Georgia Power Company
Nuclear Operations
Post Office Box 1600
Waynesboro, Georgia 30830
Telephone 404 724 8114
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ENCLOSURE 2

SEP 22 1988

Vogtle Project

Georgia Power
the southern electric system

September 21, 1988

Plant Vogtle - Units 1 & 2
Comments - 9/19/88 RO/SRO Written Examination
Log: NOT-01530
Security Code: NC

Mr. John Munro
Nuclear Regulatory Commission
101 Marietta Street, NW.
Atlanta, GA 30323

Dear Mr. Munro:

Attached are the comments compiled by the Training Department regarding the RO and SRO written examinations conducted at Vogtle Electric Generating Plant on September 19, 1988. Your consideration of these comments in grading of the examinations is appreciated.

Very truly yours,

G. Bockhold, Jr.
General Manager

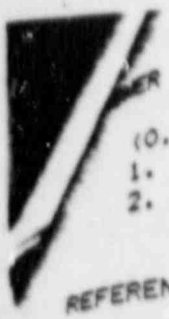
PDR:kss

xc: W. G. Hairston
C. K. McCoy

8812090145 881125
PDR ADOCK 05000424
V PDC

QUESTION 6.17 (1.00)

State the locations where the TWO Safety Grade Cold Shutdown Letdown Flowpaths discharge.



6.17 (1.00)

(0.5 each)
1. PRT
2. Excess Letdown Heatexchanger inlet

REFERENCE

VEGP Training Text, CVCS, Chapter 5.a, p. 8

(3.6/3.4)
004010A307 ..(KA's)

QUESTION [illegible] (1.00)

State the locations where the TWO SAFETY GRADE Cold Shutdown Letdown Flowpaths discharge.

---

ANSWER 2.16 (1.00)

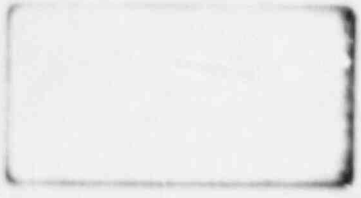
1. PRT.

2. Excess Letdown Heat Exchanger flowpath (OR upstream of the excess letdown heat exchanger).

REFERENCE

VEGP Training Text, CVCS, Chapter 5.a, p. 8
3.6/3.4

004010A307 ..(KA's)



QUESTION NUMBER: 2.16 (2)/6.17

FACILITY COMMENT: Accept these alternate answers for excess letdown:
1) Seal return flowpath to the charging pump header OR
2) Reactor coolant drain tank, since these are specific locations that excess letdown flowpath discharges to.

REFERENCES: Training Text, CVCS, Chapter 5.a, Pg. 8
1X4DB-111
1X4DB-114

BECHTEL
GROUP: POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
P & I DIAGRAM
REACTOR COOLANT SYSTEM
SYSTEM NO. 1201

BECHTEL
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

intermediate leg and passes through the excess letdown flow piping to the excess letdown isolation valves (HV8154 and HV8153). It then flows through the tube of the excess letdown heat exchanger giving up much of its heat to the ACCW System. Excess letdown is reduced from normal loop temperature to approximately 190_ F. It then passes first through a check valve and then through the excess letdown hand control valve (HV123) which provides a means of flow modulation or isolation. The flow continues to the excess letdown three-way flow direction valve (HV8143) where it can be directed to the RCDT or the RCP seal water return header. It is normally directed to the seal water return header. The excess letdown then continues through the normal seal water return flow path to the charging pump suction header. The three-way divert valve (HV8143) provides an alternate flow path to the reactor coolant drain tank for use during operations such as a plant heatup.

- Safety Grade Cold Shutdown Letdown Flow

The safety grade cold shutdown letdown flowpath provide letdown from the reactor vessel head vent through two trains of solenoid isolation valves (HV8095A and HV8096A; Train A and HV8095B and HV8096B; Train B). From these valves, the flow can be directed to the PRT through isolation valves (HV0442A and/or HV0442B), or to the excess letdown flowpath, upstream of the excess letdown heat exchanger, through a motor-operated isolation valve (HV8098).

h. Chemical Control, Purification and Makeup (Figure 5a-6)

The water chemistry, chemical shim and makeup requirements of the RCS are such that the following functions must be provided:

1) Means of addition and removal of pH control chemicals for startup and normal operation.

2) Control of oxygen concentration during normal and shutdown operation of the plant.

3) Means of purification to remove corrosion and fission products.

4) Means of addition and removal of soluble chemical neutron absorber (boron) and makeup water at concentrations and rates compatible with all phases of plant operation including emergency situations.

i. pH Control

The chemical control element employed for pH control is lithium hydroxide (LiOH). This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium systems; in addition, Li-7 is produced in the core region due to irradiation of the dissolved boron in the coolant. The LiOH is introduced to the Reactor Coolant System via the charging flow. A chemical mixing tank is provided to introduce the

QUESTION 5.03 (2.50)

Compare the calculated Estimated Critical Position (ECP) for a startup to be performed 12 hours after a trip from 100% power, to ACTUAL control rod position if the following events/conditions occurred. Consider each independently. (State your answer as Actual position is HIGHER THAN the ECP, the Actual position is LOWER THAN the ECP, or the Actual position is the SAME AS the ECP)

a. One reactor coolant pump is stopped two minutes prior to criticality.

b. The S/U is conducted 4 hours earlier than planned.

c. The steam dump pressure setpoint is reduced to 950 psig.

d. Condenser pressure is reduced from 7 psia to 2 psia.

e. All S/G levels are allowed to decrease by 5% as the rod withdrawal occurs during the startup.

ANSWER 5.03 (2.50)

a. SAME AS
b. HIGHER THAN
c. LOWER THAN
d. SAME AS
e. SAME AS

REFERENCE

VEGP Lesson Plan, LO-LP-33510-01
Learning Objective #14

001010A201 001000K534 001000K104 ..(KA's)

QUESTION NUMBER: 5.03b

FACILITY COMMENT: Answer assumes Xe peaks 8-9 hours following a reactor trip. Answer could change based on assumption for peak Xe. (i.e., Xe peaks at 10 hours - SAME AS). Recommend allowing credit for answers based on initial peaking assumptions.

REFERENCES: LO-LP-33430-02, Pg 7

# III. LESSON OUTLINE:

## NOTES

D. Xenon-135 Behavior Following Shutdown

Objective 8,9

1. $\phi$ approx. equal to 0

   a. $\sigma_a^{Xe} N_{Xe} \phi = 0$, burnout ceases

   b. $\gamma_{Xe} \phi \Sigma_f^{fuel} = 0$, production from fission ceases

2. Only production is by continuing decay of $^{135}I$

3. Only removal is by $^{135}Xe$ decaying to $^{135}Cs$

4. $\frac{dN_{Xe}}{dt} = \lambda_I N_I - \lambda_{Xe} N_{Xe}$

Objective 10

   a. Initially produced at a faster rate than it decays

      1) Lost a major removal process

      2) Only lost 5% of production

      3) $^{135}Xe$ concentration initially increases

   b. If shutdown lasts long enough

Objective 8,9

      1) Rate of removal approaches and then exceeds rate of production

      2) When the rates are equal this is the peak $^{135}Xe$ concentration

Time $\propto \sqrt{\text{power}}$ (6-10 hrs) approx.
-5150 pcm
(±500 pcm)

         a) $\lambda_I N_I = \lambda_{Xe} N_{Xe}$

         b) Time to reach peak varies from 6 to 10 hours, taking longer for higher power levels - rule of thumb is approx. equal to the square root of equil. power.

         c) $^{135}Xe$ worth large amount of -p/ρ

      3) After peak, $^{135}Xe$ decay dominates and concentration decreases to zero.

Objective 10

5. $^{135}Xe$ worth after shutdown

LOTP-33430-00-007

6. Peaking of $^{135}Xe$ concentration - important consideration for reactor startup

   a. Must be enough $+\rho_{excess}$ available to overcome -p/ρ introduced by $^{135}Xe$ - important at EOL

Refer to Reactor Engineering Manual for most recent curve

QUESTION 5.16 (1.00)

Answer the following questions in reference to subcooling margin of the plant.

a. What is the subcooling margin of the plant if the following conditions exist:

Thot = 587 F  Tavg = 572 F  Tcold = 557F
Ppzr = 2235 psig  Psg = 1033 psig

b. If power is raised from 50% to 100%, why does the subcooling margin decrease?

ANSWER 5.16 (1.00)

a. Tsat for 2250 psia (2235 psig)
= 652.67 F (0.50)

Subcooling margin = Tsat - Thot = 652.67 - 587 = 65.67 F + or - 0.05 F (0.50)

b. Subcooling margin decreases because Thot will increase as power increases (0.50)

REFERENCE

VEGP Lesson Plan, LO-LP-34110-03

001000K556 ..(KA's)

QUESTION NUMBER: 5.16a

FACILITY COMMENT: Margin of tolerance is too small (0.05°F). $T_{Hot}$ is given as 587. Answer requires 5 significant figures. Recommend expanding margin to allow for rounding off in calculation. Suggest ±2°F.

REFERENCES: N/A

QUESTION 5.18 (1.50)

Following the taking of critical data, in the intermediate range, a stable startup rate of 0.15 DPM at 0.1% of full power is established. The reactor is at no-load Tave, and the Boron concentration is 800 ppm. The plant is below the point of adding heat.

a. What will the reactor power be after 2 minutes?

b. At what power level does the point of adding heat begin? (No calculation required.)

ANSWER 5.18 (1.50)

a. $P = P_0 10^{exp(SUR)(t)}$ (0.5)

$= (0.1)10^{exp(0.15)(2)}$

$= 0.199526\%$ (0.5)

Round off to 0.2 % is acceptable.

b. 1E-6 to 3E-6 amps or 1% to 2% of full power. (0.5)

REFERENCE

VEGP Lesson Plans, LO-LP-33230-01, P. 6
Lesson Objectives, LO-LP-33230-01, #3

192008K114 ..(KA's)

QUESTION NUMBER: 5.18(b)

FACILITY COMMENT: License candidates are taught that the P0AH is equivalent to approx. $5 \times 10^{-6}$ amps which is also the point where power becomes visible in the power range (-1-2%). This is consistent with actual indications seen on the simulator during training. Recommend wider tolerance on acceptable answer.
(Cannot find reference to values listed in key.)

REFERENCES: N/A

QUESTION 6.04 (2.00)

List FOUR permissives that must be satisfied before the Diesel Generator output breaker will shut following a start by the Safeguards Sequencer.

---

ANSWER 6.04 (2.00)

(any 4 @ 0.5 each)
1. DG ready to load (>440 RPM, 90% voltage)
2. Preferred incoming breakers open.
3. No bus faults as indicated by no bus lockouts on preferred bus.
4. DG close permissive from sequencer.
5. No lockouts on Diesel engine or generator.

REFERENCE

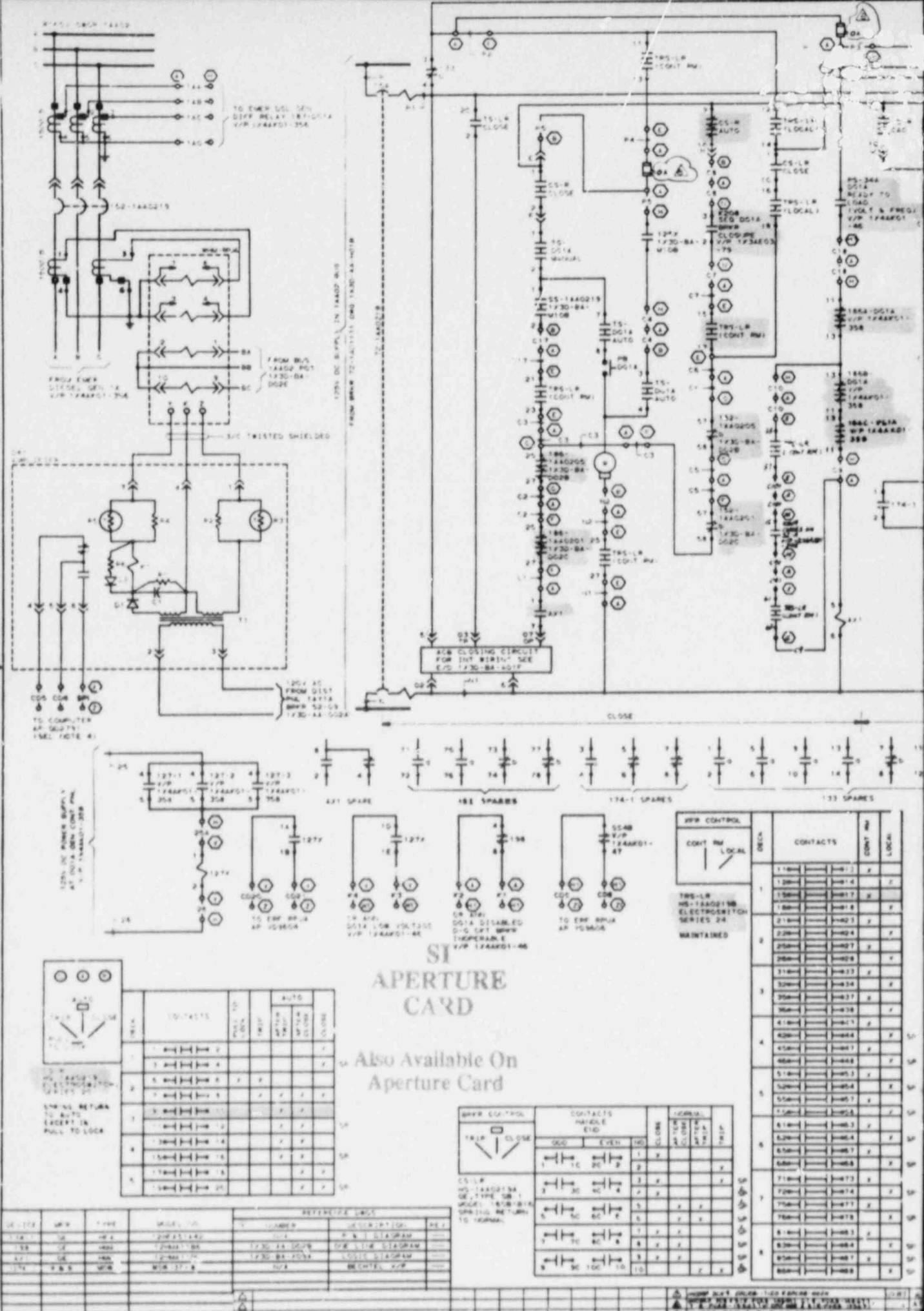
VEGP Lesson Plans, LO-LP-01401-01, p. 8

(3.5/4.0)
064000K411 ..(KA's)

QUESTION NUMBER: 6.04

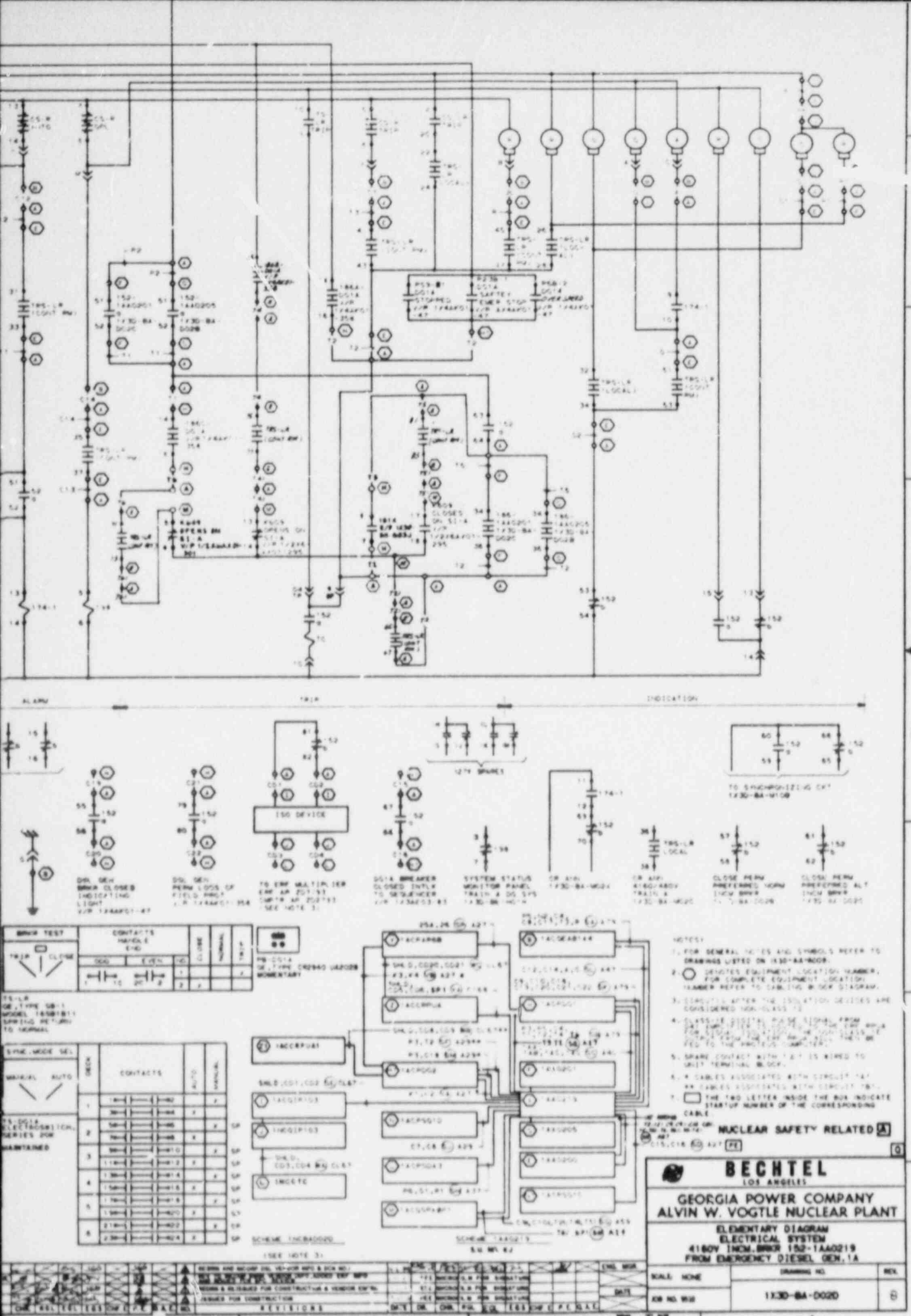
FACILITY COMMENT: Answer gives partial list of requirements for DG breaker closure. Recommend accepting additional requirements as correct answers.
- Breaker "Local-Control Room" selector switch in the control room position (See TRS-LR).
- Breaker control switch in "AUTO" (See CS-R).

REFERENCES: 1X3D-BA-D02D



SI
APERTURE
CARD

Also Available On
Aperture Card


ISO DEVICE
NUCLEAR SAFETY RELATED
BECHTEL
LOS ANGELES
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
ELEMENTARY DIAGRAM
ELECTRICAL SYSTEM
4160V 1NOM.BRKR 152-1AA0219
FROM EMERGENCY DIESEL GEN.1A
1X3D-BA-D020
8
8812090145-01

QUESTION 6.07 (1.50)

How must each of the following parameters change to achieve an increase in the Overtemperature Delta Temperature Setpoint (OTdelta T)?
Use INCREASE, DECREASE, REMAINS THE SAME.

a. Tave

b. Reactor Pressure

c. Delta Flux Penalty

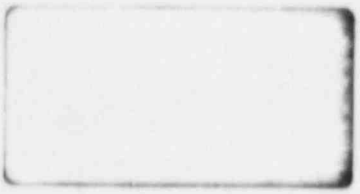
ANSWER 6.07 (1.50)

a. Decreases
b. Increase (0.5 each)
c. Decreases

REFERENCE

VEGP Training Text, Chapter 25 Logics sheet 5.

012000A101 ..(KA's)



QUESTION NUMBER: 6.07c

FACILITY COMMENT: Delta flux penalty [f (Delta I)] can only cause the OT Delta T setpoint to decrease. Changes in the parameter in either direction beyond the allowed range will decrease the setpoint.
Recommend delete part c.

REFERENCE: TS Table 2.2-1

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE $\Delta T$

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:

$\Delta T$ = Measured $\Delta T$ by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured $\Delta T$;

$\tau_1, \tau_2$ = Time constants utilized in lead-lag compensator for $\Delta T$, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured $\Delta T$;

$\tau_3$ = Time constants utilized in the lag compensator for $\Delta T$, $\tau_3 = 0$ s;

$\Delta T_0$ = Indicated $\Delta T$ at RATED THERMAL POWER;

$K_1$ $\leq$ 1.10;

$K_2$ = 0.012/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation;

$\tau_4, \tau_5$ = Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;

$T$ = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured $T_{avg}$;

$\tau_6$ = Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T'$ $\leq$ 588.5°F (Nominal $T_{avg}$ at RATED THERMAL POWER);

$K_3$ = 0.00056/psig;

$P$ = Pressurizer pressure, psig;

$P'$ = 2235 psig (Nominal RCS operating pressure);

$S$ = Laplace transform variable, $s^{-1}$;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -33.5% and + 6.5%, $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5%, the ΔT Trip Setpoint shall be automatically reduced by 1.27% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5%, the ΔT Trip Setpoint shall be automatically reduced by 0.83% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5%.

QUESTION 8.02 (1.00)

While operating in Mode 1, control power is lost to a pressurizer power operated relief valve, which one statement below is CORRECT?

a. Technical Specifications requires no action provided another PORV is operable and all pressurizer code safety valves are operable.

b. Technical Specifications requires the power supply to be removed from the associated block valve after verifying it to to open, if the PORV is not operable within 1 hour.

c. Technical Specifications requires the associated block valve to be shut and its power removed if the PORV is not made operable within 1 hour and continuous operation is desirable.

d. Technical Specifications requires action restore two PORVs to operable status within 72 hours or be in hot standby within next 6 hours and hot shutdown within following 6 hours.

ANSWER 8.02 (1.00)

d

REFERENCE

VEGP T.S. 3.4.4

0000A203 ..(KA's)

QUESTION NUMBER: 8.02

FACILITY COMMENT: For the given conditions, the following Tech Spec actions would be required:

- Within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remover power from the block valve, AND
- Restore at least a total of two PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

IAW TS 3.4.4b, the correct answer would be "c" and "d". Accept "c" since it is the most correct, or since no question is totally correct, delete question.

REFERENCES: TS 3.4.4b

## 3/4.4.4 RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve, and

   1. With only one PORV OPERABLE, restore at least a total of two PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no PORVs OPERABLE, restore at least one PORV to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s) or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.4.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION.

QUESTION 8.09 (1.00)

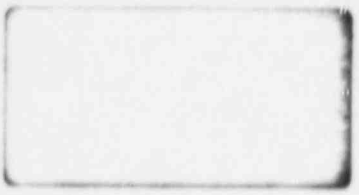
In accordance with VEGP 10003-C, "Manning the Shift", during WHAT modes must a SRO be present in the Control Room?

ANSWER 8.09 (1.00)

Modes 1 to 4 [0.5] for either unit [0.5]

REFERENCE

VEGP Operations Administrative Procedure, 10003-C, p. 1
194001A103 ..(KA's)



QUESTION NUMBER: 8.09

FACILITY COMMENT: The question only asked for the modes an SRO was required to be present in the control room and did not elicit the response "in either unit". Recommend full credit should be given for stating modes required.

REFERENCES: N/A

QUESTION 8.12 (1.50)

List SIX types of activities which are recorded in the Shift Supervisor's log except for the time when the activities occurred.

ANSWER 8.12 (1.50)

(any 6 @ 0.25 each)

1. The name and position of each operator on shift.
2. Major equipment status changes.
3. Major system and equipment testing.
4. Personnel injuries.
5. Entering and exiting a Technical Specification action statement.
6. Significant events, such as reactor trips or unexpected power changes.
7. Implementing the Emergency Plan.
8. Significant security incidents.
9. Mode changes.

REFERENCE

VEGP Operations Administrative Procedure, 10001-C, sec. 2.2.1, p. 1

194001A106 ..(KA's)

QUESTION NUMBER: 8.12

FACILITY COMMENT: Answer gives generic list of activities that are recorded in the SS's log. Recommend credit for specific examples of those events that are required to be recorded.

REFERENCES: N/A

QUESTION 8.14 (2.50)

List FIVE duties of the Emergency Director which are NON-DELEGABLE according to Emergency Response Procedure, 91102-C, Duties of the Emergency Director?

ANSWER 8.14 (2.50)

(any 5 @ 0.5 each)

1. Classifying and declaring the emergency. (including downgrading and terminating.)
2. Recommending protective actions to off-site authorities and content of messages.
3. Authorizing personnel radiation exposures in excess of 10CFR20 limits.
4. Deciding to evacuate non-essential personnel form site at Alert Classification.
5. Deciding to request assistance from federal support groups.
6. Deciding to notify off-site authorities responsible for emergency measures.

REFERENCE

VEGP Emergency Response Procedure, 91102-C, sec. 2.3, p. 2

19400[illegible]4116 ..(KA's)

QUESTION NUMBER: 8. 4

FACILITY COMMENT: Recommend full credit to be given for downgrading and terminating the emergency. These duties are separate and independent of declaration and classification.

REFERENCES: 91001-C, Pg 1, Step 2.2

| Approval | Vogtle Electric Generating Plant<br>NUCLEAR OPERATIONS | Procedure No.<br>91001-C |
|---|---|---|
| Date<br>8/11/88 | Unit COMMON<br>Georgia Power | Revision No.<br>6 |
| | | Page No.<br>1 of 12 |

REPLACEMENT

# EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

MANUAL SET NO. 57

1.0 PURPOSE

1.1 The purpose of this procedure is to provide instructions in the classification of off-normal events into one of four emergency classification levels. This procedure also provides initial implementing instructions for each emergency classification.

2.0 RESPONSIBILITIES

2.1 The On-Shift Operations Supervisor (OSOS) is responsible for initial classification of events. The OSOS shall assume the responsibilities of the Emergency Director until relieved. The OSOS then becomes responsible for recognizing changes in plant conditions and advising the Emergency Director concerning classification of events.

2.2 The Emergency Director has the following non-delegable responsibilities relative to emergency classification.

2.2.1 Classifying and declaring the emergency.

2.2.2 Declaring changes in the emergency classification, including downgrading and terminating.

2.3 The Technical Support Center (TSC) and the Emergency Operations Facility (EOF) Managers are responsible for:

2.3.1 Providing recommendations on emergency classifications to the Emergency Director.

3.0 PREREQUISITES

An off-normal event has occurred, or is in progress.

QUESTION 8.17 (1.00)

What is the Clearance Policy used when placing a clearance on a FAIL OPEN Air-Operated Valve (What must be done to Tag the valve closed)?

ANSWER 8.17 (1.00)

(0.5 each)
1. HOLD tag the handswitch in the CLOSED position.
2. HOLD tag the handwheel in the CLOSED position.

REFERENCE

VEGP-00304-C, p. 8, Equipment Clearance and Tagging

194001K102 ..(KA's)

QUESTION NUMBER: 8.17

FACILITY COMMENT: Answer only addresses a FAIL OPEN AOV with a handwheel. Recommend credit also be given for clearance policy towards FAIL OPEN AOV without a handwheel as follows.
- HOLD tag the handswitch in the closed position.
- Mechanically or hydraulically gag the valve in the closed position and HOLD tag the gagging device.

REFERENCES: 00304-C, Pg 8, 9, Steps 4.1.4.2. and 4.1.4.3.

3.8 PLANT PERSONNEL

It is the responsibility of all plant personnel to adhere to the requirements of this procedure.

3.9 INDEPENDENT VERIFIER

The Operations Department Individual who is responsible for verifying the position of a safety-related component as described on the CLEARANCE SHEET in accordance with the provisions of Procedure 00308-C, "Independent Verification Policy".

4.0 INSTRUCTIONS

4.1 CLEARANCE PHILOSOPHY

4.1.1 When clearing power supplies to solenoids remove fuses, where practical, instead of links. Fuses shall be bagged and HOLD tagged as part of the CLEARANCE. No more than two fuses per bag.

4.1.2 The scope of each CLEARANCE should be just enough to adequately clear the equipment. This is to reduce interference with other CLEARANCES, and use of PARTIAL RELEASES.

4.1.3 When HOLD tagging a Motor Operated Valve (MOV) as a fluid boundary, the handswitch shall be HOLD tagged in the position in which the valve handwheel will be HOLD tagged. The breaker shall be opened or off, as applicable, and the handwheel shall be HOLD tagged.

4.1.4 When using Air Operated Valves (AOV) as boundary valves perform the following:

4.1.4.1 For a FAIL CLOSE AOV

a. HOLD tag the handswitch in the closed position

b. HOLD tag the air supply valve closed and check that the air line to the valve is depressurized.

c. If the valve has a handwheel, HOLD tag it in the closed position.

4.1.4.2 For a FAIL OPEN AOV with handwheel

a. HOLD tag the handswitch in the closed position

b. HOLD tag the handwheel in the closed position

4.1.4.3 For a FAIL OPEN AOP without handwheel

a. HOLD tag the handswitch in the closed position

b. Mechanically or hydraulically (as appropriate) gag the valve in the closed position and HOLD tag the gagging device.

4.1.5 When restoring Air Operated Valves used as boundary valves perform the following

4.1.5.1 For a FAIL CLOSE AOV

a. If the valve has a handwheel, remove the handwheel HOLD tag and restore to the operate or open position.

b. Remove the HOLD tag from and open air supply valve

c. Remove HOLD tag from handswitch and operate as required.

4.1.5.2 For a FAIL OPEN ACV with handwheel

a. Remove HOLD tag from and open air supply valve

b. Remove HOLD tag from handwheel and place handwheel in the open position.

4.1.5.3 For a FAIL OPEN AOV without handwheel

a. Remove HOLD tag from and open air supply valve.

b. Remove HOLD tag from gagging device and remove gag.

4.1.6 The handswitch position on HOLD TAGS shall be in the same position as the controlled component, e.g. A handswitch tagged closed means the valve is closed. If a difference exists due to unforeseen circumstances an information tag shall be attached to the handswitch and the CLEARANCE stating conditions. An example may be: The motor of a MOV is burned out and manual operation is necessary.

4.1.7 HOLD TAGS should be placed in a manner that they will be easily visible to anyone preparing to operate the equipment.

QUESTION 1.02 (2.00)

State how each of the following will affect shutdown margin. Limit your answer to INCREASE, DECREASE, or NO CHANGE. Consider each case separately. Assume EOL.

a. Boron concentration is decreased 20 ppm while maintaining constant power and no rod motion.

b. Bank D rod height is increased from 125 steps to 200 steps while maintaining constant power and boron concentration.

c. Reactor trip. all rods insert.

d. While shutdown, the RCS is cooled down by 40 degrees.

ANSWER 1.02 (2.00)

a. DECREASE

b. NO CHANGE

c. INCREASE

d. DECREASE

REFERENCE

VOGTLE LO-LP-33510, Objs. 1, 5 & 6.
3.2/3.4, 3.5/3.7, 3.8/3.9

192002K114 192002K113 192002K110 ..(KA's)

QUESTION NUMBER: 1.02c

FACILITY COMMENT: Per attached reference, the defined SDM assumes a stuck rod. Under this assumption SDM prior to and immediately following a reactor trip is the same. Recommend changing answer to be NO CHANGE.

REFERENCES: LO-LP-33510-1, Obj. 1

# III. LESSON OUTLINE:

NOTES

I. INTRODUCTION

The shutdown margin (SDM) calculation is used to determine the amount of reactivity by which the reactor is or would be subcritical from any given conditions. It is necessary for the reactor operator to know how to calculate a SDM to operate the reactor safety and to ensure compliance with Technical Specifications.

Review operational modes prior to lecture

Objectives

LO-TP-33510-00-001

LO-TP-33510-00-005

II. PRESENTATION

A. Shutdown Margin (SDM)

Objective 1

1. Define in Tech Specs:

2. Shutdown margin shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

3. Purpose of SDM

Objective 3

a. Ensures reactor can be safely made subcritical from operating condition.

b. Reactivity transients during postulated accidents can be acceptably controlled.

c. Reactor remains subcritical when shutdown.

B. Tech Spec Limits (T.S. 3.1.1.1/2)

Objective 4

1. MODE 1 and 2 (Power ops, startup)

a. SDM shal be greater than or equal to 1.3% delta k/k.

b. Limit based on possible steam line rupture at EOL, no-load Tave.

MTC more negative at EOL, therefore cooldown will add more + reactivity.

2. MODE 3 and 4 (Hot standby, Hot Shutdown)

a. SDM shall be greater than or equal to ~~2.0% delta k/k.~~ the limits shown in the Tech Spec. figure 3-1-1.

LO-TP-33510-00-002

b. Limit more restrictive, at higher boron conc.

Note: at no-load Tavg, there is more mass in the S/G, therefore more heat removed as it boils dry.

QUESTION 1.10 (1.00)

If the UNIT 2 reactor were to operate continuously at approximately 100% power for the entire first fuel cycle, describe how the axial flux peak would move or behave over core life as follows:

a. State why the axial flux peak is initially located below the core centerline.

b. What causes the axial flux peak to move upward over core life?

ANSWER 1.10 (1.00)

a. The lower core inlet temperature [0.25] in combination with the negative MTC [0.25] result in more power being generated in the bottom half of the core.

b. Fuel depletion in lower regions of the core.

REFERENCE

VOGTLE LO-LP-33520-01, Obj. 3.
2.9/3.1

192005K112 ..(KA's)

[redacted]

[redacted]

QUESTION NUMBER: 1.10a

FACILITY COMMENT: Recommend accepting answers that discuss or describe operational characteristics of a negative MTC (i.e., better moderation, higher power, etc.). See attached.

REFERENCES: LO-LP-33520, Obj. 3

# III. LESSON OUTLINE:

## NOTES

B. Axial Power Distribution — Objective 3

1. Pmax at 40% core height BOL due to

   a. $T_{cold}$ entering the bottom of the Reactor, therefore

   b. Better moderation

2. EOL

   a. Fuel in high flux areas begins to burn out — LO-TP-33520-00-003

   b. Peak flattens and moves up

3. Control rod position effects

   a. Would significantly disrupt the smooth flux distribution — Objective 3

   b. Cause local power peaks

   c. Operationally attempt to run at power with <u>all</u> rods out

   d. Try to compensate for power changes using dissolved boron

4. Xenon distribution

   a. Can cause xenon oscillations

   b. ~~Talk about Xenon oscillations later~~ Periodic axial flux shifts

C. Radial power distribution — Objective 1

1. A core of uniform fuel enrichment would have a pronounced central radial power peak — Objective 3

2. To flatten the radial neutron flux fuel is radially zoned — LO-TP-33520-00-004

   a. Use three enrichments

      1) 2.1 w/o U-235
      2) 2.6 w/o U-235
      3) 3.1 w/o U-235

   b. High enrichment is used in outer region

   c. The two lower enrichments are mixed checker - board fashion — See Vogtle Text Vol. 2, Ch. 4, Section C, Page 4-82

QUESTION 4.03 (1.50)

In accordance with the Standing Orders, state SIX operations procedures that may be performed without having the procedures in hand.

ANSWER 4.03 (1.50)

CAF

REFERENCE

VOGTLE Standing Order, # CAF.
VOGTLE 00054-C, "Rules for Performing Procedures," step 4.2.4.1.
4.1

194001A10S ..(KA's)

QUESTION NUMBER: 4.03

FACILITY COMMENT: Standing order referenced in question has not been issued or written. Recommend deleting question.

REFERENCES: Standing Order Book

# TABLE OF CONTENTS

DATE ISSUED 08-05-88

VOGTLE ELECTRIC GENERATING PLANT

OPERATIONS STANDING ORDERS

UNIT ONE


| ORDER NO. | DATE ISSUED | STANDING ORDER TITLE |
|---|---|---|
| 1-87-63 | 12-03-87 | REQUIREMENT FOR IST RETESTING |
| 1-87-71 | 12-29-87 | ACCUMULATOR #3 INCREASING LEVEL |
| 1-88-10 | 02-19-88 | CRITICAL COMPONENT |
| 1-88-21 | 05-09-88 | POWER LEVEL MONITORING PROGRAM |
| 1-88-22 | 06-11-88 | FIRE PROTECTION LCO PROGRAM |
| 1-88-23 | 07-08-88 | CONTROL BLDG. CHILLED WATER OPERATIONS |
| 1-88-25 | 07-28-88 | MANUAL OPERATION OF MOTOR OPERATED VALVES |

# Table of Contents

DATE OF ISSUED 8-4-88
VOGTLE ELECTRIC GENERATING PLANT
OPERATIONS STANDING ORDERS
UNIT 2


| ORDER NO | DATE ISSUED | STANDING ORDER TITLE |
|---|---|---|
| 2-87-03 | 08-24-87 | UNIT 2 ACTION REQUIRED FOR UNIT 1 PCB RECLOSURE |
| 2-87-14 | 11-20-87 | UNIT II EMERGENCY HORN |
| 2-87-16 | 12-14-87 | FIRE PROTECTION PIV INTERFACE |
| 2-87-17 | 12-14-87 | CONTROL OF RWST OUTLET VALVE |
| 2-88-01 | 01-14-88 | TRACKING OF VALVE/BREAKER MISPOSITIONING |
| 2-88-08 | 04-14-88 | UNIT 2 ELECTRICAL BUS AND SUPPORT SYSTEM OUTAGES |
| 2-88-09 | 04-14-88 | VALID ANNUNICATOR IDENTIFICATION PLAN |
| 2-88-10 | 04-15-88 | POTENTIAL CONTAMINATION IN THE SPENT FUEL POOL |
| 2-88-16 | 05-06-88 | PREREQUISITE CONDITIONS REVIEW |
| 2-88-18 | 05-18-88 | VALID METER IDENTIFICATION PLAN |
| 2-88-19 | 05-20-88 | ROVING WATCH SYSTEM QUALIFICATION |
| 2-88-20 | 05-24-88 | ROVING WATCH PROGRAM |
| 2-88-21 | 06-02-88 | MAINTENANCE WORK ORDER REQUESTS |
| 2-88-22 | 06-06-88 | RCS LEVEL MONITORING |
| 2-88-23 | 07-13-88 | UNIT II FIRE PROTECTION RESPONSE |
| 2-88-24 | 07-27-88 | ADMINISTRATION OF OPERATING PROCEDURES DURING HOT FUNCTIONAL TESTING |
| 2-88-25 | 08-04-88 | FIELD SHIFT SUPERVISOR DUTIES |


INFORMATION ONLY

QUESTION NUMBER: 1.19
~~1.09~~/5.13

STATEMENT: Answer lists generic causes of water hammer. Allow credit for specific examples of operations that could cause water hammer.

REFERENCES: N/A

QUESTION NUMBER: 6.11 a

STATEMENT: Coincidence with causing a full feed water isolation, the SIS & P-14 signals also result in a turbine trip. Some students may include the main turbine on the list of components tripped. Recommend that this be included as an optional response.

REFERENCES: 1X5DN203-1

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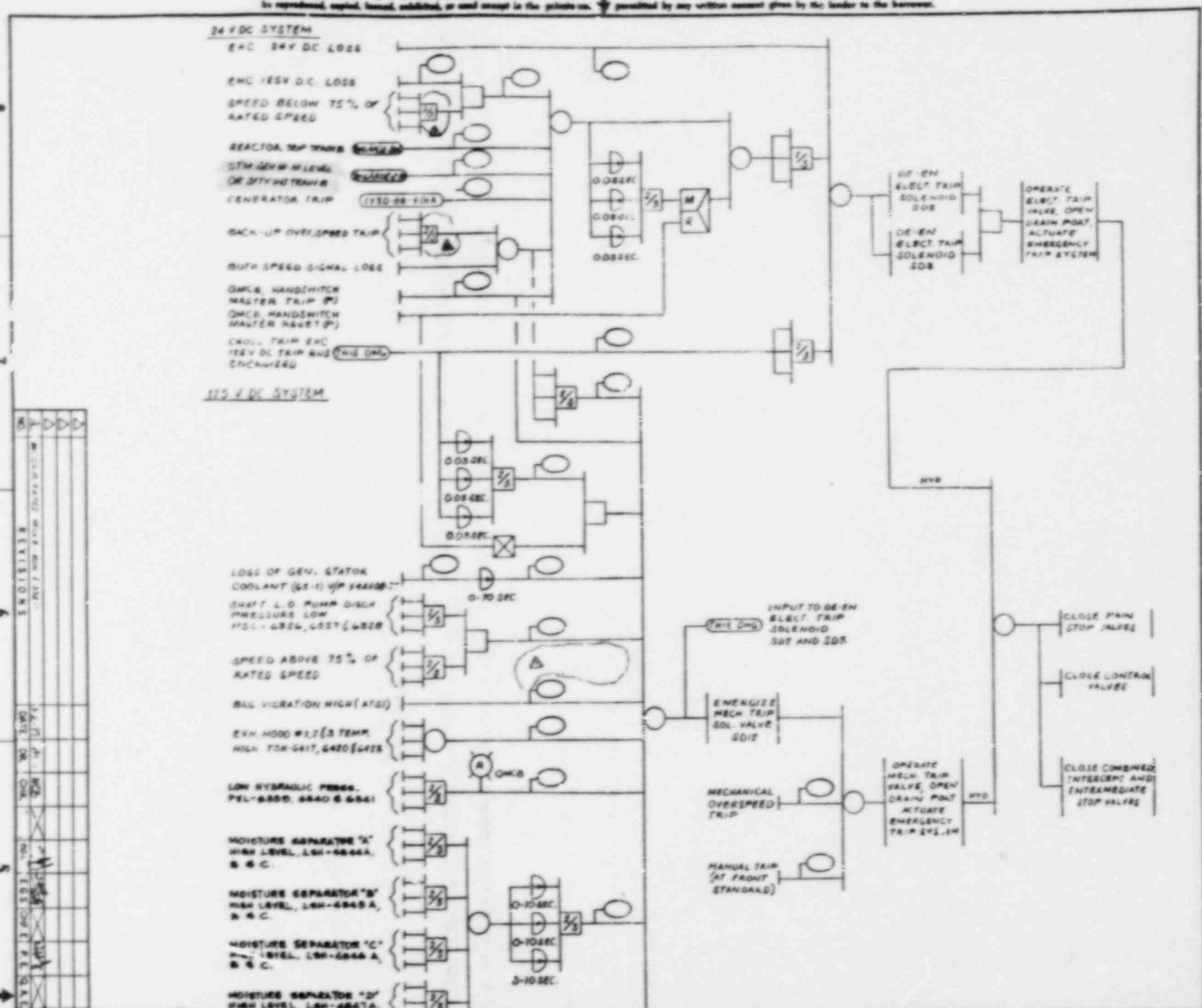


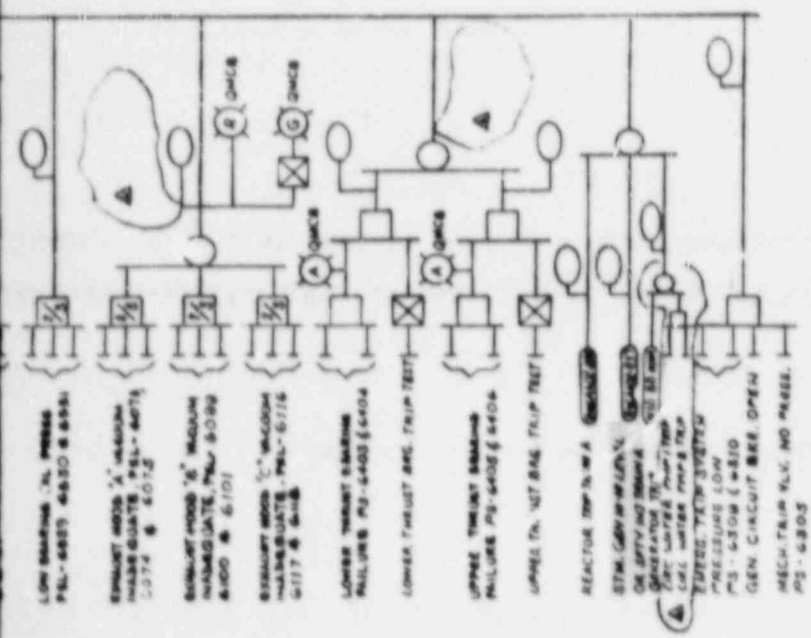

XSDN 203-1

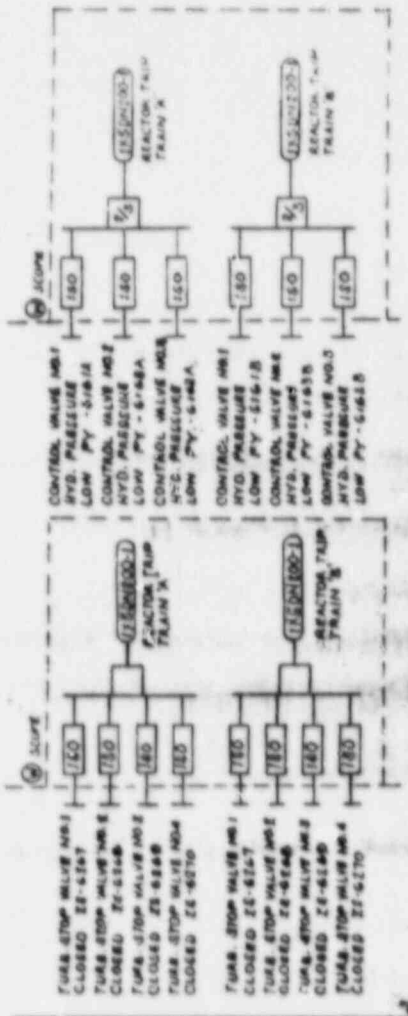
NOTES

1. THIS LOGIC DIAGRAM DEPICTS THE FEATURES OF THE GENERAL ELECTRIC MARK II BWR SYSTEM. TEST LOGIC IS NOT INCLUDED.
2. THE GENERATOR TRIPS SHOWN ARE INSTANTANEOUS. OTHER TURBINE TRIPS INITIATE A GENERATOR TRIP THROUGH THE REACTOR TRIPPING SYSTEM, WHICH IS DELAYED FOR APPROXIMATELY 30 SECONDS IN ORDER TO PROVIDE POWER FOR THE REACTOR COOLANT PUMPS.
3. MAIN STOP VALVES, CONTROL VALVES, AND COMBINED INTERCEPT STOP VALVE POSITION ARE INDICATED BY PRESSURE INDICATORS.
4. PROVIDE ELECTRICAL INPUT FOR CONTROL ROOM LOSS OF CONDENSER TRIP ISOLATION LOGIC (SEE M & E).

REF. DWGS.
[illegible]

SI
APERTURE
CARD
Also Available On
Aperture Card




| NO. | REVISIONS | DATE | DR. | CHK. | SUPV. | E&L | EGS | CHF. E | P.E. | P.E. | ENG. MGR. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 5 | DELETED DUPLICATION OF REACTOR TRIP RELAYS | 10-8-82 | P.F. | J.Mc | | | P.E. | | | | |
| 4 | CORRECTED TAG NUMBERS AND GATE TYPE | 8-2-82 | ALV | PAL | | | P.E. | | | | |
| 3 | CHANGED TAG NUMBERS | 5-10-82 | PAL | J.F. | | | P.E. | | | | DATE |
| 2 | ISSUED FOR CONSTRUCTION | 9-3-79 | PAL | PCR | BRK. | JH | Jmc | | | | |

NUCLEAR SAFETY RELATED

BECHTEL
LOS ANGELES

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

CONTROL LOGIC DIAGRAM
MAIN TURBINE TRIPPING SYSTEM

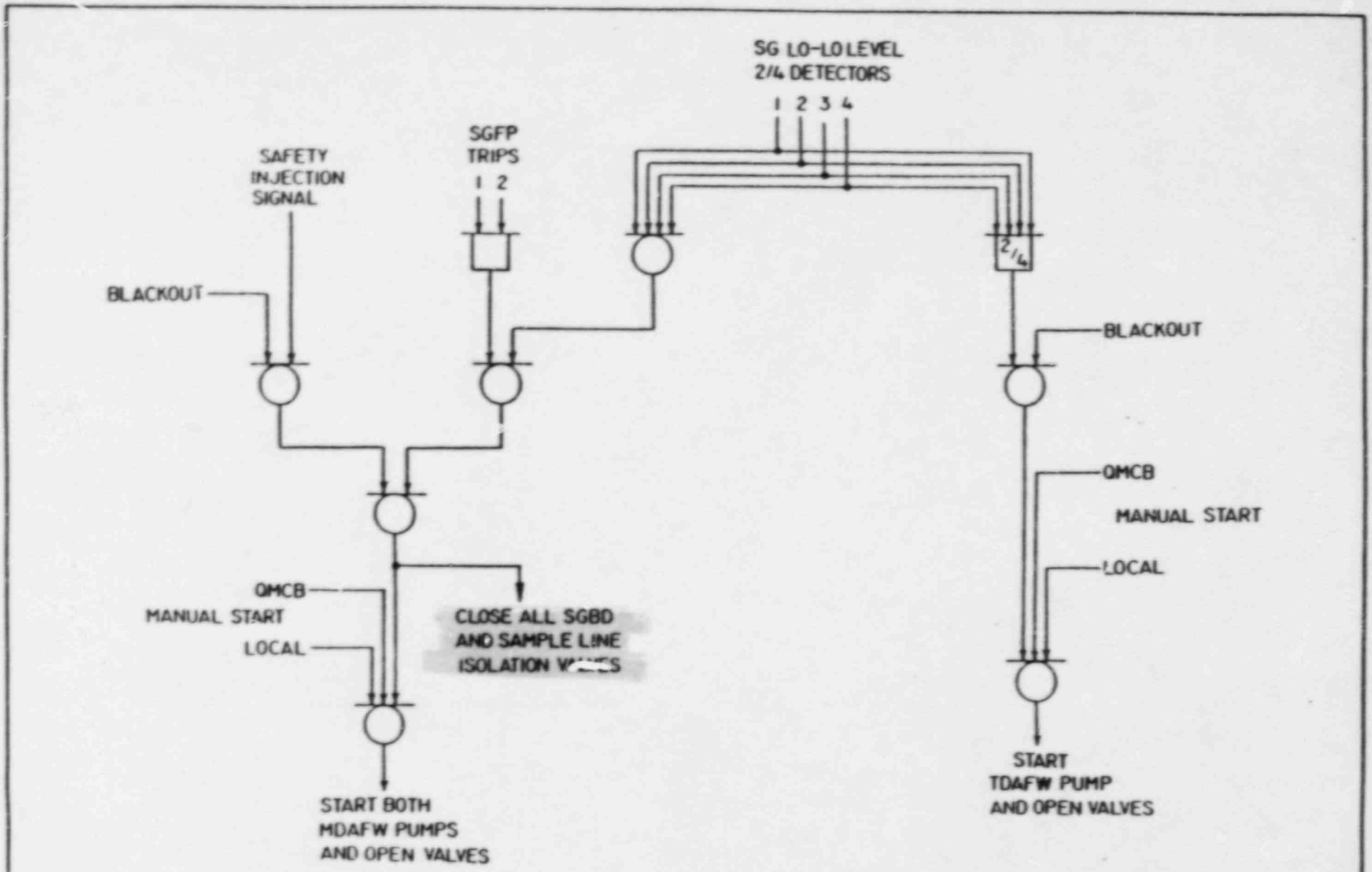
| SCALE: NONE | DRAWING NO. | REV. |
|---|---|---|
| JOB NO. 9510 | 1X5DM203-1 | 4 |

8812090145-02

QUESTION NUMBER: 6.13

STATEMENT: Some students may include the S/G blowdown isolations and S/G sample valves in their answers since they are actuated from a MDAFW actuation signal. Recommend these be included as optional responses.

REFERENCES: Figure 13d-3




REV. 1

Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AUXILIARY FEEDWATER
START LOGIC

FIGURE 13d-3

4339

QUESTION NUMBER: 7.09 a

STATEMENT: Answer a. should be changed to greater than 24°F.

REFERENCES: 19001-C, Attachment D, Rev. 5., Pg 16

Sheet 1 of 1

## ATTACHMENT D

## VERIFICATION OF NATURAL CIRCULATION

The following conditions support or indicate natural circulation flow:

- RCS subcooling monitor indication - GREATER THAN 24°F.
- SG pressures - STABLE OR LOWERING.
- RCS hot leg temperatures - STABLE OR LOWERING.
- Core exit thermcouples - STABLE OR LOWERING.
- RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE.

END OF ATTACHMENT D

QUESTION NUMBER: 7.15

STATEMENT: Answer should be expanded to include:
- At the end of each calendar year.
- Not used for 6 days.

REFERENCES: 00930-C, Steps 5.2.12.2 and 5.2.12.6, Pg 14

5.2.11.7 An individual(s) may be added as necessary to complete the work, to an existing specific RWP by the job supervisor after verifying the individual's remaining exposure limit and with the concurrence of HP supervision.

5.2.12 Termination Of Radiation Work Permits

5.2.12.1 Specific RWPs shall be terminated or suspended whenever the job is completed or cancelled, or whenever significant changes in radiological conditions occur.

5.2.12.2 General RWPs shall be terminated whenever significant changes in radiological conditions occur or at the end of each calendar year. General RWPs shall be reviewed by Health Physics supervision each calendar quarter.

5.2.12.3 HP should ensure that terminated RWPs are clearly identified as terminated and inform the Shift Supervisor of RWPs to be terminated.

5.2.12.4 HP should also ensure that terminated RWPs are removed from the job location and/or control points and returned to the Health Physics office.

5.2.12.5 Specific RWPs that are not used for 3 working days should be suspended. The RWP can be reactivated upon HP notification by the work party supervisor or Work Planning Group.

5.2.12.6 Any RWP that has not been used for 6 days shall be terminated unless HP has been apprised of the need for suspending the RWP by the work party supervisor or the Work Planning Group.

5.3 CONTAMINATION CONTROLS

5.3.1 Use Of Protective Clothing (PC)

5.3.1.1 Personnel shall wear and use protective clothing in a manner consistent with the training each has received. Dressing/undressing guidelines are given in Table 1.

5.3.1.2 HP shall ensure that PCs are commensurate with the levels and state of the contamination expected for the area entered. Suggested protection levels are shown in Table 2.

5.3.1.3 Outer street clothing and jewelry should be removed prior to donning PCs. Violations of this policy may result in the confiscation of contaminated clothing and personal effects.

QUESTION NUMBER: 8.18

STATEMENT: Question does not ask what the O ENTRY" margin is, but what limitations are placed on an individual when he/she has reached this margin. Recommend full credit for: "the individual should NOT be allowed to enter a RCA".

REFERENCES: Exam key

QUESTION NUMBER: 8.20

STATEMENT: Part c. In accordance with 13105 a sample must be taken within 6 hours after an indicated level change of 7% (67 gals) or more which corresponds to a 1% volume change as required by surveillance requirements 4.5.1.b. Recommend that answer c. be modified to meet the above condition.

REFERENCES: 13105-1, Rev. 4
TS 3/4.5.1

4.2.1.6 When the desired Accumulator level is reached, CLOSE the fill valve opened in Step 4.2.1.5. Independent verification required.

4.2.1.7 REPEAT Steps 4.2.1.5 and 4.2.1.6 if required to adjust levels in additional Accumulators.

4.2.1.8 CLOSE 1-HV-8871. Independent verification required

4.2.1.9 CLOSE 1-HV-8888. Independent verification required

4.2.1.10 STOP the running SI Pump and ENSURE its control switch is in AUTO: Independent verification required.

SI Pump A 1-HS-0998A

SI Pump B 1-HS-0999A

NOTE

The boron concentration of an Accumulator must be verified within 6 hours after an indicated level rise of 7% (67 gallons) or more per Surveillance Requirement 4.5.1.b.

4.2.1.11 NOTIFY Chemistry to sample the Accumulator as required and REPORT the boron concentration of each one sampled.

4.2.1.12 If required, ADJUST Accumulator pressure(s) per Sub-subsection 4.2.3 of this procedure.

4.2.1.13 LOG Accumulator boron concentration sample results in the Unit Control Log when reported by Chemistry.

4.2.2 Pressurizing The Accumulators

4.2.2.1 OPEN Nitrogen Supply Header Isolation 1-2402-U4-016.

CAUTION

Accumulator Nitrogen supply/vent valves (1-HV-8875A-H) will hold pressure on the Accumulator side only. Therefore, when 1-HV-8380 is opened, all Accumulators are pressurized simultaneously.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6616 (36% of instrument span) and 6854 (64% of instrument span) gallons (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B)

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open. (HV-8808A, B, C, D)

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume (67 gallons) by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

QUESTION NUMBER: 1.22c

STATEMENT: Steam flow is not used in Vogtle calorimetric calculation. Recommend delete c. from question and reassign point values to a., b., and d.

REFERENCES: 14030

## DATA SHEET 1

## MANUAL CALORIMETRIC

NOTE

The plant computer should be used when available for obtaining the following data; when not available, indications in ( ) should be used.

1.0 RECORD the following data:

1.1 Time _____ Date _____

1.2 Average Steam Pressure (See Note 1)

| | | | | |
|---|---|---|---|---|
| SG 001 | P0400A | (PI-514) | _____ | PSIG |
| | P0401A | (PI-515) | _____ | PSIG |
| | P0402A | (PI-516) | _____ | PSIG |
| SG 002 | P0420A | (PI-524) | _____ | PSIG |
| | P0421A | (PI-525) | _____ | PSIG |
| | P0422A | (PI-526) | _____ | PSIG |
| SG 003 | P0440A | (PI-534) | _____ | PSIG |
| | P0441A | (PI-535) | _____ | PSIG |
| | P0442A | (PI-536) | _____ | PSIG |
| SG 004 | P0460A | (PI-544) | _____ | PSIG |
| | P0461A | (PI-545) | _____ | PSIG |
| | P0462A | (PI-546) | _____ | PSIG |
| | | Total = | _____ | PSIG |
| | | Average Steam Pressure = | _____ | PSIG |

NOTE 1: If a data point is not acceptable (e.g., value abnormally high or low, indicator bad, total loss of indication), enter "Bad" for the value. Calculate the "Total" based on the remaining values and obtain the "Average" by dividing the "Total" by the number of remaining values.

SHEET 1

1.3 Average FW [illegible]erature (See Note 1)

| | | | | |
|---|---|---|---|---|
| SG 001 | T0 | [illegible]-15208) | _____ | °F |
| SG 002 | T043 | [illegible]I-15209) | _____ | °F |
| SG 003 | T0458A | (TI-15210) | _____ | °F |
| SG 004 | T0478A | (TI-15211) | _____ | °F |
| | | Total = | _____ | °F |
| | | Average FW Inlet Temperature = | _____ | °F |

1.4 Total FW Flow (See Note 2)

| | | | | |
|---|---|---|---|---|
| SG 001 | F0403A | (FI-510A) | _____ | MPPH |
| | F0404A | (FI-511A) | _____ | MPPH |
| SG 002 | F0423A | (FI-520A) | _____ | MPPH |
| | F0424A | (FI-521A) | _____ | MPPH |
| SG 003 | F0443A | (FI-530A) | _____ | MPPH |
| | F0444A | (FI-531A) | _____ | MPPH |
| SG 004 | F0463A | (FI-540A) | _____ | MPPH |
| | F0464A | (FI-541A) | _____ | MPPH |
| | | Sum = | _____ | MPPH |
| | | Total FW Flow = Sum/2 = | _____ | MPPH |

NOTE 1: If a data point is not acceptable (e.g., value abnormally high or low, indicator bad, total loss of indication), enter "Bad" for the value. Calculate the "Total" based on the remaining values and obtain the "Average" by dividing the "Total" by the number of remaining values.

NOTE 2: If a data point is not acceptable (e.g., value abnormally high or low, indicator bad, total loss of indication), record the indicated value of its redundant data point for that loop in its place.

## DATA SHEET 1

1.5 Blowdown Flow (Enter zero if secured) (See Note 3)

| | | | | |
|---|---|---|---|---|
| SG 001 | F0407A | (FI-1171B) | _____ | GPM |
| SG 002 | F0427A | (FI-1172B) | _____ | GPM |
| SG 003 | F0447A | (FI-1173B) | _____ | GPM |
| SG 004 | F0467A | (FI-1174B) | _____ | GPM |
| | | Total = | _____ | GPM |

1.6 Indicated Power (A Drawer)

| | | |
|---|---|---|
| N41 | _____ | % |
| N42 | _____ | % |
| N43 | _____ | % |
| N44 | _____ | % |

1.7 PROTEUS Calorimetric Power (Enter NA if not available)

U1118 _____ MWT

NOTE 3: If blowdown indication is lost, enter 0 gpm for the applicable loop. This is the default value used in U1118.

DATA SHEET 1

2.0 Determine the following steam table data:

2.1 Average Steam Enthalpy = _____ BTU/lbm
(Use Average Steam Pressure +14.7 and Saturated Steam Tables, quality factor of 1)

2.2 Average FW Enthalpy = _____ BTU/lbm
(Use Average FW Temperature and Average Steam Pressure +14.7 and Subcooled Tables)

2.3 Average Blowdown Enthalpy = _____ BTU/lbm
(Use Average Steam Pressure +14.7 and Saturated Liquid Enthalpy at this pressure, quality factor of 0)

2.4 Average Feedwater Specific Volume = _____ $ft^3/lbm$
(Use Average FW Temperature and 1000 psia and Subcooled Tables from PTDB.)

3.0 Determine the temperature corrected FW Flow:

3.1 $$\text{Correction Factor} = \left\{1 + 0.98 \times 10^{-5}\left[\left(\frac{\text{Step}}{1.3}\right) - 440\right]\right\}^2 \sqrt{\frac{0.01917}{\left(\frac{\text{Step}}{2.4}\right)}}$$

$$= \left\{1 + 0.98 \times 10^{-5}\left[\frac{\quad}{} - 440\right]\right\}^2 \sqrt{\frac{0.01917}{\quad}}$$

$$= \_\_\_\_\_ \text{ (5 decimal places)}$$

3.2 $$\text{Corrected FW Flow} = \left(\frac{\text{Step}}{1.4}\right) \times \left(\frac{\text{Step}}{3.1}\right)$$

$$= \_\_\_\_\_ \times \_\_\_\_\_$$

$$= \_\_\_\_\_ \text{ MPPH}$$

DATA SHEET 1

4.0 Determine the total reactor thermal power:

4.1 $Q = \left(\begin{matrix}\text{Step 3.2}\\ \text{MPPH}\end{matrix}\right) \times \left[\left(\begin{matrix}\text{Step 2.1}\\ \text{Btu/lbm}\end{matrix}\right) - \left(\begin{matrix}\text{Step 2.2}\\ \text{Btu/lbm}\end{matrix}\right)\right]$

$- \left(\begin{matrix}\text{Step 1.5}\\ \text{gpm}\end{matrix}\right) \times \left[\left(\begin{matrix}\text{Step 2.1}\\ \text{Btu/lbm}\end{matrix}\right) - \left(\begin{matrix}\text{Step 2.3}\\ \text{Btu/lbm}\end{matrix}\right)\right] \times \frac{495.12}{10^6} \frac{\text{MPPH}}{\text{gpm}}$

$- 55.29^{*}$ MBTU/Hr

$= (\_\_\_\_\_) \times \left[\_\_\_\_\_ - \_\_\_\_\_\right]$

$- (\_\_\_\_\_) \times \left[\_\_\_\_\_ - \_\_\_\_\_\right] \times \frac{495.12}{10^6}$

$- 55.29$

$= \_\_\_\_\_ - \_\_\_\_\_ - 55.29$

$= \_\_\_\_\_$ MBTU/Hr

*Net RCS heat input - losses determined during post core load hot functional testing.

QUESTION NUMBER: 2.02(2)

STATEMENT: SI pumps should be as follows:
850 gpm (425 each) @ 1150 psig. The pressure given in the key (1750 psig) is the design maximum pump pressure. Pump head at design flow is approx. 1150 psig (2680 ft.).

REFERENCES: LO-LP-13201-01

# III. LESSON OUTLINE:

NOTES

c. Design Pressure--1750 psig

d. Design Temp--300°F

e. Design Flow--425 gpm

f. Design Head--2680 ft

LO-TP-13201-00-004

g. Maximum Flow--650 gpm

h. Design head at max flow--1650 ft

i. Design shutoff head--1625 psig

j. Required NPSH at any flow--less than 15 ft

k. Motor has two motor coolers (30 gpm each NSCW)

l. Pump bearings have oil cooler (10 gpm NSCW) and will run only 40 seconds without NSCW flow. Pump has eleven impellors and a mechanical seal.

m. Each pump is 100% capacity for ESF analysis.

n. Handswitches on QMCB-A and respective shutdown panel.

1) Give alarm on SSMP when in Pull-to-Lock.

2. RWST--covered in LO-LP-13001

3. HV-8806 RWST to SIP Suction Isolation Valve

a. QMCB/Lockout-A

b. 480 V MCC 1 BBD

c. Out of position alarm (SSMP, Group Mon. Light)

4. HV-8923A SIP-A Suction Isolation Valve

a. QMCB-A

b. 480 V MCC 1 ABD

c. Out of position alarm (SSMP, Gp. Mon.)

QUESTION NUMBER: 2.03

STATEMENT: b. RMWST is alternate makeup
DWST is normal makeup
Change answer to reflect these sources as stated above.

REFERENCES: LO-LP-25102-02-c, Obj. 9
~~13719-1, Pg 12, Step 4.4.4~~

## III. LESSON OUTLINE:

NOTES

5) **Temperature**

a) SFP
b) SFP HX outlets

6) **Flow out of backflushable filter**

V. System Interfaces

A Refueling Water Storage Tank (RWST)

1. used primarily for refueling water purification

2. Line from RWST through refueling water purification pumps joins purification loop upstream of demineralizer

3 SFP WATER SUPPLY can be pumped or gravity fed

4. LINE joins demin line on cooling loop return ~~from SFP~~

B Reactor Makeup Water Storage Tank (RMWST)

1 Train A line only

2 Seismic Category I makeup supply

## III. LESSON OUTLINE:

NOTES

3. pumped or gravity fed

4. ~~makeup~~ for evaporative losses

4. ALTERNATE MAKEUP

C. Demineralized water

1. ~~Alternate makeup~~ NORMAL MAKEUP

2. Normally locked closed

D. Basin Recycle System (Hold up Tank-HUT)

1. used to supply water for maintenance in canal area

2. water pumped from transfer canal to SFP via portable pump

3. As SFP level rises, excess water diverted to BRS HUT via purification loop.

IV Administrative Requirements

A Technical Specifications (3/4.9.11)

1. Limiting Conditions for Operation (LCO)

QUESTION NUMBER: 2.04

STATEMENT: Also accept as correct answer "to limit AFW flow to a depressurized SG".

REFERENCES: FSAR 10.4.9.1.1

effects of hydraulic instability (water hammer). The AFW is also designed to permit testing of the pumps during normal plant operation.

The AFW is provided with flow limiting orifices to limit the auxiliary feedwater flow to a depressurized steam generator and to ensure that the required minimum flow is directed to the remaining three intact steam generators.

10.4.9.1.2 Power Generation Design Basis

The AFW supplies feedwater to the steam generators at a sufficient flowrate to support normal low-power transients such as startup, cooldown, and hot standby.

10.4.9.1.3 Codes and Standards

Codes and standards applicable to the AFW are listed in table 3.2.2-1. The AFW is designed and constructed in accordance with Safety Class 3 requirements up to the motor-operated valves in each branch line located upstream of the stop-check valves, which serve as the containment isolation valves; the remaining valves and piping to the main feedwater system are Safety Class 2.

10.4.9.2 System Description

10.4.9.2.1 General Description

The system consists of two motor-driven pumps, one steam turbine-driven pump, and associated piping, valves, instruments, and controls as shown on figures 10.4.9-1 and 10.4.1-1 (sheet 3). Table 10.4.9-1 provides data for the various active components in the AFW. Table 10.4.9-2 presents minimum system flow requirements for various modes of operation. A plan view of the AFW pumphouse is provided in figure 1.2.2-27.

Each motor-driven auxiliary feedwater pump is sized to supply the feedwater flow required for removal of 100 percent of the decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. The system capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the reactor coolant system (RCS) at 50°F/h. Following transients or accidents, cooldown may begin within 1 to 2 h at zero load hot standby. For a transient or accident condition, the minimum flow is

QUESTION NUMBER: 2.05b

STATEMENT: Accept additional correct answers as stated in AFW SOP.
- AFW piping hot to the touch.
- AFW pump starts but fails to develop adequate discharge pressure.

REFERENCES: 13610-1, Pg 11, Step 4.4.4.1

4.4.3.6 OPEN the MS SPLY TO AUX FW TD PMP 1 valves 1-HV-3009 and 1-HV-3019. Independent Verification Required.

4.4.4 Response To High Auxiliary Feedwater Pump Discharge Temperatures

NOTE

Backleakage of feedwater into the Auxiliary Feedwater System can disable the AFW pumps due to steam or vapor binding.

If backleakage through a check valve continues after attempting to reseat it then the affected pump should be inspected frequently for steam binding and overheating, and operated as necessary to prevent it from overheating.

4.4.4.1 Check valve backleakage may be identified by any of the following symptoms:

a. Auxiliary Feedwater Piping hot to the touch,

b. Auxiliary Feedwater Pump discharge temperature greater than 200°F as shown on the ERF computer terminal,

| AFW PUMP | TEMP ELEMENT | COMPUTER POINT |
|---|---|---|
| TDAFW | TE-5166 | T9887 |
| MDAFW B | TE-5211 | T9888 |
| MDAFW A | TE-5210 | T9889 |

c. Auxiliary Feedwater piping temperature rapidly increases following pump shutdown,

d. Auxiliary feedwater pump starts but fails to develop adequate discharge pressure.

4.4.4.2 If an Auxiliary Feedwater Pump is vapor bound COOL the pump as follows:

a. CLOSE the discharge valves to the Steam Generators for the affected pump,

QUESTION NUMBER: 3.04a

STATEMENT: The upper limit for a thermocouple is 2300 (page 11). However, the upper limit of indication for a thermocouple (asked for in questicn) is 2200°F (page 13). Accept either answer.

REFERENCES: LO-LP-36102-00-c, Pg 11 & 13

# III. LESSON OUTLINE:

NOTES

CONDITION IV: SIGNIFICANT CORE DAMAGE SUCH THAT DETECTOR INSERTION IS HINDERED

A. Assumptions

1. Core shutdown - all rods may or may not be fully inserted.

2. Core damage so severe as to allow only partial or no insertion of incore detectors.

a. Most likely to occur in upper, central regions

b. Amount of insertion available will give operator a general idea of amount of core damage.

B. Using MIDS to Locate Damage

1. Insert detector until stopped by damage and reading detector position to build 3-D map.

2. Picoammeter not needed for this core condition.

III. CORE EXIT THERMOCOUPLES

A. General System Description

1. 50 chromel-alumel thermocouples used to measure exit temperature

LO-TP-36102-00-C-014
LO-TP-36102-00-C-015

2. Located on upper core plate at various locations.

LO-TP-36102-00-C-016

3. Clad in stainless steel sheathed with aluminum oxide insulation.

4. Thermocouple characteristics

a. T/C output linear with respect to temperature

b. Accuracy

1) ±2°F from 0 to 530°F

2) ±3/8°F from 530 to 700°F

3) Upper limit - +2300°F

# III. LESSON OUTLINE:

## NOTES

b. $M_K$ updated monthly and is based on full flow conditions.

NOTE: Keep in mind that under accident conditions, $F_{delta\ H}$ calculations generated by computer using mixing factors are invalid.

Under post-accident conditions, the real parameter of concern is basically the thermocouple reading itself.

C. Thermocouple Indications on Plant Computer

1. Computer programmed for inadequate cooling evaluations*

*Presently Plant Vogtle SPDS system uses all T/C for monitoring ICC.

a. 5 T/C with range to 2200°F

b. Remaining TC range to 1200°F.

2. Computer corrects all thermocouple readings for reference temp. variations.

a. ±10°F variations

b. Accuracy ±5% between 750°F and 2200°F

D. Post Accident Monitoring

1. Computer inoperable

a. Expanded readings can be taken at the in-core display panel in control room

b. Reference temperature readings may be taken locally at Remote Processing Units (RPU) A3 and B3.

1) In case of off normal containment conditions.

2) Tables will be provided for conversions — LO-TP-36102-00-C-017

3) Readings taken with millivolt potentiometer

QUESTION NUMBER: 3.09(2)

STATEMENT: P-6 setpoint is $10^{-10}$ amps. Change answer to reflect correct setpoint.

REFERENCES: LO-LP-28102-01

# III. LESSON OUTLINE:

## NOTES

b) Will block automatic reinitiation of SI once reset, until P-4 signal is removed.

5) Arms Steam Dumps if C-9 is present (TR A)

6) Shifts steam dumps from load rejection to plant trip controller (TR B)

2. P-6, Source Range Block Permissive

a. Setpoint: $1 \times 10^{-10}$ ICA on any 1/2 IR NIS detectors (NR-35, NR-36)

b. Resets at $5 \times 10^{-11}$

c. Function:

1) Allows operator to manually block SR High Flux trip and de-energize SR High Voltage (both TR A and TR B switches, QMCB)

2) Loss of P-6 (either train) will automatically unblock trip

d. Permissive Status Light on BPLP (QMCB-C) comes on when P-6 is present

3. P-7, at power trip permissive

a. Setpoints - either of the following

1) greater than 10% turbine power (p-13) or — ½ coincidence

2) greater than 10% power on 2/4 PR NIS (P-10)

b. Functions: unblocks at power trips

1) Pressurizer Low Pressure

2) Pressurizer Hi Level

3) RCP Undervoltage

4) RCP Underfrequency

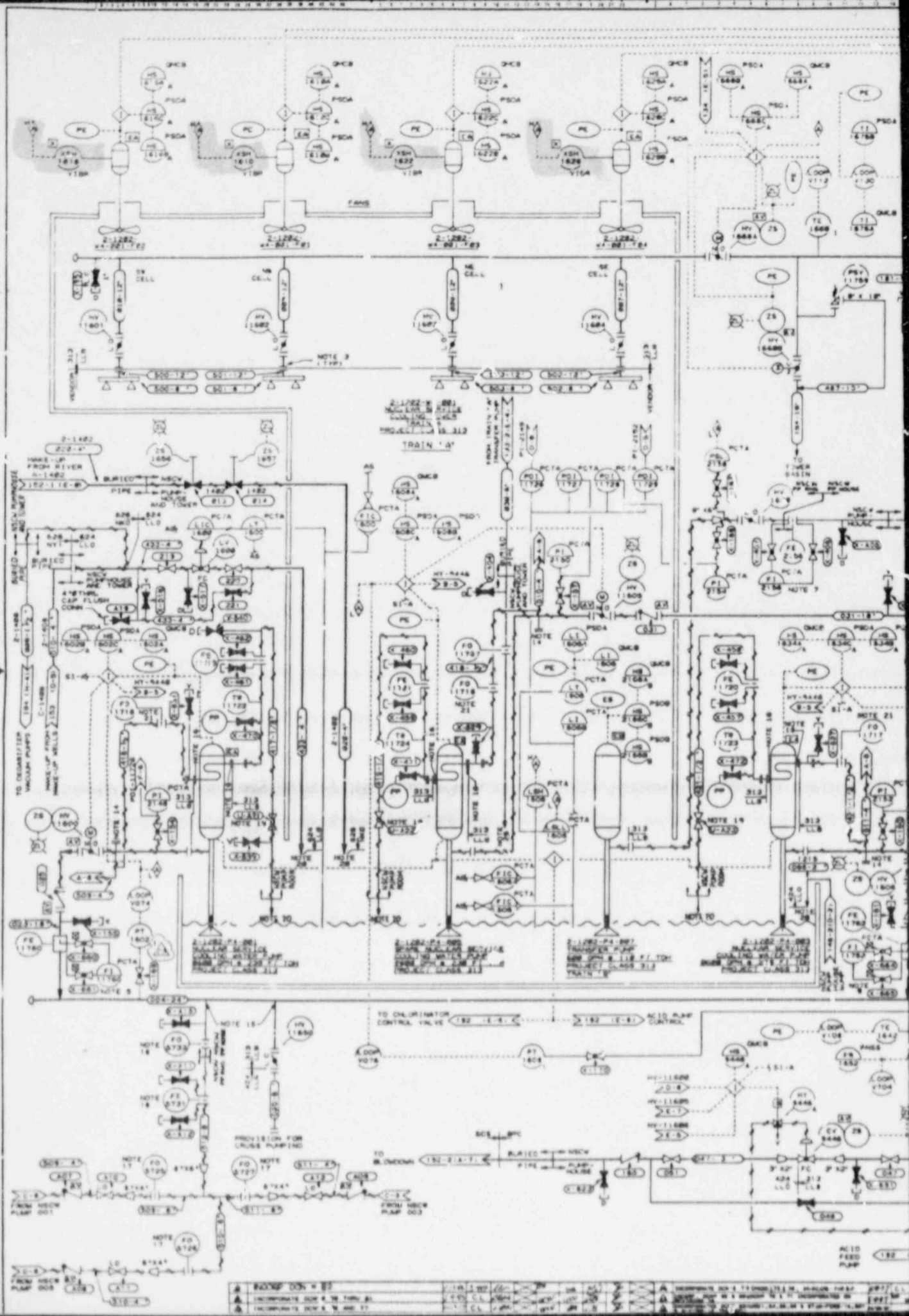
5) Two loop loss of flow trip

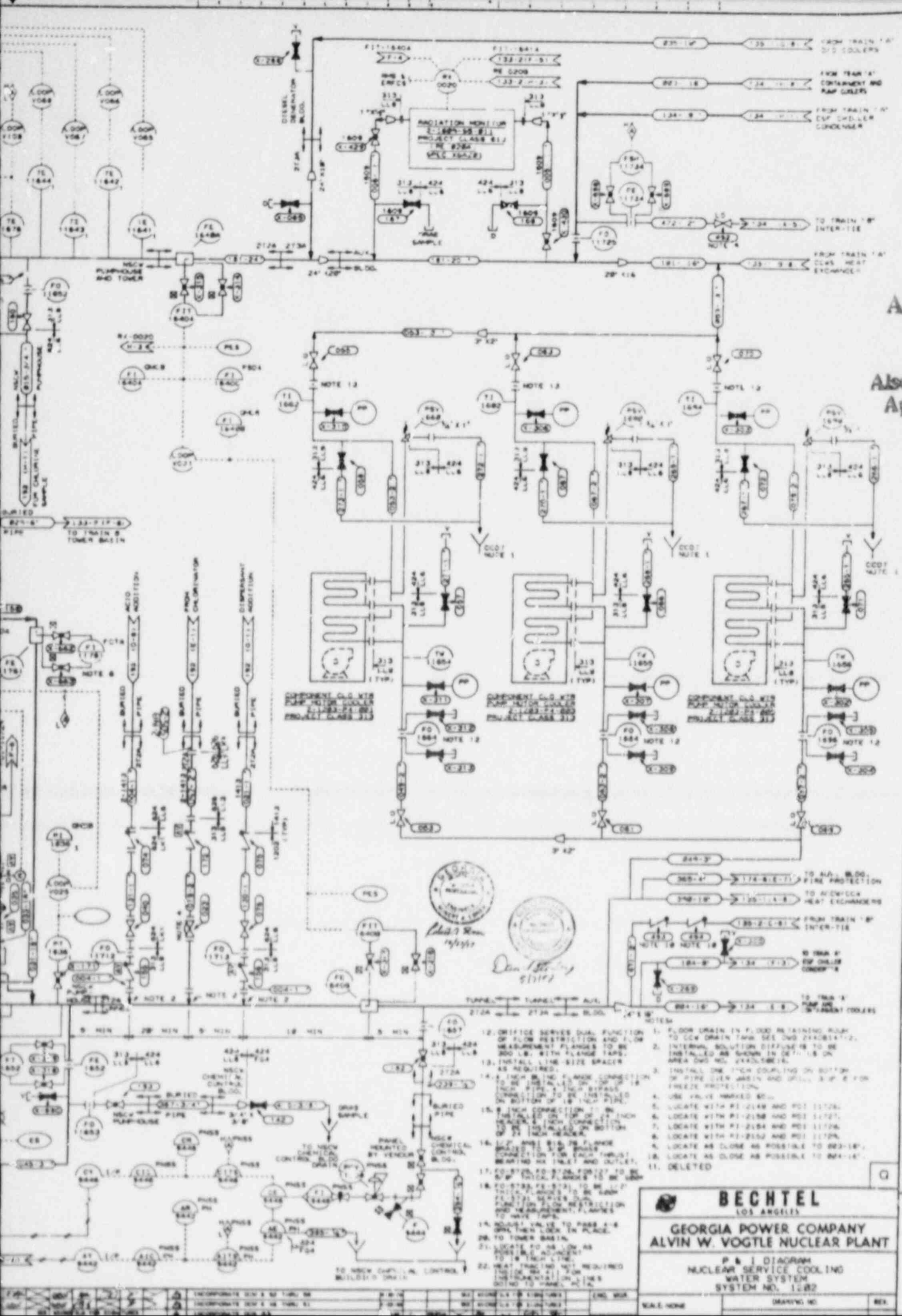
c. Permissive Status Light on BPLP goes out when P-7 is present

QUESTION NUMBER: 3.21

STATEMENT: The high vibration function is actually an alarm function only and applies to Units 1 & 2. The change that was implemented on Unit 2 was to place a 30 sec time delay on the vibration alarm on fan startup. By substituting alarm for trip in questions 3.21 (a) & (c) they remain valid. Since the alarm applies to Units 1 & 2, recommend that 3.21 (b) be deleted and point values be adjusted on 3.21 (a) & (c).

REFERENCES:
1X3D-BD-K03A
2X3D-BD-K03A
1X4DB133-1
2X4DB133-1
RQ-LP-60301-00-c


FANS
TRAIN 'A'
TRAIN 'B'
TO CHLORINATOR CONTROL VALVE
ACID PUMP CONTROL
PROVISION FOR CROSS PUMPING
ACID FEED PUMP



BECHTEL
LOS ANGELES

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
NUCLEAR SERVICE COOLING WATER SYSTEM
SYSTEM NO. 1208

SCALE: NONE

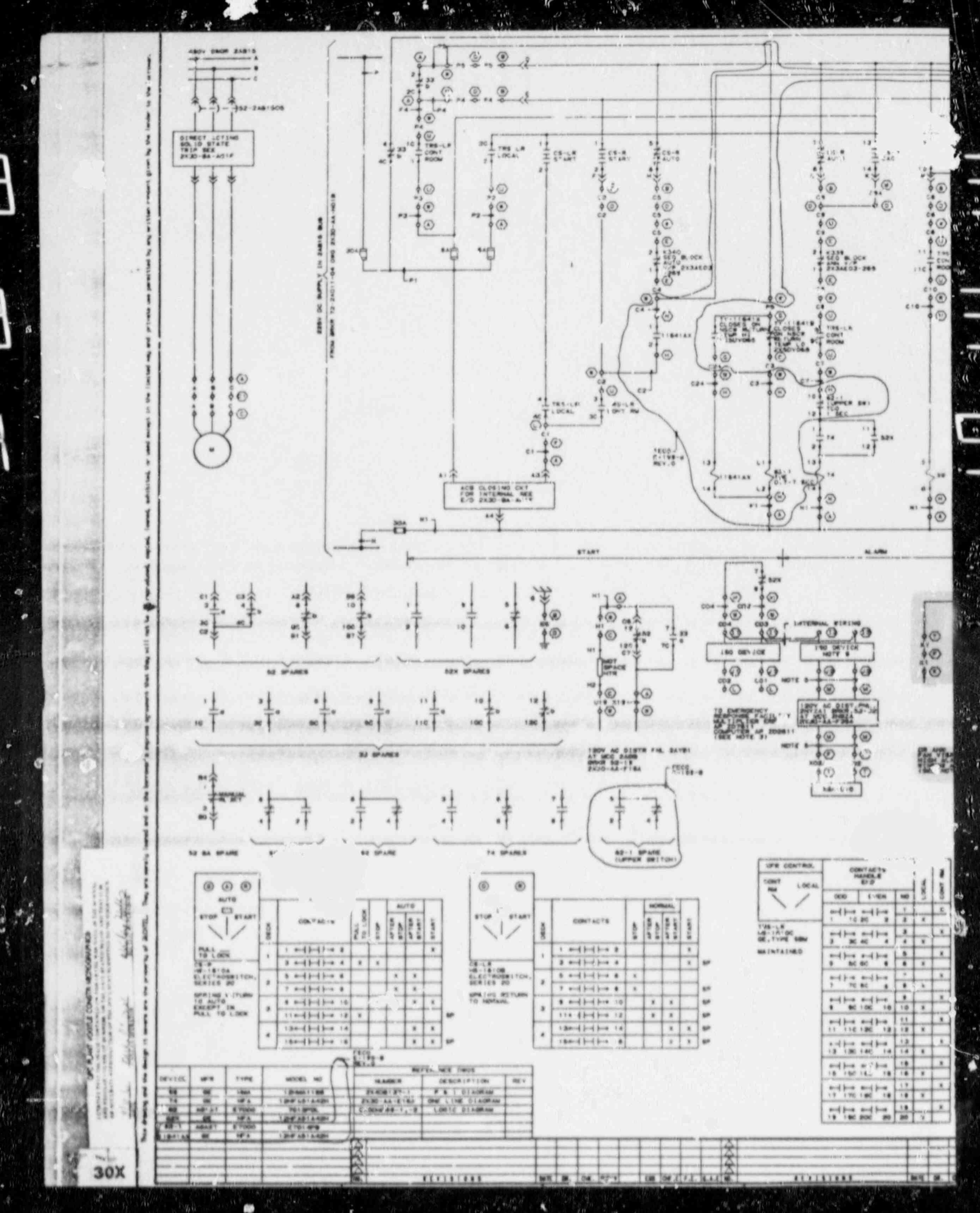
DRAWING NO.

REV.

SI APERTURE CARD

Also Available On Aperture Card

8812090145-03

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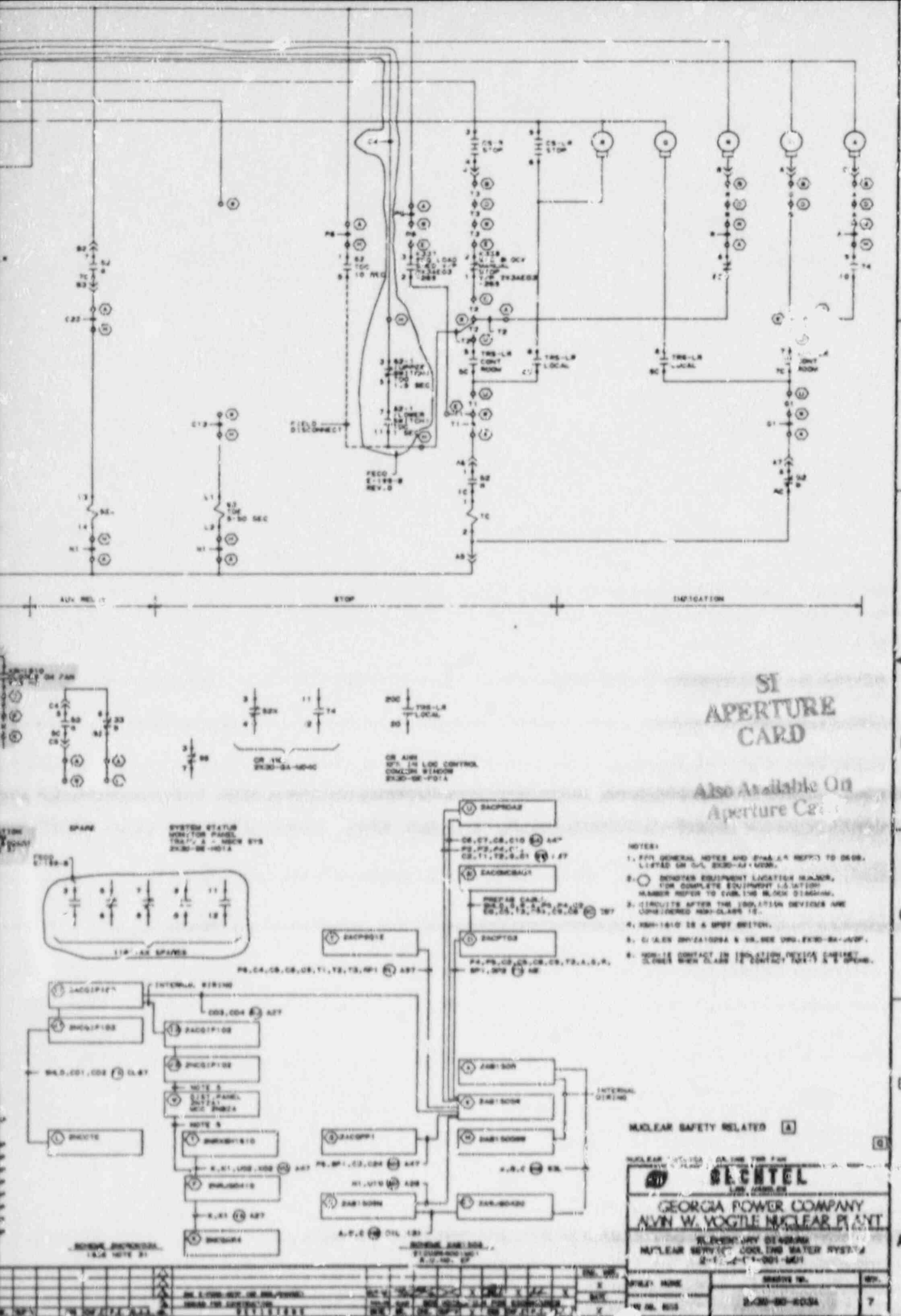



SI
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CARD

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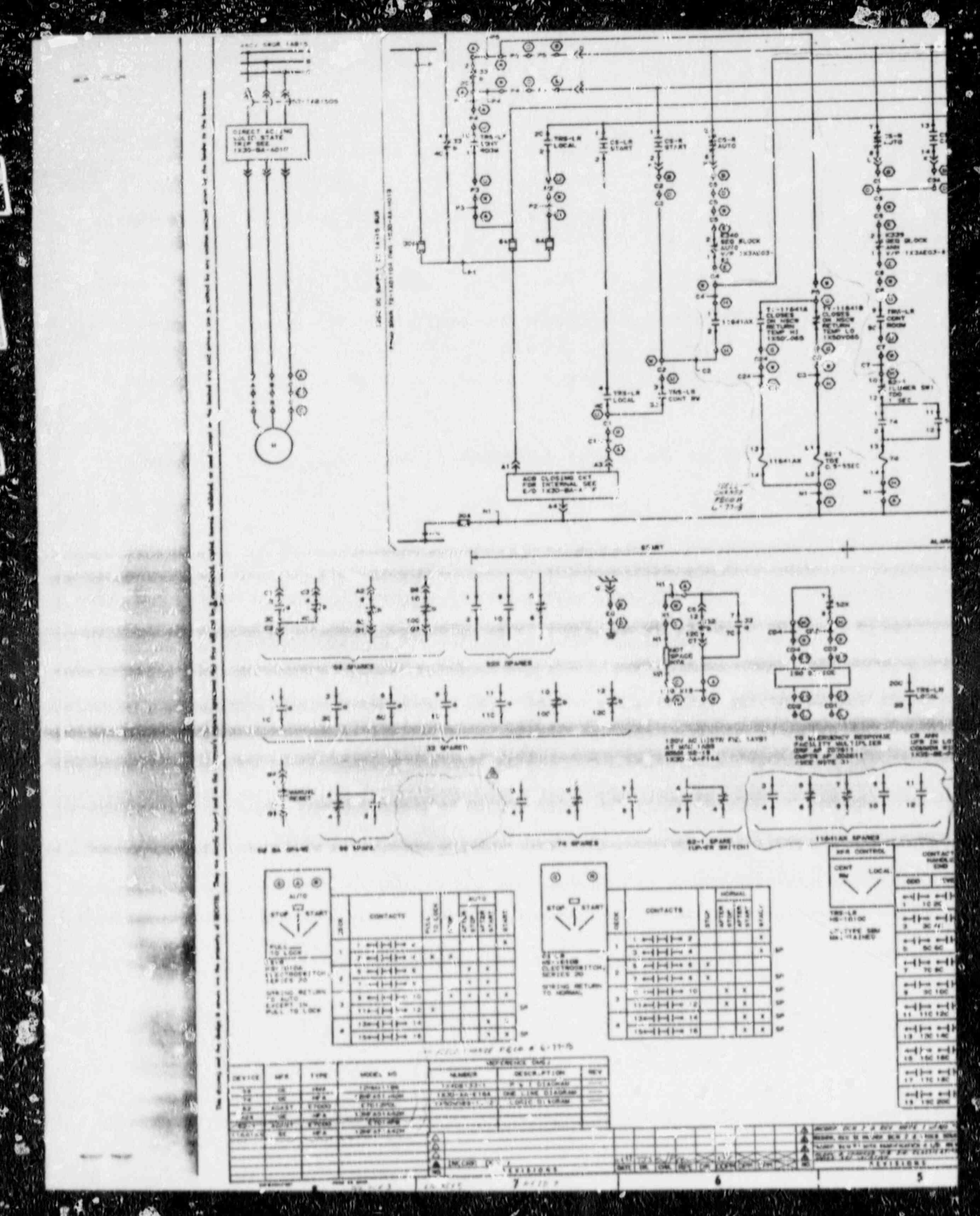

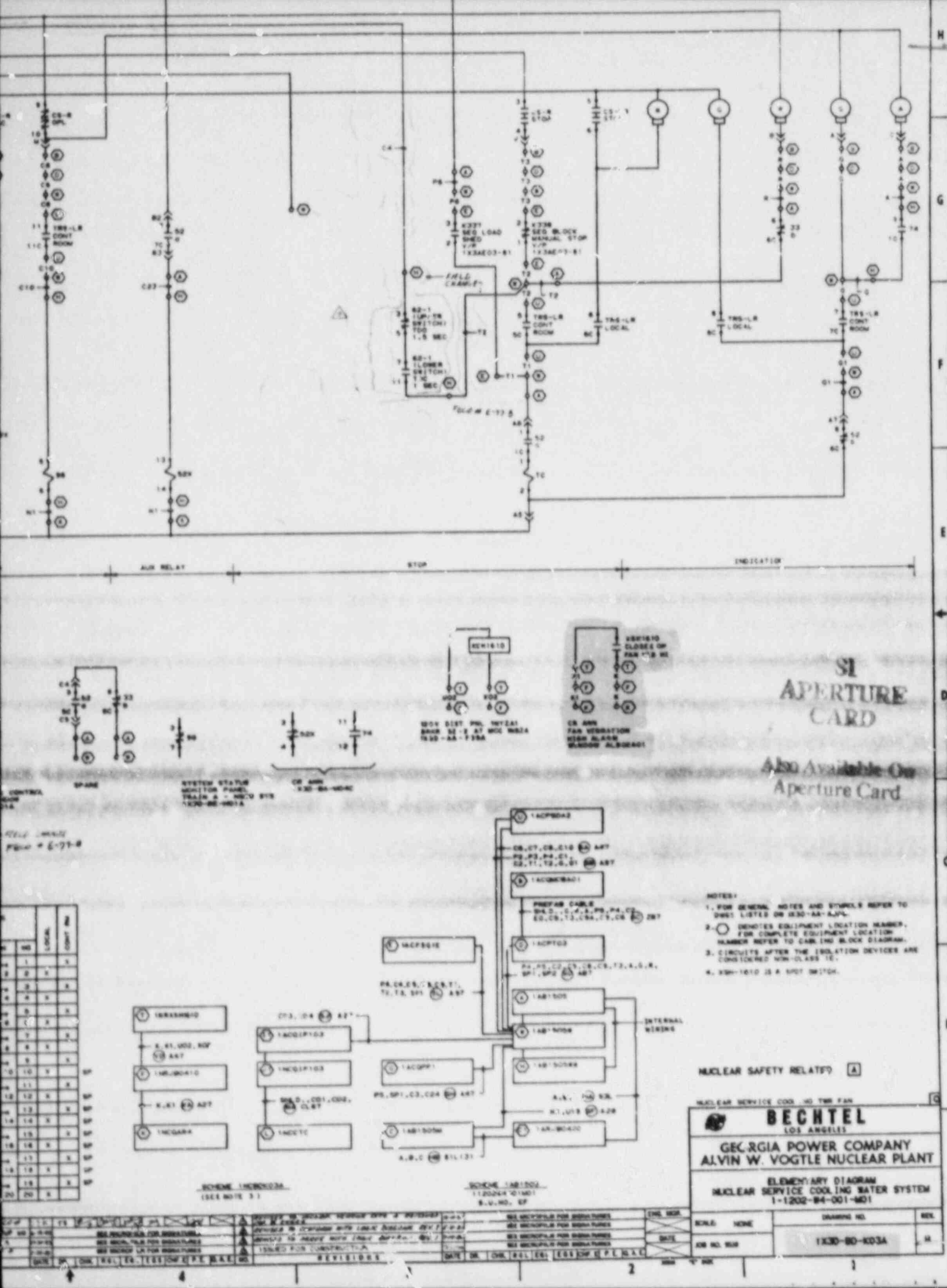


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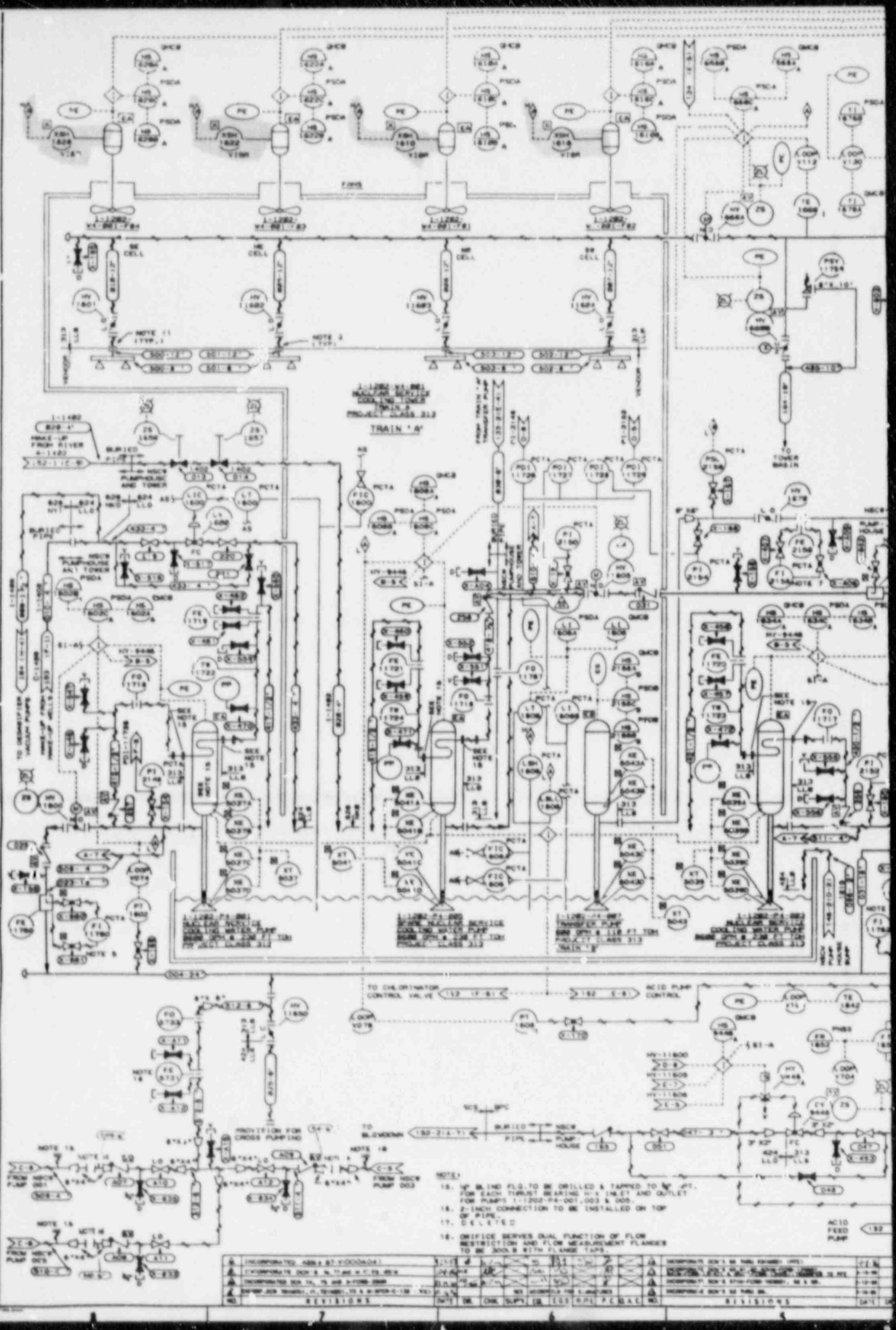
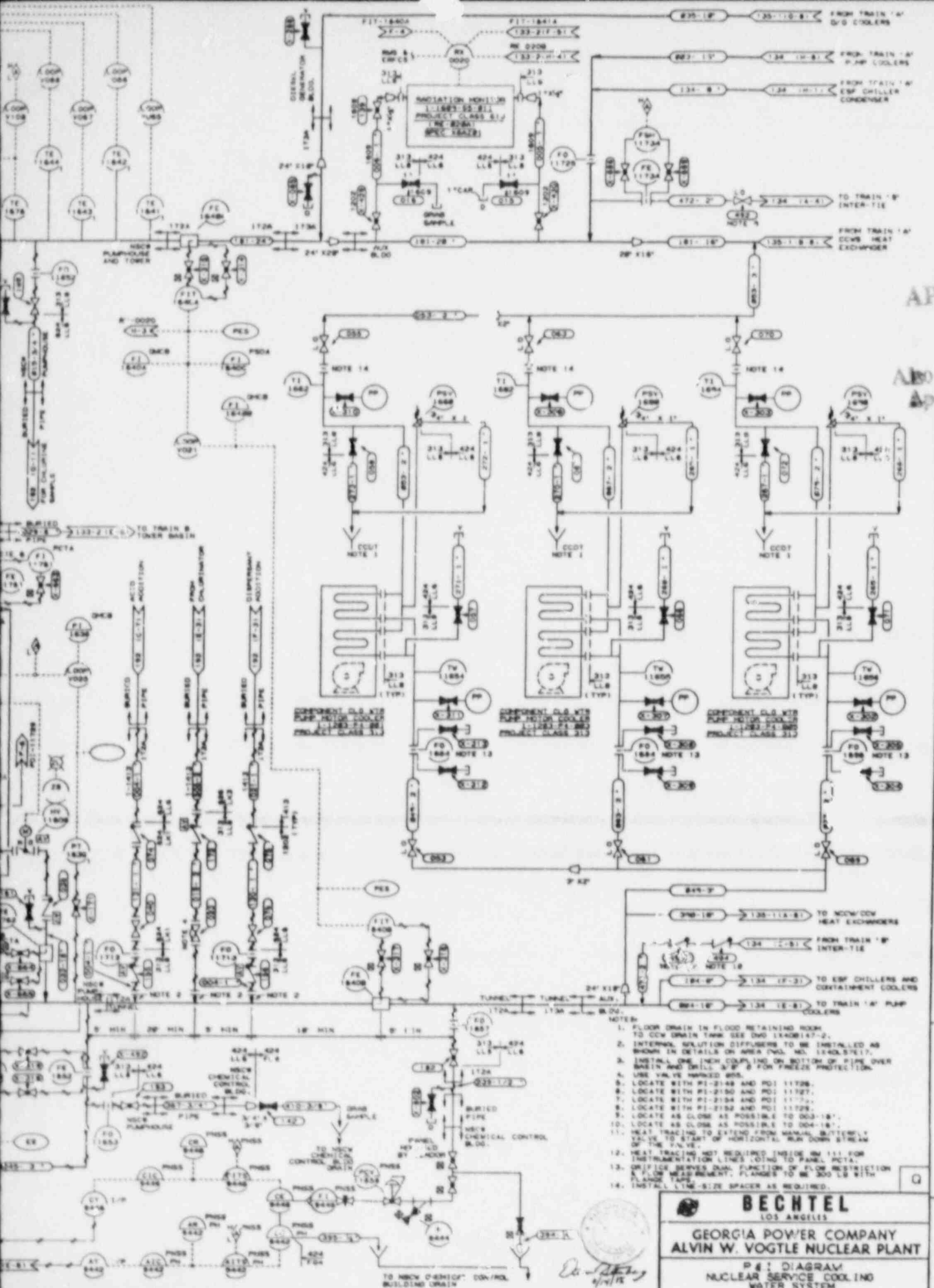

TRAIN 'A'
NUCLEAR SERVICE COOLING TOWER TRAIN 'A' PROJECT CLASS 313
NUCLEAR SERVICE COOLING WATER PUMP
SPARE NUCLEAR SERVICE COOLING WATER PUMP
TRANSFER PUMP
NUCLEAR SERVICE COOLING WATER PUMP
PROJECT CLASS 313
TO CHLORINATOR CONTROL VALVE
ACID PUMP CONTROL
PROVISION FOR CROSS PUMPING
TO BLOWDOWN
ACID FEED PUMP
TO TOWER BASIN
NOTE:
15. 1/2" BLIND FLG. TO BE DRILLED & TAPPED TO 3/4" NPT. FOR EACH THRUST BEARING H-X INLET AND OUTLET FOR PUMPS 1-1202-P4-001, 003 & 005.
16. 2-INCH CONNECTION TO BE INSTALLED ON TOP OF PIPE.
17. DELETED
18. ORIFICE SERVES DUAL FUNCTION OF FLOW RESTRICTION AND FLOW MEASUREMENT FLANGES TO BE 300LB WITH FLANGE TAPS.
REVISIONS
REVISIONS

30X




SI APERTURE CARD
Also Available On Aperture Card

9812090145-06

c. Waste Monitor Tank Capacity

Unit 2 will have 2 additional 20,000 gal tanks designated as common.

1. both units have 2 10,000 gal tanks

2. pinch points of waste water system

3. Add 2 20,000 gal tanks
   a. 2 new pumps
   b. d lvl unit 2 southwest corner
   c. cost > 800k $

4. increase surge capacity

5. being built now SW corner D lvl unit 2

d. NSCW System - Unit 2 Changes

1. NSCW Tower Fan Vibration Reset

Unit 2 will have a 30 sec time delay on the vibration trip for the NSCW tower fans.

a. 30 sec time delay
b. local reset
c. reset at breaker

2. NSCW Aux Containment Cooler Isolation Valves

Unit 2 aux cnmt cooler isolation valves are interlocked so that the outlet valves must be opened first. When opening the outlet valves they will stroke open for 3 sec, wait for 60 sec, and then finish stroking open. This allows the header to backfill before allowing flow through the system.

a. outlet opens first
b. open 3 sec, wait 60 sec, finish stroke
c. then open inlets
d. reduces waterhammer
   1. outlet header lower pressure
   2. slow backfill during 60 sec wait
   3. after test may install on unit 1

QUESTION NUMBER: 4.02

STATEMENT: Although RQ-HO-60301-00-001 specifically states rectangular for Unit 1, the general concept is square vs circular shapes. Accept either rectangular or square for Unit 1.

REFERENCES: N/A

QUESTION NUMBER: 4.06(2)

STATEMENT: The 4500 mrem/yr administrative limit is not achievable unless increases in radiation exposure have been authorized (the quarterly limit is 1000 mrem). Recommend deleting part 2 from the answer key and re-distributing points within the question.

REFERENCES: 00920-C, Pg 8

# TABLE I

## RADIATION EXPOSURE ADMINISTRATIVE LIMITS
(REF: Procedure 00920-C)

| | | | |
|---|---|---|---|
| VISITOR (< 19) | 100 mrem/qtr | Whole Body | |
| PREGNANT | 400 mrem/yr | Whole Body | |
| NO RCA ENTRY | | | |
| RCZ VISITOR (> 19) | 300 mrem/qtr | Whole Body | EXTREMITY = Hands, Forearms (Elbow to Wrist), Lower Legs (Knee to Ankle) Ankles, Feet. |
| RCA VISITOR (> 19) | 1200 mrem/yr | Whole Body | |
| PARALLEL BADGED WORKER | 7500 mrem/qtr | Skin | |
| NRC | 18750 mrem/qtr | Extremity | |
| | 10 MPC/7 days | Inhalation | |
| | 130 MPC/qtr | Inhalation | |
| PRENATAL (NOT PREGNANT) | 500 mrem/qtr | Whole Body | WHOLE BODY = Head trunk, Upper Leg to knee, Upper Arm to Elbow, Active Organs, Gonads, Lens of eye. |
| | 2000 mrem/yr | Whole Body | |
| | 7500 mrem/qtr | Skin | |
| | 18750 mrem/qtr | Extremity | |
| | 10 MPC/7 days | Inhalation | |
| | 130 MPC/qtr | Inhalation | |
| THESE LIMITS CANNOT BE EXTENDED - THE INDIVIDUAL MUST BE ON RADIATION WORKER LIMITS TO RECEIVE MORE EXPOSURE. | | | |
| RADIATION WORKER | 1000 mrem/qtr | Whole Body | MARGIN = (Quarter WB Limit) - (WB Current Exposure) |
| | 4500 mrem/yr | Whole Body | |
| | 7500 mrem/qtr | Skin | |
| | 18750 mrem/qtr | Extremity | |
| | 10 MPC/7 days | Inhalation | "NO ENTRY" MARGIN is 200 mrem - Manual RWP Sign In Required - HP Lab Supv must Authorize Entry. |
| | 130 MPC/qtr | Inhalation | |
| IF PRIOR SITE EXPOSURE IS DOCUMENTED, THESE WHOLE BODY LIMITS MAY BE EXTENDED TO 3000 MREM/QTR OR 5000 MREM/YR USING FORM 00920-1 OR FORM 00920-2. | | | |
| ALL OTHERS | NO OCCUPATIONAL EXPOSURE | | |

### INCREASE WHOLE BODY (WB) ADMINISTRATIVE LIMITS

:: Individual must be on Rad Worker Limits (1000 mrem/qtr. etc.)

UP TO 1500 mrem/qtr & 4500 mrem/yr Auth: HP Foreman/Worker's Foreman
UP TO 2000 mrem/qtr & 4500 mrem/yr Auth: HP Lab Supv/Worker's Supv
UP TO 2500 mrem/qtr & 4500 mrem/yr Auth: HP Supt/Worker's Supt

:: To exceed 2500 mrem/qtr or 4500 mrem/yr, TLD must be read and evaluated

UP TO 3000 mrem/qtr & 4500 mrem/yr Auth: HP Supt/General Manager
4500 mrem/yr & 4999 mrem/yr Auth: HP Supt/General Manager

QUESTION NUMBER: 4.09d

STATEMENT: Answer should be 6.

REFERENCES: 12004-1, Step 4.1.32
(See attached)

INITIALS

4.1.32 At approximately 30% (348 MWe) turbine power the station electrical loads should be transferred from the Reserve Auxiliary Transformers to the Unit Auxiliary Transformers.

a. TRANSFER the 13.8kV busses per 13420, "13.8kV AC Electrical Distribution System", _____

b. TRANSFER the 4160V AC bus 1NA05 per 13425, "4160V AC Non 1E Electrical Distribution System". _____

c. TRANSFER MFPT Speed Control to AUTO.

(1) ENSURE MFPT Master Speed Controller SIC-509A is in AUTO, _____

(2) ADJUST MFPT A(B) Speed Controller SIC-509B(509C) to have the same output value as the Master Speed Controller SIC-509A, _____

(3) PLACE MFPT A(B) Speed Controller SIC-509B(509C) in AUTO. _____

4.1.33 Prior to reaching 50% power (580 MWe), PERFORM the following: _____

a. ENSURE both circulating water pumps running and if necessary START the second circulating water pump per 13724, "Circulating Water System". _____

b. START a second Condensate Pump and PLACE the third in standby, _____

c. START the second Main Feed Pump per 13615, "Condensate And Feedwater System", _____

d. RAISE Turbine-Generator gas pressure to between 70 psig and 75 psig per 13810, "Generator Gas System", _____

QUESTION NUMBER: 4.22b

STATEMENT: Clarification given during exam stating flowrate (gpm) was not required. Recommend deleting point value from flowrate and reassigning.

REFERENCES: N/A

Candidate confused on this issue

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

| | |
|---|---|
| Facility Licensee: | Georgia Power Company |
| Facility Licensee Docket No.: | 50-424 and 50-425 |
| Facility Licensee No.: | NFP-68 |
| Operating Tests administered at: | Vogtle Electric Generating Plant Training Simulator |
| Operating Tests Given On: | September 20-22, 1988 |

During the conduct of the simulator portion of the operating tests identified above, the following apparent performance and/or human factors discrepancies were observed and discussed with the facility's staff instructors.

1. During the conduct of a faulted #3 steam generator (main steam line rupture inside containment) event, #1 steam generator's response in both pressure and level decrease was significantly more rapid than that of #2 and #4 steam generators.

2. Indication (meter) of NI reactor power was inconsistent with the actual power.

3. At steady state operation, turbine load indication varied significantly from load demand indication.

4. The simulator exhibitted a loss of its "ringback" function.

5. During the conduct of a failure of PT 455 (pressurizer pressure transmitter) high, the associated PORV failed open and was overridden to remain open, however, the RO took the switch to "close" and it closed.

6. Several spurious Train A SSMP alarms were exhibited.