 IF in Solid Plant conditions, **slowly fully open** the RHR LETDOWN TO CVCS ISOLATION Valve using 1-HC-128 while adjusting the Letdown Pressure Controller 1-PIC-0131 as required to obtain the desired Letdown Flow as indicated on 1-FI-0132C. ________

4.5.4.5 IF NOT in Solid Plant Condition, **adjust** the Letdown Pressure Controller 1-PIC-0131 and/or RHR Letdown Isolation using 1-HC-128 as required to obtain the desired Letdown Flow as indicated on 1-FI-0132C. ________

B INSTRUMENT SETPOINTS

PRESSURES

CHANNEL	DESCRIPTION	FUNCTION	SETPOINT
P 115	Volume Control Tank Pressure	Hi alarm Lo alarm Close valve PV-115	45 psig 18 psig 15 psig
131	Letdown Low Pressure Control	Controller Hi Alarm	370 psig 425 psig
P-140 P-141	Seal Water Injection Filter ΔP	Hi Alarm	25 psid
P-150 P-151 P-152 P-153	RCP No 1 Seal ΔP	Lo Alarm	215 psid
8155	VCT N_2 Supply	Regulator	18 psig
8156	VCT N_2 Supply	Regulator	18 psig

B. INSTRUMENT SETPOINTS

PRESSURES

CHANNEL	DESCRIPTION	FUNCTION	SETPOINT
P-614 P-615	RHR Pump Discharge	Hi alarm	576 psig

FLOWS

CHANNEL	DESCRIPTION	FUNCTION	SETPOINT
F-610 F-611	RHR Pump Miniflow	Valve open Valve close	(1)* (2)
F-618 F-619	RHR HX Bypass Flow Control	Controller	3000 gpm

(1)Open: 20.20 in.W.C.(824 gpm at 350°F,780 gpm at 100°F)
(2)Close:112.60 in.W.C.(1944 gpm at 350°F,1841 gpm at 100°F)

C. RELIEF VALVE SETPOINTS

VALVE	DESCRIPTION	SETPOINT
8708A/B	RHR Pumps Suction Header	450 ± 10* psig

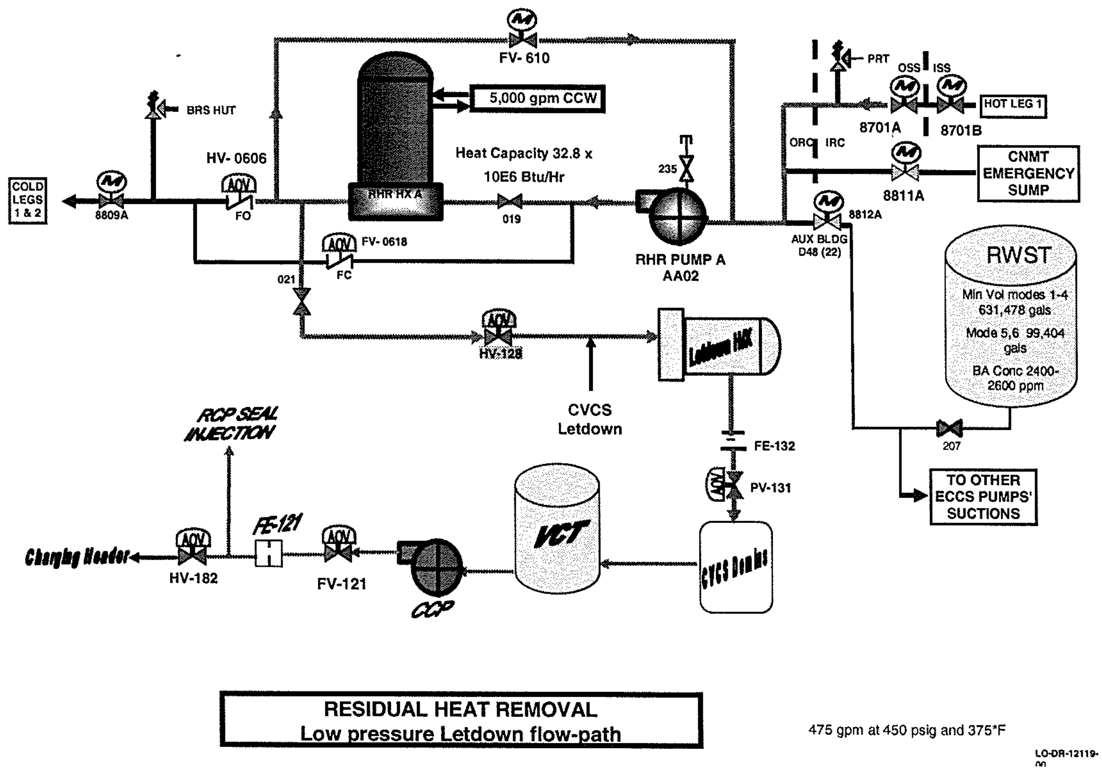
*The tolerance is required for consistency with the RHR open permissive setpoint of 365 psig (P-408, 418, 428, 438)

AX6AA04-30

6.6.3 RHR Letdown

Taking the plant to cold shutdown conditions requires the RCS to be cooled, de-pressurized and taken water solid (pressurizer bubble collapsed). The low pressure state of the RCS renders the Normal CVCS Letdown ineffective due to the inherently low DP across the letdown orifices. The RHR SOP 13011-1/2 directs RHR Letdown to be put in service when RHR is put in service for shutdown cooling. Normal CVCS Letdown may or may not still be in service.

RHR Letdown



Later, when the pressurizer is taken solid (no steam space), pressurizer sprays become ineffective in controlling RCS pressure. During solid plant conditions, RCS pressure is controlled simply by the difference between normal charging and letdown from the RHR system. The Letdown flow is taken downstream of the RHR heat exchanger and directed to a point upstream of the Letdown heat exchanger. The operator controls RCS pressure and letdown flow by manipulation of either HV-128 and/or pressure control valve PV-131 in the letdown flow path.

NOTE

Per the SOP, when the RCS is water solid, pressure should be controlled solely by PV-131. HV-128 should be maintained FULL OPEN.

51. 077AA2.02 001/1/1/GEN VOLT - GEN CURVE/C/A - 3.5/3.6/BANK-HL-17 NRC/HL-18 NRC/RO/SRO/TNT

Initial conditions:

- Unit 1 is at 100% power.
- The Main Generator is operating at 1215 MW and 40 MVARs lagging.
- Main Generator hydrogen pressure is 60 psig.

Current conditions:

- A grid disturbance causes the Main Generator to go to 1200 MW and 400 MVARs lagging.
- The crew is performing 18017-C, "Abnormal Grid Disturbances / Loss of Grid," Section A, "Degraded Grid Conditions".
- The UO is at the step to maintain the generator within the limits of the Reactive Capability Curve.

Per 18017-C, which one of the following is the action (if any) the UO will perform?

REFERENCE PROVIDED

A. No action required.

B. Establish 1215 MW using the INCREASE LOAD pushbutton on the Main Turbine Control Panel.

C. Establish 200 MVARS lagging using the "Volts/VARs" RAISE button on EX2100 Excitation Control Screen.

D✓ Establish 200 MVARS lagging using the "Volts/VARs" LOWER button on EX2100 Excitation Control Screen.

077AA2.02 Generator Voltage and Electric Grid Disturbance

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:
(CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Voltage outside the generator capability curve.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where the main generator is overexcited and exceeding the capability curve for plant conditions and asks what actions to take, an operational implication.

DISTRACTOR ANALYSIS

A. Incorrect. Main generator is overexcited and exceeds capability Main generator voltage must be reduced to within curve.

B. Incorrect. Main generator is overexcited and exceeds capability Main generator voltage must be reduced to within curve. Raising generator voltage higher would drive the excitation further out of limits. This is plausible as MWatts had lowered from 1215 MWatts to 1200MWatts and the candidate may think this is a corrective action.

C. Incorrect. Main generator is over-exicted and exceeds capability Main generator voltage must be reduced to within curve. This is plausible if the candidate thinks lagging is in the negative MVARs direction and does not notice the leading - lagging labels on the curve. This answer would be correct if MVARs were leading.

D. Correct. Main generator is over-excited and exceeds capability Main generator voltage must be reduced to within curve.

REFERENCES

Figure 1 of 18017-C to be given as reference to the candidate.
18017-C, "Abnormal Grid Disturbances / Loss of Grid",

VEGP learning objectives:

V-LO-LP-60330-04: Given main generator operational parameters, determine the operating point on the generator capability curve and identify if the point is within or outside the capabilities of the main generator.

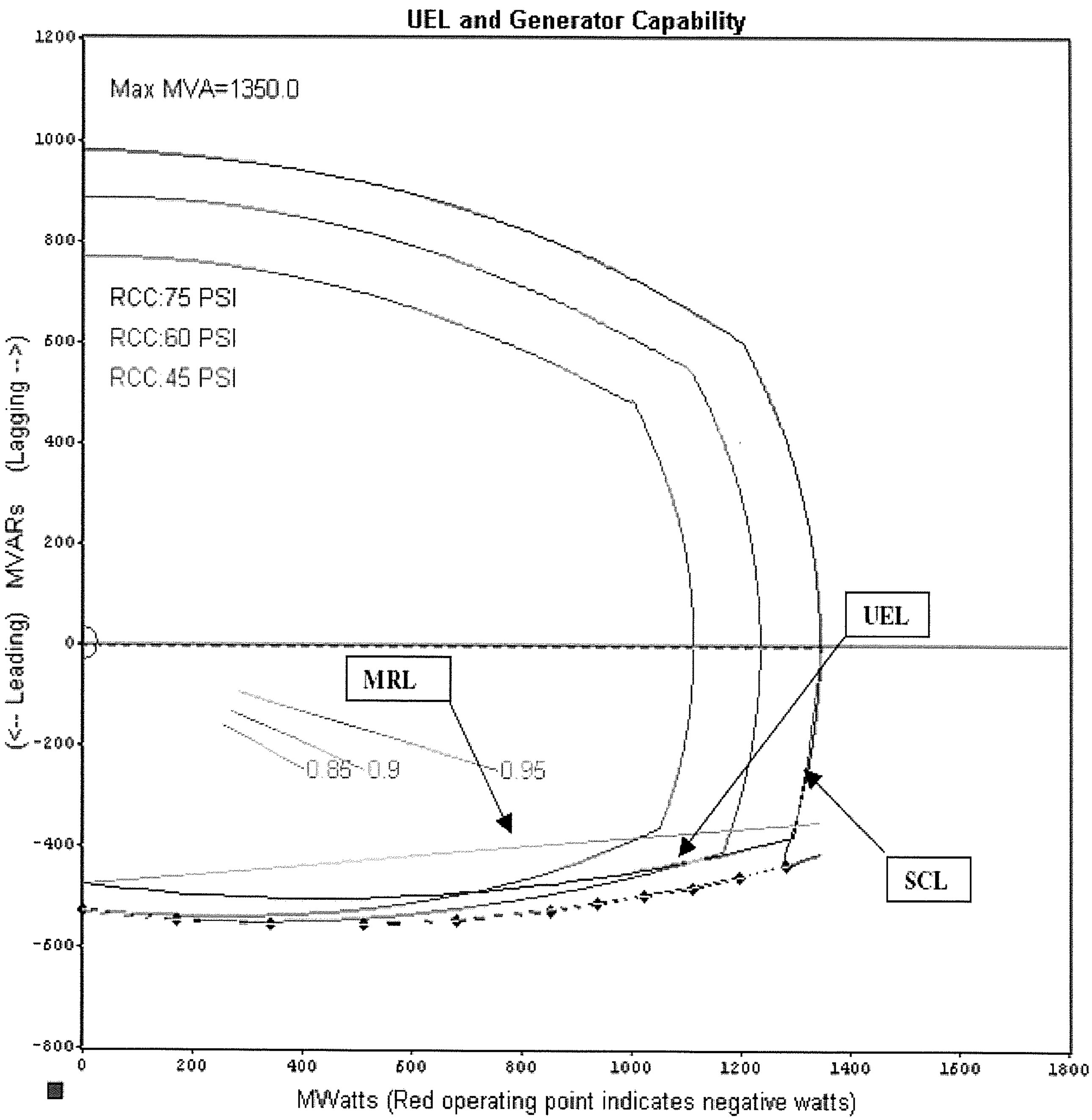
A. DEGRADED GRID CONDITIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__A1. Check Diesel Generators - IN STANDBY.	__A1. Restore Diesel Generators to operable status.
__A2. Terminate maintenance or testing activities on critical electrical distribution components.	
__A3. Check Main Generator Power System Stabilizer on COI - PSS ENABLED.	__A3. Perform actions of TABLE 1, as necessary.
__A4. Initiate the Continuous Actions Page.	
__*A5. **Maintain Main Generator - OPERATING WITHIN THE REACTIVE CAPABILITY CURVE OF FIGURE 1.**	__A5. <u>IF</u> Main Generator can <u>NOT</u> be maintained within the capability curve, <u>THEN</u> trip the reactor and initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
A6. Place the following on alternate power supply using 13800, MAIN TURBINE OPERATION: __• Main Turbine Turning Gear __• Turning Gear Oil Pump	
A7. Verify Turning Gear Oil Pump: __With turbine on line - IN AUTO. -OR- __With turbine on turning gear - IN OPERATION.	__Verify Main Turbine Auxiliary Emergency DC Lube Oil Pump is operating.

FIGURE 1

Sheet 1 of 1

REACTIVE CAPABILITY CURVE



52. 078K4.02 001/2/1/IA - XCONN SSA/MEM - 3.2/3.5/NEW/HL-18 NRC/RO/SRO/AML

Service Air to the Spent Fuel Pool gate seals is to be tagged out.

Which ONE of the following completes the following statement?

INSTRUMENT AIR from __(1)__ will __(2)__ to supply an alternate source of air to the Spent Fuel Pool gate seals.

A. (1) Unit 1

(2) automatically align

B✓ (1) Unit 1

(2) have to be manually aligned

C. (1) Unit 2

(2) automatically align

D. (1) Unit 2

(2) have to be manually aligned

078K4.02 Instrument Air System (IAS)

Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following:
(CFR 41.7)

Crossover to other air systems.

K/A MATCH ANALYSIS:

Question matches KA by testing students knowledge on the source of backup air to the gate seals and how this is accomplished.

ANSWER / DISTRACTOR ANALYSIS:

A. Incorrect - Plausible that students may think there is an automatic backup of air, instrument air from Unit 1 is the only air backup.

B. Correct.

C. Incorrect - Plausible that students may think there is an automatic backup of air, instrument air from Unit 1 is the only air backup and has to be aligned manually.

D. Incorrect - The first half of this choice is incorrect, the 2nd half for manual alignment is correct.

REFERENCES:

13713, section 4.5, pg 18

VEGP learning objectives:

LO-LP-25102-07 Describe the design and operation of the SFP gates & gate seals.

INITIALS

4.5 ALTERNATE GATE SEAL SUPPLY

4.5.1 Unit One (1) Instrument Air

NOTES

- Approximately 50 feet of air hose and a hose coupling for 1-inch line are required for this method of gate seal air supply. □
- This section is not intended for long term (permanent) system operation and should be used on an as-needed basis. The system should be restored to normal operation when Service Air is available. □

4.5.1.1 **Obtain** SS approval to install air supply hose from Unit One instrument air AND document on Checklist 1. ________

4.5.1.2 **Verify** CLOSED SERVICE AIR FHB LEVEL 1 UTILITY STA 13108 ISO 1-2401-U4-139. ________

4.5.1.3 **Install** air hose at Valve FHB LEVEL 1 UTILITY STA 13108 ISO 1-2401-U4-139. ________

4.5.1.4 **Close** INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098. ________

4.5.1.5 **Remove** cap AND **install** air coupling to 1-inch line at INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098. ________

4.5.1.6 **Connect** air hose to coupling at INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098. ________

4.5.1.7 **Open** INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098. ________

4.5.1.8 **Close** SERVICE AIR TO UTILITY STATION FHB LEVEL 1 ISOLATION 1-2401-U4-112. ________

4.5.1.9 **Open** FHB LEVEL 1 UTILITY STA 13108 ISO 1-2401-U4-139. ________

4.5.1.10 **Verify** Gate Seal pressure maintained between 45-55 psig. ________

		INITIALS
4.5.1.11	**Notify** SS of installation AND **install** Temporary Modification Tag with this procedure referenced on installed air hose and document on Checklist 1.	________
4.5.1.12	WHEN normal alignment can be restored, **obtain** SS approval to restore Gate Seals to Unit 1 service air, AND **proceed** with this section.	________
4.5.1.13	**Close** the following valves:	
	a. FHB LEVEL 1 UTILITY STA 13108 ISO 1-2401-U4-139	________
	b. INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098	________
4.5.1.14	**Open** SERVICE AIR TO UTILITY STATION FHB LEVEL 1 ISOLATION 1-2401-U4-112.	________
4.5.1.15	**Remove** air hose from 1-2401-U4-139 AND 1-2420-U4-098.	________
4.5.1.16	**Remove** air coupling AND **install** cap at INTR AIR FUEL HDLG BUILDING LEVEL 1 ISO 1-2420-U4-098 AND document on Checklist 1.	________
4.5.1.17	**Notify** SS of system restoration AND document on Checklist 1.	________

53. 079G2.1.23 001/2/2/STATION AIR/C/A - 4.3/4.4/NEW/HL-18 NRC/RO/SRO/KAJ

Initial conditions:

- Unit 2 is in Mode 5.
- The RCS is on solid-plant pressure control.
- The NCP is in service.
- 'B' RHR is in service for RCS temperature control.

Current conditions:

- Unit 2 instrument air has been lost.
- The crew is implementing 18028-C, "Loss of Instrument Air," Attachment B, "Loss of Instrument Air in Modes 4, 5, or 6."

Given the conditions above, which ONE of the following completes the following statement?

Per 18028-C, the NCP ___(1)___ required to be tripped,

and

if RCS temperature is RISING, the preferred method to limit temperature rise is by ___(2)___.

A.✓ (1) is

(2) operating available ARV(s)

B. (1) is

(2) stopping all but one RCP

C. (1) is NOT

(2) operating available ARV(s)

D. (1) is NOT

(2) stopping all but one RCP

079G2.1.23 Station Air

Ability to perform specific system and integrated plant procedures during all modes of operations:
(CFR 41.10 / 43.5 / 45.2 / 45.6)

K/A MATCH ANALYSIS:

The candidate is given a plausible scenario where instrument air is lost while the RCS is in solid plant operation with the NCP in service along with RHR Train B. The candidate must determine the correct actions to take regarding the NCP and the proper method to limit the RCS temperature rise.

DISTRACTOR ANALYSIS:

A. Correct. The NCP is required to be tripped to prevent overpressuring the RCS. 18028-C requires use of the ARVs for cooling if necessary.

B. Incorrect. The NCP is required to be tripped to prevent overpressuring the RCS. 18028-C requires use of the ARVs for cooling if necessary. It is plausible the candidate may think stopping RCPs is the correct choice since many EOPs stop RCPs to limit heatup. Stopping the RCPs is listed in the RNO for step B7 if the ARVs are NOT available.

C. Incorrect. The NCP is required to be tripped to prevent overpressuring the RCS. 18028-C requires use of the ARVs for cooling if necessary.

D. Incorrect. The NCP is required to be tripped to prevent overpressuring the RCS. 18028-C requires use of the ARVs for cooling if necessary. It is plausible the candidate may think stopping RCPs is the correct choice since many EOPs stop RCPs to limit heatup. Stopping the RCPs is listed in the RNO for step B7 if the ARVs are NOT available.

REFERENCES:

18028-C, "Loss of Instrument Air", Attachment B.

VEGP learning objectives:

LO-LP-60321-08 Describe the effects on RCS pressure due to a loss of instrument air while solid on RHR.

LO-LP-60321-11 Given the entire AOP, describe:

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

ATTACHMENT B

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

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B1. Check Instrument Air supply header pressure on PI-9361 - LESS THAN 100 PSIG.	__B1. Go to Step B12.
__B2. IF a temporary air compressor is connected to the Service Air Header. Perform Attachment C while continuing with Attachment B.	

CAUTION

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

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B3. Check RCS inventory – SOLID.	B3. Perform the following: __a. IF needed to maintain RCS level, THEN establish safety grade charging by initiating 13006, CHEMICAL AND VOLUME CONTROL SYSTEM. __b. Go to Step B6.
__B4. Trip all charging pumps.	
__*B5. **Monitor No. 1 seal leakoff temperature and flow until charging pump is restarted.**	

ATTACHMENT B

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

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION Loss of instrument air pressure will cause the RHR HX outlet valves to fail full open and the HX bypass valves to fail fully closed.	
__B6. Check plant Mode - MODE 4 OR MODE 5.	__B6. Suspend all fuel movement.
__B7. Check RCS temperatures – LOWERING.	B7. Perform the following: __a. Control temperature using ARVs. __b. IF ARVs NOT available, THEN stop all but one RCP.
__*B8. **Check RCS cooldown rate - GREATER THAN 100°F/HR.**	*B8. Perform the following: __a. Monitor cooldown rate. __b. IF cooldown rate can NOT be maintained less than 100°F/hr, THEN perform Step B9. __c. Go to Step B10.

54. 086A3.03 001/2/2/FIRE PROT - ACTUAT/C/A-2.9/3.3/BANK - VOGTLE 2005/HL-18 NRC/RO/SRO/AML

Given the following sequence of events:

- Both Units are operating at 100% power.
- DG1A is in an extended outage period and is unavailable.
- A loss of both RATs occurs.
- DG1B starts and then trips on overspeed.
- You receive a report that RAT 1A is on fire.

Which ONE of the following correctly describes (1) the automatic response of the fire detection and (2) protection systems, given the above sequence of events?

A. (1) The fire detection system will detect the fire.

(2) The diesel fire pump(s) will start, but the clapper valve(s) must be manually tripped.

B. (1) The fire detection system will NOT detect the fire.

(2) The diesel fire pump(s) must be manually started by pulling up the lever on the Primary Emergency Start Contactor.

C. (1) The fire detection system will NOT detect the fire.

(2) The diesel fire pump(s) must be manually started using the Alternate Emergency Start Contactor.

D.✓ (1) The fire detection system will detect the fire.

(2) The diesel fire pump(s) will start and deliver water to RAT 1A without operator action.

086A3.03 Fire Protection System (FPS)

Ability to monitor automatic operation of the Fire Protection System including: (CFR 41.10 / 43.5 / 45.2 / 45.6)

Actuation of the fire detectors.

<u>K/A MATCH ANALYSIS:</u>

An operator must know what to anticipate for system operation in order to effectively monitor the system. Therefore, the K/A is met because the applicant must know how the system should automatically respond given the sequence of events presented in the question. The candidate must also know the fire detection system will work on a

loss of all AC power. Fire detection system has battery backup in the event of a loss of AC power. The diesel fire pumps will also start to deliver water in a loss of power event as their control system also has battery backup.

DISTRACTOR ANALYSIS:

A. Incorrect The diesel pumps will deliver flow automatically due to battery backup. The fire detection system will also detect the fire due to battery backup.

B. Incorrect. The diesel pumps will deliver flow automatically due to battery backup. The fire detection system will also detect the fire due to battery backup. It is plausible to think the detection system will not detect the fire on loss of power. Also, if the candidate thinks the fire pumps don't auto start, using emergency contactors is a method to start the fire pumps.

C. Incorrect. The fire detection system will detect the fire due to battery backup. It is plausible to think the detection system will not detect the fire on loss of power. The diesel pumps will deliver flow automatically due to battery backup. Also, if the candidate thinks the fire pumps don't auto start, using emergency contactors is a method to start the fire pumps.

D. Correct. Battery backup exists for detection and diesel fire pump starting. The system will flow water to the RAT without any operator action.

REFERENCES:

V-LO-TX-22101, Fire Detection.
V-LO-TX-43101, Fire Protection.
Vogtle 2005 NRC Exam question # 56 (HL-13, NOT in the last 2 exams given)
17103A-C, Annunciator Response for Fire Alarm Computer

VEGP learning objectives:

LO-PP-22101-01 Describe the five electronic fire detector types used at Plant Vogtle, how each operates to generate a fire alarm signal, and the areas typically covered by that type of detection.

LO-PP-22101-04 Describe the following regarding the Excel Life Safety (XLS) Local Computer Display Cabinet:

a. Four types of alarms that can be received.
b. How an Operator can determine the type alarm received.
c. How an operator can determine the location for the point in alarm.

Section C: SYSTEM TROUBLE RESPONSE

C1.0 PROBABLE CAUSE

a. System, component, or circuit trouble.

b. System or component misalignment. (Table 2 and the appendices provide specific listings of all causes for detection, suppression, water supply, or computer system troubles.)

C2.0 AUTOMATIC ACTIONS

a. Loss of power to any LSIP, MUX, or the CPU will cause a trouble alarm and automatic swap to battery backup.

b. Loss of power to any LZIP will cause an LZIP POWER FAILURE alarm at the FAC and a loss of indication for all associated zones.

C3.0 INITIAL ACTIONS

None

C4.0 SUBSEQUENT OPERATOR ACTIONS

C4.1 **Determine** the affected LZIP and its location using the Fire Alarm Computer and or Table 2.

C4.2 **Dispatch** Operator to trouble location, to silence/acknowledge the local panel using the TROUBLE OFF or ALARM SILENCE switch.

a. **Reset** LZIP per 17103B-C, "Annunciator Response Procedures for Fire Alarm Computer," Appendix 2.

b. **Reset** the LSIP per 17103B-C, "Annunciator Response Procedures for Fire Alarm Computer," Appendix 3.

Section C: SYSTEM TROUBLE RESPONSE (Continued on next page)

INITIALS

2.2.6 Per DOEJ-SM-C070400401-001, the Portable B.5.b Pump cannot be considered a backup fire suppression system as required by FP LCO 4.3 Cond A or B. The Portable B.5.b Pump is a defense in depth contingency for the interim period between total loss of suppression capability and arrival of Burke County EMA. A Burke County EMA pumper truck is the credited backup. ________

2.2.7 Per DOEJ-SM-C070400401-001, all hot work shall be suspended and hourly Fire Watches shall be established for 1A, 1B, 2A, and 2B Cable Spreading Rooms if a backup suppression system has to be established in either FP LCO 4.3 Cond A or B. ________

2.2.8 A Condition Report should be generated anytime two jockey pumps are required running to maintain Fire Protection System pressure. ________

2.2.9 Fire pumps auto start on decreasing header pressure at:

110 psig Electric fire pump starts ________

95 psig #1 Diesel fire pump starts ________

85 psig #2 Diesel fire pump starts ________

2.2.10 Attachment 1 provides guidance for developing troubleshooting steps in the event a low header pressure condition is encountered with no obvious explanation. A troubleshooting plan in accordance with NMP-GM-002 "Problem Solving and Troubleshooting Guidelines" must still be completed using these guidelines. (LVL 3 AI 2009201678) ________

3.0 PREREQUISITES

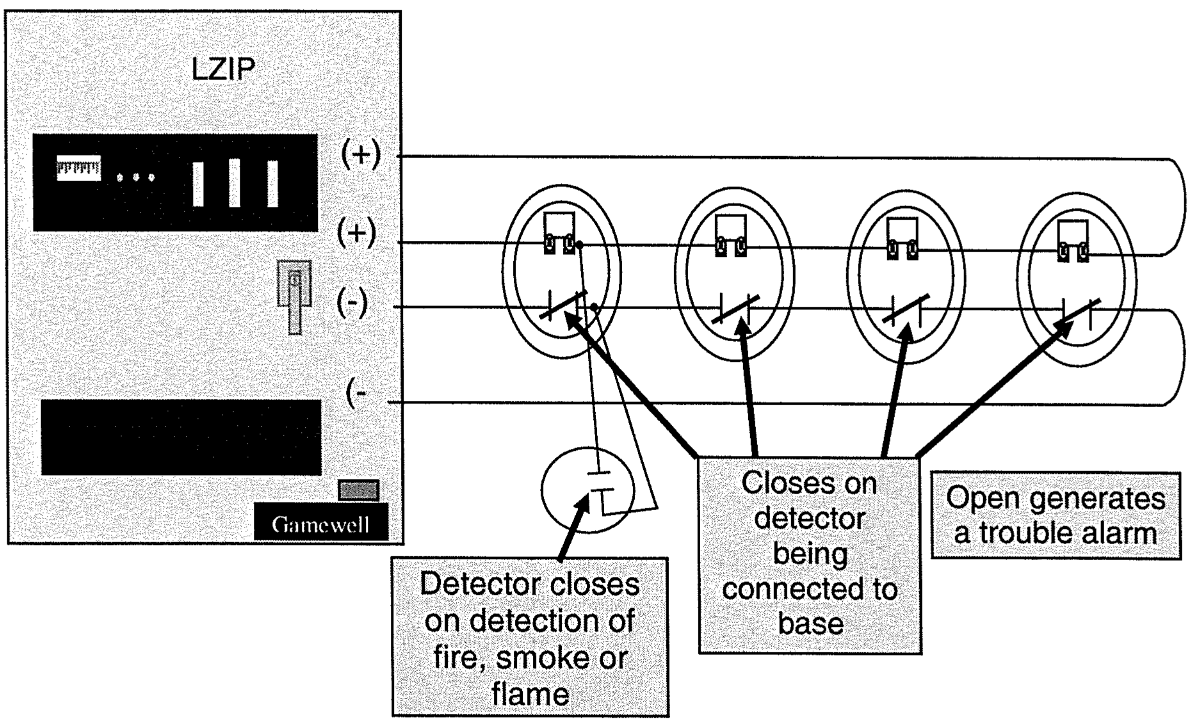
3.1 One or two Jockey Pump(s) is/are running to maintain or attempting to maintain the Fire Protection System pressure. ________

3.2 The Fire Pump House No. 1 and/or Fire Pump House No. 2 Heating, Ventilation, and Air Conditioning System (HVAC) has been aligned per 13330-C, "Outside Area Buildings HVAC System." ________

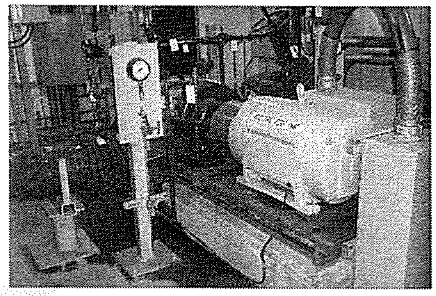
C. Local Zone Indicating Panels (LZIPs)

Local zone indicating panels (LZIPs) are found throughout the plant. Inside the LZIP, a 24VDC signal is sent out in series to up to 40 detectors in a specific zone as in Figure 5. The LZIPs process fire signal information for multiple areas (maximum of 10 zones) and generate local alarms, send remote alarm signals to computer multiplexing cabinets, and send actuation signals to local suppression indicating panels (LSIPs). LZIPs provide operators with local information concerning fire zone status. The panels are equipped with zone alarm, zone trouble, and audible signal circuit trouble displays. Power Block area LZIPs are provided with uninterruptible power source (UPS). LZIPS located outside of the power block area are provided with a backup 24 hour battery supply. The LZIPs receive their signal from detectors or manual pull stations.

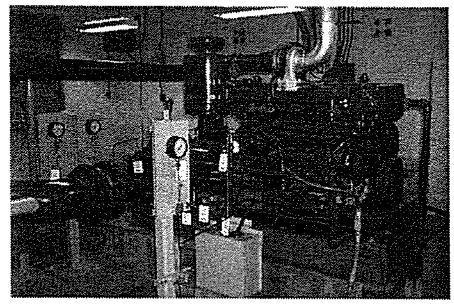
Figure 5



In the event that the header pressure starts falling, either due to a spray actuation or system leaks, at 110psi the electric fire pump (P4-002) will auto start. This pump delivers approximately 2500 gpm @ 125psi and is powered from ANA02. This pump can be started locally or from the control room at the QPCP, but can only be stopped locally.

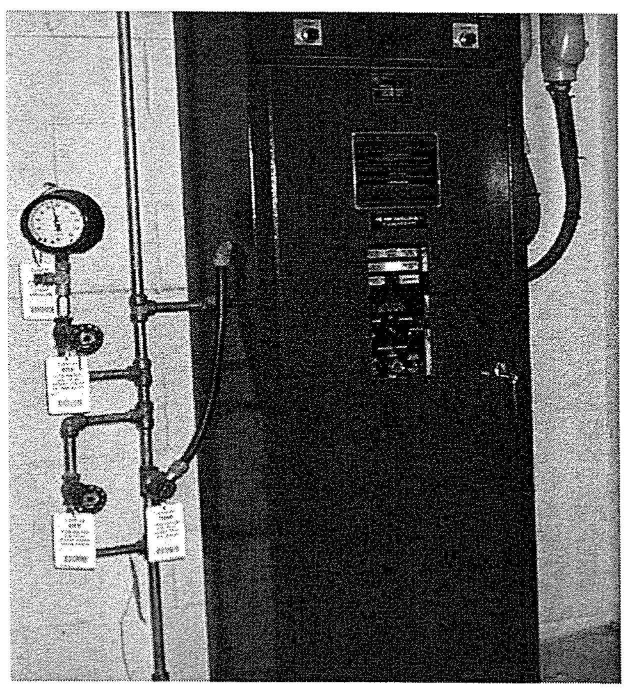


If pressure continues to lower to < 95psi, then DF pump #1 (P4-005) will auto start. And if the pressure continues to fall below 85 pounds DF pump #2 (P4-003) will auto start. Both diesel pumps are rated at 2500 gpm @125 psi.



Each of the diesel-driven fire pumps is provided with a 5-position control switch located on local control panels (pump #1 PFP1; pump #2 PFP2). The diesel-driven fire pump handswitch has the following positions:

* **OFF** - stops the engine if it is running; causes an alarm in the main control room
* **AUTO** - allows the pump to auto start on lowering Fire Protection System pressure (95 psig for pump #1 and 85 psig for pump #2)
* **TEST** - opens a solenoid drain valve (pump #1 KV-7990; pump #2 KV-7907) located on the pressure sensing line to the low header pressure switch: the pump will start on sensed low pressure. This position is selected to simulate a header low pressure at the diesel control and provides the primary means of starting the diesel locally, also the manual bypass around the solenoid.
* **MANUAL 1** - places battery #1 in service for a manual start (the start/stop pushbuttons are functional in this position)
* **MANUAL 2** - places battery #2 in service for a manual start (the start/stop pushbuttons are functional in this position)



The Electric and Diesel fire pumps can be started from the control room. However, they can **NOT** be stopped from the control room. They must be stopped locally.

1. 086A3.02 001/2/2/FIRE PROTECTION/C/A 2.9/B/VG05301/R/MAB/RSB

The following sequence of events occurs:

- Both Units were operating at 100% power.
- Diesel Generator (DG) 1A is in an extended outage period and is unavailable
- Both Unit 1 Reserve Auxiliary Transformers (RAT) trip
- DG 1B starts and then trips on overspeed
- You receive a report that RAT 1A is on fire

Which ONE of the following correctly describes the automatic response of the fire detection and protection given the above sequence of events?

A. The fire detection system will detect the fire. The diesel fire pump(s) will start, but the clapper valve(s) must be manually tripped.

B. The fire detection system will not detect the fire. The diesel fire pump(s) must be manually started by pulling up the lever on the Primary Emergency Start Contactor.

C. The fire detection system will not detect the fire. The diesel fire pump(s) must be manually started using the Alternate Emergency Start Contactor.

D. The fire detection system will detect the fire. The diesel fire pump(s) will start and deliver water to RAT 1A without operator action.

55. 103A3.01 001/2/1/CONT - ISOLATION/MEM - 3.9/4.2/MOD-BANK FARLEY/HL-18 NRC/RO/SRO/AML

Given the following conditions:

- Unit 1 is at 100% power.
- CNMT Mini-Purge Supply and Exhaust fans are running for a batch release in accordance with 13125-1, "Containment Purge System."
- During replacement of a faulty MCB CIA/CVI Actuation handswitch, a MANUAL CIA/CVI signal is inadvertently generated.

Which ONE of the following predicts the plant response and completes the statements below?

The CIA/CVI signal ___(1)___ DIRECTLY stop the CTMT Mini-Purge supply and exhaust fans,

and

the CIA/CVI signal __(2)__ DIRECTLY close ALL of the valves listed below.

<u>Valve names</u>

HV-9451, SG1 SGBD Sample Iso
HV-9452, SG2 SGBD Sample Iso
HV-9453, SG3 SGBD Sample Iso
HV-9454, SG4 SGBD Sample Iso

HV-2629B, CTB Mini Purge Exh ORC Iso Vlv-Mini
HV-2628B, Norm Purge Exh IRC Iso Vlv-Mini
HV-2627B, CTB Norm Purge Sply ORC Iso Vlv-Mini
HV-2626B, CTB Norm Purge Sply IRC Iso Vlv-Mini

	__(1)__	__(2)__
A.✓	will NOT	will NOT
B.	will NOT	will
C.	will	will
D.	will	will NOT

103A3.01 Containment System

Ability to monitor automatic operation of the containment system, including:

(CFR 41.7 / 45.5)

Containment isolation.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a spurious CIA/CVI signal is produced. The candidate has to determine if the CNMT Mini-Purge fans receive a direct signal to stop and from a list of CNMT isolation valves, whether or not the ALL receive an automatic close signal.

DISTRACTOR ANALYSIS:

A. Correct. The mini-purge fans do NOT receive a direct trip signal, however, they will trip on low flow once the supply and exhaust dampers shut. It is plausible the students may confuse this since the fans do eventually trip on low flow.

The list of 4 mini-purge valves receive a direct closure signal on CIA/CVI, however, the SGBD valves close on an AFW actuation signal.

B. Incorrect. The mini-purge fans do NOT receive a direct trip signal, however, they will trip on low flow once the supply and exhaust dampers shut. It is plausible the students may confuse this since the fans do eventually trip on low flow.

The list of 4 mini-purge valves receive a direct closure signal on CIA/CVI, however, the SGBD valves close on an AFW actuation signal.

C. Incorrect. The mini-purge fans do NOT receive a direct trip signal, however, they will trip on low flow once the supply and exhaust dampers shut. It is plausible the students may confuse this since the fans do eventually trip on low flow.

The list of 4 mini-purge valves receive a direct closure signal on CIA/CVI, however, the SGBD valves close on an AFW actuation signal.

D. Incorrect. The mini-purge fans do NOT receive a direct trip signal, however, they will trip on low flow once the supply and exhaust dampers shut. It is plausible the students may confuse this since the fans do eventually trip on low flow.

The list of 4 mini-purge valves receive a direct closure signal on CIA/CVI, however, the SGBD valves close on an AFW actuation signal.

REFERENCES:

Farley 2012 May NRC
19100-C, Loss of All AC Power, Attachment C, CVI
1X5DN011-1, Preaccess Purge Supply Unit Logic Diagram
1X6AA02-00239, Auxiliary Feedwater Actuation

VEGP learning objectives:

LO-PP-29101-21 State any auto actions that occur in the systems listed as a result of the following signals: SI, High Rad, and CVI.

d. Mini Purge

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

ATTACHMENT C

Sheet 1 of 2

CONTAINMENT VENTILATION ISOLATION

COMPUTER POINT	VALVE (LOCATION)		DESCRIPTION**
	UNIT 1	UNIT 2	
ZD9044	HV-12975 (CNMT)	HV-12975 (CNMT)	CNMT AIR RAD MON SPLY ISO IRC
ZD9046	HV-12976 (AB-B08)	HV-12976 (AB-B131)	CNMT AIR RAD MON SPLY ISO ORC
ZD9048	HV-12977 (AB-B08)	HV-12977 (AB-B131)	CNMT AIR RAD MON RTN ISO ORC
ZD9050	HV-12978 (CNMT)	HV-12978 (CNMT)	CNMT AIR RAD MON RTN ISO IRC
ZD9204	HV-2626A (CNMT)	HV-2626A (CNMT)	CTB NORM PURGE SPLY IRC ISO VLV-MAIN***
ZD9206	HV-2626B (CNMT)	HV-2626B (CNMT)	CTB NORM PURGE SPLY IRC ISO VLV-MINI
ZD9208	HV-2627A (EB-125)	HV-2627A (EB-125)	CTB NORM PURGE SPLY ORC ISO VLV-MAIN***
ZD9210	HV-2627B (EB-125)	HV-2627B (EB-125)	CTB NORM PURGE SPLY ORC ISO VLV-MINI
ZD9212	HV-2628A (CNMT)	HV-2628A (CNMT)	CTB NORM PURGE EXH IRC ISO VLV-MAIN***
ZD9214	HV-2628B (CNMT)	HV-2628B (CNMT)	CTB NORM PURGE EXH IRC ISO VLV-MINI
ZD9216	HV-2629A (EB-117)	HV-2629A (EB-116)	CTB NORM PURGE EXH ORC ISO VLV-MAIN***
ZD9218	HV-2629B (EB-117)	HV-2629B (EB-116)	CTB MINI PURGE EXH ORC ISO VLV-MINI
ZD9236	HV-2624A (CNMT)	HV-2624A (CNMT)	CTB POST LOCA PURGE EXH IRC ISO VLV***

ATTACHMENT C

Sheet 2 of 2

CONTAINMENT VENTILATION ISOLATION

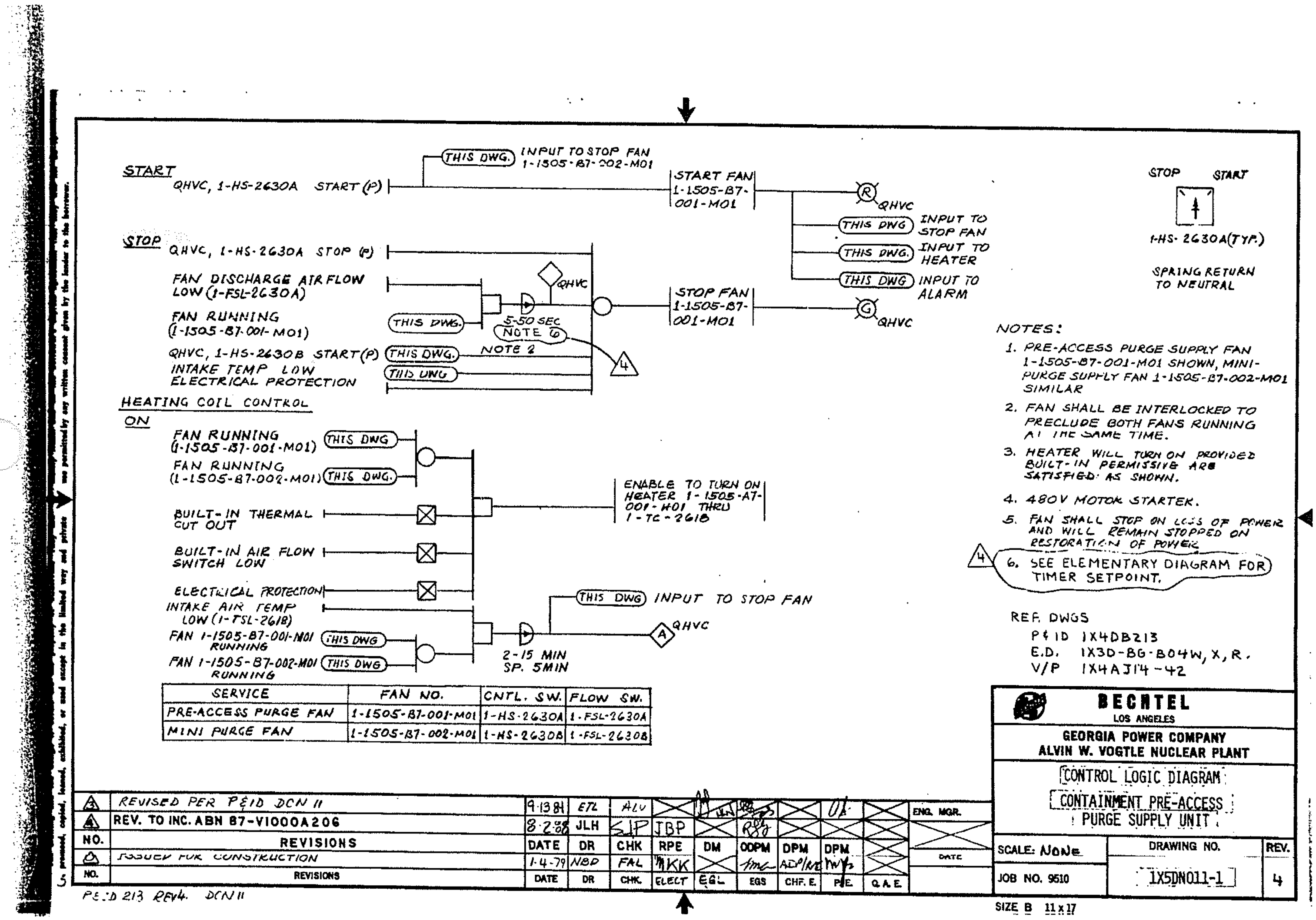
COMPUTER POINT	VALVE (LOCATION)		DESCRIPTION**
	UNIT 1	UNIT 2	
ZD9238	HV-2624B (CNMT)	HV-2624B (CNMT)	CTB POST LOCA PURGE EXH IRC ISO VLV***
ZD9583	HV-12604 (AB-209)	HV-12604 (AB-220)	PIPING PEN RM OUTLET ISO DMPR
ZD9587	HV-12605 (AB-209)	HV-12605 (AB-220)	PIPING PEN RM INLET ISO DMPR
ZD9589	HV-12606 (AB-209)	HV-12606 (AB-220)	PIPING PEN RM INLET ISO DMPR
ZD9585	HV-12607 (AB-209)	HV-12607 (AB-220)	PIPING PEN RM OUTLET ISO DMPR
NONE*	HV-12596 (AB-C83 Above Block Out To Tank)	HV-12596 (AB-C49 Above Block Out To Tank)	RECYCLE HOLD-UP TK-1 ISO VENT VLV
NONE*	HV-12597 (AB-C83 Above Block Out To Tank)	HV-12597 (AB-C49 Above Block Out To Tank)	RECYCLE HOLD-UP TK-1 ISO VENT VLV

* USE QHVC INDICATION IF AVAILABLE OR LOCALLY VERIFY.

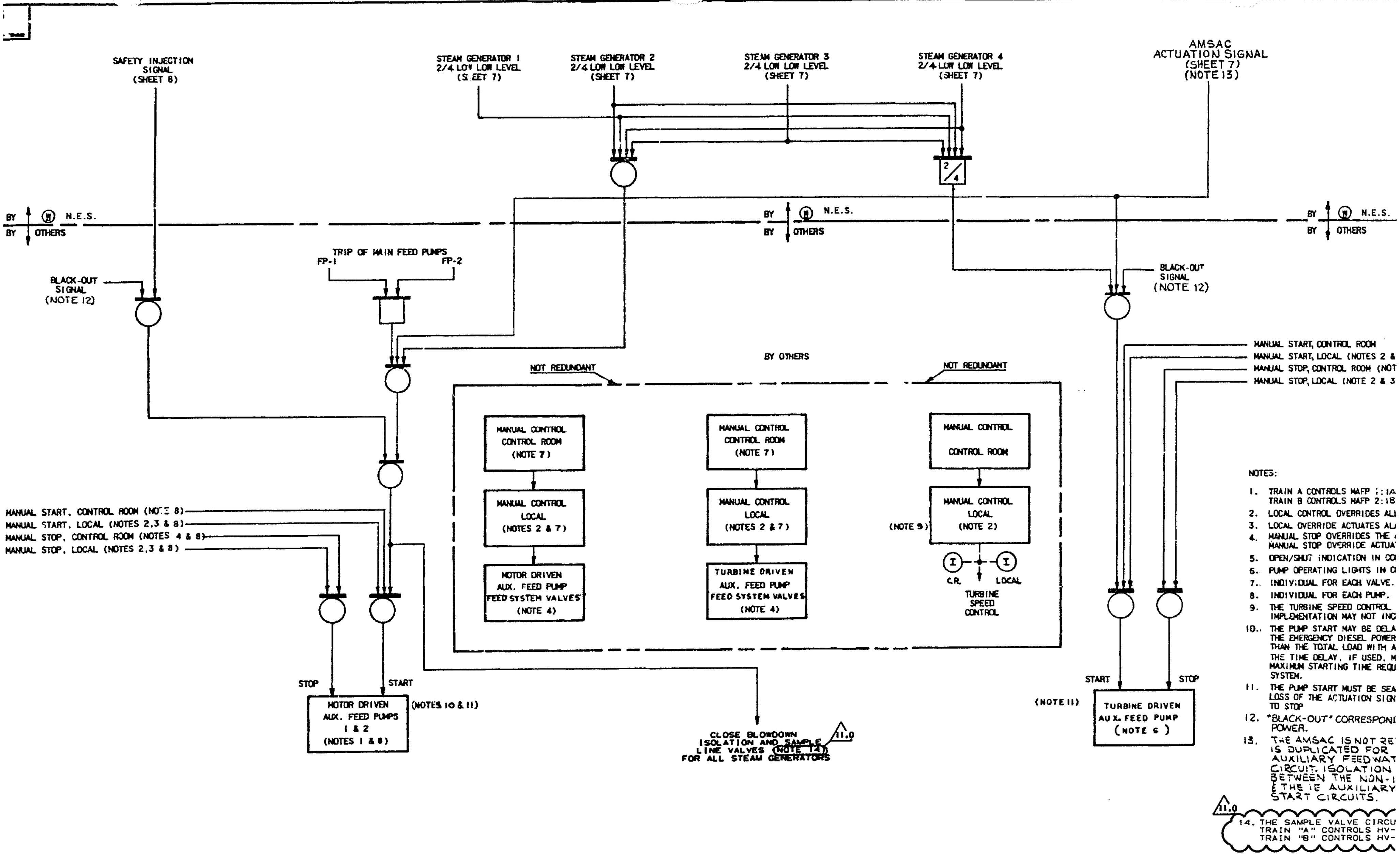
** USE COMPUTER "CNMT ISOLATION VALVES" DISPLAY FOR POSITION VERIFICATION ON VALVES AND DAMPERS AS APPLICABLE.

*** MOTOR OPERATED.

END OF ATTACHMENT C



START
QHVC, 1-HS-2630A START (P)
THIS DWG. INPUT TO STOP FAN 1-1505-B7-002-M01
START FAN 1-1505-B7-001-M01
QHVC
THIS DWG INPUT TO STOP FAN
THIS DWG. INPUT TO HEATER
THIS DWG INPUT TO ALARM
STOP
QHVC, 1-HS-2630A STOP (P)
FAN DISCHARGE AIR FLOW LOW (1-FSL-2630A)
FAN RUNNING (1-1505-B7-001-M01)
THIS DWG.
5-50 SEC
NOTE 6
NOTE 2
QHVC, 1-HS-2630B START (P)
INTAKE TEMP LOW
ELECTRICAL PROTECTION
STOP FAN 1-1505-B7-001-M01
HEATING COIL CONTROL
ON
FAN RUNNING (1-1505-B7-001-M01)
FAN RUNNING (1-1505-B7-002-M01)
BUILT-IN THERMAL CUT OUT
BUILT-IN AIR FLOW SWITCH LOW
ELECTRICAL PROTECTION
INTAKE AIR TEMP LOW (1-TSL-2618)
FAN 1-1505-B7-001-M01 RUNNING
FAN 1-1505-B7-002-M01 RUNNING
ENABLE TO TURN ON HEATER 1-1505-A7-001-H01 THRU 1-TC-2618
THIS DWG INPUT TO STOP FAN
2-15 MIN SP. 5MIN
STOP START
1-HS-2630A (TYP.)
SPRING RETURN TO NEUTRAL
NOTES:
1. PRE-ACCESS PURGE SUPPLY FAN 1-1505-B7-001-M01 SHOWN, MINI-PURGE SUPPLY FAN 1-1505-B7-002-M01 SIMILAR
2. FAN SHALL BE INTERLOCKED TO PRECLUDE BOTH FANS RUNNING AT THE SAME TIME.
3. HEATER WILL TURN ON PROVIDED BUILT-IN PERMISSIVE ARE SATISFIED AS SHOWN.
4. 480V MOTOR STARTER.
5. FAN SHALL STOP ON LOSS OF POWER AND WILL REMAIN STOPPED ON RESTORATION OF POWER
6. SEE ELEMENTARY DIAGRAM FOR TIMER SETPOINT.
REF. DWGS
P&ID 1X4DB213
E.D. 1X3D-BG-B04W, X, R.
V/P 1X4AJ14-42
SERVICE | FAN NO. | CNTL. SW. | FLOW SW.
PRE-ACCESS PURGE FAN | 1-1505-B7-001-M01 | 1-HS-2630A | 1-FSL-2630A
MINI PURGE FAN | 1-1505-B7-002-M01 | 1-HS-2630B | 1-FSL-2630B
BECHTEL
LOS ANGELES
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
CONTROL LOGIC DIAGRAM
CONTAINMENT PRE-ACCESS PURGE SUPPLY UNIT
REVISED PER P&ID DCN 11
REV. TO INC. ABN 87-V1000A206
ISSUED FOR CONSTRUCTION
NO. REVISIONS DATE DR CHK RPE DM ODPM DPM DPM
9-13-81 ETL ALV
8-2-88 JLH
1-4-79 NBP FAL
ELECT EGL EGS CHF. E. PE. Q.A.E.
ENG. MGR.
SCALE: NONE
DRAWING NO. 1X5DN011-1
REV. 4
JOB NO. 9510
SIZE B 11x17
P&ID 213 REV4. DCN 11


SAFETY INJECTION SIGNAL (SHEET 8)
STEAM GENERATOR 1 2/4 LOW LOW LEVEL (SHEET 7)
STEAM GENERATOR 2 2/4 LOW LOW LEVEL (SHEET 7)
STEAM GENERATOR 3 2/4 LOW LOW LEVEL (SHEET 7)
STEAM GENERATOR 4 2/4 LOW LOW LEVEL (SHEET 7)
AMSAC ACTUATION SIGNAL (SHEET 7) (NOTE 13)
2/4
BY N.E.S.
BY OTHERS
BLACK-OUT SIGNAL (NOTE 12)
TRIP OF MAIN FEED PUMPS
FP-1
FP-2
BY OTHERS
NOT REDUNDANT
NOT REDUNDANT
MANUAL CONTROL CONTROL ROOM (NOTE 7)
MANUAL CONTROL LOCAL (NOTES 2 & 7)
MOTOR DRIVEN AUX. FEED PUMP FEED SYSTEM VALVES (NOTE 4)
MANUAL CONTROL CONTROL ROOM (NOTE 7)
MANUAL CONTROL LOCAL (NOTES 2 & 7)
TURBINE DRIVEN AUX. FEED PUMP FEED SYSTEM VALVES (NOTE 4)
MANUAL CONTROL CONTROL ROOM
MANUAL CONTROL LOCAL (NOTE 2)
(NOTE 9)
C.R.
LOCAL
TURBINE SPEED CONTROL
MANUAL START, CONTROL ROOM (NOTE 8)
MANUAL START, LOCAL (NOTES 2,3 & 8)
MANUAL STOP, CONTROL ROOM (NOTES 4 & 8)
MANUAL STOP, LOCAL (NOTES 2,3 & 8)
STOP
START
MOTOR DRIVEN AUX. FEED PUMPS 1 & 2 (NOTES 1 & 8)
(NOTES 10 & 11)
CLOSE BLOWDOWN ISOLATION AND SAMPLE LINE VALVES (NOTE 14) FOR ALL STEAM GENERATORS
11.0
MANUAL START, CONTROL ROOM
MANUAL START, LOCAL (NOTES 2 &
MANUAL STOP, CONTROL ROOM (NOT
MANUAL STOP, LOCAL (NOTE 2 & 3
START
STOP
(NOTE 11)
TURBINE DRIVEN AUX. FEED PUMP (NOTE 6)
NOTES:
1. TRAIN A CONTROLS MAFP 1:1A TRAIN B CONTROLS MAFP 2:1B
2. LOCAL CONTROL OVERRIDES ALL
3. LOCAL OVERRIDE ACTUATES AL
4. MANUAL STOP OVERRIDES THE MANUAL STOP OVERRIDE ACTUA
5. OPEN/SHUT INDICATION IN CO
6. PUMP OPERATING LIGHTS IN C
7. INDIVIDUAL FOR EACH VALVE.
8. INDIVIDUAL FOR EACH PUMP.
9. THE TURBINE SPEED CONTROL IMPLEMENTATION MAY NOT INC
10. THE PUMP START MAY BE DELA THE EMERGENCY DIESEL POWER THAN THE TOTAL LOAD WITH A THE TIME DELAY, IF USED, M MAXIMUM STARTING TIME REQU SYSTEM.
11. THE PUMP START MUST BE SEA LOSS OF THE ACTUATION SIGN TO STOP
12. "BLACK-OUT" CORRESPONI POWER.
13. THE AMSAC IS NOT RE IS DUPLICATED FOR AUXILIARY FEEDWAT CIRCUIT. ISOLATION BETWEEN THE NON-I & THE IE AUXILIARY START CIRCUITS.
11.0
14. THE SAMPLE VALVE CIRCU TRAIN "A" CONTROLS HV- TRAIN "B" CONTROLS HV-

1. 103A3.01 061

Unit 1 is at 100% power when the following error occurs:

- CTMT Mini-Purge Supply and Exhaust fans are running for a batch release in accordance with SOP-12.2, Containment Purge and Pre-access Filtration System.
- During replacement of a faulty MCB Phase A handswitch, a MANUAL Phase A signal is inadvertently generated.

Which one of the following predicts the plant response and completes the statements below?

CTMT Mini-Purge supply and exhaust fans <u>(1)</u> automatically stop.

The <u>(2)</u> will be isolated.

<u>Valve names used below:</u>
HV-3649A/B/C, CONT RM EXH FAN INLET DMPRS
HV-3622, COMPUTER RM HVAC RTN
V-3623, COMPUTER RM HVAC RTN
HV-3328, 1A SGBD SAMPLE ISO
HV-3329, 1B SGBD SAMPLE ISO
HV-3330, 1C SGBD SAMPLE ISO

	(1)	(2)
A.✓	will <u>NOT</u>	Control Room Ventilation System (HV-3649A/B/C, HV-3622, HV-3623)
B.	will <u>NOT</u>	SGBD Sample isolations from containment (HV-3328, HV-3329, HV-3330)
C.	WILL	Control Room Ventilation System (HV-3649A/B/C, HV-3622, HV-3623)
D.	WILL	SGBD Sample isolations from containment (HV-3328, HV-3329, HV-3330)

From A181007,paragraph 2.7.1.2 states:

The containment Ventilation Isolation isolates containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on:

- completion of the SI logic,
- high radioactivity levels in the purge exhaust (RE-24A/B), or
- **MANUAL initiation** of either PHASE A or PHASE B.

From A181003, 2.2.1.3 & 2.2.1.4 states, "**Automatic containment isolation valves shall actuate to close when isolations signals** are received from [... those signals listed above...]." SEE also figure 8 Logic diagram.

56. 103K3.01 001/2/1/CONT - LOSS INTEGR/MEM - 3.3/3.7/NEW/HL-18 NRC/RO/SRO/AML

Given the following:

- The unit is in Mode 6.
- Core off-load is in progress.

Which ONE of the following situations requires immediate suspension of the core off-load? (Consider each individually)

A✓ Energized welding cables are run through the Emergency Escape Airlock doors.

B. One SR NI has been placed on a non-1E temporary power supply.

C. Inner Personnel Airlock door is open and mechanically bound.

D. Containment Purge Supply Fan is declared inoperable.

103K3.01 Containment System

Knowledge of the effect that a loss or malfunction of the containment system will have on the following:
(CFR 41.7 / 45.6)

Loss of containment integrity under shutdown conditions.

<u>K/A MATCH ANALYSIS:</u>

Question meets the KA by testing the students ability to recognize the containment malfunction (loss of integrity) and the actions required per TS requirements.

<u>DISTRACTOR ANALYSIS:</u>

A. Correct - Per 12007-C, energized cables running through the door will require an immediate suspension of Core Alterations.

B. Incorrect - One SR NI placed on a temporary non-1E power source is allowed per Tech Specs as long as the other SR NI is power from its normal power supply.

C. Incorrect - Both air locks may be open as long as at least one is capable of being closed.

D. Incorrect - Do not require operability, therefore fuel movement is not required to be stopped.

REFERENCES:

Tech Spec 3.9.4 and Bases
Tech Spec 3.9.3 and Bases
12007-C, "Refueling Operations (Entry into Mode 6)", pages 11 and 12
Tech Spec Definition 1.1 CORE ALTERATIONS
13501-1, Nuclear Instrumentation System

VEGP learning objectives:

LO-PP-25101-23 State the definition of "Core Alteration" per the Technical Specifications.

LO-PP-25101-27 State the LCO, TR, applicability, and any one hour or less actions for all refueling LCOs and TRs.

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

a. State the LCO.

b. State any one hour or less required actions.

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

a. Whether any Tech Spec LCOs of section 3.6 are exceeded.

b. The required actions for all section 3.6 LCOs.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

BASES

LCO (continued)

action to close containment penetrations to minimize potential offsite doses. The LCO requirements for penetration closure may also be met by the automatic isolation capability of the CVI system. Temporary non-1E power may be supplied to the air operated and/or solenoid operated CVI valves. The temporary non-1E power must be connected in such a way that it cannot affect the capability of the valves to close either automatically or manually from the control room handswitch.

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The personnel and emergency air locks are required by Item b of this LCO to be isolable by at least one air lock door in each air lock. Both containment personnel and emergency air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided at least one air lock door is isolable in each air lock. An air lock is isolable when the following criteria are satisfied:

1. one air lock door is OPERABLE,

2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated individual is available to close the door.

OPERABILITY of a containment air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed.

The equipment hatch is considered isolable when the following criteria are satisfied:

1. the necessary equipment required to close the hatch is available.

2. at least 23 feet of water is maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated trained hatch closure crew is available.

Similar to the air locks, the equipment hatch opening must be capable of being cleared of any obstruction so that closure can be achieved as soon as possible.

(continued)

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

NONE

2.2 LIMITATIONS

2.2.1 The Audio Count Rate Channel should be in operation during refueling operations. Reference TR 13.9.6 for the Source Range Monitor Audible Indication (MODE 6).

2.2.2 Technical Specification LCO 3.3.1 Table 3.3.1-1 and LCO 3.9.3 (MODE 6) should be referenced for the Reactor Trip System Instrumentation and Nuclear Instrumentation number of channels required to be operable.

2.2.3 Technical Specification LCO 3.3.8 should be referenced for the High Flux At Shutdown Alarm requirements.

2.2.4 During normal Operation the Source Range Shutdown Monitors should remain in the HFAS Mode of Operation.

2.2.5 The TEST/STATUS CONTROL Switch at the Optical Isolator Assembly (12CPO1B, in CB B65) is normally aligned to the CTRL ROOM position. The switch is used to select the N32/N36 Signal Processor for drawer testing. In the event of Control Room evacuation due to a fire, the switch should be selected to the APPX R position to prevent loss of Extended Range indication at Shutdown Panel B due to the fire. Under any other condition, inadvertent placement of the switch to the APPX R or CTRL ROOM positions will have no effect on instrument operability.

3.0 PREREQUISITES OR INITIAL CONDITIONS

3.1 The Nuclear Instrumentation is energized using 11431-1, "120V AC 1E Vital Instrumentation Lineup", except during Mode 6 when one Source Range may be energized from a non-safety related power supply as outlined in 3.2 below.

3.2 In Mode 6, one Source Range NI channel may be energized from a non-safety related power supply and still be considered OPERABLE, provided the other Source Range NI channel is energized from its normal 1E power supply. Reference LCO 3.9.3 Bases.

3.3 The Nuclear Instrumentation System has been energized for at least one hour.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND	The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors (NI-0031 and NI-0032) are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Temporary neutron flux detectors which provide equivalent indication may be utilized in place of installed instrumentation. The installed source range neutron flux monitors are fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers seven decades of neutron flux (1E-1 cps to 1E +6 cps) with a 2% instrument accuracy. The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is minimized by isolating all unborated water sources except as provided for by LCO 3.9.2, "Unborated Water Source Isolation Valves." The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE each monitor must provide visual indication. When any of the safety-related busses supplying power to one of the detectors (NI-0031 or NI-0032) associated with the source range neutron flux monitors are taken out of service, the corresponding source range neutron flux monitor may be considered OPERABLE when its detector is powered from a temporary nonsafety-related

(continued)

BASES

LCO (continued)

source of power, provided the detector for the opposite source range neutron flux monitor is powered from its normal source.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the operability requirements for the installed source range detectors and circuitry are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

B.1

Condition B is modified by a Note to clarify the requirement that entry into or continued operation in accordance with Condition A is required for any entry into Condition B. The Note reinforces conventions of LCO applicability as stated in LCO 3.0.2 and as reflected in examples in 1.3, Completion Times.

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in EPA Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA-520/1-88-020, September 1988.

(continued)

INITIALS

2.2.21 During core alterations or movement of irradiated fuel in containment the following conditions must be met: ________

- Greater than or equal to 23 feet of water shall be maintained over the top of the reactor vessel flange, in accordance with Technical Specification 3.9.7 (not required during control rod latching and unlatching). ________

- Direct communications shall be maintained between the Control Room and personnel at the Refueling Station. (TR 13.9.2) (1996332108) ________

- CVI instrumentation and associated alarms shall be Operable and CVI dampers capable of being manually closed upon of receipt of a valid CVI Radiation Monitor Alarm. (TS 3.3.6 and TS 3.9.4 and bases) ________

- At least one airlock door shall be closed or capable of being closed in both the emergency airlock and the personnel airlock. Both containment personnel airlock and/or emergency airlock doors may be open provided the following criteria are met: (1996332046, 1996332308, 1997235109) ________

    - A designated individual is available for each airlock opened and able to close at least one Airlock door within 15 minutes. ________

    - The airlock door seal protectors are easily removed. ________

    - No cables or hoses are run through the airlock. ________

INITIALS

- The containment equipment hatch shall be either, closed and held in place by four bolts, or open provided it is capable of being closed with a minimum of four bolts holding it in place WITHIN 25 minutes. The equipment hatch is considered "capable of being closed" when the following criteria are satisfied: (2000341449) ________

  - Designated, trained hatch closure crew is available, per 27504-C, "Preparation and Loss of Power Emergency Containment Equipment Hatch Closure." ________

  - The necessary equipment required to close the equipment hatch is available (verified once each 7 days by 28921-C, "Equipment Hatch Emergency Closure Surveillance"). ________

  In order to maintain a prompt closure time, obstructions interfering with the equipment hatch closure are permitted only when objects are being moved into or out of containment. Normally, cables or hoses are not routed through the equipment hatch. If an exception becomes necessary, quick disconnect capability will be employed in the vicinity of the hatch. ________

57. G2.1.15 001/3//CONDUCT OPS/MEM-2.7/3.4/NEW/HL-18 NRC/RO/SRO/AML

Which one of the following completes the following statement?

Temporary instructions issued to plant operating personnel addressing subjects not covered by existing plant operating procedures is the definition of a ___(1)___ Order,

and

___(2)___ required to be reviewed by BOTH the oncoming OATC and UO every time a shift turnover is performed.

	__(1)__	__(2)__
A.	Night	are
B.	Night	are NOT
C✓	Standing	are
D.	Standing	are NOT

G2.1.15 Conduct of Operations

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, operations memos, etc.
(CFR 41.10 / 45.12)

K/A MATCH ANALYSIS:

This question states the definition for a Standing Order and the candidate must determine whether it is the definition for Standing or Night Orders. The candidate also must determine if the OATC and UO are both required to review Standing Orders at every shift turnover.

DISTRACTOR ANALYSIS

A. Incorrect. This is the definition of a Standing Order and they are required to review these every shift turnover.

B. Incorrect. This is the definition of a Standing Order and they are required to review these every shift turnover.

C. Correct. This is the definition of a Standing Order and they are required to review these every shift turnover.

D. Incorrect. This is the definition of a Standing Order and they are required to review these every shift turnover.

REFERENCES

NMP-OS-007-003, "Plant Operating Orders"
10004-C, Shift Relief
11869-C, Unit Operator Relief Checklist
11872-C, Operator At The Controls Relief Checklist

VEGP learning objectives:

Not applicable.

1.0 Purpose

This instruction describes the process for handling Plant Operating Orders issued at the Southern Nuclear sites.

2.0 Applicability

This instruction applies to all Operations Department personnel at SNC sites.

3.0 References

3.1 NMP-OS-007, Conduct of Operations
3.2 NMP-OS-007-001, Conduct of Operations Standards and Expectations
3.3 NMP-AD-010, 10 CFR 50.59 Screenings and Evaluations

4.0 Definitions

4.1 **Night Orders** - Operations related information or orders issued by an Operations Superintendent/Shift Manager or designee to provide instructions to be carried out during backshifts, weekends, holidays, etc. Night Orders do not deviate from, or change the intent of existing plant operating procedures nor do they deviate from other plant design or regulatory documents, such as the FSAR, PTLR/COLR, TRM, or the Technical Specifications and its bases. Night Orders are entered in the Operations Electronic Admin Log, or Night Orders Record Book and on the Operations web page.

4.2 **Standing Orders** - Temporary instructions issued to plant operating personnel. They address subjects not covered by existing plant operating procedures. Standing Orders do not deviate from, or change the intent of existing plant operating procedures, nor do they deviate from plant design or regulatory documents, such as the FSAR, PTLR/COLR, TRM, or the Technical Specifications and its Bases. If the Standing Order includes directions to perform actions outside current approved operating procedures or at the direction of Operations' Management, a 10CFR 50.59 screening must be completed, per NMP-AD-010, 10 CFR 50.59 Screenings and Evaluations, and submitted with the Standing Order. Standing Orders are entered in the Document Management System (DMS) and the Standing Order Book. The Standing Order Book is a loose leaf type notebook for maintaining the Standing Order Log (Figure 3) and copies of effective Standing Orders.

5.0 Responsibilities

5.1 Fleet Operations Manager / Site Operations Managers - responsible for monitoring the implementation of the SNC Plant Operating Orders process.

5.2 Operations Superintendents / Shift Managers – responsible for ensuring that Night Orders and Standing Orders are written as needed.

5.3 Shift Managers / Shift Supervisors – responsible for implementing and ensuring compliance with Night Orders and Standing Orders.

5.4 All Operations personnel – responsible for viewing and complying with the instructions provided by Night Orders and Standing Orders.

1.0 PURPOSE

This procedure provides instructions for shift relief to ensure that a comprehensive exchange of information takes place between on coming and off-going shift operators.

2.0 RESPONSIBILITIES

2.1 Personnel shall conduct shift relief in a professional manner.

2.2 No person shall assume a position unless he is physically and mentally fit to competently discharge his responsibilities.

2.3 The off-going operator shall not leave his work area until he is satisfied that his relief is fully aware of existing conditions.

2.4 No person shall permit his relief to assume the shift if there is doubt that he is alert, coherent, and fully capable of performing his assigned duties.

2.5 No person shall assume a position unless he meets all licensing requirements for the position.

3.0 INSTRUCTIONS

3.1 Equipment should be in a stable condition before beginning shift relief.

3.2 Each off-going operator shall complete the narrative log, rounds sheets and off-going portion of turnover checklist for his station.

3.3 The on-coming shift should relieve the off-going shift at the normal workstations with the exception of System Operators who normally turn over in the Operator Ready Room.

3.4 Prior to assuming shift duties, each on-coming operator shall:

a. Review the narrative log, rounds sheets and checklists for his station. The review shall include narrative logs since the last shift worked or the preceding 3 days, whichever is less.

b. Discuss relevant items affecting plant operation with his off-going counterpart.

3.5 Each SM, SS, OATC, UO and SO shall complete his relief checklist prior to assuming his duties and obtain complete information on current plant status.

Approved By S. H. Chesnut	Vogtle Electric Generating Plant	Procedure Number 11869-C Rev 10.3
Date Approved 3/6/97	UNIT OPERATOR RELIEF CHECKLIST	Page Number 1 of 1

Date: ________ Unit: ____ Mode: ____ Power Rx ______ (%-CPS)

GMWe ________ MWe MVARS ______ (IN-OUT)

OFF-GOING UNIT OPERATOR ____________________ SHIFT ______

ON-COMING UNIT OPERATOR ____________________ SHIFT ______

[ ] Unit Control Log
[ ] Clearances In Progress
[ ] Walk Down Control Boards
[ ] Annunciator Status & Test
[ ] Temp Mods In Progress
[ ] LCO Status Log
[ ] Night Orders - Admin Log
[ ] Standing Orders
[ ] Special Conditions Surveillance Log
[ ] Control Room Rounds Sheet
[ ] Caution Tag Log
[ ] Components/Valves Out Of Normal Position

[ ] Procedures In Progress

[ ] Major Equipment Outages/Unusual Alignments

REMARKS

ON-COMING UNIT OPERATOR ______ / ______ / ______
INITIALS DATE TIME

OFF-GOING UNIT OPERATOR ______
INITIALS

SS REVIEW ______ / ______
INITIALS DATE

(EXAMPLE)

Approved By S. H. Chesnut	**Vogtle Electric Generating Plant**	Procedure Number Rev 11872-C 12.4
Date Approved 3/6/97	OPERATOR AT THE CONTROLS RELIEF CHECKLIST	Page Number 1 of 1

Date: ____________ Unit: ________ Mode: ______ Power Rx __________ (%-CPS)

Control Bank __________ at ________ steps Boron-RCS: __________ ppm

Tavg: ____________ °F RCS Press: ____________

OFF-GOING OPERATOR AT THE CONTROLS ________________________ SHIFT ________

ON-COMING OPERATOR AT THE CONTROLS ________________________ SHIFT ________

[ ] Unit Control Log
[ ] Clearances In Progress
[ ] Walk Down Control Boards
[ ] Annunciator Status & Test
[ ] Temp Mods In Progress
[ ] LCO Status Log
[ ] Night Orders - Admin Log
[ ] Special Conditions Surveillance Log
[ ] Operations Shift & Daily Surveillance Logs
[ ] Caution Tag Log
[ ] Components/Valves Out Of Normal Position
[ ] Standing Orders

[ ] Procedures In Progress

[ ] Major Equipment Outages/Unusual Alignments

REMARKS

ON-COMING OPERATOR AT THE CONTROLS ____________ / ____________ / ____________
INITIALS DATE TIME

OFF-GOING OPERATOR AT THE CONTROLS ____________
INITIALS

SS REVIEW ____________ / ____________
INITIALS DATE

(EXAMPLE)

Printed February 14, 2013 at 13:59

58. G2.1.29 001/3//CONDUCT OPS/MEM - 4.1/4.0/MOD - LOIT BANK/HL-18 NRC/RO/SRO/AML

You are performing a lineup verification for a valve in the shut position that will require an Independent Verification (IV).

Which ONE of the following is the correct performance of the lineup verification per NMP-OS-002, "Verification Policy"?

A. The positioner will turn the valve in the closed position with the independent verifier observing as it is completely closed.

B. The positioner will open the valve 1-2 turns, then shut it again with the independent verifier observing as it is completely closed.

C.✓ The positioner will turn the valve in the closed position. The independent verifier, being separated by time and distance, will perform the same action.

D. The positioner will open the valve 1-2 turns, then shut it again. The independent verifier, being separated by time and distance, will perform the same action.

G2.1.29 Conduct of Operations

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.
(CFR 41.10 / 45.1 / 45.12)

K/A MATCH ANALYSIS:

The question asks straight forward the policy for verifying a valve in the closed position.

DISTRACTOR ANALYSIS:

A. Incorrect. The independent verifier is not present when the manipulation is performed.

B. Incorrect. The independent verifier is not present when the manipulation is performed.

C. Correct.

D. Incorrect. The independent verifier will not open and shut the valve again, he will just turn the valve in the closed direction.

REFERENCES:

NMP-OS-002, Verification Policy, pages 7 and 8

VEGP learning objectives:

LO-LP-63308-01 Briefly describe the independent verification policy. Include a discussion of the types of verification that are available including concurrent verification.

6.1.5 Independent verification should be performed as soon as practical after the associated task is performed, but can generally wait until completion of the task unless an adverse consequence could result (plant transient, loss of safety function, etc).

NOTE: The following step applies ONLY to Operations personnel that are restricted to the main control room.

6.1.6 When independent verification is specified for activities in the main control room, independence will be maintained to the extent practical (i.e. verifier will not directly observe the performance of the step).

6.1.7 For restoration of systems which require IV, careful consideration must be given to the sequence of placing the affected components in service and restoration of the system to operable status. If desired to place a system in service prior to completion of the IV, a peer check should be used to verify critical components are properly aligned (this is to prevent damage to equipment, spilling of water, etc). The system should not be considered operable until completion of the IV.

6.1.8 Independent verification involves the following process:

6.1.8.1 The person performing the component manipulation enters the area, separated from the verifier by time and distance.

6.1.8.2 The positioner then references the lineup, procedure, tagout, or caution tag and verifies the proper component, using human performance tools such as STAR.

6.1.8.3 The positioner shall place (or check) the component in the required position per the lineup, procedure, tagout, or caution tag, as applicable.

6.1.8.4 The positioner signs or initials in the prescribed place.

6.1.8.5 The verifier enters the area, separated from the positioner performing the manipulation by time and distance.

6.1.8.6 The verifier references the lineup, procedure, tagout, or caution tag and verifies the correct component has been identified, using human performance tools such as STAR.

6.1.8.7 The verifier observes the position of the component and physically checks component position.

6.1.8.8 The verifier signs or initials in the prescribed place.

6.1.9 Independent Verification Methods

6.1.9.1 Direct Observation (preferred method)

6.1.9.1.1 Methods of performing direct observation for independent verification of valves or breakers include, but are not limited to, the following:

A. Visual observation of local breaker position indicating lights.

B. Visual observation of local breaker position indicating mechanical "flags."

C. Visual observation of breaker switch or handle position.

D. Manual valves to be independently verified open should be moved slightly in the closed direction and then moved in the open direction until the valve is considered in the fully open position, and, visual observation of the stem, i.e., grease markings indicating normal valve travel, valve stems extended on rising stem valves and mechanical position indication should also be included.

E. Valves required to be positioned slightly off "backseat" to prevent binding should be fully opened and returned to the procedurally established position during independent verification.

F. Manual valves to be independently verified closed should be moved, or attempted to be moved, only in the closed direction using normal closing torque and visually observing the stem. i.e., Grease markings indicating normal valve travel, valve stems inserted on rising stem valves, and mechanical position indication.

G. Visual observation and comparison with required stem position, local indicators, or other suitable valve component should be used to independently verify the position of throttled valves. Throttled valves shall not be moved to verify position unless specifically permitted to do so by the Shift Supervisor.

H. Control valve positions should be independently verified by ensuring that power or air, as appropriate, is available to the valve operators and that no physical obstructions which could prevent proper operation are apparent.

59. G2.1.43 001/3//CONDUCT OPS/C/A - 4.1/4.0/NEW/HL-18 NRC/RO/SRO/AML

Initial conditions:

- Unit 1 is at 100% power.
- 75 gpm letdown has been placed in service.
- RCS Boron Concentration is 920 ppm.
- The OATC inadvertently sets Letdown Temperature Controller, TIC-0130, potentiometer for 120 gpm letdown flow rate.

Current conditions:

- The UO performing his rounds finds the wrong potentiometer setting and corrects the setting per 13006-1, "Chemical Volume Control System."

Which ONE of the following completes the statement below?

In response to the initial potentiometer setting, an inadvertent RCS ___(1)___ will occur,

and

when the UO corrects the potentiometer setting, TV-0130 will ___(2)___ to control letdown temperature at the new setting.

	__(1)__	__(2)__
A.	boration	open
B.	boration	close
C.	dilution	open
D✓	dilution	close

G2.1.43 Conduct of Operations

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant temperature, secondary plant, fuel depletion, etc.:
(CFR 41.10 / 43.6 / 45.6)

K/A MATCH ANALYSIS:

The question presents a plausible scenario where TIC-0130 potentiometer has been set inadvertently too high (set for 120 gpm letdown flow). Letdown temperature will be too cool causing an RCS dilution. When the error is discovered, and the pot setting

corrected, the candidate must determine which direction TV-0130 will move to control the Letdown temperature.

DISTRACTOR ANALYSIS:

A. Incorrect. With potentiometer setting too high, TV-0130 will open more than necessary to control Letdown temperature. Letdown temperature will be too cold causing an RCS dilution. When corrected, TV-0130 will throttle closed to raise letdown temperature.

B. Incorrect. With potentiometer setting too high, TV-0130 will open more than necessary to control Letdown temperature. Letdown temperature will be too cold causing an RCS dilution. When corrected, TV-0130 will throttle closed to raise letdown temperature.

C. Incorrect. With potentiometer setting too high, TV-0130 will open more than necessary to control Letdown temperature. Letdown temperature will be too cold causing an RCS dilution. When corrected, TV-0130 will throttle closed to raise letdown temperature.

D. Correct. With potentiometer setting too high, TV-0130 will open more than necessary to control Letdown temperature. Letdown temperature will be too cold causing an RCS dilution. When corrected, TV-0130 will throttle closed to raise letdown temperature.

REFERENCES:

V-LO-LP-53401, PWR Generic Fundamentals for Demineralizers and Ion Exchangers.
13006-1, Chemical and Volume Control System, section 4.2.4 and 4.4.2

VEGP learning objectives:

V-LO-PP-09100-05, State how letdown temperature is controlled, relative to the following:

b. Demineralizer performance.

V-LO-LP-53401-15 Describe the effect of temperature on saturated ion exchangers.

INITIALS

4.2.4 Raising Letdown Flow (45 gpm to 120 gpm) or (75 gpm to 120 gpm)

NOTE

Table 1 may be reviewed for specific charging flow and letdown orifice combinations and conditions. ☐

CAUTIONS

- CCP-1B or the NCP must be used for 120 gpm letdown. CCP-1A is only capable of flows up to 102 gpm at NOPT. It may not be used with 120 gpm letdown. At higher than normal RCS pressure, the flow will be reduced even more. ☐
- Letdown temperature and pressure should be continuously monitored during the transfer. The Operator should manually intervene to control these limits during the transfer. ☐

4.2.4.1 IF CCP-A is in service, **swap** to the NCP or CCP-B per Section 4.2.1 or 4.2.3 as applicable. ________

4.2.4.2 **Maintain** Seal Injection flows between 8 and 13 gpm per RCP. ________

4.2.4.3 IF the NCP is in service, perform the following:

a. Dispatch an operator to **obtain** NCP miniflow value as indicated on 1FI-10132 NCP MINIFLOW FLOW INDICATOR ________

b. **Place** Charging Flow Control 1FIC-121 in MAN AND **adjust** charging until the sum of NCP miniflow AND normal charging flow is greater than 120 gpm. ________

NOTE

Reducing seal injection flow before closing miniflow will help keep seal flow from exceeding maximum limit when 1-HV-8109 goes closed. ☐

c. **Adjust** SEAL FLOW CONTROL 1HC-182 as necessary to maintain Seal Injection flows approximately 8 gpm. ________

d. **Close** NCP MINIFLOW 1-HV-8109. ________

INITIALS

4.2.4.4 WITH Charging Flow Control 1FIC-121 in MAN, simultaneously perform the following:

- **Adjust** 1FIC-121 until charging flow is approximately 120 to 130 gpm. ________
- **Adjust** SEAL FLOW CONTROL 1HC-182 as necessary to maintain Seal Injection flows between 8 and 13 gpm. ________

4.2.4.5 **Verify** Regenerative Heat Exchanger Outlet 1TI-127 temperature remains less than 290°F. ________

CAUTION

The time at reduced letdown pressure should be as short as possible to minimize orifice erosion. ☐

4.2.4.6 **Place** Letdown Pressure Controller 1PIC-131 pressure controller in MAN AND **adjust** LETDOWN PRESS 1PI-131A pressure to between 190 and 200 psig. ________

INITIALS

NOTE

Steps 4.2.4.7 and 4.2.4.8 should be performed concurrently to minimize the potential for orifice erosion. □

ALB07-F03 LTDN HX OUT HI FLOW

4.2.4.7 **Open** the selected LETDOWN ORIFICE isolation valve:

1HS-8149A for 45 gpm orifice isolation 1-HV-8149A. ______

OR

1HS-8149B for 75 gpm orifice isolation 1-HV-8149B (odd fuel cycles). ______

OR

1HS-8149C for 75 gpm orifice isolation 1-HV-8149C (even fuel cycles). ______

4.2.4.8 **Adjust** 1PIC-131 to maintain letdown pressure between 360 and 380 psig. ______

4.2.4.9 WHEN LETDOWN PRESS 1PI-131A stabilizes between 360 and 380 psig, **place** 1PIC-131 in AUTO. ______

4.2.4.10 **Monitor** 1LR-459 Pressurizer Actual Level and Level Setpoint. ______

4.2.4.11 **Adjust** 1FIC-121 to maintain Pressurizer Level within 1% of setpoint. ______

INITIALS

4.2.4.12 **Place** Pressurizer Level Control in automatic UNLESS it is to remain in Manual under Tagout or Caution tag:

a. **Verify** PRZR Level Controller 1LIC-459 in AUTO. ________

b. AFTER level has been stable within 1% of setpoint for approximately 3 minutes, **place** 1FIC-121 in AUTO. ________

4.2.4.13 **Adjust** LETDOWN HEAT EXCH OUTLET 1TIC-130 to maintain LETDOWN HEAT EXCH OUTLET 1TI-130 below 115°F, and **place** in AUTO. ________

4.2.4.14 **Record** the letdown orifice that was placed-in service or removed-from-service in the Unit Control Log. ________

4.2.4.15 **Notify** Chemistry that Letdown Flow has been raised. ________

4.2.4.16 **Notify** Reactor Engineering that Letdown Flow has been changed and to monitor for impacts. ________

INITIALS

4.4.2 Returning Normal Charging and Letdown to Service

4.4.2.1 IF a Charging Pump is NOT in service, **Go To** Section 4.4.13 to start the NCP OR an available Centrifugal Charging Pump, THEN **Return To** this section. ________

NOTES

- This section also applies to returning normal charging and letdown to service following termination of safety injection. ☐
- In the event letdown was isolated from the shutdown panel, a walkdown of letdown components and piping shall be performed prior to restoring normal letdown to service. AI2011202626 ☐
- Letdown is to be established as soon as possible after initiating flow through a Charging Nozzle. ☐

4.4.2.2 IF NCP is in service, **verify** NCP MINIFLOW 1-HV-8109 is open. ________

4.4.2.3 Perform the following:

a. **Close** LETDOWN ORIFICE Isolation Valves:

- 1-HV-8149A ________
- 1-HV-8149B ________
- 1-HV-8149C ________

b. **Close** LETDOWN ISOLATION VLV UPSTREAM AND DOWNSTREAM Valves:

- 1-LV-460 ________
- 1-LV-459 ________

c. **Close** PZR AUX SPRAY VALVE 1-HV-8145. ________

d. **Open** CVCS LETDOWN PIPE BREAK PROT ISOLATION 1-HV-15214. ________

e. **Open** RCS LETDOWN LINE ISO VLV IRC 1-HV-8160. ________

f. **Open** RCS LETDOWN LINE ISO VLV ORC 1-HV-8152. ________

INITIALS

g. **Place** Letdown Pressure Controller 1PIC-131 in MAN and **adjust** output to between 50% and 75%. ________

h. **Place** LETDOWN HX OUTLET TEMP 1TIC-130 in MAN and **adjust** output to the most current position as recorded on the Control Room Rounds Sheets. ________

i. **Verify** PRESSURIZER LEVEL 1LR-459 greater than 17%. ________

> NOTE
>
> If Normal Charging and Letdown are being returned to service as directed from Section 4.4.15, one of the valves in Substep j. and both valves in Step 4.4.2.4 will already be open. ☐

j. **Verify** one of the following are OPEN:

NORMAL CHARGING TO LOOP 1 1-HV-8146 (even-numbered fuel cycle) ________

<u>OR</u>

ALTERNATE CHARGING TO LOOP 4 1-HV-8147 (odd-numbered fuel cycle) ________

4.4.2.4 **Verify** CHARGING TO RCS ISOLATION Valves are OPEN:

- 1-HV-8106 ________
- 1-HV-8105 ________

4.4.2.5 Simultaneously perform the following:

- **Adjust** 1HC-182 output to maintain between 8 and 13 gpm to each RCP. ________
- **Adjust** 1FIC-121 to raise CHG FLOW 1FI-121A to between 80 and 90 gpm. ________

INSTRUCTOR GUIDE	KEY POINTS, AIDS, QUESTIONS/ANSWERS
c. Result of this characteristic is that at lower temperatures resins are more efficient at removing boron from coolant than at higher temperatures	**Objective 15**
d. A saturated resin bed will actually release boron as temperature is increased	**Objective 13 and 14**
The temperature of the coolant passed through the bed affects the boron affinity of a resin bed. At lower temperatures, the borate ion bonding to the exchange site contains three boron atoms. At higher temperatures, the borate ion contains only one boron atom. The result of this characteristic is that at higher temperatures the resins are less efficient at removing boron from the coolant than at lower temperatures. A saturated resin bed will actually release boron as temperature is increased.	**Example 4-5 / TP 4-24** The temperature of a deborating demineralizer increases 75°F. What effect will this have on demineralizer operation? Answer: Boron removal is more efficient at lower temperatures, therefore the demineralizer will not lower the boron concentration at the same rate as it did prior to the temperature increase.
6. In systems where it is possible to subject demineralizer resin to high temperatures, demineralizers have automatic features to protect against temperature damage	
a. This is usually accomplished by automatic closure of demineralizer inlet valves to isolate demineralizer from high temperature liquid, when high temperature at inlet to demineralizer is sensed	
b. These systems are typically equipped with bypass valve that can divert flow around demineralizer until normal system temperature is restored	**Objective 10**

GF@gpworldwide.com

60. G2.2.42 001/3//EQUIP CTRL/MEM - 3.9/4.6/MOD-LOIT/HL-18 NRC/RO/SRO/KAJ

Unit 1 is at 100% power.

The following is the status of ECCS accumulator # 1:

- N_2 pressure - 631 psig   Cb - 1894 ppm   Level - 30%

Which one of the following completes the following statement?

Per Tech Spec 3.5.1, "Accumulators," the ECCS accumulator parameters above are __________.

A. all within Tech Spec limits.

B. not within Tech Spec limits due to level.

C. not within Tech Spec limits due to N_2 pressure.

D✓ not within Tech Spec limits due to boron concentration.

G2.2.42 Equipment Control

Ability to recognize system parameters that are entry-level conditions for Tech Specs.
(CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

K/A MATCH ANALYSIS:

The candidate is given ECCS accumulator parameters to monitor and has to determine if the given parameters are within limits per 14000-1, Tech Spec Rounds.

DISTRACTOR ANALYSIS:

A. Incorrect-Accumulator # 1 boron is too low.

B. Incorrect-Accumulator # 1 boron is too low.

C. Incorrect-Accumulator # 1 boron is too low.

D. Correct-Accumulator # 1 boron is too low.

REFERENCES:

Tech Spec 3.5.1, Accumulators
OSP-14000-1, Operations Shift and Daily Surveillance Logs.

VEGP learning objectives:

LO-LP-39209-01 For any given item in section 3.5 of Tech Specs, be able to:

a. State the LCO.

b. State any one hour or less required actions.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1000 psig.

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. AND C.2 Reduce pressurizer 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 ≥ 6555 gallons and ≤ 6909 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 617 psig and ≤ 678 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 1900 ppm and ≤ 2600 ppm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> For each affected accumulator, once within 6 hours after each solution volume increase of ≥ 67 gallons, 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 pressurizer pressure is > 1000 psig.	In accordance with the Surveillance Frequency Control Program

DATA SHEET 1 - MODE 1 & 2 **MODE ____________** **Sheet 1 of 10**

DATE ____________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER		INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
CONTAINMENT PRESSURE SHALL BE MAINTAINED WITHIN LIMITS VERIFY PRESSURE	SR 3.6.4.1	CONTAINMENT PRESSURE (PSIG)		1PI-0935			>-0.3 PSIG AND < 1.8 PSIG AND	3.6.4
				1PI-0937				
				1PI-0934			CHANNEL CHECK REQUIRED 3* REQUIRED 4	3.3.2(D) 3.3.2(E)
ESFAS INSTRUMENTATION SHALL BE OPERABLE CHANNEL CHECK	SR 3.3.2.1 FCN 1C,2C,4C			1PI-0936				
				1PI-10945				
ACCIDENT MONITORING INSTRUMENT SHALL BE OPERABLE CHANNEL CHECK	SR 3.3.3.1 FCN 7	*1PI-0937 OR 1PI-10945 CANNOT BE USED TO SATISFY REQUIRED CHANNELS. NOTE: PI'S ON QMCB HAVE POSITIVE RANGE ONLY.					REQUIRED 2	3.3.3 (B,G,H,J)
		COMPUTER POINT (PSIG)		P9871 (1)				
EACH ACCUMULATOR SHALL BE OPERABLE VERIFY PRESSURE AND WATER LEVEL	SR 3.5.1.3	ACCUMULATOR NITROGEN PRESSURE (PSIG)	1	1PI-0960A			≥626 PSIG AND ≤678 PSIG	3.5.1
			1	1PI-0961A				
			2	1PI-0962A				
			2	1PI-0963A				
			3	1PI-0964A				
			3	1PI-0965A				
			4	1PI-0966A				
			4	1PI-0967A				
	SR 3.5.1.2	ACCUMULATOR WATER LEVEL (%)	1	1LI-0950			≥29.2% AND ≤70.7%	
			1	1LI-0951				
			2	1LI-0952				
			2	1LI-0953				
			3	1LI-0954				
			3	1LI-0955				
			4	1LI-0956				
			4	1LI-0957				
EACH ACCUMULATOR SHALL BE OPERABLE VERIFY BORON CONCENTRATION AFTER VOLUME INCREASE (ADDITION NOT FROM THE RWST)	SR 3.5.1.4	ACCUMULATOR WATER LEVEL INCREASE (INIT)	1	1LI-0950*			<7%	3.5.1 REQUEST CHEMISTRY SAMPLE
			1	1LI-0951*				
			2	1LI-0952*				
			2	1LI-0953*				
			3	1LI-0954*				
			3	1LI-0955*				
			4	1LI-0956*				
			4	1LI-0957*				
		*OBTAIN ACCUMULATOR WATER LEVELS AT LAST SAMPLE FOR EACH INDICATOR FROM CURRENT PERFORMANCE OF 14228-1, "OPERATIONS MONTHLY SURVEILLANCE LOGS", OR AT POINT IN TIME OF MOST RECENT CHEMISTRY SAMPLE, IF LATER. COMPARE PRESENT LEVEL FOR THE ACCUMULATOR LEVEL INDICATOR TO PREVIOUSLY RECORDED LEVEL FOR THE INDICATOR.						

(1) IF CONTAINMENT PRESSURE IS GREATER THAN 0.60 PSIG, INITIATE ACTION TO PERFORM PRESSURE RELIEF PER 13125-1

COMPLETED BY: DAY: __________ TIME: _____ NIGHT: __________ TIME: _____

SS REVIEW: DAY: __________ TIME: _____ NIGHT: __________ TIME: _____

DATA SHEET 1 - MODE 1 & 2 MODE 1 **Sheet 1 of 10**

DATE JAN 7 2013

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER		INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
CONTAINMENT PRESSURE SHALL BE MAINTAINED WITHIN LIMITS VERIFY PRESSURE	SR 3.6.4.1	CONTAINMENT PRESSURE (PSIG)		1PI-0935	Ø	0	>-0.3 PSIG AND < 1.8 PSIG AND	3.6.4
				1PI-0937	Ø	0	CHANNEL CHECK	
				1PI-0934	Ø	0	REQUIRED 3* REQUIRED 4	3.3.2(D) 3.3.2(E)
ESFAS INSTRUMENTATION SHALL BE OPERABLE CHANNEL CHECK	SR 3.3.2.1 FCN 1C,2C,4C			1PI-0936	Ø	0		
				1PI-10945	0.464	0.455		
ACCIDENT MONITORING INSTRUMENT SHALL BE OPERABLE CHANNEL CHECK	SR 3.3.3.1 FCN 7	*1PI-0937 OR 1PI-10945 CANNOT BE USED TO SATISFY REQUIRED CHANNELS. NOTE: PI'S ON QMCB HAVE POSITIVE RANGE ONLY.					REQUIRED 2	3.3.3 (B,G,H,J)
		COMPUTER POINT (PSIG)		P9871 (1)	0.45	0.45		
EACH ACCUMULATOR SHALL BE OPERABLE VERIFY PRESSURE AND WATER LEVEL	SR 3.5.1.3	ACCUMULATOR NITROGEN PRESSURE (PSIG)	1	1PI-0960A	640	640	≥626 PSIG AND ≤678 PSIG	
				1PI-0961A	635	635		
			2	1PI-0962A	640	640		
				1PI-0963A	635	635		
			3	1PI-0964A	645	645		
				1PI-0965A	640	640		
			4	1PI-0966A	640	640		
				1PI-0967A	640	640		3.5.1
	SR 3.5.1.2	ACCUMULATOR WATER LEVEL (%)	1	1LI-0950	52	52	>29.2% AND <70.7%	
				1LI-0951	51	51		
			2	1LI-0952	50	50		
				1LI-0953	48	48		
			3	1LI-0954	49	48		
				1LI-0955	49	48		
			4	1LI-0956	55	55		
				1LI-0957	54	54		
EACH ACCUMULATOR SHALL BE OPERABLE VERIFY BORON CONCENTRATION AFTER VOLUME INCREASE (ADDITION NOT FROM THE RWST)	SR 3.5.1.4	ACCUMULATOR WATER LEVEL INCREASE (INIT)	1	1LI-0950*	[illegible]	[illegible]	<7%	3.5.1 REQUEST CHEMISTRY SAMPLE
				1LI-0951*	[illegible]	[illegible]		
			2	1LI-0952*	[illegible]	[illegible]		
				1LI-0953*	[illegible]	[illegible]		
			3	1LI-0954*	[illegible]	[illegible]		
				1LI-0955*	[illegible]	[illegible]		
			4	1LI-0956*	[illegible]	[illegible]		
				1LI-0957*	[illegible]	[illegible]		

*OBTAIN ACCUMULATOR WATER LEVELS AT LAST SAMPLE FOR EACH INDICATOR FROM CURRENT PERFORMANCE OF 14228-1, "OPERATIONS MONTHLY SURVEILLANCE LOGS", OR AT POINT IN TIME OF MOST RECENT CHEMISTRY SAMPLE, IF LATER. COMPARE PRESENT LEVEL FOR THE ACCUMULATOR LEVEL INDICATOR TO PREVIOUSLY RECORDED LEVEL FOR THE INDICATOR.

(1) IF CONTAINMENT PRESSURE IS GREATER THAN 0.3 PSIG, INITIATE ACTION TO PERFORM PRESSURE RELIEF PER 13125-1

COMPLETED BY: DAY: [signature] TIME: 0732 NIGHT: [signature] TIME: 1931

SS REVIEW: DAY: [signature] TIME: 0843 NIGHT: [signature] TIME: 2110

61. G2.2.44 001/3//EQUIP CTRL/C/A - 4.2/4.4/BANK-HL15/HL-18 NRC/RO/SRO/AML

Given the following:

- Reactor power is 30%.
- A reactor shutdown is in progress.
- A HIGH failure of IR NIS channel N-36 occurs.

Which ONE of the following is correct regarding the effects of this failure as reactor shutdown progresses with no operator action?

	Power < P-10	Power < P-6
A.	Reactor trip occurs	SR High flux trip automatically resets
B.✓	Reactor trip occurs	SR High flux trip must be manually reset
C.	Reactor does NOT trip	SR High flux trip automatically resets
D.	Reactor does NOT trip	SR High flux trip must be manually reset

G2.2.44 Equipment Control

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:
(CFR: 41.5 / 43.5 / 45.12)

K/A MATCH ANALYSIS:

The question gives a plausible scenario where an IR NIS fails high during a plant shutdown. The candidate has to determine the effects on the plant when P-10 resets and how SR hi flux reset is accomplished.

DISTRACTOR ANALYSIS

A. Incorrect. Reactor trips on P-10 reset, SR hi flux trip requires manual reset.

B. Correct.

C. Incorrect. Reactor trips on P-10 reset, SR hi flux trip requires manual reset.

D. Incorrect. Reactor trips on P-10 reset. OATC reset of SR hi flux trip is correct.

REFERENCES

HL-15 NRC question 64
12005-C, Reactor Shutdown to Hot Standby (Mode 2 to Mode 3), step 4.4.2.b
1X6AA02-00227
1X6AA02-00228

VEGP learning objectives:

LO-PP-17201-01 Discuss the operation of the Source & Intermediate Range Detectors to include:

d. All Reactor Trip signals
e. All Permissives & Interlocks

UNIT NO. _____ Date ____/____/____

INITIALS

4.2.4 BELOW approximately 7E-6% IR:

a. **Observe** the following status lights are extinguished:

(1) IR P6 NC35D (TSLB-4, 3.1) ________

(2) IR P6 NC36D (TSLB-4, 3.2) ________

(3) SOURCE RANGE BLOCK PERMISSIVE P6 (BPLP) ________

(4) SR TRAIN A TRIP BLK'D (BPLP) ________

(5) SR TRAIN B TRIP BLK'D (BPLP) ________

b. IF NEITHER Source Range instrument tripped Blocked status lights clear BELOW approximately 7E-6% IR: (1985305687, 1986209841, 1986209843, 1986210224)

(1) **Place** both SR BLOCK RESET switches, HS-40030 and HS-40031, to RESET. ________

(2) **Verify** at least one Source Range NI channel trip is active, (BPLP trip blocked light reset). ________

OR

Immediately **open** both reactor trip breakers. (TS LCO 3.3.1) ________

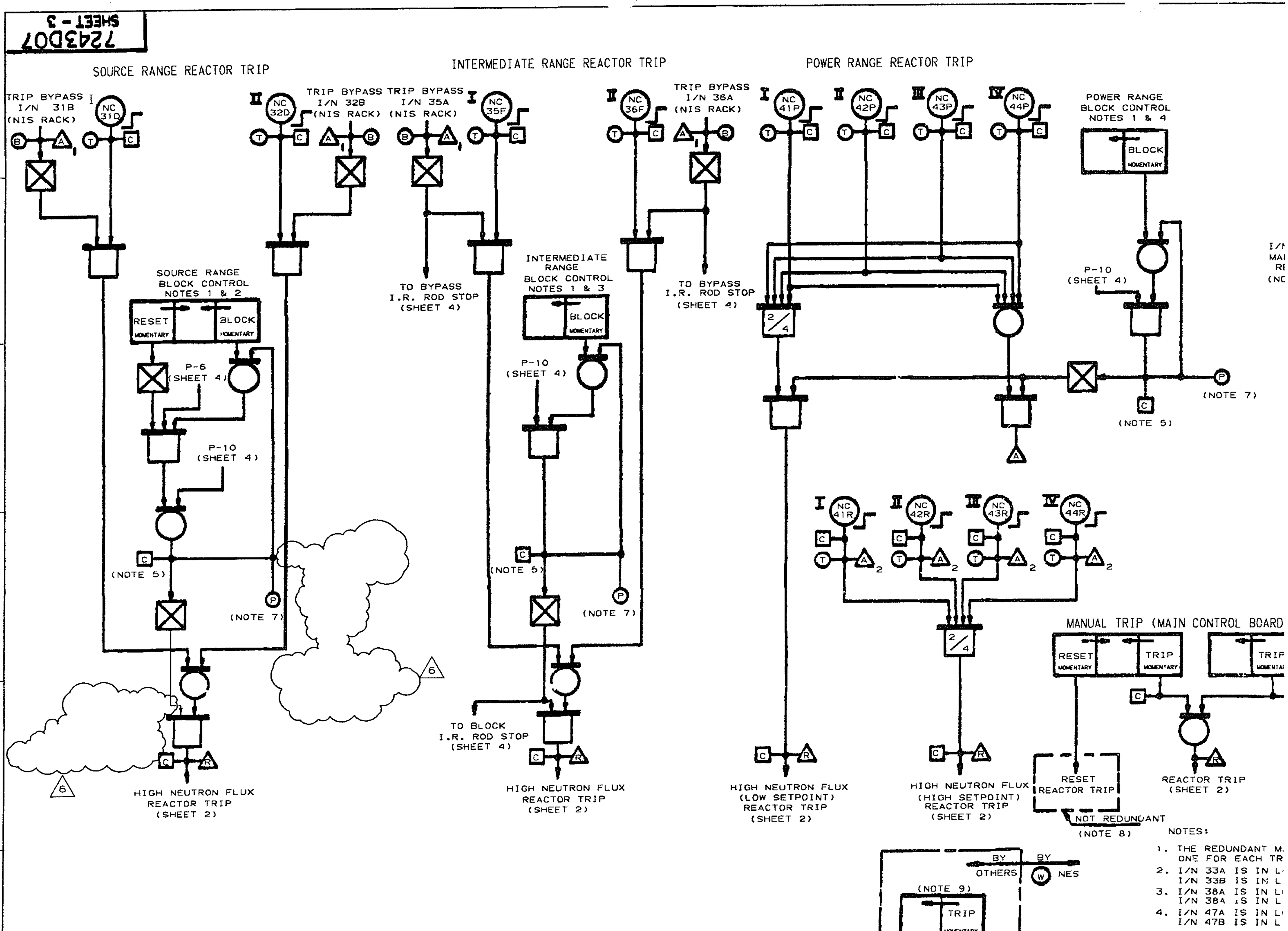
(3) IF EITHER Source Range NI channel trip block light fails to reset:

(a) Immediately **suspend** all actions that may add positive reactivity to the core. (TS LCO 3.3.1) ________

AND

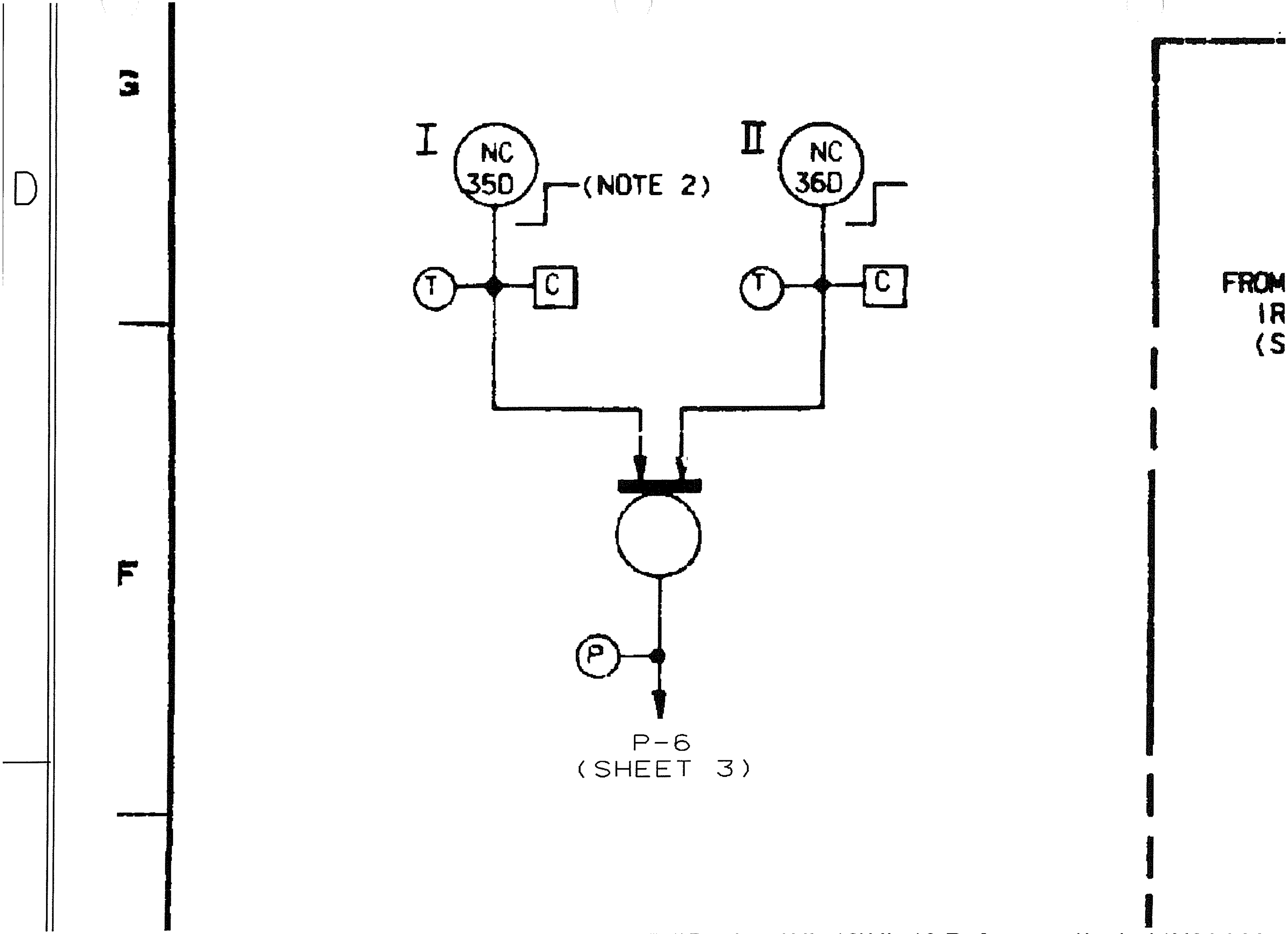
(b) WITHIN 1 hour, **verify** SDM per OSP 14005, "Shutdown Margin Calculation." (TS LCO 3.3.8) ________*

(4) **Notify** I&C to immediately **restore** the affected channels to service. (1986209841, 1986210224) ________



NOTES:

1. THE REDUNDANT M... ONE FOR EACH TR...
2. I/N 33A IS IN L... I/N 33B IS IN L...
3. I/N 38A IS IN L... I/N 38A IS IN L...
4. I/N 47A IS IN L... I/N 47B IS IN L...
5. TWO COMPUTER IN...


D
G
F
I
NC 35D
(NOTE 2)
II
NC 36D
T
C
T
C
P
P-6
(SHEET 3)
FROM
IR
(S

QUESTIONS REPORT

for Vogtle 2009 (HL15) June NRC RO Questions

1. G2.2.44 001/3/N/A/EQUIPMENT CONTROL/C/A - 4.2 / 4.4/MOD - LOIT BANK/HL-15/RO/TNT/DS

Given the following:

- Reactor power is 30%
- A reactor shutdown is in progress
- A high failure of IR NIS channel N-36 occurs

Which **ONE** of the following is **CORRECT** regarding the effects of this failure as reactor shutdown progresses?

	Power < P-10	Power < P- 6
A.	Reactor trip occurs	SR High flux trip automatically unblocks
B.✓	Reactor trip occurs	SR High flux trip must be manually unblocked
C.	Reactor does **NOT** trip	SR High flux trip automatically unblocks
D.	Reactor does **NOT** trip	SR High flux trip must be manually unblocked

62. G2.3.11 001/3//RAD CTRL/C/A - 3.8/4.3/LOIT BANK/HL-18 NRC/RO/SRO/AML

During a release of Waste Monitor Tank (WMT) #9:

- 1-RX-0018 DPM TROUBLE light illuminates.
- 1-RE-0018 is reading downscale LOW.

Which ONE of the following would be correct regarding the release of the WMT?

A. The release will automatically isolate.
The WMT may still be released as long as ODCM requirements are met.

B. The release will automatically isolate.
The WMT may NOT be released until the radiation monitor is repaired.

C.✓ The release will have to be manually isolated.
The WMT may still be released as long as ODCM requirements are met.

D. The release will have to be manually isolated.
The WMT may NOT be released until the radiation monitor is repaired.

G2.3.11 Radiation Control

Ability to control radiation releases:
(CFR: 41.11 / 43.4 / 45.10)

K/A MATCH ANALYSIS:

Question gives a plausible scenario where a Liquid Radwaste radiation detector fails LOW. The candidate must know whether to allow the release to continue or halt the release and what would be required to release the WMT.

DISTRACTOR ANALYSIS

A. Incorrect. No auto isolation on LOW radiation or trouble.

B. Incorrect. No auto isolation on LOW radiation or trouble, meeting ODCM would requirements would allow release with the radiation monitor inoperable.

C. Correct. Manual isolation is required, the ODCM does allow for release with a radiation monitor inoperable as long as the LCO actions are taken.

D. Incorrect. Meeting ODCM would requirements would allow release with the radiation monitor inoperable.

REFERENCES

13216-1, "Liquid Waste Release",
Offsite Dose Calculation Manual, Table 2-1 Radioactive Liquid Effluent Monitoring Instrumentation

VEGP learning objectives:

LO-PP-47101-08 Describe the major steps required for Operations to release a WMT.

LO-PP-47101-09 State the conditions that require immediate termination of a Liquid waste release.

LO-PP-47101-10 State the ODCM, TR, applicabilities, and any one hour or less actions required for the Liquid Waste Processing System.

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 The Liquid Waste Processing System is potentially radioactive. Caution should be exercised to avoid spillage and to minimize exposure. ________

2.1.2 Once a Waste Monitor Tank (WMT) has been placed on recirculation for sampling, the tank shall remain isolated to prevent introduction of liquids that could alter the concentration of the contained volume. ________

2.1.3 Radiation Monitor 1-RE-0018 reading should be observed at least once every 2 hours during the release to assure that the activity does not exceed the setpoint on the "Batch Liquid Release Permit". ________

2.1.4 If a high alarm is received from 1-RE-0018 while releasing a tank, the release shall be stopped immediately and the Shift Supervisor and Chemistry notified. ________

2.1.5 If 1-RE-0018 reads less than expected, release can continue provided Chemistry is notified and 1-RX-0018 does not show a trouble condition. ________

2.1.6 DO NOT release more than one Waste Monitor Tank per plant site at the same time, unless authorized by the Chemistry Manager. ________

2.1.7 If a high alarm is received from 1-RE-0018 while flushing with tank water, flush with demin water per Section 4.8. ________

2.1.8 If required to reset Dilution Flow Totalizer A-FQI-7620 prior to starting a release, Chemistry should be notified and Dilution Flow Totalizer A-FQI-7620 value recorded in the Electronic Log for the purpose of tracking tritium. ________

INITIALS

4.1.11 IF 1-RE-0018 is inoperable, **perform** Step 4.1.13 and **mark** Step 4.1.12 n/a. ________

4.1.12 IF 1-RE-0018 is operable:

a. **Pulse check** 1-RE-0018 as follows:

(1) **Verify** Blowdown Sump dilution flow is at least 12,000 gpm and greater than flow required by the "Batch Liquid Release Permit". ________

ALB05-B3 INTMD RADIATION ALARM ☐
ALB05-C3 HIGH RADIATION ALARM ☐

(2) **Notify** the Control Room to expect an alarm on 1-RE-0018 on the Digital Radiation Monitor System. ________

NOTE
1-HS-0018 must be held in the open position until 1-RV-0018 is fully open. ☐

(3) **Open** LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 using 1-HS-0018. ________

(4) **Request** Chemistry to activate the pulse test on channel 1-RE-0018. ________

(5) **Verify** LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 closes and Hi Radiation alarm annunciates. ________

(6) **Position** handswitch 1-HS-0018 to OPEN and **verify** LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 remains closed. ________

(7) **Request** Chemistry to restore channel 1-RE-0018 to normal. ________

b. IF Chemistry request to lower background, **flush** 1-RE-0018 with Demin water per Section 4.8. ________

INITIALS

> NOTE
>
> The following steps will verify the setpoints on the release permit agree with the 1-RI-0018 readings. ☐

c. **Flush** Tank water thru 1-RE-0018 as follows:

(1) **Throttle** open LWPS LIQUID WASTE DISCH RAD MON DRAIN, 1-1901-X4-145 approximately one turn. (RD60) ______

(2) **Close** LWPS CLEAN WASTE DISCH 1-RE-0018 INBOARD RT, 1-1901-X4-144. (RD59) ______

(3) **Unlock** and **open** LWPS WST MON TANK PUMP 09 TO CLN WST DISCH, 1-1901-U4-238. (RD59) ______

(4) **Open** LWPS WST MON TANK PUMPS TO CLN WASTE DISCH, 1-1901-U4-175. (RD59) ______

> NOTE
>
> To prevent over running floor drain 1-1901-X4-145 or 1-1901-U4-175 may be used to control flow. ☐

(5) **Throttle** open LWPS WASTE MONITOR TANK PUMP 09 DISCHARGE, 1-1901-U4-229, approximately 1/2 turn. (RD58) ______

> NOTE
>
> 1-HS-0018 must be held in the open position until 1-RV-0018 is fully open. ☐

(6) <u>AFTER</u> 5 minutes <u>OR</u> <u>WHEN</u> 1-RE-0018 is reading below the trip setpoint set by the "Batch Liquid Release Permit", **Open** LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 using 1-HS-0018. ______

(7) **Open** LWPS CLEAN WASTE DISCH RE-0018 INBOARD ROOT, 1-1901-X4-144. (RD59) ______

INITIALS

(8) **Slowly close** LWPS LIQUID WASTE DISCH RAD MON DRAIN, 1-1901-X4-145. (RD60) (IV REQUIRED) ________

4.1.13 IF 1-RE-0018 is inoperable:

a. **Verify** that two independent samples have been taken and analyzed by Chemistry "Action Statement 37 Sheet" attached to the release permit or by contacting the Shift Supervisor (SS). **Document** verification and method in the SO Logbook. ________

b. **Open** LWPS WST MON TANK PUMPS TO CLN WASTE DISCH, 1-1901-U4-175, (RD59) (IV REQUIRED) ________

Critical

c. **Unlock** and **open** LWPS WST MON TANK PUMP 09 TO CLN WST DISCH, 1-1901-U4-238. (RD59) ________

____
CV

d. **Request** the Control Room adjust Blowdown Sump dilution flow to at least 12,000 gpm and/or greater than flow required by the "Batch Liquid Release Permit" per procedure 13727-C. ________

NOTE

1-HS-0018 must be held in the open position until 1-RV-0018 is fully open. ☐

e. **Open** LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 using 1-HS-0018. (IV REQUIRED) ________

Table 2-1. Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	OPERABILITY Requirements[a]	
	Minimum Channels Operable	ACTION
1. Radwaste Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE–0018)	1	37
b. Steam Generator Blowdown Effluent Line (RE-0021)	1	38
c. Turbine Building Effluent Line (RE–0848)	1	38
2. Radwaste Monitors Providing Alarm, but Not Automatic Termination of Release		
NSCW Effluent Line (RE-0020 A)	1	39
NSCW Effluent Line (RE-0020 B)	1	39
3. Flowrate Measurement Devices		
a. Liquid Radwaste Effluent Line (FT–0018), (FT-1084A/B), or (FT-1085A/B)	1	40
b. Steam Generator Blowdown Effluent Line (FT-0021)	1	40
c. Flow to Blowdown Sump (AFQI-7620, FI7620A)	1	40

a. All requirements in this table apply to each unit.

Table 2-1 (contd). Notation for Table 2-1 — ACTION Statements

ACTION 37 — With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. The local radiation monitor reading (if functional) is recorded at least once per 12 hours during the release or at least two independent samples are analyzed in accordance with Section 2.1.2.3, and

b. At least two technically qualified members of the Facility Staff independently verify the discharge line valving and the release rate calculations.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 38 — With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the local radiation monitor reading (if functional) is recorded at least once per 12 hours or grab samples are analyzed for gross radioactivity at a MINIMUM DETECTABLE CONCENTRATION no higher than 1×10^{-7} µCi/mL using gross beta/gamma counting or 5×10^{-7} µCi/mL for the principal gamma emitters using gamma-ray spectroscopy.

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 µCi/gram DOSE EQUIVALENT I–131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 µCi/gram DOSE EQUIVALENT I–131.

ACTION 39 — With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, the local radiation monitor reading (if functional) is recorded or grab samples are collected and analyzed for radioactivity at a MINIMUM DETECTABLE CONCENTRATION no higher than 1×10^{-7} µCi/mL using gross beta/gamma counting or 5×10^{-7} µCi/mL for the principal gamma emitters using gamma-ray spectroscopy.

ACTION 40 — With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 4 hours during actual releases. Pump curves generated in place may be used to estimate flow.

63. G2.3.12 001/3//RAD CTRL/MEM - 3.2/3.7/NEW/HL-18 NRC/RO/SRO/AML

Given the following:

- The unit is in Mode 6 for a Refueling outage.
- The dummy fuel assembly is to be raised to the surface in the West (Unit 2) new fuel elevator.

Which one of the following completes the following statement?

Per 93210-C, "Fuel Elevator Operating Instructions," to raise the new fuel elevator, a ___(1)___ must be used to override the interlock,

and

permission must be obtained from ___(2)___.

A. (1) Bypass Interlock Pushbutton

(2) both the Fuel Handling Coordinator and the Shift Supervisor

B. (1) Bypass Interlock Pushbutton

(2) the Shift Supervisor only

C✓ (1) Key Bypass Switch

(2) both the Fuel Handling Coordinator and the Shift Supervisor

D. (1) Key Bypass Switch

(2) the Shift Supervisor only

G2.3.12 Radiation Control

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:
(CFR: 41.12 / 45.9 / 45.10)

K/A MATCH ANALYSIS:

The question presents a plausible scenario where the dummy fuel assembly must be raised for inspection. The candidate must determine the correct method to bypass the interlock to allow the assembly to raise and whose permission is specifically required.

DISTRACTOR ANALYSIS

A. Incorrect. Key Bypass Switch is required, NOT an Interlock Bypass Pushbutton. The 2nd half is correct as BOTH the Shift Supervisor AND Fuel Handling Coordinator permission is required.

B. Incorrect. Key Bypass Switch is required, NOT an Interlock Bypass Pushbutton. The 2nd half is incorrect as BOTH the Shift Supervisor AND Fuel Handling Coordinator permission is required, NOT the Fuel Handling Supervisor.

C. Correct. Key Bypass Switch is required. BOTH the Shift Supervisor AND Fuel Handling Coordinator permission is required.

D. Incorrect. Key Bypass Switch is required. The 2nd half is incorrect as BOTH the Shift Supervisor AND Fuel Handling Coordinator permission is required, NOT the Fuel Handling Supervisor.

REFERENCES

93210-C, "Fuel Elevator Operating Instructions", Section 4.2, section 4.2, Raising A New or Dummy Assembly in the New Fuel Elevator (Unit 2), pg 5.

VEGP learning objectives:

LO-PP-25101-06 Identify the interlocks and bypasses associated with the following:

d. New fuel elevator

INITIALS

4.2 RAISING A NEW OR DUMMY FUEL ASSEMBLY IN THE NEW FUEL ELEVATOR (UNIT 2)

4.2.1 **Obtain** the bypass interlock key from the Support Shift Supervisor to actuate the key bypass switch. ________

> **CAUTION**
>
> Do not raise a spent fuel assembly in the elevator. Serious radiation exposure to operating personnel may result

4.2.2 **Obtain** permission from the Fuel Handling Coordinator and the Shift Supervisor to actuate the key bypass switch. ________

> **CAUTION**
>
> The following steps should never be utilized without direct supervisory control of the evolution.

4.2.3 **Insert** a new or the dummy fuel assembly in the elevator. ________

4.2.4 **Turn** the Key Bypass Switch to the ON position. The red warning lamp adjacent to the switch flashes with the switch on. ________

4.2.5 **Request** Health Physics personnel monitor radiation levels in the area of the elevator as it is raised. ________

<u>IF</u> monitor indicates an unsafe increase in radiation level:

a. Immediately **release** UP pushbutton to stop upward travel of elevator. ________

b. **Depress** and **hold** DOWN pushbutton <u>UNTIL</u> elevator travel stops. ________

c. **Determine** reason for increased radiation level before proceeding with evolution. ________

64. G2.4.09 001/3//EMER PROC/C/A - 3.8/4.2/LOIT BANK/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- The unit is currently in Mode 4.
- RCS temperature is 290°F with RHR in the shutdown cooling mode of operation.
- PRT level has started to rise.
- RCS pressure indicates 285 psig.
- Pressurizer level has started to lower in an uncontrolled manner.
- Containment pressure and radiation levels are normal.

Which ONE of the following describes the mitigative action required by 18004-C, "Reactor Coolant System Leakage," for the given conditions?

A. Lower RCS pressure to reseat the RHR suction relief valve.

B✓ Stop RHR pumps and place in PTL if pressurizer level is less than 9%.

C. Open an RWST suction to the running RHR pump to increase PZR level.

D. Stop RHR pumps and place in PTL if RCS subcooling is less than 38°F.

G2.4.09 Emergency procedures / Plan

Knowledge of low power / shutdown implications in accident (e.g., loss of coolant or loss of residual heat removal) mitigation strategies (CFR 41.10 / 43.5 / 45.13)

K/A MATCH ANALYSIS:

Question meets the KA by testing the students ability to determine mitigation strategies to perform for the conditions given.

DISTRACTOR ANALYSIS:

A. Incorrect - Plausible because of the PRT level starting to rise. RHR suction relief valve discharges to the PRT at 450 psig. Lowering RCS pressure would reseat the valve. RCS temperature at 290 F is below saturation for 450 psig.

B. Correct-RHR pumps are stopped if PRZR level drops below 9%(37% adverse) or subcooling < 24F(38F adverse).

C. Incorrect - Plausible to open RWST path to raise PRZR level, but not with RCS temperature at 290 F.

D. Incorrect - Plausible if subcooling values between normal and adverse are reversed. Setpoint is 24 F (adverse 38 F). Conditions are not adverse in question.

REFERENCES:

18004-C, "Reactor Coolant System Leakage", step B4.

VEGP learning objectives:

LO-LP-60304-06 Describe why RHR pumps are checked as a first operator action before raising charging flow in Sections B and C of the "RCS Leakage" procedure.

LO-PP-60304-09 Given the entire AOP, describe:

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

B. RCS LEAKAGE (MODE 3 <1000 PSIG AND 4)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

RCS depressurization steps should be initiated as necessary to maintain RCS to PRZR liquid differential temperature less than 270°F and meet RCS leak before break criteria.

B1. Check plant conditions:

__In Mode 3 with RCS pressure less than 1000 psig.

-OR-

__In Mode 4.

B1. Go to the appropriate section of this procedure:

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

-OR-

__SECTION C. RCS LEAKAGE (MODE 5).

__B2. Initiate NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION AND INITIAL ACTION.

__B3. Initiate the Continuous Actions Page.

*B4. **Check if RHR pumps should be stopped:**

a. Check the following:

__ PRZR level - LESS THAN 9% [37% ADVERSE].

-OR-

__ RCS Subcooling - LESS THAN 24°F [38°F ADVERSE].

__a. Go to Step B4.c.

Step 4 continued on next page

65. G2.4.27 001/3//EMER PROC/MEM - 3.4/3.9/BANK - HL-17 AUDIT/HL-18 NRC/RO/SRO/TNT

Following a Control Room Evacuation due to a fire, 18038-1, "Operation From Remote Shutdown Panels," is in progress.

The crew will perform 18038-1, Attachment 'G' for Fire Emergency Operation of SG ARVs.

Which ONE of the following completes the following statement?

This operation will be performed locally ___________.

A. in the North Main Steam Valve Room

B. in the South Main Steam Valve Room

C. inside Remote Shutdown Panel 'A'

D✓ inside Remote Shutdown Panel 'B'

G2.4.27 Emergency procedures / Plan

Knowledge of "fire in the plant" procedures:
(CFR 41.10 / 43.5 / 45.13)

K/A MATCH ANALYSIS:

The candidate is given a plausible scenario where a fire in in the control room. He must know where the local operation of ARVs in the Fire Emergency Mode is performed per Attachment G of 18038-1.

DISTRACTOR ANALYSIS:

A. Incorrect. Fire Emergency Operation of SG ARVs is performed inside Remote Shutdown Panel B. It is plausible the candidate may think ARVs are operated locally in the Main Steam Valve rooms. They have to know which ARVs are fire emergency qualified and which valve room they are operated if they feel they are operated locally in the valve rooms.

B. Incorrect. Fire Emergency Operation of SG ARVs is performed inside Remote Shutdown Panel B. It is plausible the candidate may think ARVs are operated locally in the Main Steam Valve rooms. They have to know which ARVs are fire emergency qualified and which valve room they are operated if they feel they are operated locally in the valve rooms.

C. Incorrect. Fire Emergency Operation of SG ARVs is performed inside Remote Shutdown Panel B. It is plausible the candidate may think the ARVs are controlled inside shutdown panel A depending on which ARV he feels is the fire qualified.

D. Correct. Fire Emergency Operation of SG ARVs is performed inside Remote Shutdown Panel B.

REFERENCES:

18038-1, Attachment G for Fire Emergency Operation of SG ARVs.

VEGP learning objectives:

LO-PP-60327-02 List the instruments and controls that are "Fire event" qualified and how they are identified.

LO-PP-60327-03 In the event of a Control Room fire, which Remote Shutdown Panel is preferred for fire event operation? Why?

ATTACHMENT G

Sheet 1 of 1

FIRE EMERGENCY OPERATION OF SGARVS FROM SHUTDOWN PANEL B

1. Contact the I&C Shop and request an I&C Technician be sent to Unit 1 Shutdown Panel B to operate the two DC current sources.

NOTES

- ARVs for Steam Generators 2 and 3 (1-PV-3010 and 1-PV-3020) are operated remotely from PSDB. These ARVs have the capability to be controlled with a portable temporary current source. ☐
- Two DC current sources (RIS Model CL-2134 or equivalent) having 4-20mA continuously adjustable output and test leads suitable for connection to banana jacks in the shutdown panel have been pre-staged in the Shutdown Panel B storage box. ☐

2. Position 1-HS-3010A (3020A) to the FIRE EMERGENCY position at Shutdown Panel B.

3. Connect 4-20 mA current source via banana plugs inside Shutdown Panel B for the ARV to be operated.

4. Adjust the current signal to the valve to maintain RCS temperature as desired.

5. Upon completion of Fire Emergency Operation, perform the following; independent verification required:

   a. Adjust the current signal to CLOSE the ARV being operated.

   b. Disconnect the DC current source from inside Shutdown Panel B for the ARV being operated.

   c. Position 1-HS-3010A (3020A) to the NORMAL position on Shutdown Panel B.

END OF ATTACHMENT G

66. G2.4.39 001/3//EMER PROC/MEM-3.9/3.8/MOD - MNS 2010/HL-18 NRC/RO/SRO/TNT

Given the following conditions on Unit 1:

- A Site Area emergency has been declared.

- The initial page announcement has been performed.

- Site assembly and accountability is in progress in accordance with 91401-C, "Assembly and Accountability."

Which ONE of the following completes the following statement?

In accordance with NMP-EP-111, "Emergency Notifications," the plant page announcement for the Site Area emergency shall be repeated every ___(1)___ minutes during the first two hours of the declared emergency,

and

follow-up emergency messages using the Emergency Notification form must be completed and transmitted to State, Local, and Federal authorities ___(2)___ .

A. (1) 15

(2) as a minimum, every hour

B. (1) 15

(2) only when significant changes to plant conditions occur

C✓ (1) 30

(2) as a minimum, every hour

D. (1) 30

(2) only when significant changes to plant conditions occur

G2.4.39 Emergency Procedures / Plan

Knowledge of RO responsibilities in emergency plan implementation. (CFR 41.10 / 45.11)

K/A MATCH ANALYSIS:

The question presents a plausible scenario where a site area emergency has been declared, the candidate must determine how often the plant page announcement for site area must be performed and how often follow-up messages to the state, local, and federal authorities must be performed.

DISTRACTOR ANALYSIS:

A. Incorrect. 30 minutes is the requirement for the page announcement. The part for follow-up notifications to state, local, and federal authorities is correct.

B. Incorrect. 30 minutes is the requirement for the page announcement. Follow-up notifications to state, local, and federal authorities are required every 60 minutes as a minimum.

C. Correct.

D. Incorrect. 30 minutes is the requirement for the page announcement. Follow-up notifications to state, local, and federal authorities are required every 60 minutes as a minimum.

REFERENCES:

NMP-EP-111, "Emergency Notifications"

VEGP learning objectives:

LO-LP-40101-15 State the individual responsibilities for making emergency notifications.

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. REPEAT 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: ____________

6.1.5 If the notification of a higher emergency classification cannot be made within 15 minutes of the lesser emergency classification then, the notification of the lesser emergency classification should be completed within 15 minutes of the declaration of the lesser event. The notification for the higher emergency classification should be prepared and an additional notification should be performed within 15 minutes of the higher emergency declaration.

NOTE

Electronic notification provides a mechanism to perform emergency notification near live-time. Electronic notification utilizes standardized data as delineated in Figure 1, Emergency Notification Form. This data is supplemented with additional data provided in WebEOC to enhance the notification process. However, electronic notification and/or communication via WebEOC are not required to satisfy the regulatory requirement to notify offsite agencies of emergency conditions. To expedite availability of WebEOC in an emergency, the crew members responsible for completing Figure 1 and making electronic notifications should login to WebEOC as soon as possible using Attachment 3 and remain logged-in.

6.1.6 Follow-up Emergency Messages, using the Emergency Notification Form, Figure 1 should be completed and transmitted to state, local and federal authorities as designated in Table 1, during an Alert or higher classification (The expectation for follow-up notification is that these notifications will be performed when there is a significant change in plant conditions, or at least every hour).

6.1.7 PAR changes should be communicated to appropriate agencies as soon as possible following PAR development and approval. Notification of PARs to applicable agencies is required within 15 minutes following PAR development and approval.

6.1.8 If this procedure is initiated as part of an emergency preparedness drill or exercise, all verbal communications (radio, telephone, etc.) shall be preceded and followed by the statement: "This is a drill". All electronic notifications shall be clearly marked indicating that the information is drill related.

6.1.9 All GENERAL Emergency notifications will contain PARs. PARs are only applicable for GENERAL Emergency conditions. Guidance for the development of PARs is provided in NMP-EP-112, Protective Action Recommendations.

6.1.10 Dose assessment information and emergency release status are developed utilizing site specific procedures. Results from dose assessment calculations affecting PARs or reflecting a change in the status of radiological releases should be communicated as soon as practicable following the approval of the dose assessment results by the dose assessment supervisor. The expectation is that this information will be communicated to the ED as soon as possible following determination by Dose Assessment Supervision that radiological conditions have changed significantly to warrant notification of offsite agencies.

67. WE02EA2.1 001/1/2/SI TERM - FUNCTIONS/C/A - 3.3/4.2/BANK-FARLEY/HL-18 NRC/RO/SRO/AML

The following conditions exist on Unit 1:

- A LOCA is in progress.
- Main Steam Line Isolation has occurred due to Containment pressure. 15
- 19010-C, "Loss of Reactor or Secondary Coolant," is in progress.

The crew is at the step to, "Check if ECCS flow should be reduced," with plant parameters as follows:

- RCS pressure is 1725 psig and stable.
- CETCs indicate 570°F.
- Total available AFW flow is 580 gpm.
- SG NR levels are all between 12 - 15%. 32-65
- PZR level is 30% and slowly rising. 37

Based on the current conditions, which ONE of the following actions are the operators required to take?

A. Continue in 19010-C.

B. Transition to 19011-C, "SI Termination."

C. Transition to 19012-C, "Post-LOCA Cooldown and Depressurization."

D. Transition to 19231-C, "Response to Loss of Secondary Heat Sink."

WE02EA2.1 SI Termination

Ability to determine and interpret the following as they apply to the (SI Termination):
(CFR: 43.5/ 45.13)

Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A MATCH ANALYSIS:

Question gives a plausible scenario where SI termination is in progress during performance of 19010-C, candidate must choose the correct procedure for the given plant conditions. PRZR level is too low during adverse conditions, SI termination criteria are not met requiring the crew to continue with 19010-C.

ANSWER / DISTRACTOR ANALYSIS

A. Correct - With containment pressure high enough for a Main Steam Line Isolation,

met. The proper course of action will be to remain in 19010-C.

B. Incorrect- With containment pressure high enough for a Main Steam Line Isolation, Adverse Containment conditions apply, therefore, SI termination criteria are NOT met. However, if the student does not correlate the high containment pressure to Adverse Containment conditions, and uses normal values, he would think a transition to 19011-C, SI Termination is the correct transition.

C. Incorrect- This will occur later in the procedure when LHSI parameters are checked, but it is not done at this time. A transition to 19012-C is plausible from 19010-C but this transition is not checked until near the end of the procedure.

D. Incorrect- With Adverse Containment conditions, SG NR levels are too low, if the candidate thinks SG NR levels are too low and does not correlate AFW flow as adequate, he may think a red path on Heat Sink is present making a transition to 19231-C plausible.

REFERENCES

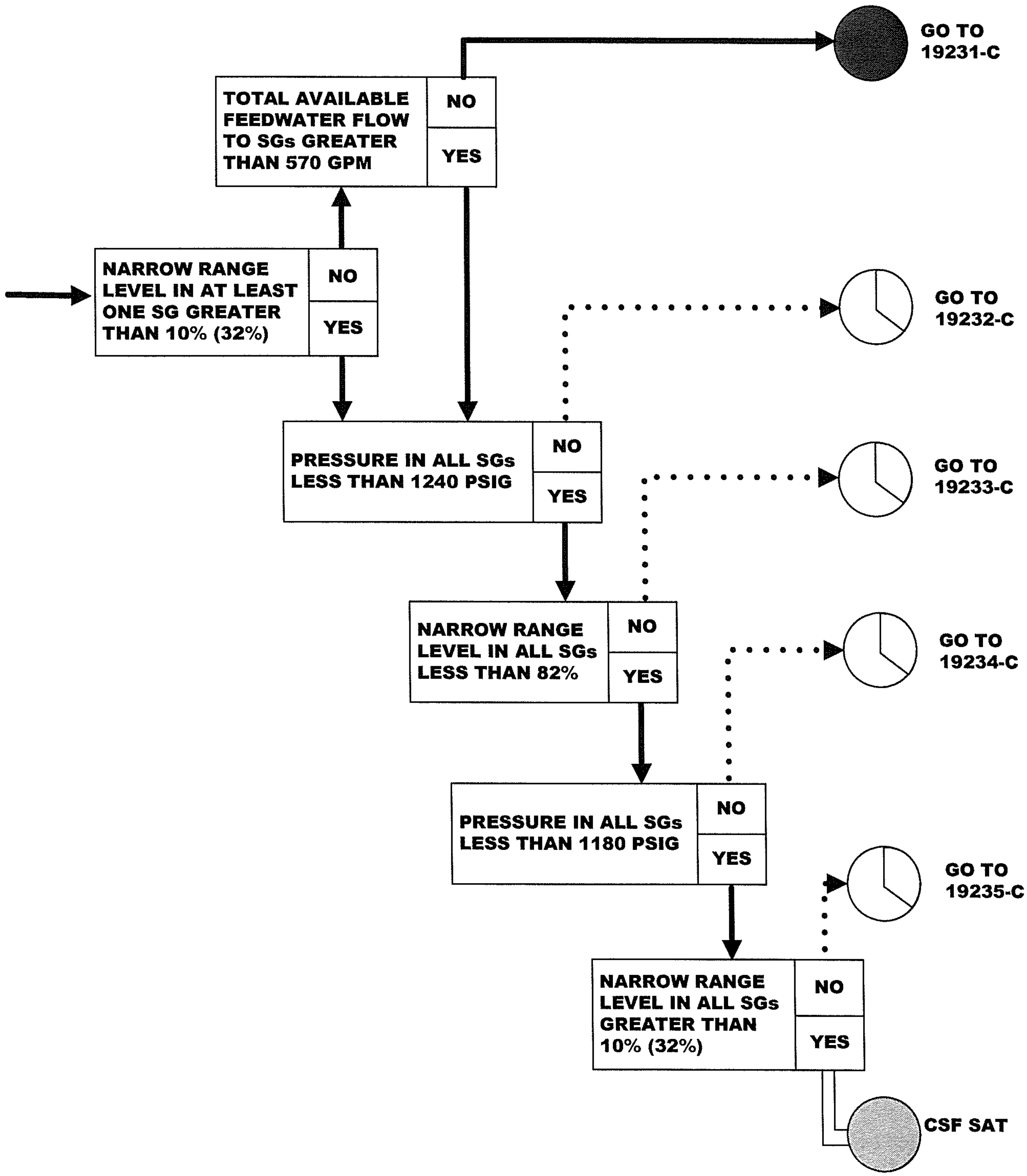
19010-C, "Loss of Reactor or Secondary Coolant".
19200-C, "F-0 Critical Safety Function Status Trees".
Farley 2004 NRC

VEGP learning objectives:

LO-LP-37111-C, Using EOP 19010-C as a guide, briefly describe how each step is accomplished.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***11. Check if ECCS flow should be reduced:**	
__a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE].	__a. Go to Step 12.
b. Secondary Heat Sink:	__b. Go to Step 12.
__ Total feed flow to intact SG(s) - GREATER THAN 570 GPM.	
-OR-	
__ NR level in at least one intact SG - GREATER THAN 10% [32% ADVERSE].	
__c. RCS pressure - STABLE OR RISING.	__c. Go to Step 12.
__d. PRZR level - GREATER THAN 9% [37% ADVERSE].	d. Try to stabilize RCS pressure: __• Use Normal PRZR Spray if Instrument Air to Containment available. __• Do NOT use PRZR PORVs to stabilize RCS pressure. __Go to Step 12.
__e. Go to 19011-C, ES-1.1 SI TERMINATION.	

F- 0.3
HEAT SINK



68. WE03EK3.2 001/1/2/LOCA C/D - PROCEDURE/C/A - 3.4/3.9/NEW/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- A LOCA has occurred.
- Crew is performing 19012-C, "Post-LOCA Cooldown and Depressurization."
- RCS pressure is 2100 psig and lowering.

Which one of the following completes the following statement?

When pressure drops below 2000 psig, the UO blocks the low steam line pressure SI/SLI signal to __(1)__,

and

this __(2)__ block the high steam pressure rate steam line isolation.

A. (1) prevent SI from re-actuating, causing the cooldown rate to be exceeded

(2) will

B. (1) allow cooldown using Steam Dumps to condenser

(2) will

C. (1) prevent SI from re-actuating, causing the cooldown rate to be exceeded

(2) will NOT

D✓ (1) allow cooldown using Steam Dumps to condenser

(2) will NOT

WE03EK3.2 LOCA Cooldown and Depressurization

Knowledge of the reasons for the following responses as they apply to the LOCA Cooldown and Depressurization:
(CFR: 41.5 / 41.10 / 45.13)

Normal, abnormal, and emergency operating procedures associated with LOCA Cooldown and Depressurization.

<u>K/A MATCH ANALYSIS:</u>

Question meets the KA by testing the students knowledge of the procedural step that blocks SI and the reason for performing this action.

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect - Plausible to believe that students will think cooldown from SI reinitiating is the cause for blocking. Students many times invert the plant response for blocking low steam line pressure SI/SLI below P-11.

B. Incorrect - 1st half is correct, 2nd half is plausible that by blocking the SI signal you will also block the steam rate

C. Incorrect - Plausible to believe that students will think cooldown from SI reinitiating is the cause for blocking. Students many times invert the plant response for blocking low steam line pressure SI/SLI below P-11.

D. Correct.

<u>REFERENCES:</u>

19012-C, ES-1.2 Post LOCA Cooldown and Depressurization
WOG ES-1.2 Post LOCA Cooldown and Depressurization

<u>VEGP learning objectives:</u>

LO-LP-37112-01 Using EOP 19012 as a guide, briefly describe how each step is accomplished.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

When the low steamline pressure SI/SLI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***10. Check if low steamline pressure SI/SLI should be blocked:**	
a. Steam Dumps – AVAILABLE.	a. Go to Step 12.
b. PRZR pressure - LESS THAN 2000 PSIG.	b. WHEN PRZR pressure is less than 2000 psig, and the high steam pressure rate alarms are clear, THEN block low steamline pressure SI/SLI by performing Step 10.d. Go to Step 11.
c. High steam pressure rate alarms – CLEAR.	
d. Block low steamline pressure SI/SLI using the following: • HS-40068 • HS-40069	

NOTE: Low steamline pressure SI signal should be blocked when PRZR pressure decreases to less than (A.05) psig.

PURPOSE: To prevent main steamline isolation valve (MSIV) closure on low compensated steamline pressure during controlled RCS cooldown

BASIS:

The SI actuation signal on low steamline pressure can be blocked during cooldown once the PRZR pressure decreases to the P-11 setpoint (approximately 2000 psig). This prevents MSIV closure, thus allowing cooldown by (the preferred method of) steam dump to condenser.

ACTIONS:

- o Determine if PRZR pressure decreases to less than (A.05) psig
- o Block low steamline pressure SI signal

INSTRUMENTATION:

PRZR pressure indication

CONTROL/EQUIPMENT:

Controls to block low steamline pressure SI signal

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

(A.05) PRZR pressure permissive to block low steamline pressure SI (P-11).

NOTE: After the low steamline pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

PURPOSE: To alert the operator to the potential for inadvertent steamline isolation during the subsequent steam generator depressurization

BASIS:

An automatic protection feature is provided to close the main steamline isolation valves when the steam pressure rate signal is exceeded. In the following step, the operator is instructed to dump steam from the intact steam generators which may result in exceeding the rate setpoint. Therefore, this note is intended to alert the operator of this possibility.

ACTIONS:

N/A

INSTRUMENTATION:

MSIV position indication

CONTROL/EQUIPMENT:

Atmospheric steam dump valve controls

KNOWLEDGE:

The rapid cooldown should be continued using the atmospheric steam dumps if MSIV closure occurs

PLANT-SPECIFIC INFORMATION:

The note may be written to warn the operator not to exceed a certain cooldown rate to prevent MSIV closure.

69. WE04EK2.1 001/1/1/LOCA - OUT CONT/C/A - 3.5/3.9/BANK - HL-17 NRC/HL-18 NRC/RO/SRO/AML

Initial conditions:

- The crew is performing 19112-C, "LOCA Outside Containment."

Current conditions:

- RCS pressure is 1500 psig.

Which ONE of the following completes the following statement?

The FIRST system to be isolated from the RCS to attempt leak isolation is __(1)__,

and

the instrument that will be used to determine isolation of the leak is ___(2)___.

A. (1) SI

(2) PRZR pressure

B.✓ (1) RHR

(2) RCS WR pressure

C. (1) SI

(2) RCS WR pressure

D. (1) RHR

(2) PRZR pressure

WE04EK2.1 LOCA Outside Containment

Knowledge of the interrelations between the (LOCA Outside Containment) and the following: (CFR: 41.7 / 45.7)

Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

<u>K/A MATCH ANALYSIS:</u>

The Candidate must know the parameter used to determine leakage is isolated and the parameter used to determine if the leak isolation is successful.

DISTRACTOR ANALYSIS

A. Incorrect. First column is incorrect, RHR is the system isolated.
Second column is incorrect, RCS WR pressure is the correct parameter, WR RCS pressure is a PAMS instrument and PRZR Pressure will not read below 1700 psig and is NOT PAMS qualified.

B. Correct. First column is correct, RHR is the system isolated.
Second column is correct, RCS WR pressure is the correct parameter, WR RCS pressure is a PAMS instrument and PRZR Pressure will not read below 1700 psig and is NOT PAMS qualified.

C. Incorrect. First column is incorrect, RHR is the system isolated.
Second column is incorrect, RCS WR pressure is the correct parameter, WR RCS pressure is a PAMS instrument and PRZR Pressure will not read below 1700 psig and is NOT PAMS qualified

D. Incorrect. First column is correct, RHR is the system isolated.
Second column is incorrect, RCS WR pressure is the correct parameter, WR RCS pressure is a PAMS instrument and PRZR Pressure will not read below 1700 psig and is NOT PAMS qualified

REFERENCES

19112-C, "LOCA Outside Containment"
HL-17 NRC RO exam question # 71 (THIS IS A RE-USE FROM LAST 2 EXAMS)

VEGP learning objectives:

LO-PP-37116-02: Describe the steps taken to isolate a LOCA outside containment.

LO-PP-37116-03: Describe the indications used to confirm that a LOCA outside containment was successfully isolated.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. SI Pump Hot Leg injection valves - CLOSED: • HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV • HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV	c. Dispatch an Operator to close affected Unit valves: 1-HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV (AB-A09) 1-HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV (FHB-A10) 2-HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV (AB-A103) 2-HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV (FHB-A01)
3. Try to identify and isolate RHR Cold Leg injection break:	
a. Close RHR PMP-A TO COLD LEG 1&2 ISO VLV HV-8809A.	
b. Check RCS pressure – RISING.	b. Open RHR PMP-A TO COLD LEG 1&2 ISO VLV HV-8809A. Go to Step 3.d.
c. Go to Step 3.f.	
d. Close RHR PMP-B TO COLD LEG 3&4 ISO VLV HV-8809B.	
e. Check RCS pressure – RISING.	e. Open RHR PMP-B TO COLD LEG 3&4 ISO VLV HV-8809B. Go to Step 4.

Step 3 continued on next page

1. WE04EA1.3 001/1/1/LOCA OUTSIDE/F-3.8/4.0/MOD - LOIT/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

- The crew is performing 19112-C, "ECA 1.2 LOCA Outside Containment".

Current conditions:

- RCS pressure is 1500 psig.

Which one of the following correctly completes the following statement?

The FIRST system to be isolated from the RCS to attempt leak isolation is __(1)__,

and

the instrument that will be used to determine isolation of leak is ___(2)___.

A. (1) SI

(2) PRZR pressure

B.✓ (1) RHR

(2) RCS WR pressure

C. (1) SI

(2) RCS WR pressure

D. (1) RHR

(2) PRZR pressure

70. WE05EK2.2 001/1/1/LOSS HEAT SINK/C/A - 3.9/4.2/NEW/HL-18 NRC/RO/SRO/AML

Given the following:

- 19231-C, "Response to Loss of Secondary Heat Sink," is in progress.
- RCS Bleed and Feed had been initiated when AFW capability was restored.
- All Steam Generators indicate 8% WR level and approximately 90 psig.
- Core Exit Thermocouples are stable at 552 °F.

Per 19231-C, which ONE of the following identifies the required method of re-establishing feed flow under these conditions and the reason why?

A. Feed ONLY ONE SG at a rate of 30 gpm to prevent MSIV closure due to negative rate signal.

B✓ Feed ONLY ONE SG at a rate of 30 gpm to minimize thermal stresses to the SG components.

C. Feed ALL SGs at maximum rate to establish the minimum SG level requirements to allow termination of bleed and feed.

D. Feed ALL SGs at maximum rate to ensure the minimum AFW flow required for heat sink is established to allow termination of bleed and feed.

WE05EK2.2 Loss of Secondary Heat Sink

Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following:
(CFR: 41.7 / 45.7)

Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

K/A MATCH ANALYSIS:

The question presents a plausible scenario where RCS bleed and feed has been initiated during a LOHS. The candidate has to determine the rate at which to feed the SGs and the reason for the feed rate.

DISTRACTOR ANALYSIS

A. Incorrect. The flow rate is correct for the conditions, however, the reason is to prevent or minimize thermal stresses on the SG components.

B. Correct. The flow rate and the bases are correct.

C. Incorrect. With core exits stable, one SG should be fed at 30 gpm to minimize stresses on the SGs.

D. Incorrect. With core exits stable, one SG should be fed at 30 gpm to minimize stresses on the SGs.

REFERENCES

19231-C, "Response To Loss Of Secondary Heat Sink"
FR-H.1 WOG Background Document

VEGP learning objectives:

LO-LP-37051-08: Using EOP 19231 as a guide, briefly describe how each major step is accomplished. Describe the bases for each. (commitment)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__*8. **Check CST level - GREATER THAN 15%.**	__*8. Swap to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM.
9. Verify SG Blowdown isolated:	
__• SG Blowdown Isolation Valves - CLOSED WITH HANDSWITCHES IN CLOSE POSITION.	
__• SG Sample Isolation Valves - CLOSED.	
*10. **Try to establish MDAFW flow to at least one SG:**	
a. Check MDAFW Pump - AVAILABLE:	a. Perform the following:
__• Power available __• Suction pressure __• Discharge pressure	• Initiate actions to restore an MDAFW Pump: __ Reference 13610, AUXILIARY FEEDWATER SYSTEM __• WHEN MDAFW Pump is started, THEN go to Step 10.b. __• Go to Step 11.

Step 10 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*11. **Try to establish TDAFW flow to at least one SG:**	
a. Check TDAFW Pump - AVAILABLE: __• Steam admission valve HV-5106 - OPEN. __• Trip & Throttle valve PV-15129 - OPEN (HS-15111). __• Governor valve SV-15133 - OPERATING PROPERLY (PDIC-5180a)	a. Perform the following: __• Initiate 13610, AUXILIARY FEEDWATER SYSTEM to operate TDAFW Pump as necessary. __• <u>WHEN</u> TDAFW Pump is started, <u>THEN</u> go to step 11.b. __• Go to Step 12.
b. Verify TDAFW pump throttle valves open: __• HV-5122 TDAFW Pump to SG 1 __• HV-5125 TDAFW Pump to SG 2 __• HV-5127 TDAFW Pump to SG 3 __• HV-5120 TDAFW Pump to SG 4	

Step 11 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	__c. Cycle Reactor Trip Breakers.
	__d. Reset FWI.
	__Go to Step 15.
14. Verify SI actuated and reset SI as follows:	
__a. After 60 seconds, reset SI.	__a. IF SI will NOT reset, THEN initiate ATTACHMENT E.
__b. Cycle Reactor Trip Breakers.	
__c. Reset FWI.	
*15. **Try to establish main FW flow to at least one SG:**	
__a. Check condensate system - IN SERVICE.	__a. Place condensate system in service by initiating 13615, CONDENSATE AND FEEDWATER SYSTEMS. __ IF Condensate can NOT be placed in service, THEN go to Step 33.
16. Verify the following:	
__• MFRVs CLOSED AND CONTROLLERS AT 0% DEMAND IN MANUAL.	
__• BFRVs CLOSED AND CONTROLLERS AT 0% DEMAND IN MANUAL.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
27. Check SG levels:	
__a. NR level - AT LEAST ONE GREATER THAN 10% [32% ADVERSE].	__a. IF feed flow to at least one SG verified, THEN maintain flow to restore NR level to greater than 10% [32% ADVERSE]. __ IF feed flow to at least one SG can NOT be verified, THEN go to Step 28.
__b. Return to procedure and step in effect.	

NOTE

When the low steamline pressure SI/SLI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*28. **Try to establish feed flow from the condensate system to one SG:**	
__a. Check condensate system - IN SERVICE.	__a. Place condensate system in service by initiating 13615, CONDENSATE AND FEEDWATER SYSTEMS. __ IF condensate can NOT be placed in service, THEN go to Step 33.

Step 28 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__c. Check Core Exit TCs - STABLE OR LOWERING.	__c. Do NOT limit feed flow to the selected SG if Core Exit TCs are rising and go to Step 49.f.
d. Restore feed flow to selected SG - BETWEEN 30 GPM AND 100 GPM: __• IPC Point - UF5403.	
__e. Check Dry SG WR level - GREATER THAN 9% [31% ADVERSE].	__e. WHEN Dry SG WR level is greater than 9% [31% ADVERSE] THEN raise feed flow to restore NR level greater than 10% [32% ADVERSE]. __ Go to Step 70.
__f. Raise feed flow to restore NR level greater than 10% [32% ADVERSE] and go to Step 70.	
50. Verify MDAFW Pump throttle valves open for selected SG(s): __HV-5139 MDAFW Pump A to SG 1 __HV-5137 MDAFW Pump A to SG 4 __HV-5132 MDAFW Pump B to SG 2 __HV-5134 MDAFW Pump B to SG 3	50. Perform the following as necessary to establish MDAFW feed flow: • Open MDAFW Pump crosstie valves: __1) 1302-U4-055 __2) 1302-U4-056 __• Limit flow rate to avoid pump runout - LESS THAN 600 GPM.

If bleed and feed has been initiated and RCS temperature is increasing, the re-establishment of feedwater flow should be limited to one steam generator and the flow rate used should be as high as can be made available due to the urgency of the situation. If RCS temperatures are stable or decreasing when feedwater flow is restored the flow should be directed to one steam generator and the rate should be limited to the plant-specific equivalent of 25 - 100 gpm until wide range level is established. With stable or decreasing RCS temperatures, the feedwater flow rate is limited to minimize the potential impact of excessive thermal stresses since a direct measure of the steam generator temperature is not available. Once an indicated wide range level is achieved in the affected steam generator, feedwater flow can be adjusted as necessary to restore level into the narrow range and thereby satisfying the requirements for a secondary heat sink.

Once feedwater is established, the feeding process should continue until the RCS temperature indications are decreasing. At that time the active steam generator should be checked for symptoms indicating a faulted or ruptured condition. If the active steam generator is faulted or ruptured, then feedwater should be established to another intact steam generator. If an intact steam generator does not exist, then a decision should be made to use the best available steam generator, which may be the active steam generator. Once the heat load has been transferred to a backup steam generator, the original steam generator should be isolated to prevent further radiation releases.

Thus, the process of initiating feedwater to a dry steam generator, as described here, is one that accounts for the fact that the steam generator temperature may be above 550 F. The number of steam generators that may be fed in a hot, dry condition are limited and if RCS temperature is decreasing the flow rate is also limited so as to limit the thermal shock to the steam generator being fed. Subsequent to securing SI and exiting FR-H.1 the remaining dry steam generators may have their levels recovered at the direction of the plant engineering staff in a manner that will minimize thermal shock to the steam generators. This evaluation should consider steam generator materials and properties, Technical Specification considerations, etc.

71. WE06EG2.4.20 001/1/2/EP - DEGRADED CORE/C/A - 3.8/4.3/NEW/HL-18 NRC/RO/SRO/AML

Procedures list:

19222-C, "Response to Degraded Core Cooling."
19241-C, "Response to Imminent Pressurized Thermal Shock Condition."

Given the following plant conditions:

- Large break LOCA is in progress.
- RCPs are STOPPED.
- RCS subcooling is 15°F.
- CETCs are 744°F.
- Containment pressure is 1.2 psig.
- RVLIS is 68%.

Crew determines that an ORANGE path on Core Cooling exists and transitions to 19222-C. Upon initiating SG depressurization to 200 psig, they receive a RED path on Integrity.

Which one of the following completes the following statement?

Based on the above conditions, the crew is required to ____________.

A. immediately transition to 19241-C

B✓ complete 19222-C, then transition to 19241-C

C. stop the SG depressurization until the RED path on Integrity has cleared, and then continue in 19222-C

D. complete the SG depressurization step, then transition to 19241-C, if the RED path condition on integrity still exists.

WE06EG2.4.20 Emergency Procedures / Plan

Knowledge of the operational implications of EOP warnings, cautions, and notes:
(CFR: 41.1 / 43.5 / 45.13)

Degraded Core Cooling

K/A MATCH ANALYSIS:

Question meets the KA by presenting a plausible orange path condition and testing the students knowledge about the caution discussing the accumulator injection and subsequent red path on integrity.

DISTRACTOR ANALYSIS:

A. Incorrect - Per CAUTION 19222-C is to be completed before transition to 19241-C.

B. Correct - Per CAUTION 19222-C is to be completed before transition to 19241-C.

C. Incorrect - During the depressurization the RED path is expected, the depressurization should NOT be stopped due to the RED path.

D. Incorrect - The depressurization step should not be stopped, 19222-C should be completed before a transition to 19241-C is made.

REFERENCES:

19222-C, pg 10, Caution prior to step 14
WOG Background FR-C.2

VEGP learning objectives:

Not applicable

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

After the low steamline pressure SI is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

CAUTION

The following step will cause Accumulator injection which may cause red path condition in F-0.4, INTEGRITY CSFST. This procedure should be completed before transition to 19241-C, FR-P.1 RESPONSE IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***14. Depressurize all intact SG(s) to 200 psig.**	
__a. Maintain cooldown rate in RCS Cold Legs - LESS THAN 100°F/HR.	
__b. Dump steam to Condenser from intact SG(s) using Steam Dumps.	__b. Dump steam from intact SG(s) using SG ARV(s). __ Go to Step 14.d.
c. Check if low steamline pressure SI/SLI should be blocked:	
__1) PRZR pressure - LESS THAN 2000 PSIG.	__1) Return to Step 14.b.
__2) High steam pressure rate alarms - CLEAR.	__2) Lower steaming rate to clear alarm.

Step 14 continued on next page

CAUTION: The following step will cause accumulator injection which may cause a red path condition in F-0.4, INTEGRITY Status Tree. This guideline should be completed before transition to FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK.

PURPOSE: To alert the operator to complete entire guideline FR-C.2 even if a red path occurs in the Integrity Status Tree, F-0.4.

BASIS:

Once the RCS is cooled/depressurized in step 10 to the point at which the accumulators inject, the RCS cold leg temperature could be reduced such that a transition to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, is required via the red path of Status Tree F-0.4. The operator would stop the cooldown after entering FR-P.1. While the operator is allowing the thermal shock to soak out, the core will continue to boil away the injected accumulator water and begin to uncover once again. Eventually, core exit temperatures and/or RVLIS level values could exist which would require the operator to transfer to FR-C.1, Response to Inadequate Core Cooling, via one of the red paths on Status Tree F-0.2. Thus, by going from FR-C.2 to FR-P.1 and stopping the cooldown and soaking, a degraded core cooling condition could be allowed to deteriorate to an inadequate core cooling condition. Therefore, this caution will require the operator to complete guideline FR-C.2 to ensure core cooling even if a red path condition occurs in the Integrity Status Tree, F-0.4.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

72. WE12EK1.1 001/1/1/UCD PRESS - COMP/C/A - 3.4/3.8/MOD - HL-16 NRC/HL-18 NRC/RO/SRO/AML

During the performance of 19121-C, "Uncontrolled Depressurization of All Steam Generators," the following conditions exist:

- RCS cooldown rate is determined to be 125 °F/hr.
- All SG NR levels are off-scale low.

Which ONE of the following describes how the crew is directed to control AFW flow and the reason for monitoring wide range temperatures?

A.✓ Flow is reduced to 30 gpm to each SG.

WR **Hot** Leg temperatures are monitored to prevent steam generator dryout.

B. Flow is reduced to 30 gpm to each SG.

WR **Cold** Leg temperatures are monitored for conditions that may result in Pressurized Thermal shock.

C. Total flow is reduced to 30 gpm to only one SG.

WR **Hot** Leg temperatures are monitored to prevent steam generator dryout.

D. Total flow is reduced to 30 gpm to only one SG.

WR **Cold** Leg temperatures are monitored for conditions that may result in Pressurized Thermal shock.

WE12EK1.1 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:
(CFR: 41.8 / 41.10/ 45.3)

Components, capacity, and function of emergency systems.

K/A MATCH ANALYSIS:

The controlling of the Safety Related system of AFW flow is important due to the cooldown affect of the depressurization of all SG's. The student must know the value to minimize cooldown and establish a secondary heat sink.

DISTRACTOR ANALYSIS

A. Correct-With RCS cooldown rate >100 °F/HR AFW flow is lowered to 30 gpm/SG and maintained at 30 gpm each until NR level is > 10%. WR hot Leg Temps are monitored and feed flow controlled as necessary to stabilize temperature and prevent SG dryout.

B. Incorrect-Plausible because flow is maintained at 30 gpm to only one SG in 19231-C, "Loss of Secondary Heat Sink".

C. Incorrect-30 gpm to one SG is plausible versus to all SGs. WR Hot leg monitoring is correct.

D. Incorrect-30 gpm to one SG is plausible versus to all SGs. WR CL temperatures are monitored for the arming of COPS and not PTS.

REFERENCES

19121-C, "Uncontrolled Depressurization of All Steam Generators"
WOG Background for ECA-2.1, Uncontrolled Depressurization of All SGs.
HL-16 NRC Exam question # 65

VEGP learning objectives:

N/A

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***4. Control feed flow to minimize RCS cooldown:**	
__a. Monitor shutdown margin by initiating 14005, SHUTDOWN MARGIN AND KEFF CALCULATIONS.	
__b. Check cooldown rate in RCS Cold Legs - LESS THAN 100°F/HR.	__b. Lower feedflow to 30 gpm to each SG. __ Go to Step 4.d.
__c. Check NR level in all SGs - LESS THAN 65%.	__c. Control feed flow to maintain NR level in all SGs less than 65%.
__d. Check NR level in all SGs - GREATER THAN 10% [32% ADVERSE].	__d. Maintain a minimum feed flow of 30 gpm to each SG with less than 10% [32% ADVERSE] NR level.
__e. Check RCS WR Hot Leg temperatures - STABLE OR LOWERING.	__e. Control feed flow or dump steam to stabilize RCS WR Hot Leg temperatures.
***5. Check if RCPs should be stopped:**	
a. ECCS Pumps - AT LEAST ONE RUNNING: __• CCP or SI Pump	__a. Go to Step 6.

Step 5 continued on next page

STEP: Control Feed Flow To Minimize RCS Cooldown:

PURPOSE: To control feed flow to minimize the effects of the cooldown due to the secondary depressurization and to subsequently control the transient.

BASIS:

Depending upon the size of the effective break areas for the steam generators, the cooldown rate experienced after reactor trip could exceed 100°F/hr. A reduction of feed flow to the steam generators has three primary effects:

1) To minimize any additional cooldown resulting from the addition of feedwater,
2) To prevent steam generator tube dryout by maintaining a minimum feed flow to the steam generators and,
3) To minimize the water inventory in the steam generators that eventually is the source of additional steam flow to containment or the environment.

The minimum feed flow of (S.04) gpm represents the value in plant specific units corresponding to 25 gpm. The 25 gpm value is representative of a minimum measurable feed flow to a steam generator. Plant specific values may depend upon flow instrumentation and the sensitivity of the controls on the feed flow.

As steam flow rate drops, the feed flow will eventually increase the steam generator inventory. Feed flow is controlled to maintain steam generator narrow range level less than 50% to prevent overfeeding the steam generators.

In addition, as SG pressure and steam flow rate drop, RCS hot leg temperatures will stabilize and start increasing. The operator controls feed flow or dumps steam to stabilize the RCS hot leg temperatures. This allows the safety injection flow to establish conditions for SI termination and minimizes thermal stresses that may be generated.

ACTIONS:

QUESTIONS REPORT

for Vogtle 2011 (HL16) March RO NRC Bank

1. WE12EK2.1 002/1/1/MSL RUPTURE/3.4/3.7 MEM/MOD BANK WOLF CRK/RO/SRO/NRC/GCW

During the performance of 19121-C, "Uncontrolled Depressurization Of All Steam Generators", the following conditions exist:

- RCS cooldown rate is determined to be 125 °F/HR.
- All SG NR levels are off-scale low.

Which ONE of the following describes how the crew is directed to control AFW flow and the basis for the action?

A✓ Flow is reduced to 30 gpm to each SG.

WR Hot Leg temperatures are monitored to ensure secondary heat sink is maintained.

B. Total flow is reduced to 30 gpm to only one SG.

WR Cold Leg temperatures are monitored for conditions that may result in PTS.

C. Total flow is maintained > 570 gpm until ANY SG NR level is > 10%.

WR Hot Leg temperatures are monitored to ensure secondary heat sink is maintained.

D. Total flow is maintained > 570 gpm until ANY SG NR level is > 10%.

WR Cold Leg temperatures are monitored for conditions that may result in PTS.

73. WE13EA1.2 001/1/2/SG OVPRES - BEHAVIOR/MEM - 3.0/3.2/BANK - LOIT/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- A Reactor trip concurrent with a loss of offsite power has occurred.
- The crew has entered 19232-C, "Response to Steam Generator Overpressure," based on YELLOW condition on the Heat Sink CSF Status Tree.
- SG #3 pressure is 1245 psig.
- SG #1, #2, and #4 pressures are at 1210 psig.
- SG #3 NR level is 85% and slowly rising.
- SG #1, #2, and #4 NR levels are 65% and slowly rising.

Which ONE of the following is an action to mitigate the SG overpressure condition per 19232-C?

A.✓ Locally open the ARV for SG #3.

B. Initiate minimum AFW flow to SG #3.

C. Open the steam supply to the TDAFW pump.

D. Open the steam dumps in Steam Pressure Mode.

WE13EA1.2 Steam Generator Over-Pressure

Ability to operate and/or monitor the following as they apply to Steam Generator Over-Pressure:
(CFR: 41.7 / 45.5 / 45.6)

Operating behavior characteristics of the facility.

K/A MATCH ANALYSIS:

The question present a plausible scenario where the crew is in 19232-C, Steam Generator Overpressure YELLOW path due to SG # 3 pressure at 1245 psig. The candidate must determine the proper method to lower SG # 3 pressure in accordance with the YELLOW path FRP.

DISTRACTOR ANALYSIS:

A. Correct. Dumping steam from an ARV as necessary is a directed method. The procedure does not specify how to dump the steam with an ARV so local would be an acceptable method.

B. Plausible to inject cold AFW flow to lower pressure, however, this is not directed per 19232-C. In addition, the procedure directs the isolation of all FW to the SG (MFW and AFW). While giving temporary relief, the long term effect of injecting cold water in the SG would be over time to heat up and expand, making the overpressure condition worse.

C. Incorrect. Opening the TDAFW steam supply valve is a procedural step, however, the TDAFW steam is not supplied from SG # 3 and will not aid in reducing pressure from SG # 3.

D. Incorrect - Plausible manually dumping steam with the steam dumps in steam pressure mode would relieve the pressure. However, with a loss of offsite power and the 13.8 kV buses de-energized, steam dumps would not be available due to a loss of C-9.

REFERENCES:

19232-C, FR-H.2 Response to Steam Generator Overpressure.

VEGP learning objectives:

Not applicable.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. Try to dump steam from the affected SG(s) using the following as necessary:	☐ 5. Go to Step 7.
☐ SG ARV(s).	
-OR-	
☐ BYPASS STEAM ISOLATION VALVES.	
-OR-	
☐ STEAM SUPPLY VALVES TO TDAFW PUMP.	
* 6. **Check affected SG(s) pressure:**	
☐ a. SG(s) pressure - LOWERING	☐ a. Go to Step 7.
☐ b. SG(s) pressure - LESS THAN 1240 PSIG	☐ b. Return to Step 4.
☐ c. **Control steam release to maintain SG(s) pressure less than 1240 psig.**	
☐ d. Return to procedure and step in effect.	

74. WE14EK2.1 001/1/2/HI CONT PRESS - FUNC/MEM - 3.4/3.7/BANK-LOIT/HL-18 NRC/RO/SRO/AML

Given the following conditions:

- Containment Spray actuation is required, but did not automatically occur.

Which ONE of the following completes the following statements?

The OATC will manually actuate Containment Spray using __(1)__ on 1 of 2 QMCB locations.

As a result of the manual action, ___(2)___ will receive an actuation signal.

A. (1) 1 of 2 handswitches

(2) Containment Spray only

B. (1) 2 of 2 handswitches

(2) Containment Spray only

C. (1) 1 of 2 handswitches

(2) both Containment Spray and CVI

D✓ (1) 2 of 2 handswitches

(2) both Containment Spray and CVI

WE14EK2.1 High Containment Pressure

Knowledge of the interrelations between (High Containment Pressure) and the following:
(CFR: 41.7 / 45.7)

Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A MATCH ANALYSIS:

The question gives a plausible scenario where the candidate manually actuates Containment Spray. The candidate is required to know that 2 of 2 QMCB handswitches at either board location is required to actuate C. Spray and that a manual actuation would result in a CVI actuation as indicated (monitored) by the CVI Actuation alarm.

DISTRACTOR ANALYSIS

A. Incorrect. Takes 2 of 2 handswitches to manually actuate C. Spray, CVI actuates on manual CS actuation.

B. Correct. Takes 2 of 2 handswitches to actuate, however, CVI actuates on manual CS actuation.

C. Incorrect. Takes 2 of 2 handswitches to actuate. Manual C. Spray gives a CVI actuation signal.

D. Incorrect. Takes 2 of 2 handswitches to actuate. Manual C. Spray gives a CVI actuation signal.

REFERENCES

19000-C, Reactor Trip or Safety Injection, OATC Initial Actions
Simulator images
1X6AA02-232-17

VEGP learning objectives:

LO-PP-15101-02, Describe what will actuate the Containment Spray System, including coincidence and set point.

LO-PP-15101-04, List all components that receive a Containment Spray Actuation signal and their change in status.

WINDOW D06

CNMT SPRAY ACTUATION

ORIGIN	**SETPOINT**
2 out of 4 1-PT-0934A 1-PT-0935A 1-PT-0936A 1-PT-0937A	21.5 psig (2/4 channels) (relay K643)
or both 1-HS-40010 1-HS-40011	Not Applicable
or both 1-HS-40004 1-HS-40005	Not Applicable

1.0 PROBABLE CAUSE

1. Manual actuation of the Containment Spray System.
2. Containment HI-3 setpoint reached on 2 or more Containment pressure channels.

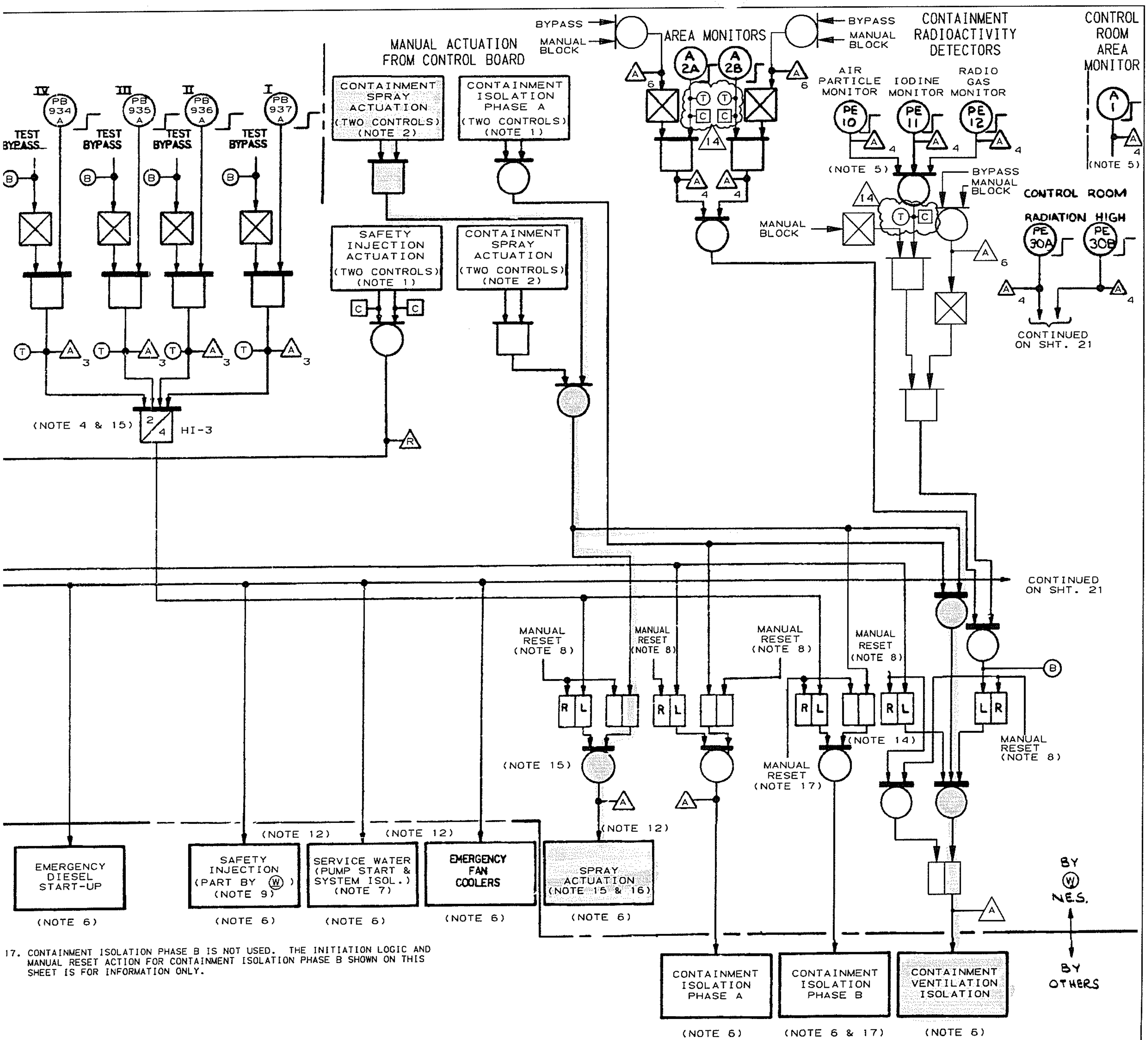
2.0 AUTOMATIC ACTIONS

1. Containment Spray Pumps start.
2. Containment Spray Isolation Valves 1-HV-9001A and 1-HV-9001B open.

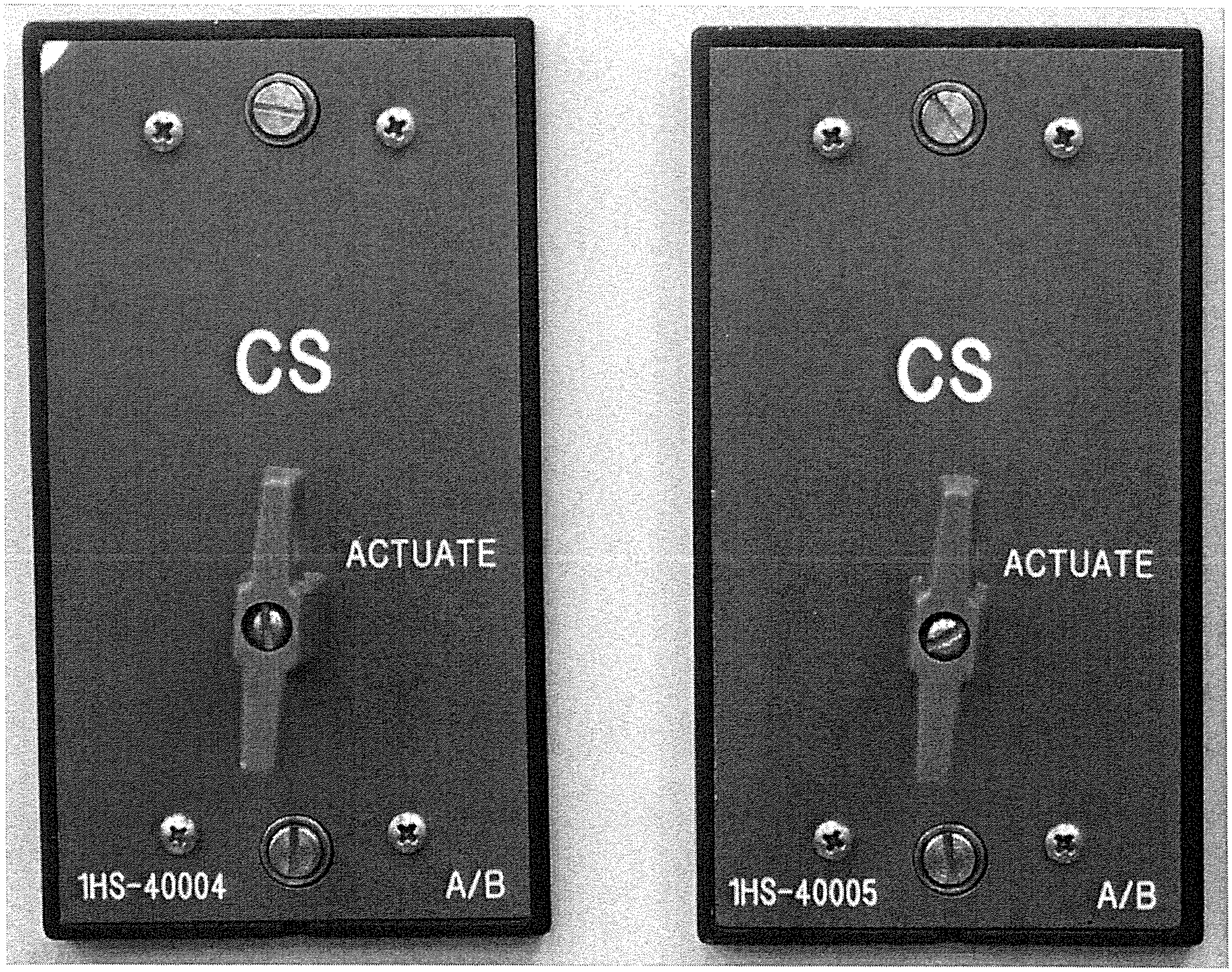
3.0 INITIAL OPERATOR ACTIONS

NOTE

Actions for a containment spray actuation are contained in Emergency Operating Procedures. ☐



17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.


CS
ACTUATE
1HS-40004
A/B
CS
ACTUATE
1HS-40005
A/B

75. WE16EK3.2 001/1/2/HI CONT RAD - PROCED/MEM - 2.9/3.3/NEW/HL-18 NRC/RO/SRO/AML

Given the following plant conditions:

- The unit is in Mode 3.
- An RCS leak has occurred in Containment.
- The crew has entered 19253-C, "Response to High Containment Radiation Level," YELLOW path.

Which ONE of the following describes the reason for operating the Containment Pre-access Filter Units in 19253-C?

A. To reduce ONLY the iodine activity level inside Containment.

B. To reduce ONLY the particulate activity level inside Containment.

C✓ To reduce the iodine and particulate activity levels inside Containment.

D. To prevent the release of airborne activity from Containment to the Aux Building.

WE16EK3.2 High Containment Radiation

Knowledge of the reasons for the following responses as they apply to the High Containment Radiation:
(CFR: 41.5 / 41.10 / 45.6 / 45.13)

Normal, abnormal, and emergency operating procedures associated with High Containment Radiation.

K/A MATCH ANALYSIS:

Question meets the KA by questioning the students knowledge on the lineup through the prefilters during an emergency condition.

DISTRACTOR ANALYSIS:

A. Incorrect - The pre-filters will remove both gaseous and particulate activity.

B. Incorrect - The pre-filters will remove both gaseous and particulate activity.

C. Correct - Pre-filters remove both gaseous and particulate activity.

D. Incorrect - Plausible if the students believe they are preventing or eliminating the release outside of containment. Starting Pre-access units may reduce activity levels in Containment but will NOT prevent a release to the Aux. Building.

REFERENCES:

Westinghouse Background documents FR-Z.3 pg 8
19253-C, Response to High Containment Radiation Level

VEGP learning objectives:

LO-PP-29101-01 State the purpose of each of the following Containment Environment

Control Systems:

c. PRE-ACCESS FILTER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Verify Containment Ventilation Isolation:	
a. Dampers and Valves - CLOSED:	
__ CVI MLB indication	
-OR-	
__ Reference ATTACHMENT A as necessary.	
__2. Check Piping Penetration Filtration and Exhaust Units - BOTH RUNNING.	__2. Start fans by initiating 13305, AUXILIARY BUILDING HVAC SYSTEM.
__3. Place the Containment Preaccess Filter units in service by initiating 13125, CONTAINMENT PURGE SYSTEM.	
__4. Notify TSC of Containment radiation level to obtain recommended action.	
__5. Return to procedure and step in effect.	

END OF PROCEDURE TEXT

STEP: Determine If Containment Atmosphere Filtration System Should Be Placed In Service

PURPOSE: To place containment atmosphere filtration system in service if criteria are satisfied

BASIS:

This step instructs the operator to place the containment atmosphere filtration system in service if possible. The containment atmosphere filtration system, which is part of the reference plant design, will reduce by filtration the radioactivity of the containment atmosphere. In the plant specific design, any available method of reducing the containment atmosphere radioactivity level should be considered to be included in this step. For certain plant designs, fan coolers have filtration capability and could be used in this step.

It should be noted that the use of containment spray to reduce radioactivity has been considered. However, since containment spray is designed for containment heat removal at high containment pressures, it has been determined that it would not be appropriate to use containment spray in this guideline to reduce radioactivity at low containment pressure.

ACTIONS:

- o Determine if plant specific criteria for placing containment atmosphere filtration system in service are satisfied
- o Place containment atmosphere filtration system in service

INSTRUMENTATION:

Plant specific instrumentation to determine if containment atmosphere filtration system can be placed in service

CONTROL/EQUIPMENT:

Plant specific containment atmosphere filtration system controls

KNOWLEDGE:

N/A

D. Air is supplied to the purge system from an intake at el 226 ft 0 in. The containment purge high flowrate system is designed to maintain the airborne radioactivity below the level required for personnel occupancy during refueling. The exhaust from the purge filtration unit is ducted to the plant vent.

The minipurge system is designed to:

- Operate during normal operation. Air is supplied through the same intake plenum used for normal purge during refueling.
- Maintain the airborne radioactivity below the level required for personnel occupancy during reactor power operation.
- Control pressure buildup caused by heat load imposed on the containment atmosphere during startup and the operation of pneumatic controllers.

This system and the preaccess filter system minimize iodine, particulate, and noble gas concentration during the entire period of containment building occupancy.

The purging operation is initiated manually from the control room. The outside air supply is introduced through a prefilter and preheated to 60°F. The containment is maintained at atmospheric pressure during the purge cycle. Two flow paths exist from the containment to the minipurge exhaust system depending on containment pressure. At normal ambient pressure, ± 0.30 psig, the minipurge exhaust fan draws through 14-in. piping and ductwork.

When internal containment pressure is from (+) 0.3 psig to the containment safety injection pressure of (+) 4.4 psig, the 14-in. shutoff damper is closed with the minipurge exhaust fan shut off. Flow is then to be directed through an 8-in. bypass line which contains a flow orifice. This provides overpressurization protection of downstream ductwork and equipment. The exhaust from the minipurge system is routed through the filtration unit before being routed to the plant vent.

In the event that the concentration in the containment of airborne particulates is higher than desired levels, air cleaning is accomplished by activating the recirculation filtration unit. This unit is equipped with charcoal and HEPA filters to reduce containment airborne radioactivity to acceptable levels. The operation of this unit is initiated from the control room by manually energizing the fan.

The containment penetrations of the normal supply and normal purge exhaust are equipped with motor-operated isolation valves inside and outside the containment.

The containment isolation valves of the minipurge supply and the minipurge exhaust are equipped with air operators inside and outside the containment. The containment penetrations, including the isolation valves and appropriate seismic restraints, are designed in accordance with Seismic Category 1 and Quality Class 2 requirements as defined in section 3.2. The air-operated valves are designed to fail closed in the event of loss of power. The valves are controlled automatically by the containment isolation system (discussed in subsection 6.2.4), which overrides all manual signals. The minipurge isolation valves shut within 5 s of receiving an actuator signal. The valves are designed to shut against the containment pressure following a DBA. Table 9.4.6-4 is a comparison of the minipurge system with Branch Technical Position CSB 6-4.

QUESTIONS REPORT

for VC Summer 2009 RO audit exam RETAKE

1. WE16 EK3.2 002/MODIFIED/VCS/MEM (SYS)//RO///

Given the following plant conditions:

- EOP-17.2 *Response to High Reactor Building Radiation Level* has just been entered by the control room team.

Which ONE (1) of the following describes the Bases for operation of RBCU HEPA filters?

A. To reduce the gaseous activity level in the Reactor Building atmosphere.

B. To prevent the release of radioactivity from the Reactor Building to the evironment.

C. To prevent the release of radioactivity from the Reactor Building to the Auxiliary Building.

D✓ To reduce the particulate activity levels in the Reactor Building atmosphere