ENCLOSURE 3

March 18, 1988

Regional Administrator
Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street
Suite 2900
Atlanta, Georgia 30323

Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
WRITTEN LICENSE EXAMINATION COMMENTS

In accordance with NUREG-1021, Section ES-201, the following comments are submitted concerning the Reactor Operator and Senior Reactor Operator written examinations administered at Surry on March 14, 1988.

REACTOR OPERATOR EXAMINATION

QUESTION 2.11 (b)

List TWO instances when an operator may want to initiate flow through the RHR Letdown Valve (HCV-1142).

ANSWER:

1) To cleanup the coolant and/or for chemistry control when on RHR cooling.

2) For RCS solid plant pressure control.

Reference: Surry ND-88.2-LP-1, pg. 1.15
Surry ND-88.2-LP-2, pg. 2.5
Surry ND-88.2-LP-2, pg. 2.18

COMMENTS:

Add to answer key as alternate acceptable answers:

- RHR System heatup
- Boron concentration equalization

Reference: 1-OP-14.1, pp. 7 and 8

For the caution prior to step 5.19 of OP-14.1, the method to equalize boron concentration is to flow RCS through the RHR System and into the CVCS Letdown System via HCV-1142 during the RHR System heatup.

QUESTION 3.05 (a)

List the FOUR signals that will automatically start the Motor Driven Auxiliary Feedwater Pumps. Include logic, setpoints not required.

ANSWER:

1) S/G Lo-Lo level (any S/G)
2) Both main feed pumps tripped
3) Any SI Signal (after 50 sec. T.D.)
4) Loss of voltage on 2 of 3 RSS Transformer Buses

Reference: Surry ND-89.3-LP-4, pp. 4.5, 4.9, and 4.10

COMMENTS:

Accept as acceptable answer "loss of voltage on 2/2 RSS Transformer Buses for affected unit." This is synonymous with the answer key response of "loss of voltage on 2/3 RSS Transformer Buses."

Reference: Unit 1 Station Electrical Distribution System Drawing

QUESTION 3.16

List the TWO conditions that will AUTOMATICALLY close the CVCS orifice isolation valves (HCV-1200A, B, C) if the selector switch is in the REMOTE position.

ANSWER:

1) Valves will auto close on SI Train A.
2) Valves will auto close if NO Charging Pumps are running.

Reference: Surry ND-88.3-LP-2, pg. 2.18

COMMENTS:

Remove from the answer key the "TRAIN A" requirement of SI as trainees are generally not required to differentiate between Train A or B actions. Also, for future reference, modify the question's reference to the "local/remote" function of the HCV-1200 valves as this capability has been removed.

Reference: ND-91-LP-3, SI Operations Handout ND-91/H-3.8, pg. 2
Operator Training Bulletin #230, referencing EWR-85-551

QUESTION 4.10

For certain LOCA's, it is required to trip the RCP if the trip criteria are met. If forced flow through the core promotes cooling, why are the RCPs tripped?

ANSWER:

Better decay heat removal rate is achieved since stopping the RCPs results in a faster transition to having only steam flowing out the break instead of two phase (water/steam mixture) flow escaping out the break.

Reference: Surry ND-95.3-LP-7, pg. 7.8

COMMENTS:

Replace answer key answer with the following from the trainee lesson plan handout dealing with RCP Trip Criteria:

- The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident.

Reference: ND-95.2-LP-7, OBJ-E, pp. 7.49 - 7.55
Statement from trainee handout ND-95.2/H-7.23, pg. 31

QUESTION 4.13

EP-4.00, "Steam Generator Tube Rupture," directs the operator to adjust the ruptured SG PORV controller setpoint to 1035 psig (pot setting 7.5). Explain why the ruptured SG PORV setpoint is INCREASED to 1035 psig.

ANSWER:

Increasing the SG PORV setpoint above normal provides a method of isolating the ruptured SG while the setpoint is low enough to prevent challenges to the code safety valves.

Reference: Surry ND-95.3-LP-13, pg. 13.71

COMMENTS:

Recommend removing from answer key the statement of "Increasing the SG PORV setpoint above normal." This is because the normal setpoint of the SG PORV is 1035 and therefore, no "increase" of setpoint occurs.

Reference: ND-95.3-LP-13, pg. 13.71

SENIOR REACTOR OPERATOR EXAMINATION

QUESTION 5.13

Select the statement below which most correctly describes the effect of ignoring blowdown flow on a secondary calorimetric.

a) The calculated reactor power will be the SAME as actual reactor power since main steam enthalpy is approximately equal to blowdown fluid enthalpy.

b) The calculated reactor power will be LESS than the actual reactor power since the heat used to increase the enthalpy of the blowdown fluid is not accounted for.

c) The calculated reactor power will be GREATER than the actual reactor power since the enthalpy of the blowdown fluid would be assumed to have increased to the enthalpy of the main steam.

d) The calculated reactor power will be GREATER than the actual reactor power since the heat used to increase the enthalpy of the blowdown fluid is not accounted for.

ANSWER:

c) The calculated reactor power will be GREATER than the actual reactor power since the enthalpy of the blowdown fluid would be assumed to have increased to the enthalpy of the main steam.

Reference: SO ND-93.2-LP-4D
ND-93.2-LP-4, pp. 9 - 18

COMMENTS:

Answers "b" and "c" are both correct responses depending upon which calorimetric is used (i.e., feedwater flow or steam flow).

Reference: ND-93.2-LP-4, pp. 4.15 - 4.17
ND-93.2/T-4.3
ND-93.2/T-4.7

QUESTION 6.05 (b)

State the purpose of the design features of the following components with respect to the consequences of a main steam line break.

b) TDAFW pump steam supply line check valve

ANSWER:

b) Prevents the loss of steam supply to the TDAFW pump (prevents loss of TDAFW pump).

Reference: SO ND-89.1-LP-2C
ND-89.1-LP-2, pp. 5, 7, 8, and 11

COMMENTS:

Add to answer key that it also "prevents blowdown of all three S/Gs in the event of a break in any one."

Reference: ND-89.1-LP-2, pg. 2.7
ND-89.1/T-2.1

QUESTION 6.08

Two signals which cause a turbine runback are high OT Delta T and high OP Delta T signals. State two additional signals which cause a turbine runback. Include any applicable setpoints and coincidences.

ANSWER:

a) NIS rod drop signal
1 of 4 NIS channels, and
5% decrease in Rx power in 2 sec

b) RPI rod bottom signal
2 of 2 impulse pressure channels > 70% turbine load, and
1 of 4 NIS channels indicate a 5% decrease in Rx pwr in 2 sec.
OR
Any rod bottom signal is received from RPI.

Reference: SO ND-89.2-LP-8G
ND-89.2-LP-8, pp. 19 - 22

COMMENTS:

Answer key answer for b) is incorrect. Suggested rewording as follows:

- 1 of 48 IRPIs less than 20 steps from bottom.

Reference: ND-89.2-LP-1, pp. 8.19 - 8.21
ND-93.3-LP-4, pg. 4.7

QUESTION 6.11 (c)

Consider the following plant evolutions separately with the Steam Dumps aligned normally for the specified condition. State the Steam Dump AUTOMATIC RESPONSE(s) (i.e., arm, actuate, or none), and state the final plant conditions (i.e., temperature and pressure values) specifying the reference parameter used for controller input (e.g., auctioneered Tave).

c) Plant Tave at 550F, NO LOAD conditions.

ANSWER:

c) Steam dumps are armed in the Steam Pressure mode and will actuate.

Steam pressure will be controlled at the normal controller setpoint of 1005 psig.

Reference: SO ND-93.3-LP-9E, F
ND-93.3-LP-9, pp. 11 - 16
ND-93.3-LP-1, pg. 6, T-1.2

COMMENTS:

Answer key refers to the Controller setpoint of 1005. The automatic function of this controller has been removed and is now manual only.

Reference: ND-93.3-LP-9, pp. 9.10 and 9.15

QUESTION 6.15 (a)

For the following NIS permissive signals, state the SOURCE, SETPOINT, LOGIC (i.e., 2 of 4), and FUNCTIONS (blocks) for each. List THREE functions for b.

a) P-8 permissive

ANSWER:

a) Source - Power Range NI
Setpoint - ' 35%
Logic - 2/4
Function - automatically blocks RCS low flow trip (2/3 loops)

Reference: SO ND-93.2-LP-4C, L
ND-93.2-LP-4, pg. 7
ND-93.2-LP-3, pp. 8 and 11

COMMENTS:

Modify answer key function to read:

- "automatically blocks RCS low flow (or RCP breakers open) trip (2/3 loops)."

Reference: ND-93.3/H-11.1

QUESTION 7.07

Unit 1 is operating at 100% power. You are the shift supervisor on shift and you observe the following uncontrolled and unexplained symptoms:

a) Excessive makeup
b) Pressurizer level decreasing
c) Pressurizer pressure decreasing
d) Containment pressure increasing

WHAT are FIVE required immediate actions?

ANSWER:

1) Isolate letdown
2) Control charging flow to maintain pzr level
3) Verify adequate charging/SI pump suction flow
4) Stop containment sump pumps
5) Check if SI is not required

Reference: AP-16, pp. 1 and 2

COMMENTS:

Add to answer key as one of the immediate actions:

- Verify leak greater than 25 gpm.

Reference: Revision 0.01 to AP-16

QUESTION 7.16

In accordance with the foldout page for EP-2 series procedures, STATE the SI Reinitiation Criteria. (Assume no adverse containment)

ANSWER:

Manually operate SI pumps as necessary if EITHER condition listed below occurs:

a) RCS subcooling based on core exit TCs - less than 30F.

b) Przr level - cannot be maintained greater than 13%.

Reference: SO ND-95.3-LP-8B
ND-95.3-LP-8, pp. 5, 8, and 19

COMMENTS:

Question asks for the "criteria" for SI Reinitiation and not how the reinitiation is performed. Recommend removal from answer key of "manually operate SI pumps as necessary."

Reference: Foldout Page for EP-2 Series Procedures

QUESTION 8.05 (c)

Who authorizes temporary procedure changes to operating procedures that change the intent of an approved procedure?

ANSWER:

Superintendent of Operations

Reference: SO ND-95.5-LP-2D, I
ND-95.5-LP-2, pp. 2 and 11
SO ND-100-LP-2C
ND-100-LP-8, pp. 5 - 7
SUADM-ADM-02, pg. 5
SUADM-ADM-21, pg. 21
SUADM-0-12, pp. 4 and 5, Appendix A
SEP, pg. 5.7

COMMENTS:

Modify answer key to read:

- Superintendent of Operations with concurrence of SNSCC.

Reference: Memorandum from D. Benson (Station Manager) to all Supervisors, January 26, 1988 (Subject: Procedure Deviations)

QUESTION 8.11

As stated in the Technical Specifications, WHAT are the THREE (3) conditions in which a control rod would be considered INOPERABLE?

ANSWER:

1) Rod cannot be moved by the CRDM.
2) Rod misaligned from its bank by more than 12 steps.
3) Rod exceeds its rod drop time limit of 1.8 seconds.

Reference: SO ND-93.3-LP-3M
ND-93.3-LP-3, pg. 31

COMMENTS:

Modify answer key answer number 3 to be 2.4 seconds versus 1.8 seconds

Reference: Tech Spec Change #116, pg. 3.12-8

Very truly yours,

D. L. Benson

D. L. Benson
Station Manager

H. F. McCallum
Supervisor, Training - PSO

Enclosure

cc (w/o enclosure):

Mr. W. E. Holland
NRC Senior Resident Inspector
Surry Power Station

bc (w/o enclosure):

Mr. W. L. Stewart - OJRP5
Mr. J. L. Wilson - OJRP5
Ms. N. E. Hardwick - OJRP5
Mr. G. L. Pannell - OJRP5
Mr. D. A. Sommers - OJRP5
Mr. H. L. Miller - SPS
Dr. T. M. Williams - RP1E
Superintendent Nuclear Training - Surry
GOV 02-54B
NOD Tech. Library (bc original)

TIALS

5.0 Procedure [continued]

PLACING THE RHR SYSTEM IN SERVICE [continued]

RHR SYSTEM HEAT-UP

NOTE: 1. Heat-up is accomplished by slowly flowing coolant from the RCS through the RHR System to the letdown line, controlling the flow rate with HCV-1142. An increase in flow of approximately 20 gpm on (FI-1-150) after opening MOV-1700 and 1701 will give a controlled heat-up.

2. There is a delay of a number of minutes before the heat-up will be seen on the recorder "RHR TEMP" (TR-1-604).

3. Minor adjustments to HCV-1758 and/or HCV-1142 may be required for precise heat-up control.

______ 5.11 Monitor letdown line flow and pressure and simultaneously open MOV-1700 and 1701. (check)

MOV-1700 ______ MOV-1701 ______

______ 5.12 Set HCV-1142 to control the letdown line flow at approximately 20 gpm greater than the indicated flow in step 5.10.2.

______ 5.13 Test MOV-1700 and 1701 IAW 1-PT-30.2.

______ 5.14 MOV-1700 is open.

______ 5.15 MOV-1701 is open.

______ 5.16 Lock the breaker open for MOV-1700.

______ VERIFIED

______ 5.17 Lock the breaker open for MOV-1701.

______ VERIFIED

______ 5.18 Control the heat-up rate $\leq$ 150°F/hr as indicated on (TR-1-604).

_TIALS

5.0 Procedure [continued]

PLACING THE RHR SYSTEM IN SERVICE [continued]

RHR FLOW TO RCS

CAUTION: THF RHR SYSTEM C_B MUST BE $\geq$ THE RCS C_B PRIOR TO OPENING MOV-1720A OR 1720B.

_____ 5.19 When the RHR and RCS temperatures are near equal or "RHR HX OUTLET TEMP" (TR-2-604) has reached its maximum attainable value:

_____ 5.19.1 Sample both systems for C_B.

RHR C_B ________ ppm RCS C_B ________ ppm

CAUTION: ENSURE SUFFICIENT SW FLOW THROUGH THE CC HEAT EXCHANGERS PRIOR TO OPENING EITHER MOV-1720A OR 1720B.

_____ 5.20 Set the controller for FCV-1605 at 30% open in "MAN".

_____ 5.21 Monitor the "RHR HX BYP FLOW" (FI-1-605) and simultaneously open MOV-1720A and 1720B. (check)

MOV-1720A ________ MOV-1720B ________

_____ 5.22 Place FCV-1605 in "AUTO" (FI-1-605 approximately 4000 gpm).

_____ 5.23 Monitor a stable temperature < 120°F on "RHR HX A CC OUTLT HDR A" (TI-CC-109A).

_____ 5.24 Slowly open HCV-1142 to 100%.

_____ 5.25 Adjust the letdown pressure (PCV-1145 in "AUTO") to establish maximum letdown purification flow.

_____ 5.26 Using HCV-1758 continue to maintain RCS temperature at approximately 345°F.

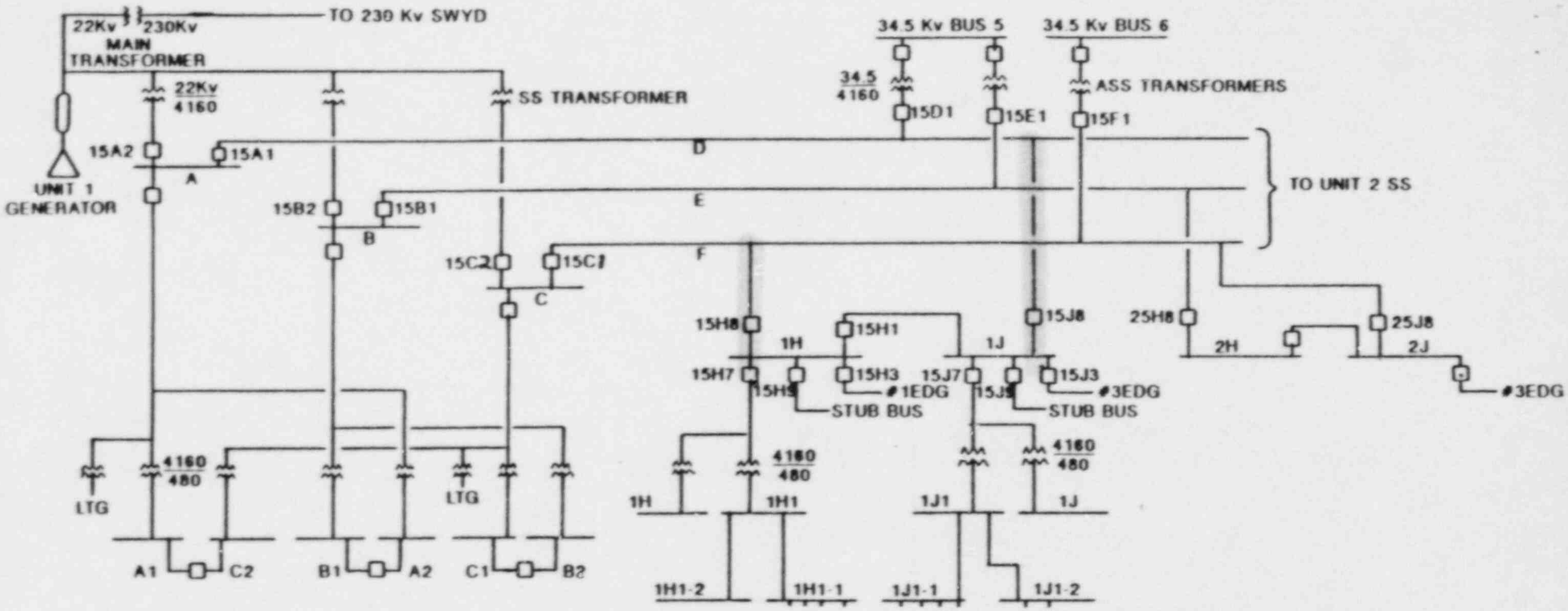
_____ 5.27 "CC RETURN HDR TEMP, HDR A" (TI-CC-110A) < 120°F.

_____ 5.28 Test MOV-1720A and 1720B IAW 1-PT-30.2.

_____ 5.29 MOV-1720A is open.

_____ 5.30 MOV-1720B is open.

QUESTION 3.05a



UNIT 1 STATION
ELECTRICAL DISTRIBUTION
SYSTEM

APPLICABLE PORTION
OF SPS-2A PRINT

Question 3.14

ACTIONS ON SI INITIATION

1. Reactor Trip
2. Turbine Trip
3. EDGs start - Do not come on the bus unless UV condition exists on the emergency bus.
4. Closes Main Feed Reg. Valves and Bypass Valves and Trips MFPs.
5. Starts Motor-driven Auxiliary Feed Pumps after 50 second time delay.
6. Starts LHSI Pumps.
7. Starts HHSI Pumps.
8. Auxiliary Feed MOVs receive "open" signal for 45 seconds.
9. Containment Vacuum Pumps Trip.
10. Control Room Ventilation is isolated and Aux. Vent System re-aligns as necessary.
11. Energizes H_2 Analyzer Heat Tracing (if SI signal remains in for 8 minutes).
12. The following valves receive signals to open:

MOV-867C	-	Old Boron Injection Tank Outlet	Lineup HHSI to T_c.
MOV-867D	-	Old Boron Injection Tank Outlet	
MOV-862A	-	RWST to Lo Hd SI Pump Suction N/O	Insure LHSI lined up.
MOV-862B	-	RWST to Lo Hd SI Pump Suction N/O	
MOV-115B	-	RWST to Chg. Pump Suction	Line up RWST to HHSI pps.
MOV-115D	-	RWST to Chg. Pump Suction	
MOV-865A	-	Accumulator Discharge (1A) N/O	Insure ACCs lined up.
MOV-865B	-	Accumulator Discharge (1B) N/O	
MOV-865C	-	Accumulator Discharge (1C) N/O	

13. The following valves receive a signal to close:

MOV-115C	-	VCT to Chg. Pump Suction
MOV-115E	-	VCT to Chg. Pump Suction
MOV-289A	-	Normal Charging Header
MOV-289B	-	Normal Charging Header
MOV-381	-	Seal Water Return
HCV-200A	-	Letdown Orifice Isolation
HCV-200B	-	Letdown Orifice Isolation
HCV-200C	-	Letdown Orifice Isolation
TV-SI-101A	-	Accumulator N_2 Relief Line
TV-SI-101B	-	Accumulator N_2 Relief Line
TV-SI-100	-	Accumulator N_2 Supply Line
TV-VG-109A	-	Primary Drain Xfer Tank Vent
TV-VG-109B	-	Primary Drain Xfer Tank Vent
TV-VG-108A	-	Primary Drain Xfer Pump Disch.
TV-VG-108B	-	Primary Drain Xfer Pump Disch.
TV-CC-109A	-	Component Cooling From RHRS
TV-CC-109B	-	Component Cooling From RHRS
TV-SS-100A	-	Pressurizer Liquid Sample
TV-SS-100B	-	Pressurizer Liquid Sample
TV-SS-101A	-	Pressurizer Vapor Sample
TV-SS-101B	-	Pressurizer Vapor Sample
TV-SS-103	-	RHRS Sample
TV-SS-106A	-	Reactor Coolant Hot Leg Sample
TV-SS-106B	-	Reactor Coolant Hot Leg Sample
TV-SS-102A	-	Reactor Coolant Cold Leg Sample
TV-SS-102B	-	Reactor Coolant Cold Leg Sample

Question 3.16

No 87024070

MEMORANDUM

TO Operations Department October 14, 1986

FROM D. A. Christian Surry Power Station

OTB #230

EWR'S VARIOUS

EWR 85-551

EWR 85-551 is the removal of local/remote switches for SOV's 1200 A, B, and C, and 2200 A, B, and C. These switches are located in containment and are not environmentally qualified. During the Fall, 1986 outage, Unit 2 containment work will be done. The EWR will be completed after the outage by removing associated wiring outside the containments.

EWR 86-235D

Earlier EWR's modified 1-MS-HCV-104 and 2-MS-HCV-104 resulting in their actuators being increased from Fisher Controls size 45 to size 70. This EWR directs the adjustment of existing spring hangers and the installation of new spring cans to account for the heavier actuators.

EWR 86-260

This EWR directs the replacement of failed RTD TE-2432D, TC protection in the "C" loop.

EWR 86-264

Presently installed, Unit 2 Steam Generator wide range local level indicators are not environmentally qualified. It is possible a failure of the local indicators could feed back and disrupt control room (remote) indications. Therefore, local steam generator wide range level indicators will be disconnected. This allows the steam generator wide range level transmitter loops to comply with Regulatory Guide 1.97.

EWR 86-325

EWR 86-325 directs an environmental qualification inspection of Unit 2 MOV's. Based on the results of the inspection/walkdown corrective action will be taken as required.

DAC

D. A. Christian

DAC/MW/dhc

QUESTION 4.10

Prolonged RCP operation and the resultant additional liquid mass depletion can greatly affect the degree and duration of core uncovery. Depending on plant type and break size, a range of RCP trip times may yield PCTs greater than the FSAR case result. The effect of RCP trip time on calculated PCTs is illustrated on handout H-7.22.

If RCPs remain operational throughout the transient (Case H on figure 1) depletion of primary liquid mass is maximized. Nevertheless, PCTs remain well below FSAR case results due to enhanced core cooling caused by the high core steam flow rates indicative of RCP operation. However, continuous operation of the RCPs during a LOCA cannot be guaranteed since tripping of the RCPs would occur upon a loss of off-site power or other essential support conditions which can be postulated to occur at any time. **The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident.** The RCPs should be tripped before RCS liquid inventory is depleted to the point where tripping of the pumps would cause the break to immediately uncover.

In most non-LOCA accidents, it is advantageous to have the RCPs in operation. This provides either additional margin to safety criteria limits or makes operator actions during recovery easier. However, whether or not the RCPs remain in operation or are tripped, safety criteria must be met and plant operators are provided with guidance to mitigate and to recover from the accident. For accidents involving loss of secondary coolant, control of RCS pressure, RCS temperature, and pressurizer level are the major concerns, rather than core cooling. For the various types of SGTR events, (either single or multiple ruptures) control of the leak rate, RCS pressure, RCS temperature, and pressurizer level are important. In all cases, RCP operation provides enhanced core heat removal and makes RCS pressure control by the operator a more straight forward matter. In general, for non-LOCA accidents, it is desirable to have the RCPs in operation throughout the event.

QUESTION 4.18

The fact that the S/G may overfill due to leakage, thus making a water hammer likely if steamed; Increased threat of radioactive release to the public.

-For Faulted S/Gs

The size and location of the break must be considered. The break, if in containment, may cause instrument errors, or reliability problems. If located outside containment, personnel hazards may be the paramount concern.

-In both cases if the break is large enough to cause a challenge to the Integrity orange or red path, it should not be used for cooldown.

Summarize the method of ruptured S/G isolation given in the text of EP-4.00.

- Adjust PORV to 1035#,
- Close ruptured S/G MSTV & bypass valve,
- Isolate affected steam supply to TDAFW pump,
- Verify affected S/G BD TVs closed
- Verify feed isolated after level reaches 9% NR.

Explain why the ruptured S/G PORV is set to a pot setting of 7.3.

The PORV should be set up to open at a value above the normal operating pressure but before the first safety valve setpoint is reached. The pot setting of 7.3 corresponds to a pressure setpoint of 1035 psig. This allows the PORV to open and prevent challenges to the code safeties which begin opening at 1085 psig.

QUESTION 5.13

WRITE on chalkboard:

h_s steam enthalpy

$-$ h_{bd} blowdown enthalpy

$=$ delta h_2

(2) The enthalpy of the blowdown is determined by taking the data point for steam pressure and using the enthalpy for saturated liquid curve in the curve book. This calorimetric is based on feed flow. Not all of the feed flow is converted to steam; some of it is lost to blowdown. For this reason, the h_{bd} is subtracted from h_s.

WRITE on chalkboard:

$Q_{S/G} = m \times \text{delta } h_1 - m \times \text{delta } h_2$

(3) The heat input for each loop is calculated by calculating the mass flow rate of feedwater, multiplied by the delta h_1 and subtracting the quantity of the blowdown mass flow rate multiplied by delta h_2.

The m_{bd} is calculated by taking the B/D meter indication (gpm), and through a series of conversion factors, converting it to Lbm/Hr.

(4) This solves for the heat transferred in the S/G for one loop. This calculation is performed for each loop and the Q for all the loops are added together.

DISPLAY ND-93.2/T-4.4 and 4.5.

WRITE on chalkboard:

$$Q_{sec\ total} = Q_{S/G\ A} + Q_{S/G\ B} + Q_{S/G\ C}$$
$$Q_{rx} = (Q_{S/G\ A} + Q_{S/G\ B} + Q_{S/G\ C}) - \text{Pzr heat} - \text{RCP heat}$$

DISPLAY ND-93.2/T-4.6.

(5) The pressurizer heat input is converted from Kw to BTU/hr by multiplying it by 3413.

The RCP heat input as shown on this sheet is 36 x 10^6 BTU/hr.

The reactor heat output is now converted to MW_{th} by dividing Q_t by 3413000.

The reactor % power is calculated by dividing the MW_{th} value by 2441 (licensed power limit) and multiplying that by 100.

8. Steam Flow Program - PT-35.2

a. The steam flow program is performed the same as the feed flow program except steam flow is used as the multiplier of delta h instead of feed flow.

b. The only other major difference is in the blowdown calculation. Since steam flow is used for the calculation, the blowdown flow that is removed from the S/G is not sensed in the total S/G flow. (It was in the feed flow calculation.)

WRITE on chalkboard:

h_{bd} blowdown enthalpy - h_{fw} feedwater enthalpy = delta h_2

DISPLAY ND-93.2/T-4.7.

Since blowdown flow is not taken into account in the steam flow, the heat added to the blowdown must be added to the delta h calculated across the S/G.

c. Other than the differences stated above, the steam flow calorimetric is performed the same as the feed flow calorimetric.

9. Feed Flow Calorimetric - Manual Data Collection - PT-35.1

   a. This method of performing a calorimetric is used when the computer is unavailable.

   b. The methodology of performing the calorimetric is the same as for feed flow with the computer; just the method of obtaining the data is different.

      (1) S/G pressures and blowdown flow are obtained from meters in the control room as specified in Appendix B of PT-35.1.

      (2) Feedwater temperature is obtained from an RTD in each feedwater line downstream of the FRV as specified in PT-35.1.

      (3) Feedwater differential pressure is obtained from the feed flow nozzle delta p gage (Barton) every five minutes.

'A' LOOP CALCULATIONS

FEED FLOW CALCULATION

DATA SHEET A

S/G Pressure (PSIG)	+	Line Loss and PSIA Conversion	=	Corrected S/G Pressure (PSIA)	→	Enthalpy Steam H_s (BTU/lbm)
		27.0				

Feedwater Temperature °F	→	Enthalpy FW H_f (BTU/lbm)

$\Delta H_1 = H_s - H_f$ [BTU/lbm]	x	Feedwater Flow M_F (lbm/Hr)	=	$M_f x \Delta H_1$ [BTU/HR]

Blowdown Flow (GPM)

x

Specific Density lbm/ft^3		Enthalpy Steam H_s (BTU/lbm)

x

Conversion Factor	-	Enthalpy BD H_{BD} (BTU/lbm)
8.022		

=

Blowdown Flow [lbm/HR]	x	$\Delta H_2 = H_s - H_{BD}$ [BTU/lbm]	=	$M_{BD} \times \Delta H_2$ (BTU/HR)

$Q_{LA} = (M_F x \Delta H_1) - (M_{BD} x \Delta H_2)$ (BTU/HR)

Completed by: ______________

Date:

STEAM FLOW CALCULATION

JUN 17 1986

A LOOP CALCULATIONS

S/C Pressure (psig)	+	Line Loss and PSIA conversion	=	Corrected S/G Pressure (PSIA)	→	Enthalpy Steam H_s (BTU/lbm)
		27.0				

−

Feedwater temperature °F	→	Enthalpy FW H_F (BTU/lbm)

=

$\Delta H_1 = H_s - H_F$ [BTU/lbm]	x	Mainsteam Flow M_s (lbm/HR)	=	$M_s \times \Delta H_1$ [BTU/HR]

B[illegible]		

[illegible]ensity lbm/ft³	→	Enthalpy BD H_{BD} (BTU/lbm)

x −

[illegible] Factor		Enthalpy FW H_{FW} (BTU/lbm)
.022		

= =

Blowdown Flow [lbm/HR]	x	$\Delta H_2 = H_{BD} - H_{FW}$ [BTU/lbm]	=	→	$M_{BD} \times \Delta H_2$ (BTU/HR)

+ =

$Q_{LA} = (M_S \times \Delta H_1) + (M_{BD} \times \Delta H_2)$ (BTU/HR)

Completed by: ____________________

Date: ____________________

QUESTION 6.05 b

a. One decay heat release valve - each S/G has a line that feeds a common header to the decay heat release valve.

b. Provides a flow path for long term decay heat release to the atmosphere.

c. Capacity - can release steam at a sufficient rate to remove 100% of the decay heat about 30 minutes after shutdown from full power.

d. Location - Upper level safeguards, steamside. The line taps off the safety valve header which taps off upstream of the trip valve.

SAME VALVE ARRANGEMENT FOR TDAFW PUMP

e. A non-return valve is provided in each line connecting the main steam lines to the common decay heat release header to prevent reverse flow of steam in case of a steam line break.

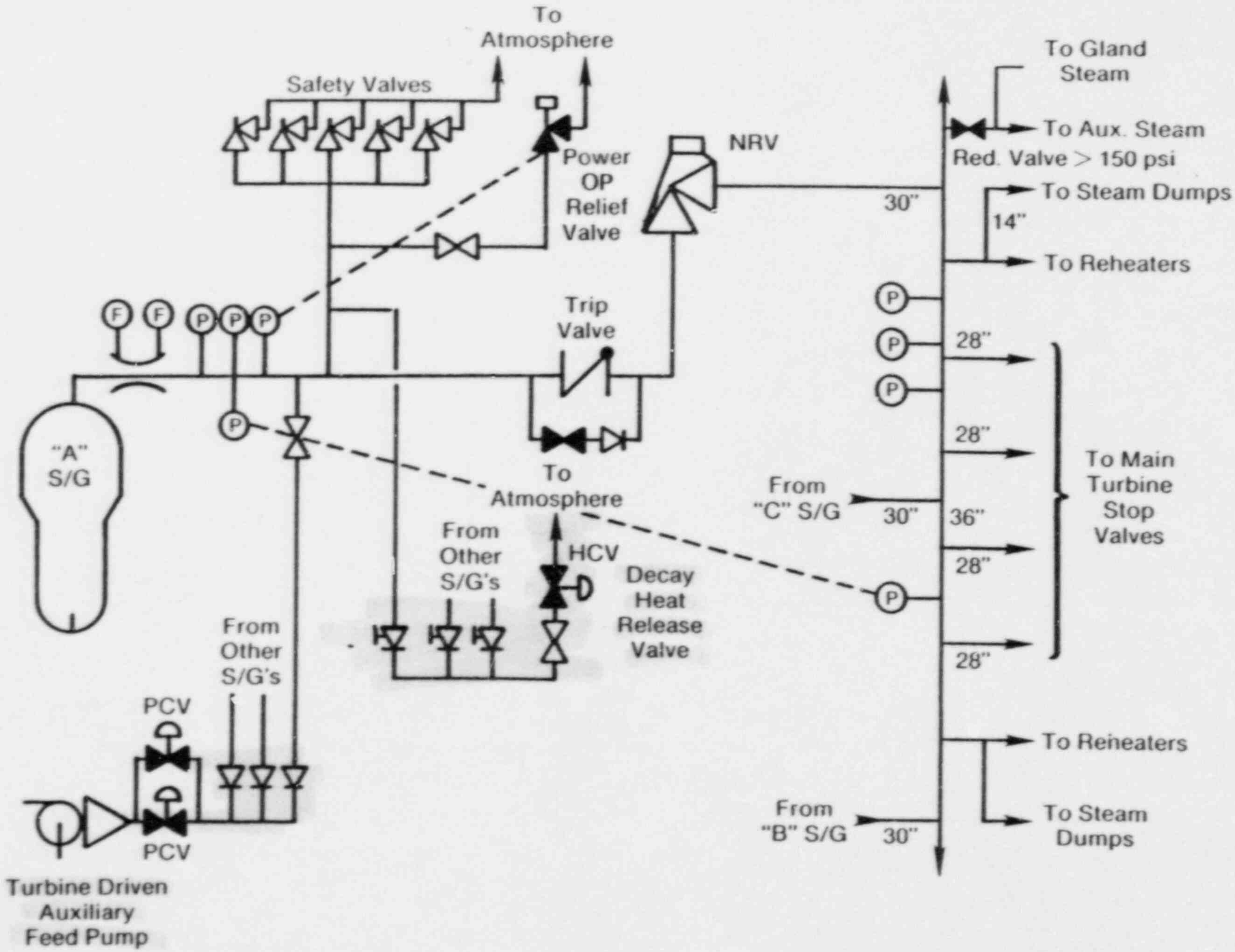
4. Turbine Driven Aux Feed Water Pump (TDAFWP)

a. Each steam line feeds a common header for steam supply to the TDAFWP.

ASK: What is the purpose of TDAFWP?

ANSWER: To provide Aux Feedwater in the event of a station blackout.

b. A check valve in each line prevents the loss of steam to the TDAFWP in the event of a MSLB.



MAIN STEAM SYSTEM ONE-LINE DIAGRAM

ND-89.1/T-2.1

QUESTION 6.08

3. Physically, pushing the latch pushbutton causes the following to occur:

   Auto Stop Oil Reset
   Overspeed Trip Reset
   Vacuum Trip Reset
   33 RO Reset
   Opens: Stop Valves
   Reheat Valves
   Intercept Valves
   Close signal to governor valves
   Positions air pilot valves
   Resets steam dumps

G. Turbine Runback

1. In the event of an approach to overpower ΔT, overtemperature ΔT, or dropped rod conditions, the Turbine Runback Subsystem provides a signal to the AGVC and the MGVC to automatically reduce turbine load at a safe yet rapid rate, to a level which maintains safe margins in the reactor core. One series of runbacks is initiated if the following conditions are met:

   - Two of the three overtemperature ΔT channels indicate greater than 2 percent below the OTΔT trip setpoint, or

   - Two of the three overpower ΔT channels indicate greater than 2 percent below the OPΔT trip setpoint, or

   - One of the four NIS rod drop signals indicate a 5 percent decrease in reactor power in 2 seconds.

2. When the conditions are met, the following events occur:

   a. The reference counter, the MGVC, and the setter counter (which controls the SETTER display) are pulsed at a very rapid rate. If an OTΔT or OPΔT condition initiates the runback, the reference counter is pulsed so that, for every 1.5 seconds out of every 30 seconds, the turbine load is reduced at a rate of ~~10~~ 200 percent/minute. The pulsing continues until the runback condition clears. If an NIS dropped rod signal initiates the runback, the same devices are pulsed at the 200 percent/minute rate for one 9-second period.

   b. The OVERPOWER ΔT TURBINE RUNBACK AND ROD STOP CH I alarm (window 1G-F4), the OVERTEMPERATURE ΔT TURBINE RUNBACK AND ROD STOP CH I alarm (window 1G-F3), or the NIS DROPPED ROD STOP AND TURBINE RUNBACK alarm (window 1G-H1) annunciates. Channels II and III have similar alarms.

   c. All automatic and manual rod withdrawal signals are blocked.

   d. The RUNBACK OPERATOR indicator light on the operator panel illuminates.

3. There is another type of runback which operates to reduce turbine load by use of the valve position limit circuits. This runback provides a maximum open signal to the governor valve servo mechanism. The load limit runback is initiated under the following conditions:

- Both channels of impulse pressure (PT-MS1446 and -1447) indicate turbine load is above 70 percent, and

- Either of the following:

  a. One of four NIS rod drop channels indicate a 5 percent decrease in reactor power in 2 seconds, or

  b. Any rod bottom signal is received from the rod position indicator (RPI) circuits.

With the conditions satisfied, the runback circuits decrease the valve position limit circuits at 120 percent per minute until one impulse pressure channel indicates less than 70 percent.

The load limit runback is annunciated by the NIS DROPPED ROD STOP AND TURBINE RUNBACK alarm (window 1G-H1) or the RPI ROD BOTTOM STOP AND TURBINE RUNBACK alarm (window 1G-H2). The RPI-initiated runback can be defeated within the Rod Control System; however, such action annunciates the RPI TURBINE RUNBACK DEFEATED alarm (window 1G-C5).

After the runback, the condition must be reset before the valves can be repositioned.

H. Overspeed Protection Controller (OPC)

1. The OPC circuit - called the "auxiliary governor" accomplishes the following:

Question 608

6. The output is then directed to the main control board position indicators. These indicators, one for each full length rod, give the actual position indicated by the LVDT. It receives the rod position DC analog signal as adjusted from the signal conditioning module.

7. The plant P-250 computer also receives the individual rod position signals.

8. The next component to which the signal is supplied is the Rod Bottom Bistable. This bistable is a simple level detector which receives its input signal from the signal conditioning module. The bistable output is used to operate a control relay which generates the "RPI ROD BOTTOM ROD STOP AND TURBINE RUNBACK" alarm and light the rod bottom indicating light. The rod bottom alarm actuates when any rod drops below 20 steps.

9. The RPI rod bottom rod stop, the turbine runback, and the rod bottom alarm can be automatically overridden by control banks B, C, and/or D step counters being less than 35 steps. For example, if all rod banks, with the exception of CBD, were above 35 steps, the annunciator would not be activated by a control bank D rod dropping below 20 steps. However, should a rod in any bank other than CBD drop to less than 20 steps, the alarm would occur. In other words, the alarm will not actuate when any rod in control banks B, C, or D is less than 20 steps, provided the associated step counter for the rod bank is less than 35 steps.

DISPLAY transparency T-4.3, RBB/SCM, and point out pertinent items of interest and the fact that this is from the IRPI cabinet.

QUESTION 6.11c

WRITE on chalkboard:

- RESET - Removes load reject arming signal. Spring return to T_{avg} position.

- T_{avg} - Allows dumps to operate with arming signal from either the load reject or the turbine trip circuits.

- STEAM PRESSURE - Allows steam dump operation with signals from the steam header pressure controller in manual only. The automatic circuitry has been removed.

E. Steam Dump Arming

1. Steam Dump "arming" means that instrument air has been made available to operate the dump valves when a demand signal is generated.

DISPLAY ND-93.3/T-9.5, Steam Dump Arming Circuit.

DISTRIBUTE ND-93.3/H-9.5, Steam Dump Arming Circuit Worksheet.

2. There are three interlocks which must be satisfied in order to arm the dumps when an arming signal is activated. These are the "condenser available" interlock, the "condenser cooling" interlock, and the RCS temperature interlock.

   a. The condenser available interlock is satisfied by 2/2 condenser pressure transmitters sensing condenser vacuum at > 26 inches Hg for Unit 1 (condenser pressure chart recorder on vertical

c. The error signal also goes to high and high-high error bistables which operate similar to those discussed earlier. The setpoints of these bistables are different than those of the Load Reject mode. The high error setpoint is 10° and the high-high error setpoint is 20°.

d. The turbine trip arming signal is reset when the main turbine is re-latched following the trip.

RE-DISPLAY ND-93.3/T-9.6, Steam Dump Modulating Circuit.

7. The Steam Pressure Mode of operation arms the dumps whenever the Steam Dump Mode Select Switch is placed in the "Steam Pressure" position and all the interlocks are met.

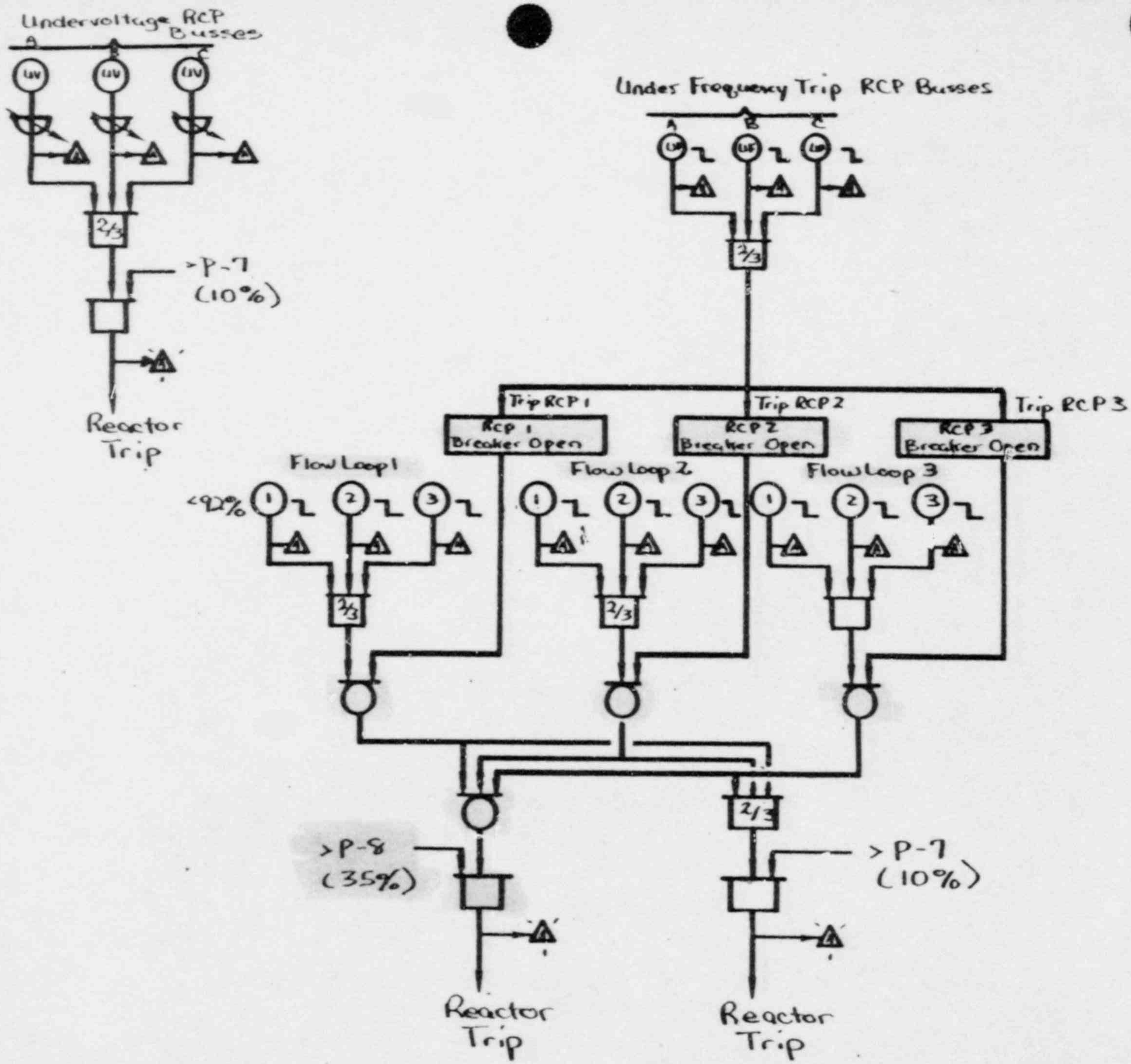
   a. This mode of control is used for plant cooldown and for maintaining the RCS temperature while at Hot Standby conditions.

   b. The dump demand signal is developed from the steam header pressure controller. The MANUAL/AUTO control station for this controller is located on benchboard 1-2 above and slightly to the right of the Steam Dump Control Select Switch.

   c. The steam header pressure controller operates only in manual. The demand signal is controlled by using the "increase" and "decrease" pushbuttons on the controller.

EXPLAIN ND-93.3/T-9.6 as necessary to verify trainee understanding.

QUESTION 6.15a



RCS FLOW TRIP LOGICS

ND-93.3-H-11.1

QUESTION 7.07

No. 97887210

NUMBER	PROCEDURE TITLE	REVISION
AP-16.00	EXCESSIVE RCS LEAKAGE	00.01
		PAGE
		2 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	VERIFY LEAK - GREATER THAN 25 GPM	GO TO Step 8.
[2]	ISOLATE LETDOWN: * Close LCV-( )460 A and B * Close RHR isolation valve HCV-( )142	
[3]	CONTROL CHARGING FLOW TO MAINTAIN PRZR LEVEL: * Control FCV-( )122 manually	
[4]	VERIFY ADEQUATE CHG/SI PUMP SUCTION FLOW: * VCT level being maintained by blender	Align CHG/SI pump suction to RWST: a) OPEN MOV-LCV-115 B and D, b) Close MOV-LCV-115 C and E.
[5]	STOP CONTAINMENT SUMP PUMPS: * ( )-DA-P-4A and B	
[6]	CHECK SI - NOT REQUIRED a) PRZR Level - STABLE or INCREASING AND PRZR Pressure - STABLE or INCREASING b) RCS leakage - LESS THAN 150 GPM	Initiate SI. GO TO EP-1.00, Reactor Trip/Safety Injection.

QUESTION 7.16

FOLDOUT FOR EP-2 SERIES PROCEDURES

1. SI REINITIATION CRITERIA

   Manually operate SI pumps as necessary if EITHER condition listed below occurs:

   * RCS subcooling based on core exit TCs - LESS THAN 30 [80]°F
   * PRZR level - CANNOT BE MAINTAINED GREATER THAN 13 [49]%

2. RED PATH SUMMARY

   a. SUBCRITICALITY - Nuclear power greater than 5%

   b. CORE COOLING - Core exit TCs greater than 1200°F

      OR

      Core exit TCs greater than 700°F AND RVLIS full range less than 42% with no RCPs running

   c. HEAT SINK - Narrow range level in all SGs less than 32% AND total feedwater flow less than 350 [492] GPM

   d. INTEGRITY - Cold leg temperature decrease greater than 100°F in last 60 minutes AND RCS cold leg temperature less than 285°F

   e. CONTAINMENT - Containment pressure greater than 60 PSIA

3. SECONDARY INTEGRITY CRITERIA

   Go to EP-3.00, Faulted Steam Generator Isolation, Step 1, if any SG pressure is decreasing in an uncontrolled manner or has completely depressurized, and has not been isolated.

4. EP-4.00, TRANSITION CRITERIA

   Go to EP-4.00, Steam Generator Tube Rupture, Step 1, if any SG level increases in an uncontrolled manner or any SG has abnormal radiation.

5. COLD LEG RECIRCULATION SWITCHOVER CRITERION

   Go to EP-2.03, Transfer to Cold Leg Recirculation, Step 1, if RWST level decreases to less than 22%.

6. AFW SUPPLY SWITCHOVER CRITERION

   Switch to alternate AFW water supply if CST level decreases to less than 20%.

No. 97024020 QUESTION 8.05C

MEMORANDUM

TO All Supervisors January 26, 1988

FROM D. L. Benson Surry Power Station

PROCEDURE DEVIATIONS

As a result of our efforts to upgrade our 10CFR50.59 review process and to minimize the probability of deviating approved procedures without the proper review, we have changed the procedure deviation approval process outlined in SUADM-ADM-21. Supervisors should discuss these changes with your respective groups. All procedure deviations initiated on or after February 2, 1988, must use the new procedure and the revised deviation form.

The major changes are:

1. All procedure deviations will be screened by the cognizant supervisor for 1) a change to the intent of the procedure and 2) the necessity for a safety analysis and 50.59 review. The cognizant supervisor for the respective procedures is identified in Table 5.4.1 of the procedure. The procedure revision now requires supervisory titled personnel to do the screening.

2. A deviation which changes the procedure intent has been defined as, at a minimum, one which changes the procedure 1) title, 2) initial conditions, 3) precautions or limitations or 4) purpose.

3. If the supervisor's screening determines that the deviation requires a safety analysis/10CFR50.59 review, SNSOC must approve it prior to implementation.

4. Two paths are available when the supervisor's screening determines that a safety analysis/10CFR50.59 review is not required.

   a. If the intent is not changed, the deviation can be approved by the cognizant supervisor identified in Table 5.4.1 and a licensed SRO (Shift Supervisor or Superintendent of Operations).

   b. If the intent is changed, the deviation must be approved by the cognizant supervisor (in most cases the superintendent level) shown in Table 5.4.2 and the SNSOC prior to implementation. Shift Supervisor or Superintendent of Operations approval is also required prior to actually using the deviated procedure.

5. The cognizant supervisor signing the deviation is responsible for the intent and safety analysis/50.59 review screening and the technical adequacy of the deviation. The Shift Supervisor signature indicates that the appropriate approvals have been obtained and that the work can be safety integrated with other ongoing shift activities.

Although this strengthening of our method of controlling plant changes will undoubtedly result in some delays initially, it is necessary to ensure the excellence in station activities that we are all working towards. To prevent the continuing need for a large number of procedure deviations, we need to initiate permanent procedure changes when the deviation is identified. SNSOC review will stress identification of procedural improvements in order to reduce the need for continuing or repetitive procedure deviations.

It is important for all employees to understand that full compliance with procedures is required and expected. If a procedure as written, including initial conditions, cannot be followed, a properly approved procedure deviation is required before work can continue.

David L Benson

D. L. Benson

DLB/pms

copy: Mr. H. L. Miller
Mr. E. S. Grecheck
Mr. H. H. Blake
Mr. P. W. Tucker
Dr. A. E. Friedman

QUESTION 8.11

ΔT and Overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism or the assembly remains misaligned from its group step demand position by more than ±12 steps. Additionally, a full-length control rod shall be considered inoperable if its rod drop time is greater than 2.4 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod drive mechanism, (i.e. programming circuitry), the provisions of Specifications 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period, the reactor will be brought to hot shutdown conditions.

4. The provisions of Specifications 3.12.C.1 and 3.12.C.2 shall not apply during physics tests in which the assemblies are intentionally misaligned.

5. Power operation may continue with one rod inoperable provided that within one hour either:

   a. the rod is no longer inoperable as defined in Specification 3.12.C.1, or

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee:	Virginia Electric and Power Company
Facility Licensee Docket No.:	50-280 and 50-281
Facility Licensee No.:	DPR-32 and DPR-37
Operating Tests administered at:	Surry Power Station
Operating Tests Given On:	March 15-16, 1988

During the conduct of the simulator portion of the operating tests identified above, no significant performance and/or human factors discrepancies were observed.