UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

EXAMINATION REPORT 50-280/OL-84-01

Facility Licensee: Virginia Electric and Power Company
P. O. Box 2666
Richmond, VA 23261

Facility Name: Surry 1 & 2

Facility Docket No.: 50-280 and 50-281

Written and oral examinations were administered at Surry Nuclear Station near Surry, Virginia.

Chief Examiner: Timothy L. Norris — 10/9/84
Timothy L. Norris — Date Signed

Approved by: Bruce A. Wilson — 10/15/84
Bruce A. Wilson, Section Chief — Date Signed

Summary:

Examinations on June 18-22, 1984

Oral and written examinations were administered to fifteen candidates, nine of whom passed.

REPORT DETAILS

1. Persons Contacted:

SRO Candidates	RO Candidates
Bailey, Jack A.	Dials, David G.
Gwaltney, Robert L.	Early, James C., II
Quinn, William F.	Johnson, Randy L.
Marshall, William W.	Jones, Irving L., III
Moore, William H.	
Ringler, Terry N.	Knowlton, Donald
Small, Michal A.	Masingo, Donald K., Jr.
Wheeler, Andrew D., Jr.	Nelson, Arthur I.

Other Facility Employees Contacted:

*Jack L. Wilson, Station Manager
*David A. Christian, Superintendent, Operations
*Larry L. Edmonds, Superintendent, Nuclear Training
*Harold F. McCallum, Supervisor, Training-Power Station Operations
*Larry Gardner, Supervisor, Training-Simulator
*L. Richard Buck, Instructor
Cathy Curfman, Senior Instructor

*Attended exit meeting

2. Examiners:

*Timothy L. Norris
Mark E. Baldwin
William E. Eldridge
Ronald H. Thornton
Odra W. Burke

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with Larry Edmonds, Harold McCallum, Cathy Curfman, and Larry Gardner to review the written examination and answer key. The following comments were made by the facility reviewers:

a. SRO Exam

(1) Question 5-14

Facility Comment: Change answer key to allow for a T-hot operating "band" of ±2-3°F.

NRC Resolution: Answer key was changed as suggested.

(2) Question 6-4

Facility Comment: Item 10 - XFER SWITCH - should not be required for full credit. The question did not ask for it.

NRC Resolution: Item 10 was deleted from the answer.

(3) Question 6-8

Facility Comment: Delete Item 4 from answer key. Backup heaters are normally run with the switch in the "ON" position verses "AUTO."

NRC Resolution: Answer key will be changed as suggested. The operation of the heater switch in the "ON" position was verified in the control room.

(4) Question 7-8

Facility Comment: Specific setpoints included in APs of minor safety significance should not be required to be memorized.

NRC Resolution: Disagree. This question is addressed in 10 CFR 55.20, Scope of Examinations, paragraphs 55.22, 55.21(d), and 55.21(e). A more detailed explanation of the scope of written questions is found in NUREG 1021, ES-402, paragraph A.3. The question and answer will remain unchanged.

(5) Question 8-10

Facility Comment: Delete this question. S.O. 15 was deleted in May, 1984.

Reference: Security procedures.

NRC Response: This question and answer were deleted as suggested.

(6) Question 8-12

Facility Comment: Change Item 6 of answer key to:

Safety valve position-temperature indicator.

Reference: Minimum equipment list and T.S. 3.7-8 and 9.

NRC Resolution; Answer key was changed as suggested.

b. RO Exam

(1) Question 1-3

Facility Comment: Change to answer key:

(a) Setpoint for source range reinitiation is 5×10^{-11} amps

(b) Answer should accept use of $P = P_o 10^{SURT}$ and a SUR of -1/3 dpm.

Reference: PT1 1, Page 17

NRC Resolution: Answer key changed as requested. The method of solving the problem using SUR was also accepted.

(2) Question 1-7

Facility Comment: Change to answer key:

(a) "Provide adequate shutdown margin" should be an acceptable answer.

Reference: PLS book, Page 2, Item C.

NRC Resolution: The answer key was changed to accept the facility comment as an additional answer.

(3) Question 1-8

Facility Comment: Change to answer key:

(a) Approximately 40 hrs. ± 10 hrs.

Reference: Surry Unit 1 Cycle 7 Curve Book

NRC Resolution: The answer key was changed as requested.

(4) Question 1-9

Facility Comment: Change to answer key:

(a) Acceptable values for β_{eff} should be in a range of 0.006 to 0.005.

NRC Resolution: The answer key was changed to accept the suggested tolerance band.

(5) Question 1-12

Facility Comment: Change to answer key:

(a) Answer should allow for a tolerance of ±5 Btu/lbm for Δh, based on the difficulty of reading the Mollier Diagram.

NRC Resolution: The answer key was changed to accept the suggested tolerance band.

(6) Question 1-13

Facility Comment: Change to answer key:

(b) Answer should also include that excessive steaming rate could cause cold-block.

(c) Answer should also include that RCS pressure decrease may cause flashing in the flow path which will cause a loss or reduction of natural circulation flow rate.

NRC Resolution: The facility's suggestions were added to the answer key for additional clarification and alternate answer.

(7) Question 2-1

Facility Comment: Change to answer key:

b.5. Answer should be RHR pump seal cooler.

Reference: Plant Maintenance procedure for RHR pump.

NRC Resolution: Answer key changed as suggested.

(8) Question 2-2

Facility Comment: Change to answer key:

(a) Add time delay (5 min) for ORS pumps (allows adequate water level in sump).

(b) Add time delay (2 min) for IRS pumps (allows adequate water level in sump).

Reference: FSAR System Description.

NRC Resolution: Suggested change was added to the answer key as additional acceptable answers.

(9) Question 2-3

Facility Comment: Change to answer key.

(a) Additional services provided - specific loads to be supplied.

Reference: lesson plan

NRC Resolution: The following loads were provided by the facility and verified by the Chief Examiner and added to the answer key as additional correct answers.

(3) Governer Motor, 4 Gov. motor booster pumps
(4) Annunciator panel
(5) Various controls and alarms
(6) Start circuits

(10) Question 2-4

Facility Comment: Change to answer key:

(a) Answer should be 480V Emerg. Bus 1H.

(h) Answer should be 480V Emerg. Bus 1H. The answer on the answer key for (a) and (h) 480V MCC 1H1 was not a choice on the exam.

NRC Resolution: The question parts (a) and (h), and their respective answers were deleted. The points per item was changed to 0.4 each.

(11) Question 2-5

Facility Comment: Change to answer key:

(c) Answer should be 1, 3, 4. Answer 2 is not logical (how can level be "normal" and high?) - reference material is from logic diagrams and does not conform to actual operation.

NRC Resolution: Items 1, 3, & 4 were accepted for full credit.

(12) Question 2-6

Facility Comment: Change to answer key:

(b) Change valve numbers to H.P. and L.P. turbines as appropriate, valve numbers are not specifically asked for by question.

NRC Resolution: The answer key was changed as suggested.

(13) Question 2-9

Facility Comment: Change to answer key:

(a) Answers 5 and 6 should be deleted - they are not protective trips.

(b) Answers 1, 2, 3, 4 should be accepted for full credit.

NRC Resolution: Answer key Items 1-4 were accepted for full credit.

(14) Question 2-10

Facility Comment: Change to answer key:

(1)a. The answer Reactor trip, turbine trip should be counted as <u>two</u> possible answers.

(2) An acceptable answer is Hydrogen analyzer heat tracing switches.

Reference: EP 1

NRC Resolution: The answer key was changed as requested.

(15) Question 3-1

Facility Comment: Change to answer key:

b.2. Center region - margin to saturation in <u>°F</u>.

Reference: To be verified in control room.

NRC Resolution: The margin to saturation was verified to be in °F. The answer key was changed as suggested.

(16) Question 3.3

Facility Comment: Delete question. Terminology and permissive numbers are not applicable to this facility.

NRC Resolution: Question deleted after verification.

(17) Question 3-8

Facility Comment: Change to answer key:

(b) Charging flow control valve (FCV-1122, 2122) is more appropriate terminology.

NRC Resolution: Answer key changed to accept terminology.

(18) Question 3-10

Facility Comment: Change to answer key:

(a)1) 2% below trip setpoint

2) 2% below trip setpoint

3) Delete "with power >70%"

4) 1/48 control rods within 20 steps of bottom for power >70%

(b) More complete answers for types of turbine runbacks will be supplied from lesson plans and other reference material.

NRC Resolution: Runbacks do occur at 3% below trip setpoint, therefore, answer for Items a.1 and a.2 remain the same. The answer key was changed to reflect the remaining comments supported by plant reference material.

(19) Question 4-5

This question is the same as question 7-8 on SRO exam. See Comment a.4 above.

4. Exit Meeting

At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the results of the examination. Those individuals who clearly passed the oral examination were identified.

There were no generic weakness (greater than 75 percent of candidates giving incorrect answers to one examination topic) noted during the oral examination. An area of below normal performance was communications among the operators during the simulator examinations. The cooperation given to the examiners was also noted and appreciated

Mr. Larry Edmonds, VEPCO Superintendent of Nuclear Training, read a letter, Enclosure 4, of additional facility comments concerning the examination. The contents of the attachment to his letter was included in "Examination Review Meeting," above. Specific NRC responses to key comments of Enclosure 4 are as follows:

Comment: "... it is our opinion that there were too many questions on systems and topics of minor or no safety significance."

Response: The scopes of the reactor operator and senior operator written examinations are specified in 10 CFR 55.21 and 55.22, respectively. Examiner Standard, ES-202 and ES-402 of NUREG 1021 provide guidance to the examiners as to the content on the individual categories. Each of the Surry written examinations was reviewed in Region II and found to comply with the requirements of the Examiner Standards and the Federal Regulations.

A review of the systems questions in Category 2 of the RO and Category 6 of the SRO exam show a proper distribution of topics with regard to major, auxiliary and engineered safety systems. Instruments and controls are topics covered in Categories 3 of the RO exam and 6 of the SRO. Again, we believe there were no questions on "systems and topics of minor or no safety significance." In fact, the Category 3 questions covered the following topics:

Question	Topic
3.1	Core Cooling Monitor (Saturation Meter)
3.2	RCS process instrument failures
3.3	Deleted
3.4	Rod Control System
3.5	Steam Generator Level Control System
3.6	Main Feedwater Control System
3.7	Emergency Core Cooling System
3.8	Pressurizer Level Control and Protection System
3.9	Reactor Control and Protection System
3.10, 3.11, 3.12	Turbine Control System
3.13	Reactor Protective System
3.14	CVCS (VCT Level Control)

Comment: "Too much memorization was required of steps and setpoints in minor and insignificant Abnormal and Operating Procedures."

Response: Examiner Standards ES-202 and ES-402 provide guidance as to the content of the procedural categories, 4 on the RO and 7 on the SRO. These standards state that "In general, the candidate must demonstrate complete knowledge and understanding of the symptoms, automatic actions, and immediate action steps specified by abnormal and emergency procedures."

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Appendix A specifies "typical safety-related activities that should be covered by written procedures."

An analysis of Category 7 of the SRO exam shows no basis for your comment. All questions and responses asked for fell within the scope of the examiner standards and concerned written procedures listed in Appendix A of Regulatory Guide 1.33.

Question Number	Memorization of Steps and Setpoints	Procedure Title (Topic)
7.1	Yes (ECP Limits)	Reactor Startup
7.2	No	Plant Startup
7.3	Yes (SI Termination Criteria)	LOCA
7.4	Yes (Restoration Actions)	Delta Flux Control
7.5	Yes (Identification & Isolation)	SG Tube Rupture
7.6	No (Symptoms)	RCP seal failure
7.7	Yes (Automatic & Immediate Actions)	Fuel Handling Accident
7.8	Yes (Immediate Actions)	Circulating Water Pump Trip
7.9	No	Radiological Control
7.10	No (Exposure Limits)	Radiological Control
7.11	No	Tech Spec LCO
7.12	Yes (Immediate Actions)	Reactor Trip

Question 7.8 was raised as a concern during the examination review. This question asked for the immediate actions following the trip of a circulating water pump. Since this event would cause a loss of condenser vacuum, it is included in paragraph 6, Appendix A of Regulatory Guide 1.33, "Procedures for Combating Emergencies and other Significant Events." We believe the above chart shows that all other procedures asking for immediate actions (such as LOCA, SG Tube rupture, fuel handling accident and reactor trip) were not minor nor insignificant.

Comment: "We feel that it is very important that the written exam review be conducted prior to/during administration of the exam."

Response: The NRC believes that the perceived integrity of the examination process is of utmost importance and outweighs advantages of the review policy which you advocate.

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

ENCLOSURE 3
(1 OF 3)

Facility: Surry

Reactor Type: Westinghouse (PWR)

Date Administered: 6/ /84

Examiner: M. E. Baldwin

Applicant: ____________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple questions sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
25.0	25	______	______	5. Theory of Nuclear Power Plant Operation, Fluids & Thermodynamics
25.0	25	______	______	6. Plant Systems: Design, Control & Instrumentation
25.0	25	______	______	7. Procedures-Normal, Abnormal, Emergency & Radiological Control
23.5 ~~25.0~~	25	______	______	8. Administrative Procedures, Condit and Limitations
98.5 ~~100.0~~	100	______	______	TOTALS

Final Grade ______ %

All work done on this exam is on my own, I have neither given nc. received aid.

____
Applicant's Signature

5-1 An estimated critical position has been calculated for a reactor startup that is to be performed 15 hours after a trip from a 60 day full power run. How would each of the following events or conditions affect the actual critical rod position compared to the estimated critical position? In your answer, select whether the actual position would be: higher than estimated, lower than estimated, or no significant difference.

a. A steam generator's level is increased significantly. (0.5)

b. The startup is delayed for approximately two (2) hours. (0.5)

c. The steam dump pressure setpoint is increased. (0.5)

d. A new boron sample is ten (10) ppm lower than the sample used for the ECP calculation. (0.5)

e. Condenser vacuum decreases by two (2) inches Hg. (0.5)

5-2 The reactivity worth of samarium at 25% equilibrium power is (greater than/less than/or equal to) the reactivity worth at 100% equilibrium power. (1.0)

5-3 For a reactor operating at a constant power and temperature the thermal neutron flux near EOL will be (greater than/smaller than/or the same as) the flux near BOL? (1.0)

5-4 In the Surry reactors, the moderator temperature coefficient (MTC) varies with certain plant conditions. The MTC: [choose the correct answer.] (1.0)

a. Becomes more negative as boron concentration is increased.

b. Varies due to temperature (Tavg) because of the non-linear density changes of water as temperature changes.

c. Causes axial flux distribution to be tilted towards the top of the core at the beginning of life.

d. Would be expected to introduce a large negative reactivity in the event of a major steam line break.

e. Is not permitted by Technical Specifications to be positive in any normal plant operating modes.

(continued on next page)

5-5 Your reactor is critical and is on a 1 DPM SUR. You are currently at 10^{-8} amps. After 30 seconds you should be at 5×10^{-8} amps. TRUE OR FALSE? (1.0)

5-6 Figure 5-1 is a sketch of Xenon concentration vs time. On the same figure sketch the approximate power history that would be associated with this Xenon concentration plot. Assume that all power changes are made as <u>step</u> changes. (2.0)

5-7 A reactor is critical at 10^4 CPS when a steam generator PORV fails open. Assuming BOL conditions, no rod motion, and no reactor trip, choose the answer below that best describes the values of Tave and nuclear power for the resulting <u>new steady state</u>. (1.0)

a. Final Tave $\geq$ initial Tave, Final Power $\geq$ point of adding heat (POAH)
b. Final Tave $\geq$ initial Tave, Final Power $\leq$ POAH
c. Final Tave $\leq$ initial Tave, Final Power $\leq$ POAH
d. Final Tave $\leq$ initial Tave, Final Power $\geq$ POAH

5-8 During a reactor startup, the operator stops rod pull #9 at 144 steps on Bank C. The Source Range Monitor (SRM) count rate levels off at 1857 cps. The initial SRM count rate was 400 cps at 0 steps withdrawn on control Bank A with $K_{eff} = 0.940$.

a. Calculate the 1/M value for this control position. (1.5)

b. What is the new value of K_{eff} at this condition? (1.5)

5-9 The reactor is critical at 5×10^{-9} amps on the intermediate range. Assume an instantaneous 15 ppm dilution of the RCS occurs. Answer the following, showing all work and all assumptions:

a. What is the resultant stable SUR from this dilution? (1.5)

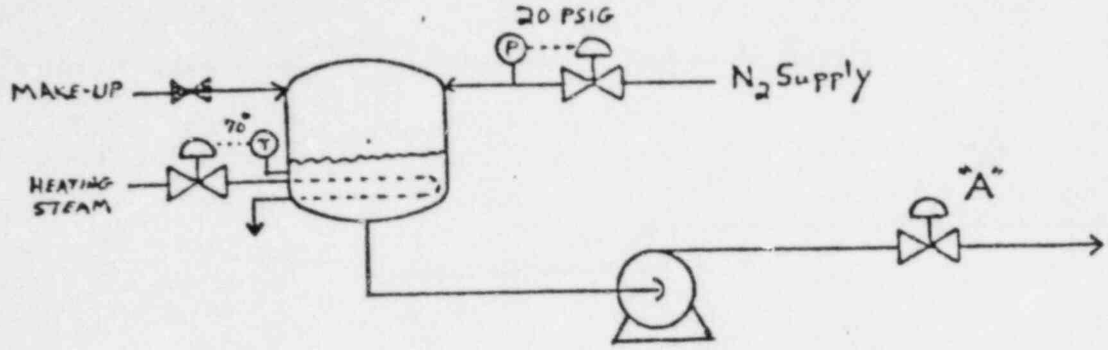
b. How long will it take for the reactor power to reach the point of nuclear heat (2×10^{-5} amps)? (1.0)

5-10 Indicate how the following changes in plant conditions would individually affect DBNR (increase, decrease, or have no effect).

a. Pressurizer pressure decreases (0.5)
b. Tc decreases (0.5)
c. Reactor power decreases (0.5)
d. RCS flow decreases (0.5)

(continued on next page)

5-11



For the system above, how would each of the following operations effect the available NPSH for the pump (increase, decrease, or remain the same)? Consider each one separately and assume short term effects only.

a. double the temperature of the water in the tank (0

b. double the level of the water in the tank (0

c. double the flow through valve "A" (pump discharge-throttle valve) (0

d. double the pressure on the tank (0

5-12 Refer to Figure 5-2 to answer the following.

a. The system flow when pumps 1 and 2 are running is greater than, less than, or essentially the same as the system flow when only pump 2 is running. (0

b. Assume that pump 3 is running at a certain speed (the speed that correlates to the graph given). What would the pump discharge pressure be if the pump speed was increased 30%? (0

(continued on next page)

5-13 Your reactor has been operating at full power for three months when a manual reactor trip occurs. All systems are operational and the steam dumps are immediately placed in the steam pressure control mode. Ten (10) minutes after the reactor trip all RCP's are tripped. Twenty (20) minutes after the reactor trip one (1) RCP is jogged momentarily.

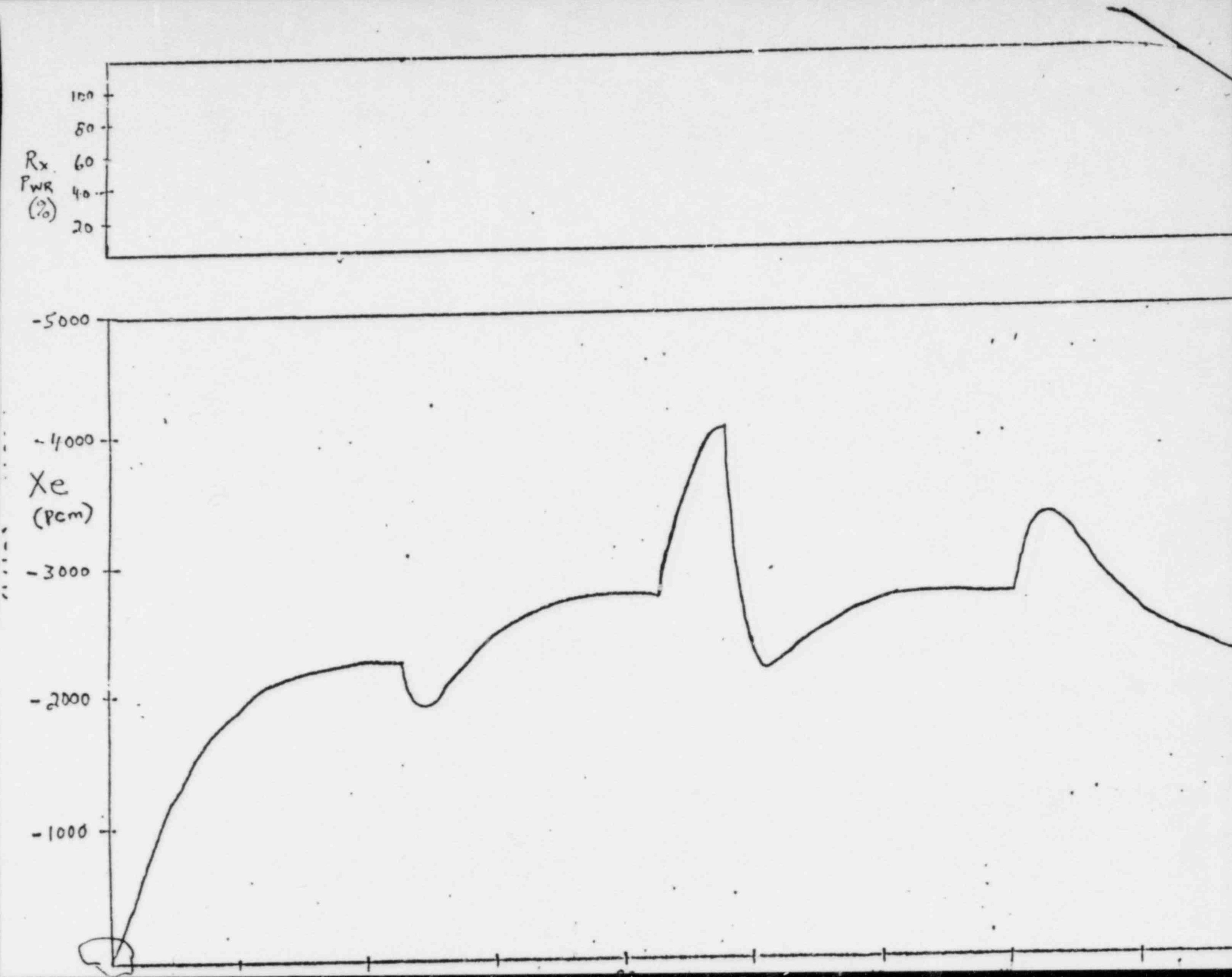
a. On Figure 5-3 indicate and explain the approximate response (trace) of the average core exit thermocouple temperature and Tc Wide Range temperature for the previously described events. Assume that the plant is maintained at hot standby until one (1) hour after the reactor trip. (2.25

b. One (1) hour after the reactor trip commence a cooldown at the maximum allowable cooldown rate (assume all CRDM Shroud Cooling fans are in service). Indicate and explain the temperature traces for the next three (3) hours. (0.75

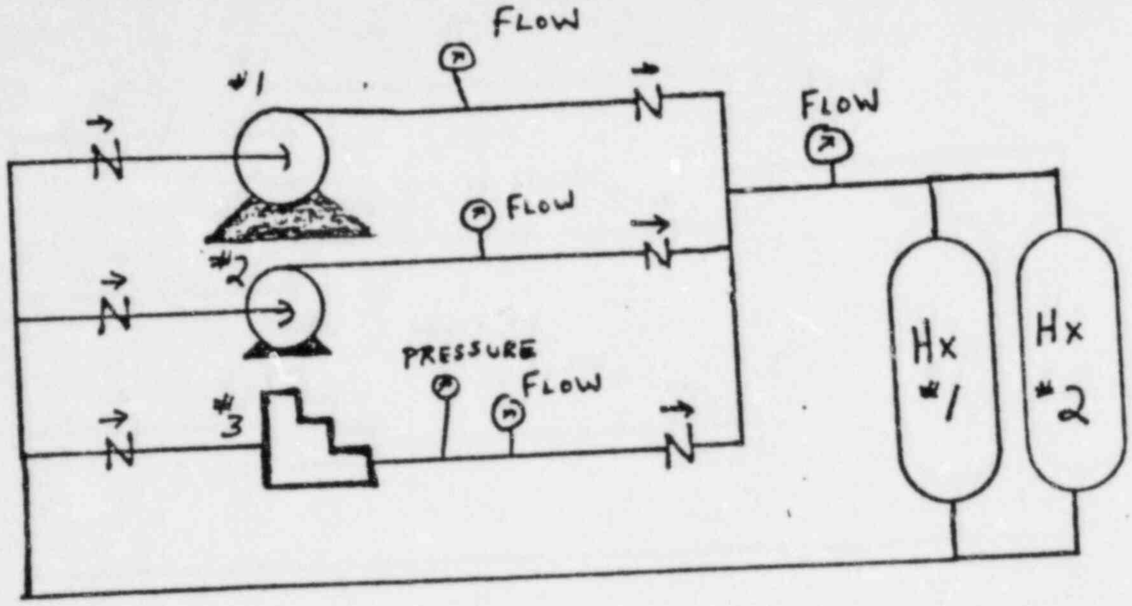
5-14 State the values that represent the amount of subcooling that exists in your reactor at full power and at zero power. Include all assumptions and calculations. (1.5)

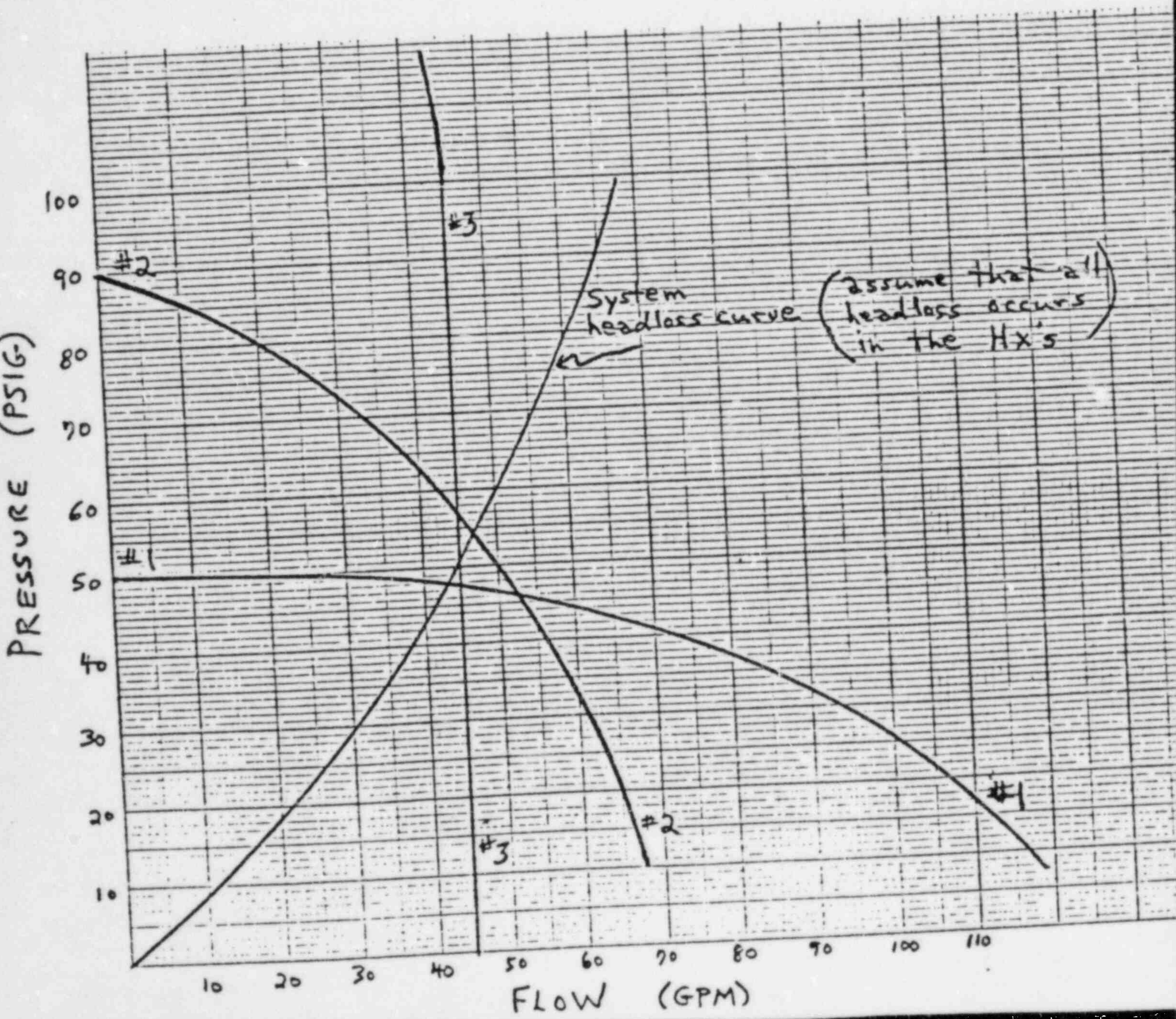
END OF CATEGORY 5

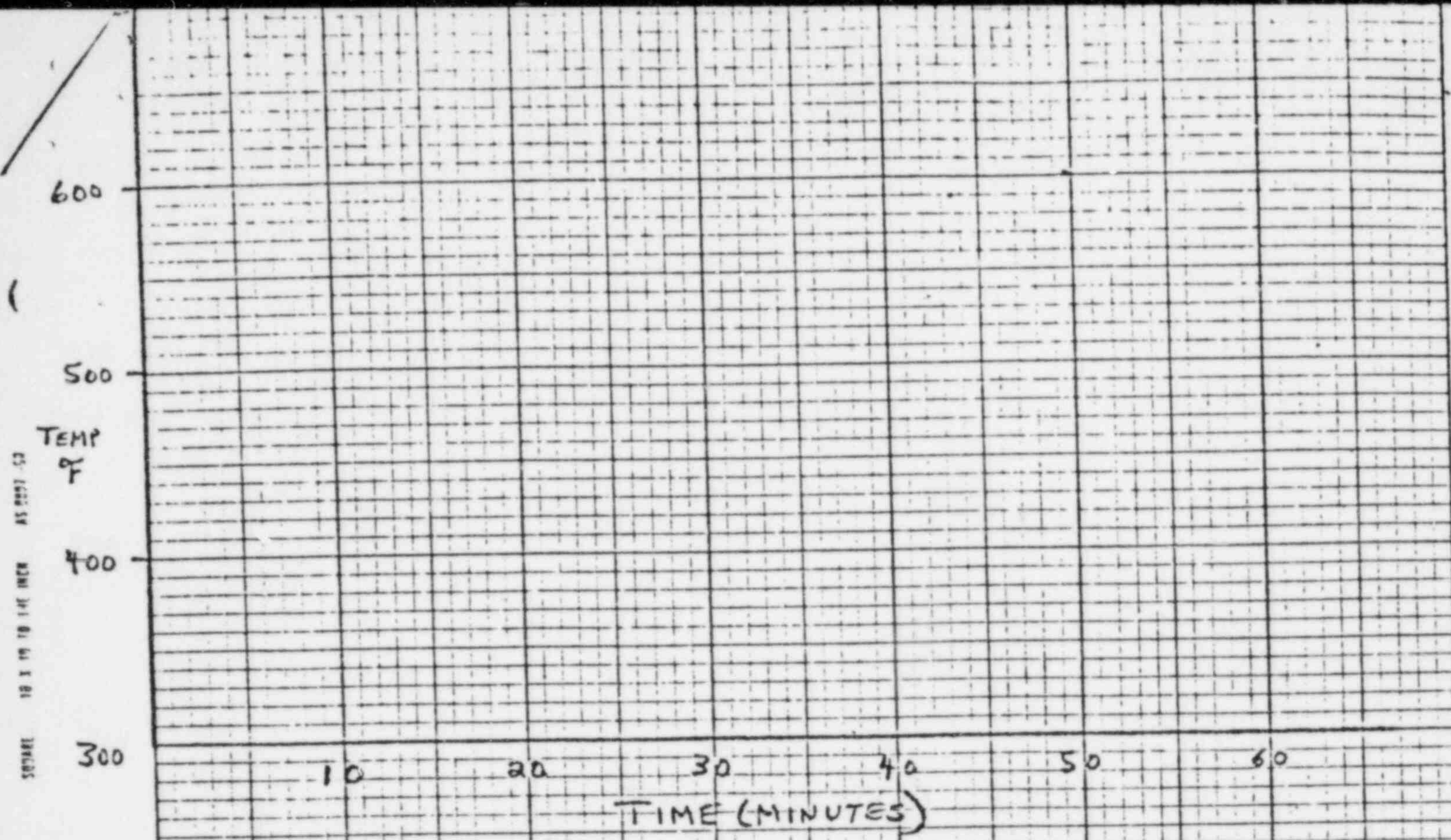
(Write "End of Category 5" on your answer sheet)



Rx PWR (%)
100
80
60
40
20
Xe (pcm)
-5000
-4000
-3000
-2000
-1000


FLOW
#1
FLOW
FLOW
#2
PRESSURE
FLOW
#3
Hx #1
Hx #2


PRESSURE (PSIG)
100
90
80
70
60
50
40
30
20
10
#2
#3
#1
System headloss curve
(assume that all headloss occurs in the Hx's)
#1
#2
#3
10
20
30
40
50
60
70
80
90
100
110
FLOW (GPM)


600
500
TEMP °F
400
300
10
20
30
40
50
60
TIME (MINUTES)


700
600
500
400
300
60
120
180
240
300
TIME (MINUTES)

6-1 a. A break in the reference leg in a pressurizer level indicator will cause the indicated level to be higher than the actual level. True or False? (0.75

b. List all of the locations that have indication for LT-462 (Pzr level-cold calibrated). (0.5)

6-2 Place an "x" in the appropriate boxes of the table on Fig. 6-1 to indicate where the systems connect to the Reactor Coolant System (RCS). (2.5)

6-3 a. What chemical is used in the containment spray chemical addition tank? (0.5)

b. Why is this chemical added to the spray (i.e., what gets accomplished that would not be accomplished by a borated water spray alone)? (0.75)

c. What will initiate containment spray (include setpoint and coincidences)? (0.75)

6-4 Sketch a portion of the electrical distribution system so as to indicate the preferred method by which power would be supplied to the Vital Bus I-III following a station blackout. Do not include breakers and do not include portions of the system that are not directly related to the flow path requested. Label all buses, transformers, and power sources. Use actual alphanumeric designations where applicable. (3.0)

6-5 Identify the seven (7) Reactor Trips for which there are no associated blocks or permissives. For each of the above trips also give the setpoint and coincidence. (2.7)

6-6 Refer to Fig. 6-3 (Reactor Makeup System) and fill in the boxes to indicate the valve positions (open, closed, or modulated) for the various modes of operation. (1.6)

(continued on next page)

6-7 For the following questions refer to Fig. 6-4 (core cooling monitor-front panel).

a. What pressure signals are inputs to the Core cooling Monitor (noun name or transmitter number)? (0.75)

b. What is the significance of the lines drawn between the core thermocouple status lights? (0.5)

c. Which calculation(s) is/are made when the "ΔT loop 1" button is pushed? (1.0)

1) Highest thermocouple minus T_H

2) Highest thermocouple minus T_C

3) Lowest thermocouple minus T_H

4) Lowest thermocouple minus T_C

5) Average thermocouple minus T_H

6) Average thermocouple minus T_C

7) T_H minus T_C

8) Highest thermocouple minus lowest thermocouple

6-8 Assume that your plant has been operating at a 100% steady state power level and Pressurizer Pressure Transmitter PT-445 just failed in the high direction. Describe the sequence of primary plant events that occurs assuming no operator actions. Extend your description to the point where a new steady state or equilibrium is reached. (2.8)

6-9 Assume that your plant ~~is operating~~ at full power when a pipe rupture occurs in the Auxiliary Feedwater System. The rupture is a complete severance of the AFW ~~connection~~ piping to the main feed line for the 'C' steam generator. An automatic SI initiates. S/G 'C' level is decreasing due to the rupture and ~~S/G's A and~~ B are decreasing as essentially all of the AFW flow is exiting through the rupture. You decide to isolate the rupture by attempting to shut MOV-FW-151 A and B. Valve B shuts however valve A will not shut due to a motor failure. List the valves (by functional name or valve number) that would need to be repositioned (by an operator) in order to provide AFW flow to S/G's A and B (assuming that valve MOV-FW-151-A remains open). (1.5)

1st [illegible] between the check valve (FW-89) and the [illegible] vent [illegible]

6-10 The recirculation spray coolers use service water to cool the spray water. Which of the two fluids (service water or spray water) operates at the highest pressure? Why? (1.0)

(continued on next page)

6-11 The unit has been operating at 45% power with all systems in automatic. State the direction of rod motion that would initially take place for each of the following events (consider each one separately).

A. A steam generator power operated relief valve fails open. (0.55)

B. A feedwater heater string becomes isolated. (0.55)

C. The lower detector of the power range channel N44 fails high. (0.55)

6-12 The manipulator crane has three (3) interlocks with inputs from the Dillon Load Cell.

a. Describe these interlocks. Include setpoints, indications, and the limitations that these interlocks impose if they are activated. (1.5)

b. For each of the three (3) Dillon load cell interlocks indicate whether it can be bypassed, and if so by what method. For the interlock(s) that can be bypassed state the indication that would be given when the interlock has been bypassed. (0.75)

c. The Dillon Load Cell has two (2) scales "A" (0-3,000) and "B" (3,000-6,000). The equipment is normally operated on the "A" scale. How is the operation of the manipulator crane affected when the "B" scale is used? (0.5)

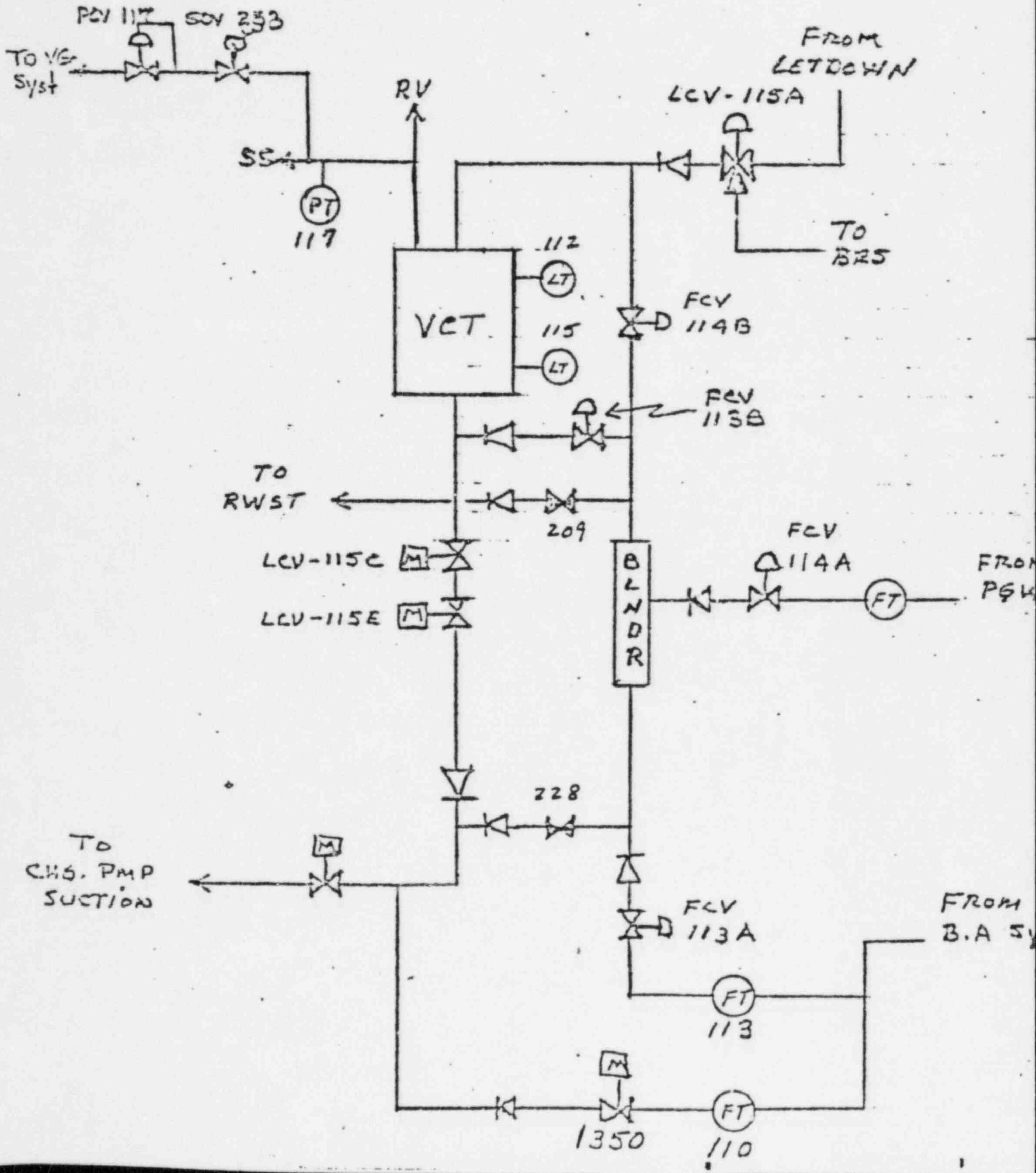
END OF CATEGORY 6

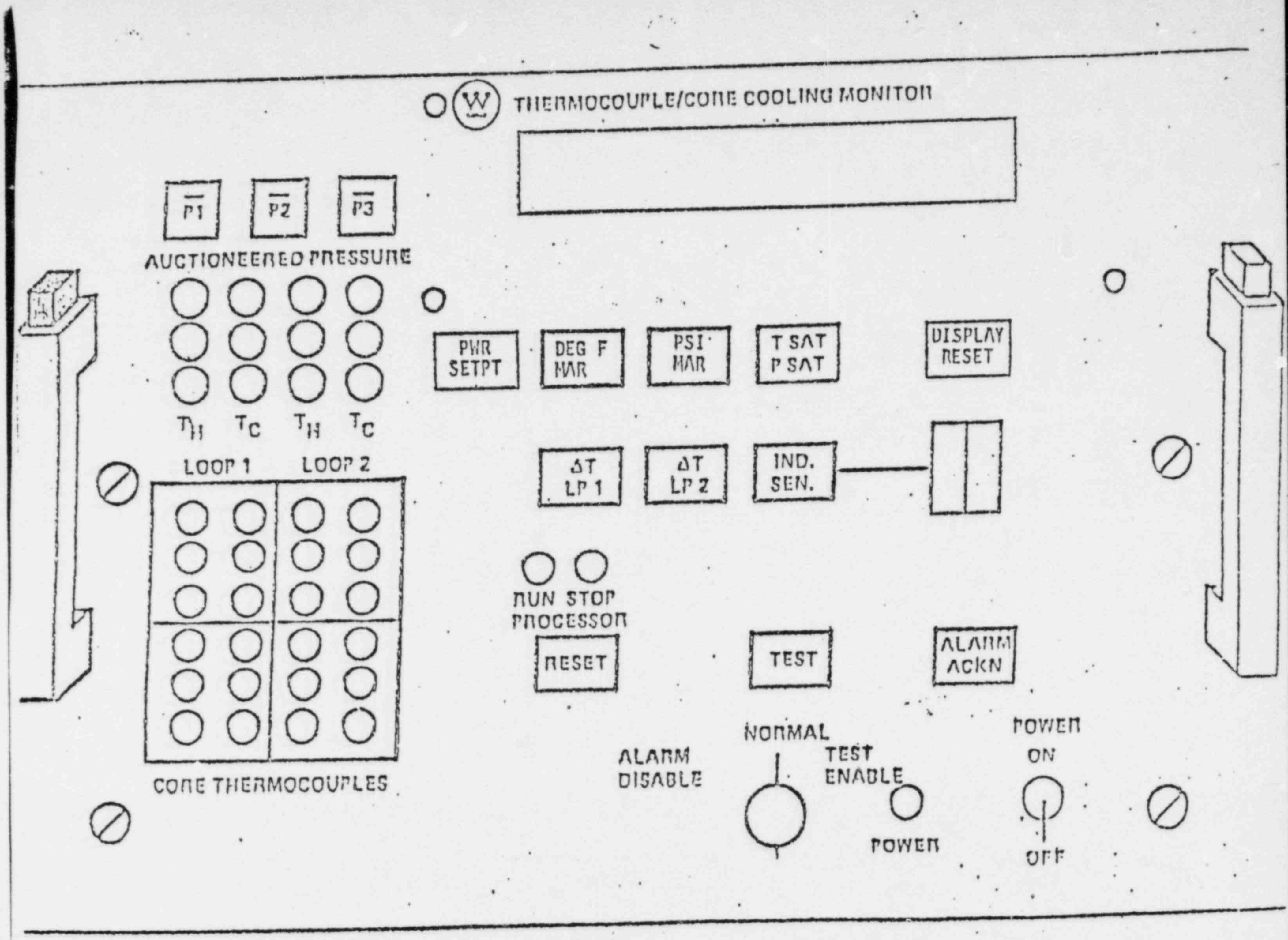
(Write end of category 6 on your paper)

	Loop 1			Loop 2			Loop 3		
	Hot Leg	Int. Leg	Cold Leg	Hot Leg	Int. Leg	Cold Leg	Hot Leg	Int. Leg	Cold Leg
SI Accum.									
PZR Surge									
PZR Spray									
RHR Suction									
RHR Return									
Normal Charging									
Loop Fill Conn.'s									
Normal Letdown									
Excess Letdown									
Process Sampling System									

FIG. 6-1

	MODE	113A	114A	113B	114B	
a.	AUTO					(0.4 pts)
b.	DILUTE					(0.4 pts)
c.	ALT. DILUTE					(0.4 pts)
d.	BORATE					(0.4 pts)




THERMOCOUPLE/CORE COOLING MONITOR
$\overline{P1}$
$\overline{P2}$
$\overline{P3}$
AUCTIONEERED PRESSURE
T_H T_C T_H T_C
LOOP 1
LOOP 2
CORE THERMOCOUPLES
PWR SETPT
DEG F MAR
PSI MAR
T SAT P SAT
DISPLAY RESET
ΔT LP 1
ΔT LP 2
IND. SEN.
RUN STOP
PROCESSOR
RESET
TEST
ALARM ACKN
ALARM DISABLE
NORMAL
TEST ENABLE
POWER
POWER
ON
OFF

7. PROCEDURES-NORMAL, ABNORMAL, EMERGENCY & RADIOLOGICAL CONTROL (25.0)

7-1 a. OP-1C states the amount of reactivity by which actual critical rod position may differ from the ECP. What are the allowable deviations (pcm) above and below the ECP? (0.5)

b. What three (3) operator actions are required if it appears that criticality will be achieved at a point below the allowable limit (but still above the minimum insertion limit)? (1.5)

7-2 During a reactor plant startup the reactor coolant system should not be pressurized to $\geq$ 2000 psig if the secondary side steam pressure is $\leq$ 485 psig. TRUE or FALSE? (1.0)

7-3 Assume that a LOCA and safety injection have occurred on your plant.

a. What are the safety injection termination criteria? (2.0)

b. What are the criteria for which SI must be manually reinitiated? (1.2)

7-4 If your plant was operating at 75% power and ΔI was outside of the target flux band on the negative side (beyond the left limit), what <u>control action(s)</u> would you take to restore ΔI to within limits? (1.5)

MEANS RUPTURED.

7-5 a. How is a faulted S/G identified during a S/G tube rupture? (List at least three (3) methods.) (0.9)

MEANS RUPTURED.

b. How is the faulted S/G isolated? (1.5)

c. What methods would be used to reduce RCS temperature to the no-load condition if the faulted S/G MSTV and NRV failed to completely shut? (0.6)

7-6 a. List two (2) symptoms of a #1 seal failure for a RCP. (1.2)

b. List two (2) symptoms of a #2 seal failure for a RCP. (1.2)

c. List one (1) symptom of a #3 seal failure for a RCP. (0.6)

(continued on next page)

7-7 Assume that your plant is in the process of refueling when the Manipulator Crane Area Monitor alarms (at both the alert and high alarm levels).

a. What are the five (5) automatic actions initiated by the high alarm? (1.2

b. What are the six (6) immediate operator actions required? (1.2

7-8 Assume that your plant is operating at 100% power when a Circulating Water Pump trips. What are your three (3) immediate operator actions? Include the numerical limitations that are part of the actions. (1.8

7-9 Does the biological effect of 100 rem dose depend on whether it is a neutron or gamma dose? Explain. (1.2

7-10 Answer the following questions as they relate to you as an operator at Surry Power Station.

a. What are your quarterly administrative limits for radiation doses to the whole body, hands and skin? (1.2

b. What is the maximum quarterly whole body administrative limit that can be approved by station management ? (0.4

c. According to the HP Radiation Protection Manual + G.E.T. a person must meet four ~~three (3)~~ 4 requirements in order to wear a respirator. List these four ~~three (3)~~ 4 requirements. Include time frame if applicable. (1.0

7-11 Assume that you are operating at 100% power and a "Limiting Condition of Operation" is violated which requires the reduction of unit power. State the minimum rate of power reduction that is required OR state the procedure name (or number) that would be used to reference this value. (1.4

7-12 List (in detail) the eleven (11) immediate operator actions and/or verification items for a reactor trip (without SI initiation). Assume that all actions and expected responses are completed without problems. (2.

END OF CATEGORY 7

(Write end of category 7 on your paper)

8-1 Assume that your plant is at 250°F.

a) According to Tech. Specs. what is the maximum number of charging pumps that may be operable? (0.7)

b) What is the basis for the requirement in part 'a' above? (1.0)

8-2 Assume that it is 0300 on 7-20-84 and reactor power is presently at 30%. Considering the ΔI history listed below, at what clock time (and date) are you allowed to increase power above 50%? (3.0)

DATE	TIME (leaving band)	TIME (re-entering band)	POWER(%)
7-19-84	0300	0318	85
7-19-84	1757	1833	55
7-19-84	2238	2400	10
7-20-84	0148	0300	30

8-3 a. Whose approval is required to make a temporary change to an Electrical Maintenance procedure (assuming that the change does not alter the intent of the original procedure)? (1.0

b. Whose approval is required to make a temporary change to an Emergency procedure (assuming a change of intent is involved)? (1.0)

8-4 a. Give an example of a "Class I" Reactor Trip. (1.0)

b. Give an example of a "Class II" Reactor Trip. (1.0)

8-5 Instructions on the use of temporary jumpers indicate that if safety-related equipment is involved then additional approval is required before installing the jumper. What reference (procedure name and/or number should be used to determine whether or not a piece of equipment is safety-related? (1.5)

8-6 a. Service water leaks within the containment (when the RCS is >200°F) require Immediate NRC notification. What is the basis for this requirement? (1.0)

b. List (or describe) six (6) additional events that require Immediate NRC notification. (2.4)

(continued on next page)

8-7 ADM-38 indicates that entry into the reactor containment during reactor operation exposes personnel to four (4) distinct hazards.

a. List these four (4) hazards. (2.4)
b. Whose permission is required for containment entry? (1.0)

8-8 List the three (3) requirements for placing a "temporary Control Board Marker" on a control board OR describe the reference book/manual and procedure title that you would use to determine these requirements. (1.5)

8-9 Assume that an unplanned radiological airborne release is in progress due to an accident on Unit 1.

a. If the Control Room Meterological wind direction recorder indicates a reading of 180° in what direction would the plume begin traveling (N,NE,E,SE,S,SW,W, or NW)? (1.0)

b. Assuming the same conditions as in "a" above, what would be the entry for the Status Board "wind direction"? (1.0)

8-10 ~~How often are Vital Area Keys inventoried and by whom?~~ DELETED (AFTER EXAM) (1.5)

8-11 Tech. Spec. 3.21 requires that all fire barrier penetrations in the fire zone boundaries protecting safety related areas shall be functional. What action would be required if a firedoor was required to be open for a two hour period so that an air hose could be routed for maintenance work? (1.5)

8-12 List five (5) unique instruments that are considered part of the Tech. Specs. "Accident Monitoring Instrumentation". (1.5)

END OF CATEGORY 8

(Write end of category 8 on your paper)

EQUATION SHEET

$\dot{Q} = \dot{m}\,\Delta h$

$\dot{Q} = UA\,(T_{avg} - T_{stm})$

$\dot{Q} = \dot{m}\,c_p\,\Delta T$

$h_L = k\,\dot{V}^2$

$DNBR = \dfrac{Q_c}{Q_x}$

$P = P_o\,10^{SUR(t)}$

$P = P_o\,e^{t/\tau}$

$SUR = \dfrac{26.06}{\tau}$

$\tau = \dfrac{\beta - \rho}{\lambda\,\rho}$

$\tau = \dfrac{\ell^*}{\rho} + \dfrac{\beta - \rho}{\bar{\lambda}\,\rho}$

$\rho = \dfrac{K_{eff} - 1}{K_{eff}}$

$\Delta\rho = \dfrac{K_2 - K_1}{K_2\,K_1}$

$\dfrac{CR_1}{CR_2} = \dfrac{1 - K_{eff_2}}{1 - K_{eff_1}}$

$RR = \Sigma_f\,\phi_{th}$

$SCR = \dfrac{S}{1 - K_{eff}}$

$M = \dfrac{CR_1}{CR_0}$

$A = \lambda N$

$\lambda = \dfrac{\ln 2}{t_{1/2}}$

$N = N_o\,e^{-\lambda t}$

5-1 a. lower than estimated
b. lower than estimated
c. higher than estimated
d. lower than estimated
e. no difference

REF.: Surry OP-1C

5-2 Equal to

REF.: Westinghouse Reactor Physics, Section I-5

5-3 Greater than; $\overset{\leftrightarrow}{RR} = \overset{\downarrow}{\Sigma_f}\ \overset{\uparrow}{\phi_{th}}$

REF.: Westinghouse Reactor Physics, Section I-2

5-4 b.

REF.: Westinghouse Reactor Theory, Section I-5

5-5 False (power will be 3.16×10^{-8})

REF.: Westinghouse Reactor Physics, Section I-3

5-6 See Figure 5-1

REF.: Westinghouse Reactor Physics I-5 & Surry Curve Book

5-7 d.

REF.: Westinghouse Reactor Physics, Section I-5

5-8 a. $1/M = CR_1/CR_2$

$= 400/1857$

$\doteq 0.215$

b. $1/M = 1\text{-}Keff_2/1\text{-}Keff_1$

$0.215 = (1-K_{eff_9})/(1-.94)$

$(0.215)(0.06) = 1- K_{eff_9}$

$1-0.0129 = K_{eff_9}$

$0.9871 = K_{eff_9}$

REF.: Westinghouse Reactor Physics, Section I-4

(continued on next page)

5-9 Assume: -9 pcm/ppm boron worth
$\lambda eff = 0.1$
$\beta eff = 0.007$

a. (-9 pcm/ppm) (-15ppm) =135pcm = .00135
$\tau = \beta eff - \rho/\lambda eff\ \rho$
$\tau = .007 - .00135/.1\ (.00135)$
$\tau = .0057/.000135$
$\tau = 41.85$ sec.

SUR = $26.06/\tau$
SUR = 26.06/41.85
<u>SUR = .62 DPM</u>

b. $P = P_o 10^{SUR(t)}$
$2 \times 10^{-5} = (5 \times 10^{-9})(10)^{(0.62t)}$
$4000 = 10^{0.62t}$
$3.602 = 0.62t$
$t = 5.81$

REF.: Westinghouse Reactor Theory, Sections I-3 & I-5

5-10 a. decrease
b. increase
c. increase
d. decrease

REF.: General Physics HT & FF, Section II-C

5-11 a. decrease
b. increase
c. decrease
d. increase

REF.: General Physics HT & FF, Section III-B

5-12 See Fig. 5-2
a. essentially the same
b. appx. 76 psig

REF.: General Physics HT & FF, Section III-B

5-13 See Fig. 5-3.

REF.: Surry EP.1.02A & Shearon Harris Simulator Data

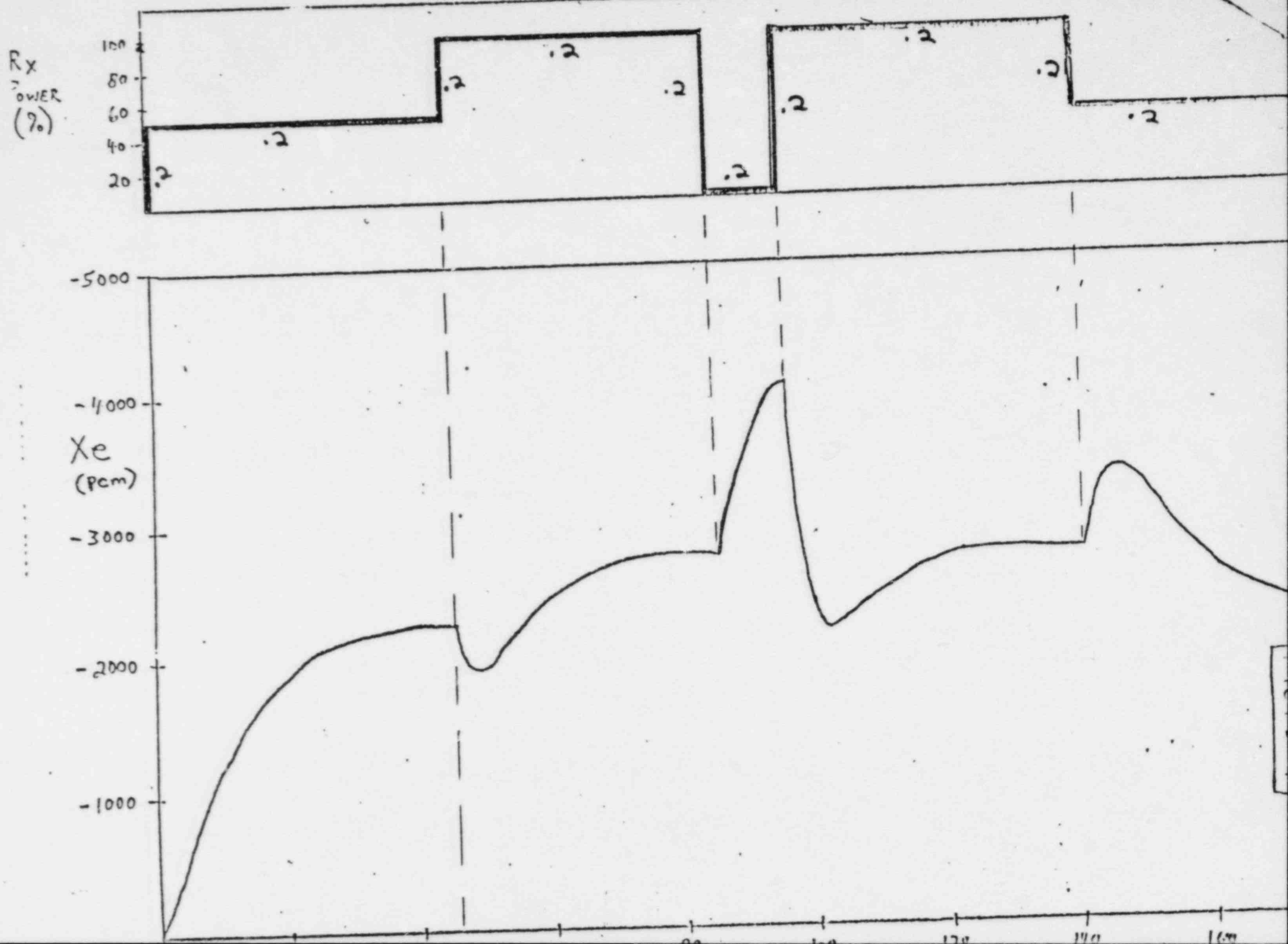
(continued on next page)

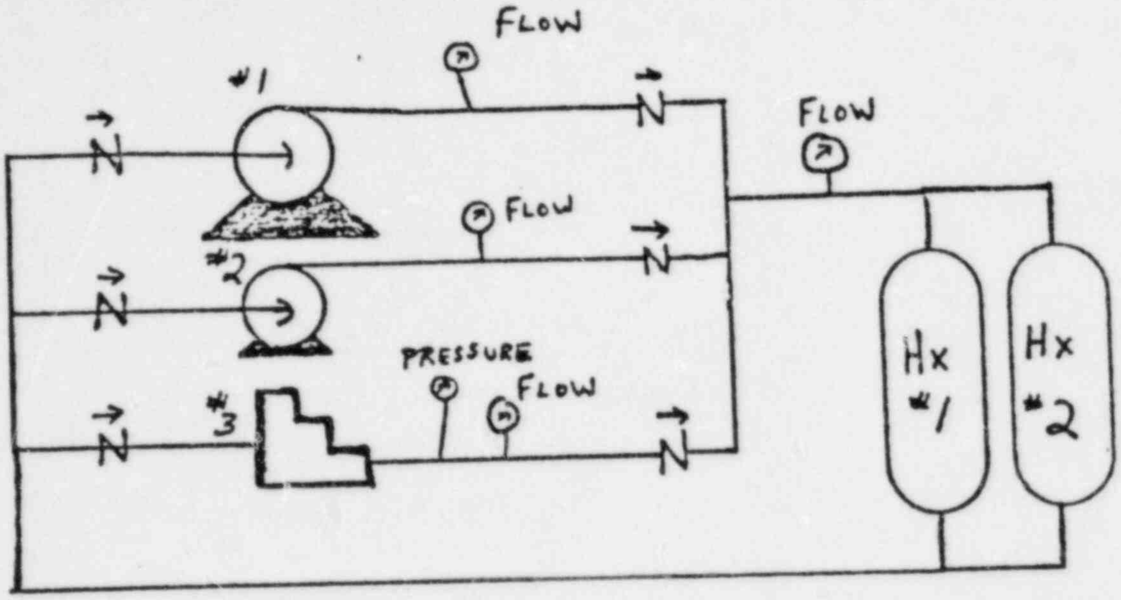
5-14 652 Sat. temp. for 2250 psia
-547 Tavg at zero power ($\neq T_H$)
105° Subcooled at zero power

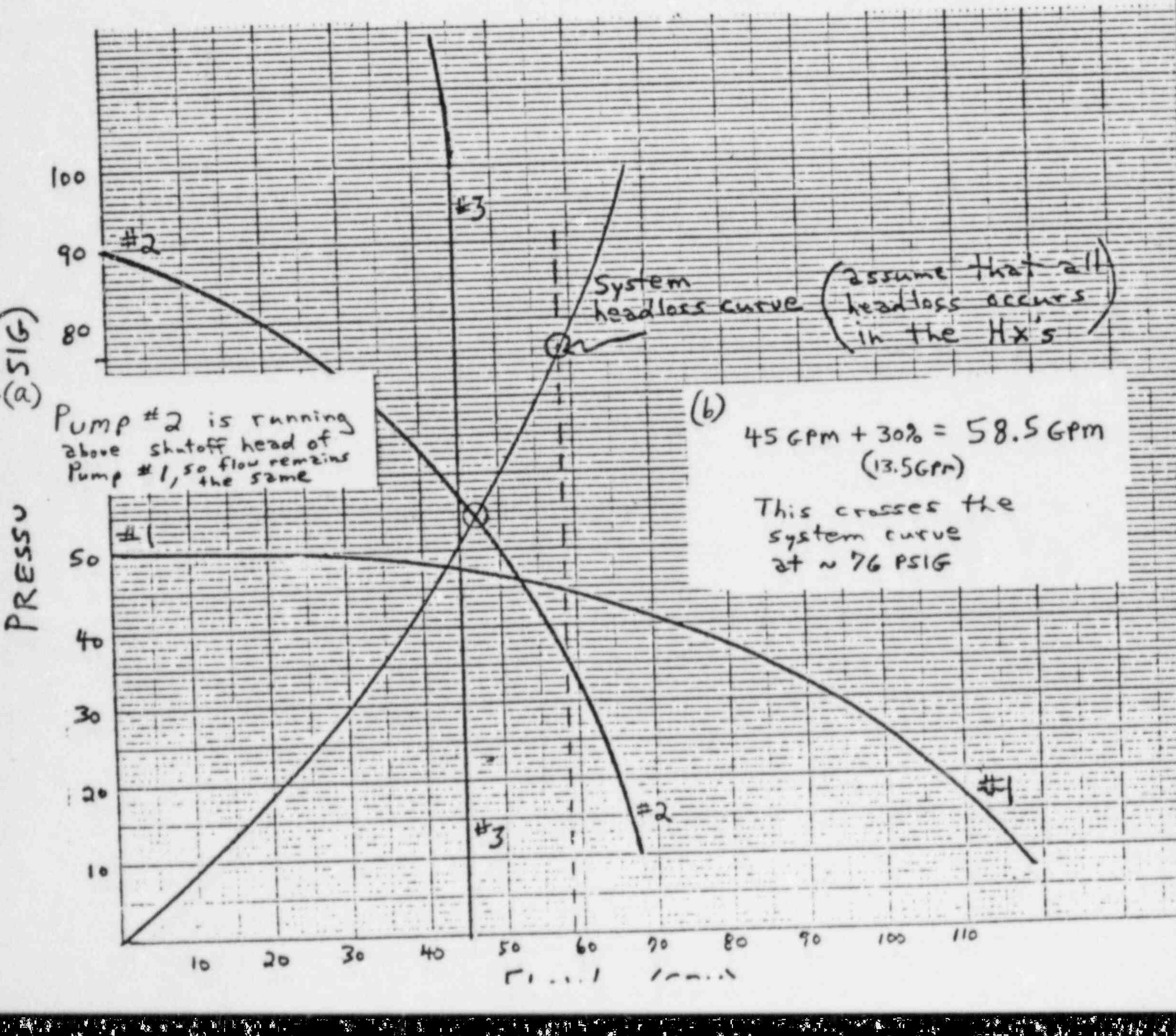
652 Sat. temp. for 2250 psia
-606 Operating T_H at 100% power
46°F Subcooled at 100% power

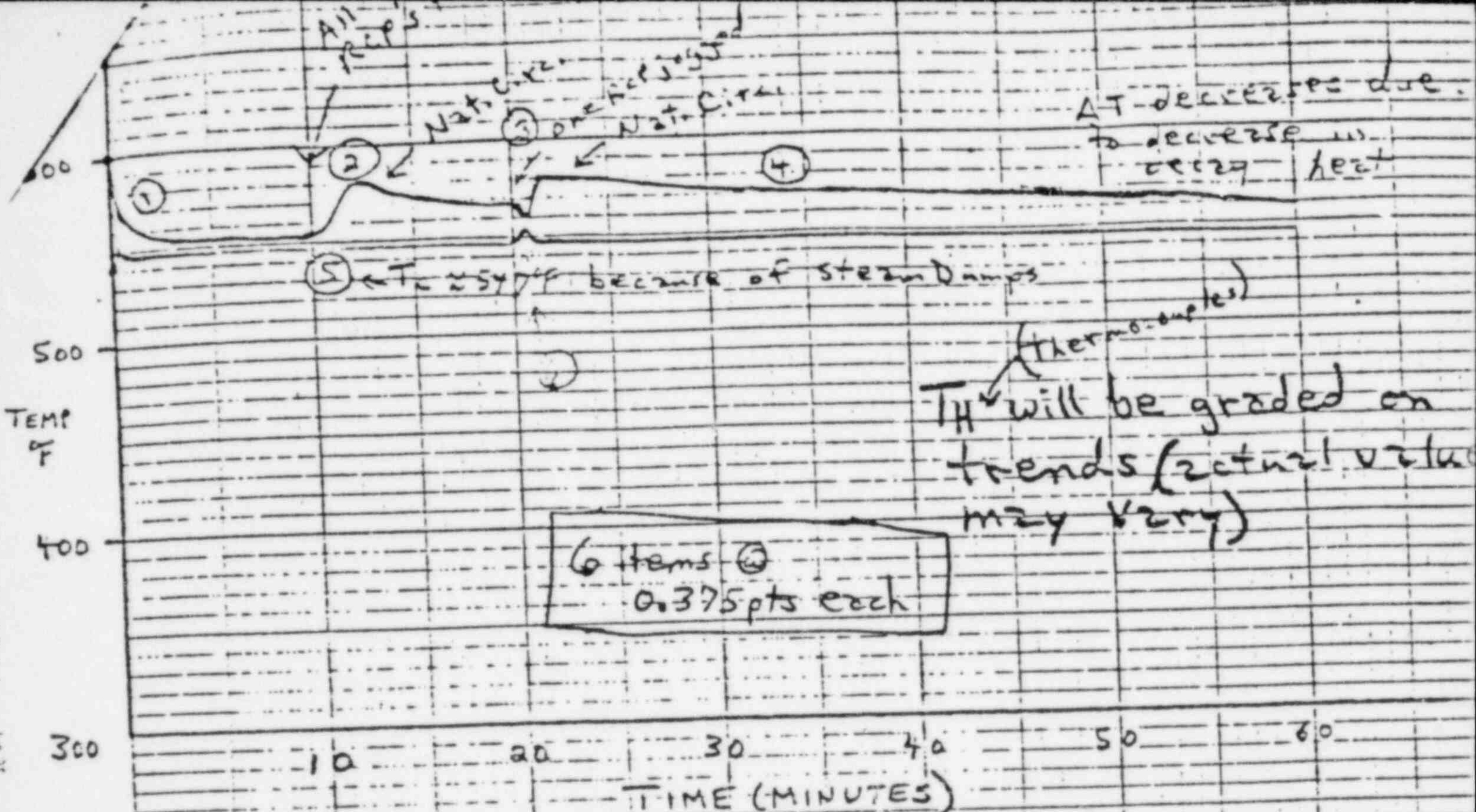
REF.: Steam Tables and Surry System Des. Chap. 1

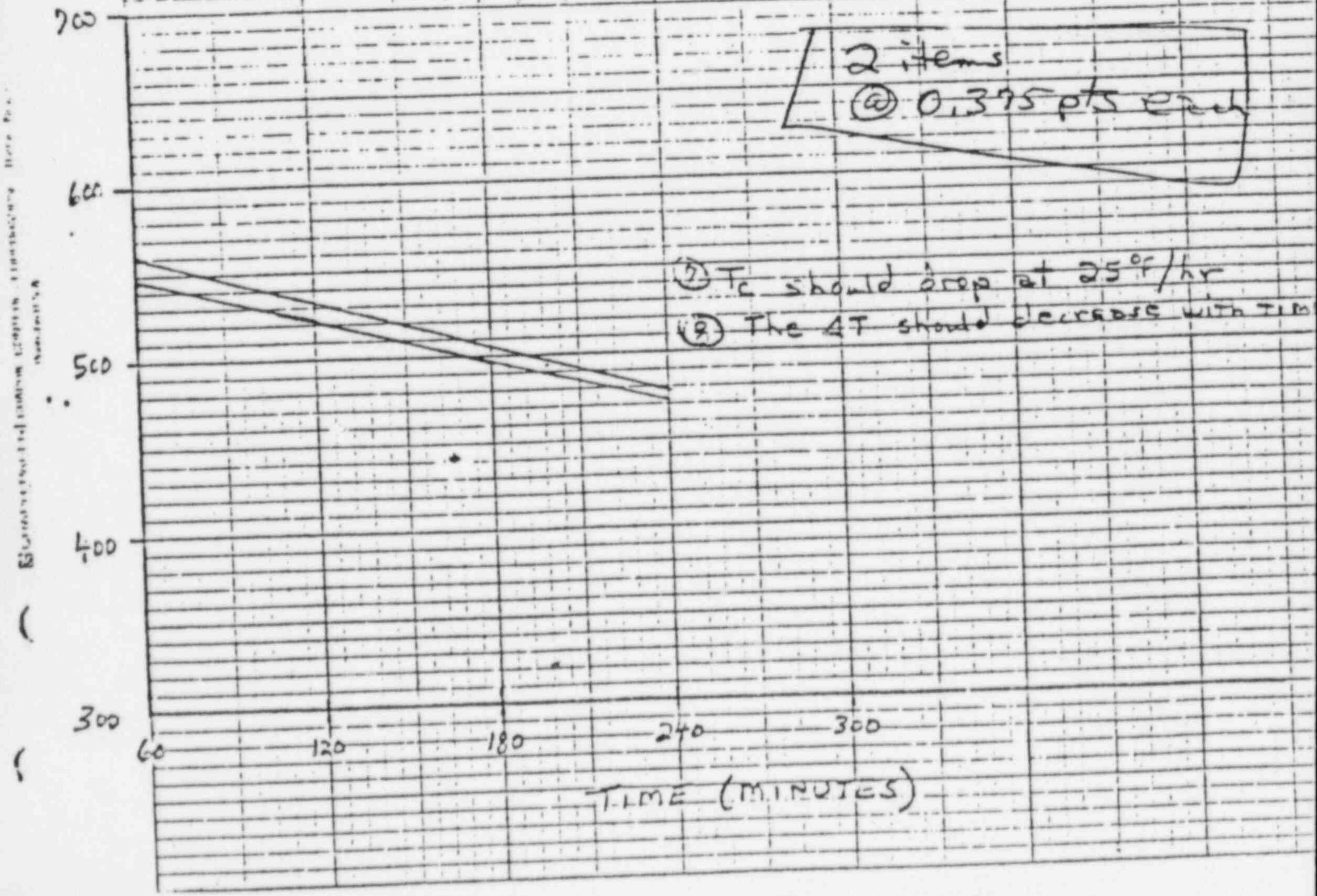
(ACTUAL VALUES NOT REQ'D, APPROXIMATE VALUE




FLOW
#1
FLOW
FLOW
#2
PRESSURE
FLOW
#3
Hx #1
Hx #2


PRESSU (PSIG)
100
90
80
50
40
30
20
10
10 20 30 40 50 60 70 80 90 100 110
#1
#2
#3
System headloss curve
(assume that all headloss occurs in the Hx's)
(a)
Pump #2 is running above shutoff head of Pump #1, so flow remains the same
(b)
45 GPM + 30% = 58.5 GPM
(13.5 GPM)
This crosses the system curve at ~ 76 PSIG


All RCP's
Nat. Circ.
Nat. Circ.
ΔT decreases due to decrease in decay heat
1
2
3
4
600
5 ← Tc ≈ 547°F because of steam Dumps
(thermocouples)
TH will be graded on trends (actual value may vary)
500
TEMP °F
400
6 items @ 0.375 pts each
300
10
20
30
40
50
60
TIME (MINUTES)


700
2 items @ 0.375 pts each
600
7 Tc should drop at 25°F/hr
8 The ΔT should decrease with time
500
400
300
60
120
180
240
300
TIME (MINUTES)

6-1 (.75) a. True
(.75) .5 b. CAF - Reference not available Control Room (left hand side of backboard - "VB 2")

REF.: Surry Instrumentation Manual, chap. 1, pgs. 14 & 15

6-2 (2.5) See Fig. 6-1

REF.: Surry Sys. Des. Vol. I., RCS, pgs. 1 & 1B; Vol. II, Chap. 10 pgs 1 & 2

6-3 (0.5) a. NaOH
(0.75) b. Helps remove iodine from the containment atmosphere
(0.75) c. Hi-Hi containment pressure 3/4 detectors $\geq$8.3 psig

REF.: Surry Sys. Des. Vol I, CLS-pg1 & Cont. Spray -pg2

6-4 (3.0) See Fig. 6-2

REF.: Surry OP-26.4 pg2; AP-11 pgs 1-3; & Sys. Des. Vol II-Emerg. Power Distribution

6-5 (2.7)

	Trip (0.2 pt)	Stpt. (0.1 pt)	Logic (0.1 pt)
1.	Manual		1/2
2.	PRM Hi Flux High Setpoint	107%	2/4 det.
3.	PZR High Pressure	2370#	2/3 det.
4.	Feed Flow/ Steam Flow Mismatch with Low S/G Water Level	20% level $.709 \times 10^6$ lbm/hr	1/2 det 1/2 channels
5.	S/G Low-Low Level	17%	2/3 det.
6.	OTΔT	Variable	2/3 channels
7.	OPΔT	Variable	2/3 channels

also acceptable SI signals from Hi Cont. Press (3/4) & manual (1/2)

REF.: Surry Instrumentation Manual Chap. 7, pg. IV-1.12; Chap. 8, pg IV-8.7; Chap. 9, pg. IV-5.7; Chap. 11, pgs. 2 & 3; Chap. 14, pgs. 1,2,4, 5, & 6

(continued on next page)

6-6 (1.6) See Fig. 6-3

REF.: Surry Sys. Des. Vol I, CVCS pgs 48 & 52

1402 (C loop)
1402-1 (B loop)

B? ✓ C? ✓

6-7 (0.75) a. Loop (CAF) wide range pressure, loop (CAF) wide range pressure, and pressurizer pressure (Prot. channels II + III considered narrow range) [one pzr + one WR per monitor channel]

(0.5) b. The divisions indicate quadrant location in the core (two TC's per quadrant).

(1.0) c. 5 & 6 (Avg. minus T_H) & (Avg. minus T_C)

REF.: Surry Subcooling Monitor Sys. Lesson Plan from North Anna and Surry OP 5.5. Surry Sys Des. RCS pg 39 & DWG 11448-FM-86A

6-8 (2.8) 1. Pzr Hi Press. Alarm (2310 psig on PT-445)

Info. in parenthesis not required for full credit

2. PCV-456 opens causing pressure to drop. (PORV)

3. Proportional heaters are full on (appx. 2220 psig).

already on → 4. ~~Backup heaters on (appx. 2210 psig).~~ (delete from exam)

5. As pressure drops below 2000 psig (as sensed by 2/3 of the protection channel pressure detectors) an interlock will cause PCV-456 to shut.

~~7 items @ 0.4 pts each~~

6. As pressure increases above 2000 psig PCV-456 will re-open

6 items @ 0.47 each

7. The pressure should continue cycling around 2000 psig. (controller action will cause the cycling effects).

REF.: Surry Sys. Des. RCS pgs 39, 41, 69, & 70 and Inst. Manual Chap. 10 attachment I, and Chap. 14 pgs 5 & 10

6-9 (1.5) See Fig. 6-5

5 valves (0.3 pts) e

The AFW pump discharge valves that tie into the same header as MOV-FW-151-A must be shut. CAF for valve numbers. FW-141, 156, + 171 + MOV-FW-15A-C + E must be shut to prevent backflow from other train.

REF.: Surry Sys. Des. Main Feed and AFW; AP-21, pg 4; EP-30, pgs. 4 & 5; FRP-H.1, pg. 3 Vlv# Ref: DWG 11448-FM-68A, 68B, + 18A

6-10 (1.0) Spray water, so that no dilution of the borated water can occur.

REF.: Surry Sys. Des. Spray Subsystem Recirculation, pg. 2

6-11 Only the underlined words are required for full credit.

a. Steam flow increases causing increased removal of heat from the RCS; Tave decreases. Reactor Control System moves the rods out because of the Tref-Tave deviation.
(.55)

b. This causes decreased efficiency in the secondary plant cycle for the same turbine load output; Tave will decrease because of greater heat removal. The Tref-Tave deviation causes a rod withdrawal.
(.55)

c. The nuclear power input signal to rod control increases and will cause the rods to insert attempting to reduce reactor power, while the channel is failing. This is due to the power mismatch between Nuclear power and Turbine Power (impulse press).
(.55)

REF.: Surry Instrumentation Manual Chap. 4. fig. 1

6-12 a) 1) Slack cable: stops downward motion at < 600 lbs., lights the slack cable lamp (0.5)

2) Gripper - weight indicators: the air line to the gripper control knob is closed by a solenoid valve until the load drops below 1200 lbs. (0.5)

3) Overload : Stops the hoist in the up direction and lights the overload lamp when load is >2700 lbs. (0.5)

b) 1) Slack cable: not bypassable (0.25

2) Gripper-weight indicators: bypassable with the "gripper bypass" (which is a valve in the pneumatic system that delivers air directly to the grippers around the weight indicator interlock). A "Gripper By-Pass" lamp will light. (0.

3) Overload: bypassable with the "overload bypass". An "Overload By-Pass" lamp will light. (0.2

c. The interlocking switches in the weight indictor trip in relation to position of the needle on the scale and not by absolute weight. If the meter is switched to the "B" scale the switch settings increase by 3000 lbs. (0.5)

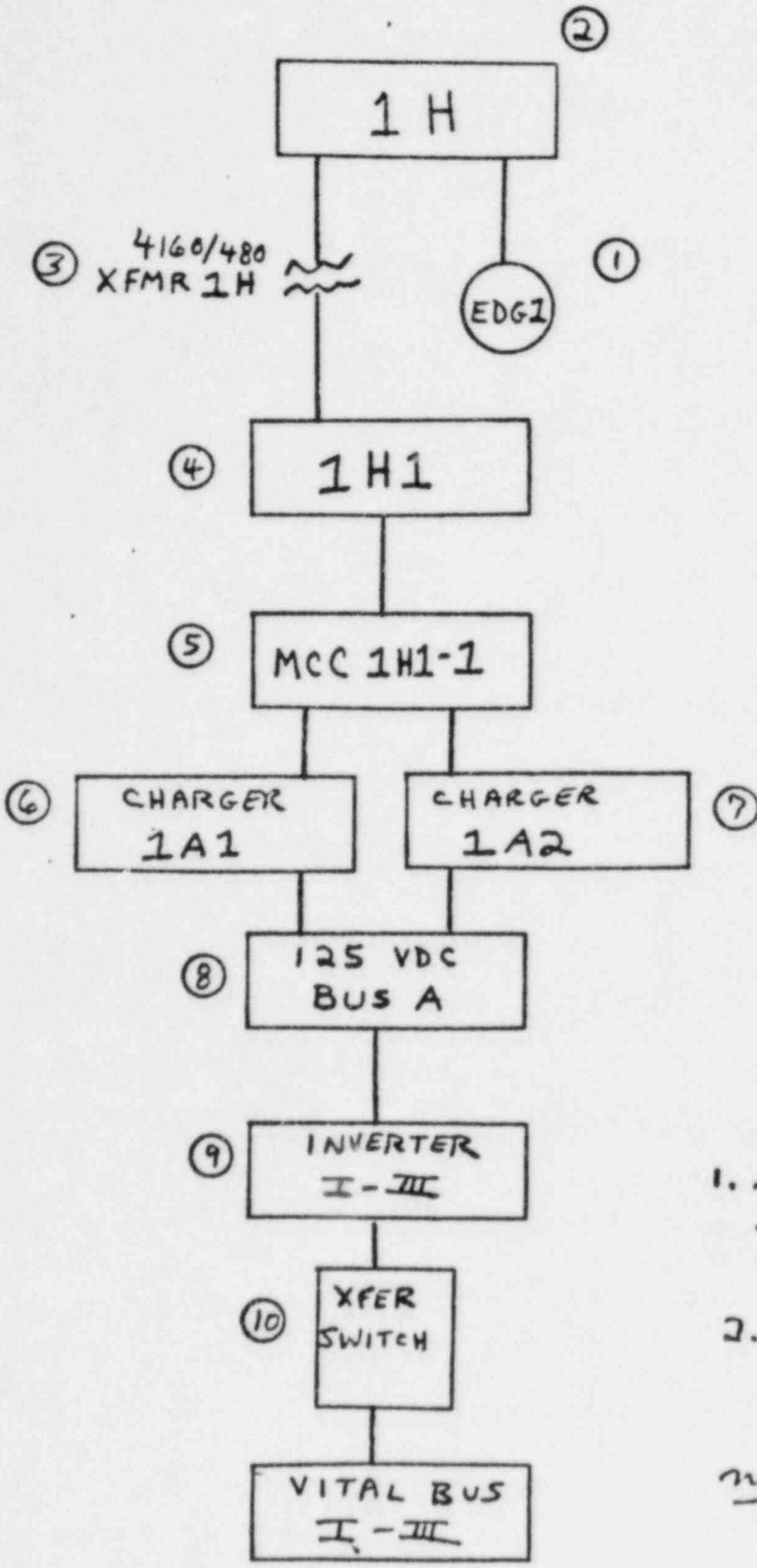
REF.: Surry Sys. Des. Fuel Handling Sys. (pgs 9B, 11, & 15) and OP-4.15 (pgs 3 & 6)

	Loop 1			Loop 2			Loop 3			
	HOT LEG	INT. LEG	COLD LEG	HOT LEG	INT. LEG	COLD LEG	HOT LEG	INT. LEG	COLD LEG	
SI ACCUM.			X			X			X	(0.25)
PZR SURGE							X			(0.25)
PZR SPRAY			X						X	(0.25)
RHR SUCTION	X									(0.25)
RHR RETURN						X			X	(0.25)
NORMAL CHARGING						X				(0.25)
LOOP FILL CONN.'S		X			X			X		(0.25)
NORMAL LETDOWN			X							(0.25)
EXCESS LETDOWN		X			X			X		(0.25)
PROCESS SAMPLING SYSTEM	X		X	X		X	X		X	(0.25)

FIG. 6-1

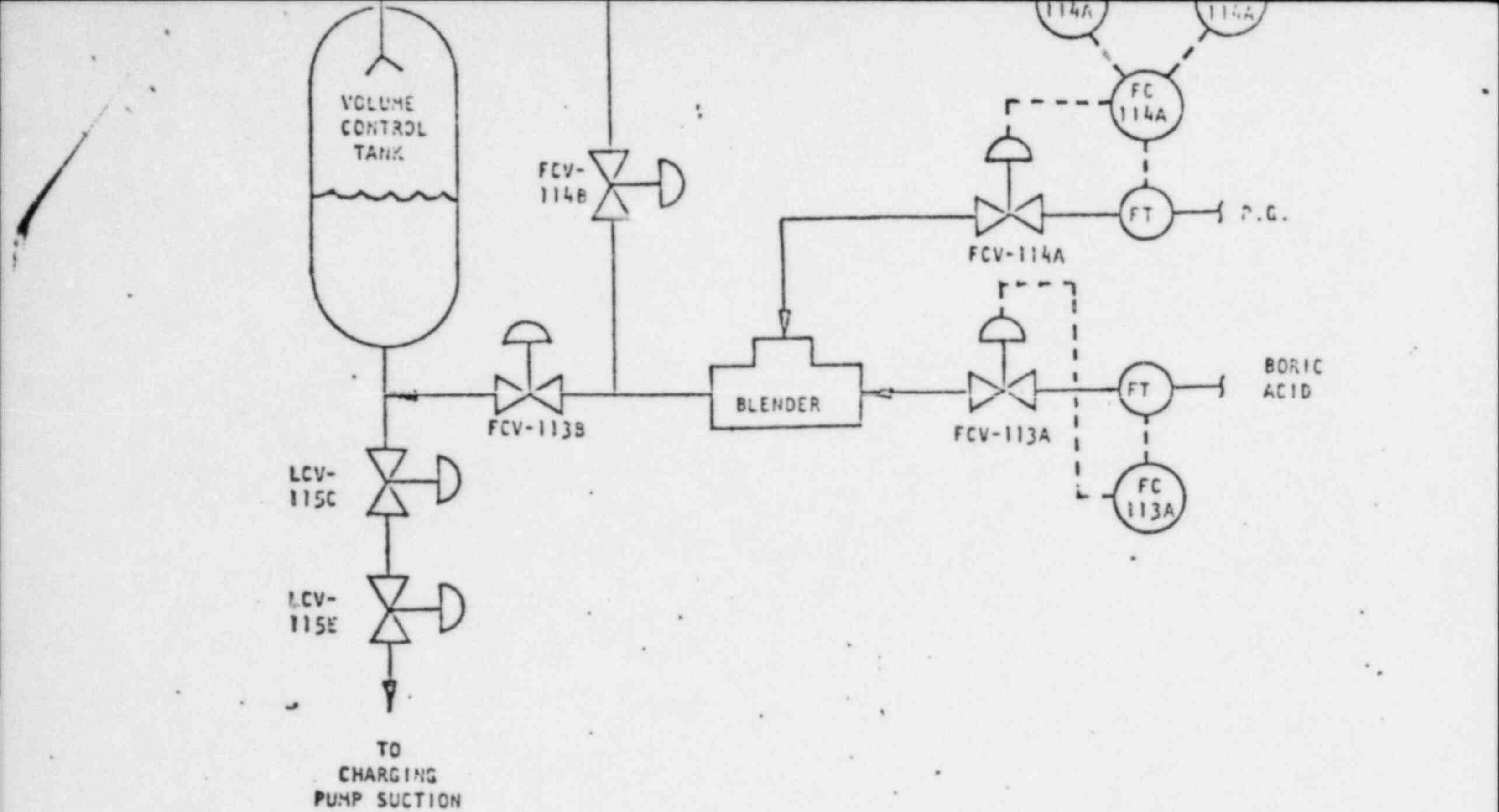
9 ~~10~~ items
@ ~~0.3~~ pts each

0.33 pts each



1. must be correctly labeled (ie proper alphanumeric designation and component name)
2. must appear in the correct sequence order

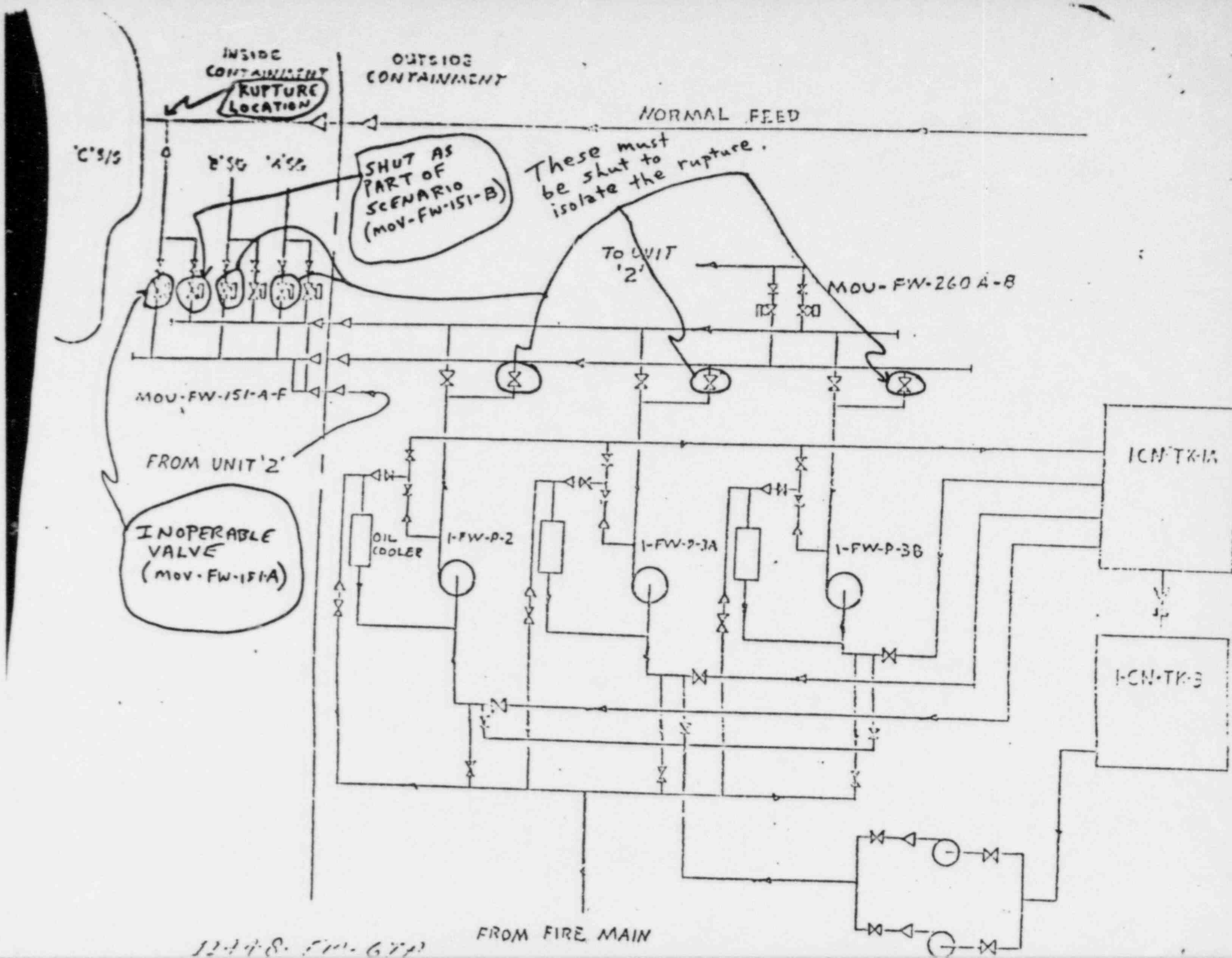
note: no single error (such as one item out of sequence) will be graded so as to cause double jeopardy.



	MODE	113A	114A	113B	114B	
a.	AUTO	MODULATED	MODULATED	OPEN	CLOSED	(0.4 pts)
b.	DILUTE	CLOSED	MODULATED	CLOSED	OPEN	(0.4 pts)
c.	ALT. DILUTE	CLOSED	MODULATED	OPEN	OPEN	(0.4 pts)
d.	BORATE	MODULATED	CLOSED	OPEN	CLOSED	(0.4 pts)

Fig. 6-3

For credit in each of the parts a → d all four valves must indicate the correct position. No partial credit given.


INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
RUPTURE LOCATION
NORMAL FEED
'C' S/G
'B' SG
'A' SG
SHUT AS PART OF SCENARIO (MOV-FW-151-B)
These must be shut to isolate the rupture.
TO UNIT '2'
MOV-FW-260 A-B
MOV-FW-151-A-F
FROM UNIT '2'
INOPERABLE VALVE (MOV-FW-151-A)
OIL COOLER
1-FW-P-2
1-FW-P-3A
1-FW-P-3B
1-CN-TK-1A
1-CN-TK-3
FROM FIRE MAIN

7-1 a. 400 pcm above, 250 pcm below (0.5)

b. 1. Insert all controlling group rods. (0.5)
2. Re-evaluate the ECP. (0.5)
3. If no ECP error is detected, obtain the operations superintendent's permission to approach criticality using a 1/M plot. (0.5)

REF.: Surry 1-OP-1C page 5

7-2 (1.0) True (ΔP SI will occur)

REF.: Surry OP-1.3 pg. 9 and Instrumentation Manual

7-3 a. 1. Containment Conditions - Normal, and
2. RCS Pressure - >2000 PSIG, and
3. RCS Subcooling - $\geq$50°F, and
4. Pzr Level - $\geq$50%, and
5. Heat sink - S/G Level - >65% WR or 17% NR
or AFW Flow - >540 GPM

5 items @ 0.4 pts

b. 1. RCS Pressure - <1715 PSIG or
2. RCS Subcooling - <50°F, or
3. Pzr Level - <20%

3 items @ 0.4 pts each

REF.: Surry EP-2.00 pgs. 4 and 16

7-4 (1.5) Borate to drive the rods out which will cause power to shift toward the top of the core or if rods are all the way out, reduce load. As load decreases power shifts toward the top of the core.

7-5 a. 1. Unexpected rise in any S/G narrow range level.
2. High radiation from any S/G blowdown line.
3. High radiation from any MS line monitor.
4. High radiation as determined by sampling and analysis.

3 items req'd @ 0.3 pts ea.

b. 1. Shut AFW to ruptured S/G if NR level >50%
2. Shut ruptured S/G MSTV
3. Verify that ruptured S/G PORV is shut
4. Shut ruptured S/G steam supply to the TDAFW pump
5. Verify ruptured S/G blowdown TV's are shut

5 items @ 0.3 pts ea.

c. 1. Shut non-ruptured S/G MSTV's
2. Use non-ruptured S/G PORV's for steam dump

2 items @ 0.3 pts each

REF.: Surry EP-4.0 pgs. 3 and 4

(continued on next page)

7-6 a. 1. RCP seal leakoff high flow
(1.2) 2. RCP seal water return low ΔP

b. 1. RCP Vapor seal tank high level
(1.2) 2. RCP seal leakoff low flow

c. Vapor seal tank low level
(0.6)
REF.: Surry Annuciator Procedures (1C-4,1C-5,1B-8, 1C-44, & 1B-32).

7-7 a. 1. purge air supply valves (MOV-VS-100A and B) close
2. purge air exhaust valves (MOV-VS-100C and D) close
3. supply fans (VS-F-4A- and B) trip
4. TV-1A-101A and B close
5. Alternate suction (ADV-1A-103) opens
AOV

5 items @ 0.25 pts each

Info. in parenthesis not req'd for full credit

b. 1. Verify auto actions occurred
2. Advise all personnel to leave area (and report at personnel change and decontamination area)
3. Start contamination air recirc. fans (VS-F-1A-and B)
4. Start iodine filtration fans (VS-F-3A and 3B)
5. Notify H.P. for survey
6. Turn control selector switches for supply fans (1-VS-F-4A- and B) to "OFF"

6 items @ 0.2 pts each

REF.: Surry AP-5.8

7-8 1. Throttle condenser outlet valves (0.4 pts) manually to maintain normal level (0.2 pts) of 22 feet, do not exceed 15°F ΔT. (0.2 pts)

2. (0.4 pts) Reduce generator load to allow condenser vacuum to be maintained (0.2 pts) above the trip point (18"-22") and temp. at the groin not to exceed 98°F. (0.2 pts)

3. (0.2 pts) Notify the System Operator of the reason for reduced load.

REF.: Surry AP-13

7-9 0.25 ← No. | → 1.0 pts 100 rem in neutron or gamma is the same because the quality factor of the ionizing radiation is already figured in rem, because rem = rad x quality factor.

REF.: Surry H. P. Rad. Prot. Manual page 1.2-2

(continued on next page)

7-10 a. WB - 0.75 rem/qtr. (0.4 pts)
Hands - 18.75 rem/qtr. "
Skin - 5.00 rem/qtr. "

b. 1.75 rem/qtr. (0.4)

c. 1. WB Count ≤ 12 months ago
2. Satisfactory Respirator Fit Test ≤ 12 months ago
3. Satisfactory Pulmonary Function Test ≤ 12 months ago
4. Sat. Respiratory Protection Training ≤ 12 months ago

Time frame ~~(0.1)~~ 0.2
~~3~~ 4 Requirements ~~(0.3 pts)~~ each 0.2 pts

REF.: Surry HP Manual - Section I pgs. 1.2-3 and 4
G.E.T. manual pg 40

7-11 150 Mwe/hour OR OP-2.1 (Unit Power Operation)
(1.0)

REF.: Surry OP-2.1

7-12
1. Manually trip reactor.
2. Verify rod bottom lights are lit.
3. Verify rod position indicators are indicating zero.
4. Verify that neutron flux is decreasing.
5. Manually trip the turbine.
6. Verify turbine stop valves are closed
7. Verify rapid load decrease to zero.
8. Open turbine drains
9. "Reset" reheaters.
10. Verify normal voltages on buses H and J.
11. Verify SS buses energized.

11 items @ 0.2 pts each

REF.: Surry EP-1.00

8-1 (0.?) a. one (1)
(1.0) b. Provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or equivalent.

REF.: Surry T.S. 3.1 - 23& 25

8-2
18 x 1.0 = 18 penalty minutes (0.5) pts
36 x 1.0 = 36 penalty minutes (0.5) pts
82 x 0.0 = 0 penalty minutes (0.5) pts
72 x 0.5 = 36 penalty minutes (0.5) pts

90 total penalty minutes

@ 0318 on 7-20-84 there would be 72 penalty minutes
@ 1809 on 7-20-84 there would be 60 penalty minutes } 1.0 pts
and power could be increased to >50%

REF.: Surry T.S. 3.12-4 & 5

8-3 (1.0) a. Electrical Foreman and a licensed SRO
(1.0) b. Superintendent - Operations

REF.: Surry T.S. 6.4 - 4 & 5

8-4 a. Any trip which meets the following:

(1.0) The trip is clearly understood and corrected and no significant malfunctions of safety-related or important equipment occurred.

b. Any trip which meets the following:

(1.0) The cause of the trip is not clearly or completely known or safety-related and/or other important equipment functioned in an abnormal or degraded manner during or following the trip.

REF.: Surry ADM-14 page 2

8-5 (1.5) Use ADM-73, Classification of Systems, Components, and Structures. Charts are available which list most equipment and when a direct determination cannot be made then use the evaluation process described in the procedure.

(Only unde portion for full

REF.: Surry ADM-73 and ADM-29, pg. 16

(continued on next page)

8-6 (1.0) a. Degradation of containment boundary
(2.4) b. Any six (6) of the twelve (12) items listed on figs. 8-1 thru 8-3 (CAF for recent revisions)

REF.: Surry ADM-29, pgs 40-44

8-7 a. (2.4)
1. ionizing radiation
2. heat stress
3. differential pressure
4. potential oxygen deficiency

(1.0) b. Station Manager, Asst. Sta. Manager, Supt. Ops., or SRO on call. (Any one of the four receives full credit.)

REF.: Surry ADM-38

8-8 (1.5)
1. cannot hide any indicator
2. must be dated
3. must have name of person posting it.

OR

1. Use Standing Order Book (S.O. #12) "Use of Temporary Control Board Marker".

REF.: Surry S.O. #12

8-9 (1.0) a. ~~X~~ N
(1.0) b. ~~X~~ S (or to prevent double jeopardy, 180°F from answer 'a' will be accepted)

REF.: Surry EPIP-2.01

8-10 (1.5) ~~Once each shift, by the Shift Supervisor or S.S. Asst.~~ DELETED

REF.: Surry S.O. #15

8-11 (1.5) Within one hour establish a continuous fire watch on at least one side of the affected penetration.

REF.: Surry T.S. 3.21-3 & 4

8-12 (1.5)
1) AFW flow rate
2) RCS subcooling monitor
3) PORV position - acoustic
4) PORV position - light
5) Safety valve position - acoustic
6) Safety valve position - ~~light~~ temp. indicator

REF.: Surry T. S. 3.7-8 and 3.7-21

FIG. 8-1

Full credit for any 6 of the 17 items listed in figs 8-1 thru 8-3

1. Any event requiring initiation of the licensee's emergency plan or any section of that plan.
2. The exceeding of any Technical Specification Safety Limit.
3. Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shutdown.
4. Any act that threatens the safety of the nuclear power plants or site personnel, or the security of special nuclear material, including instances of sabotage or attempted sabotage.
5. Any event requiring initiation of shutdown of the nuclear power plant in accordance with Technical Specification Limiting Conditions for Operation.

FIG. 8 2

6. Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems, and components important to safety that are needed to: (i) shutdown the reactor safely and maintain it in a safe shutdown conditions, or (ii) limit the release of radioactive material to acceptable levels or reduce the potential for such release.
7. Any event resulting in manual or automatic actuation of Engineered Safety Features, including the Reactor Protection System.
8. Any accidental, unplanned, or uncontrolled radioactive release. (Normal or expected release from maintenance or other operational activities are not included.)

FIG. 3-3

9. Any fatality or serious injury occurring on the site and requiring transport to an off-site medical facility for treatment. Item (9) of 50.72(a) requires notification of: "Any fatality or serious injury occurring on the site and requiring transport to an offsite medical facility for treatment." Serious injury is considered to be an injury that the judgement of the licensee representative will require admission of the injured individual to a hospital for treatment or observation for an extended period of time (greater than 48 hours). Injuries that only require treatment and/or medical observation at a hospital or offsite medical facility, but do not meet the conditions specified above, are not requi ed to be reported.
10. Any serious personnel radioactive contamination requiring extensive onsite decontamination or outside assistance.
11. Any event meeting the criteria in Part 20 for immediate or 24-hour notification of incidents.
12 Strikes of operating employees or security guards, or honoring of picket lines by these employees.

3.0 Instructions (continued)

3.6 Reporting

3.6.1 The Superintendent of Operations or the S.R.O. on call shall be notified in the event of:

A. Reactor Trip

B. Loss of Site Power

C. Major Equipment Failure

D. Personnel injury requiring medical attention

E. Any abnormal water chemistry changes

F. Reportable Occurrence

G. Any significant oil spill to the environment

H. Unusual Safety Related Event

I. Or any significant occurrence.

3.6.2 Immediate NRC Notification (10CFR50.72)

(a) The Shift Supervisor shall notify Operations Center via the Emergency Notification System whenever any Emergency or Non-Emergency event, as specified below occurs.

(b) Whenever the NRC is notified pursuant to this section, a "station deviation" report will be initiated. When making this notification, review and utilize Checklist No. 67 to provide the NRC with applicable information. Attach this checklist to the "station deviation" report.

(c) Whenever the NRC is notified, the Shift Technical Advisor (STA) shall be summoned to the control room to initiate a preliminary investigation of the event.

3.0 Instructions (continued)

Emergency Events

The NRC shall be notified within one hour following the declaration of any of the Emergency Classes, i.e. Unusual Event, Alert, Site Emergency, or General Emergency.

Non-Emergency Events

(A) Notify the NRC within one (1) hour of any of the following:

(1) The initiation of any plant shutdown required by Technical Specifications.

(2) Any event or condition during operation that results in the condition of the plant, including its principal safety barriers, being seriously degraded; or results in the plant being:

(A) In an unanalyzed condition that significantly compromises plant safety;

(B) In a condition that is outside the design basis of the plant; or

(C) In a condition not covered by operating and emergency procedure.

(3) Any natural phenomenon or other external condition that poses an actual threat to the safety of the plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.

(4) Any event that results or should have resulted in Safety Injection System discharge into the reactor coolant system as a result of a valid signal.

3.0 Instructions (continued)

Non-Emergency Events (continued)

(5) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability

(6) Any event that poses an actual threat to the safety of the plant or significantly hampers site personnel in the performance of duties necessary for safe operation of the plant inluding fires, toxic gas releases, or radioactive releases.

(b) Notify the NRC within four (4) hours of any of the following:

(1) Any event found while the reactor is shutdown, that, had it been found while the reactor was in operation, would have resulted in the plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.

(2) Any event or condition that results in manual or automatic, actuation of any Engineered Safety Feature (ESF), including The Reactor Protection System (RPS). Actuations that result from preplanned sequences are excluded.

(3) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

ENCLOSURE 3
(2 of 3)

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: Surry

Reactor Type: Westinghouse 3 Loop

Date Administered: June 1984

Examiner: R. H. Thornton

Applicant: Proctor Copy & MASTER

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple questions sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
25.0	25.575 ~~25.0~~			1. Principles of Nuclear Power Plant Operations, Thermodynamics, Heat Transfer and Fluid Flow
24.75 ~~25.0~~	25.32 ~~25.0~~			2. Plant Design Including Safety and Emergency Systems
23.0 ~~25.0~~	23.53 ~~25.0~~			3. Instruments and Controls
25.0	25.575 ~~25.0~~			4. Procedures -- Normal, Abnormal, Emergency & Radiological Control
97.75 ~~100.0~~	100.0			TOTALS

Final Grade ______ %

All work done on this exam is on my own, I have neither given nor received aid.

____
Applicant's Signature

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, & FLUID FLOW (25.0)

1-1 Considering Doppler effects for an increase in reactor fuel temperature, which of the following choices (increases, remains the same, decreases) applies to each parameter below?

a. Resonance Escape Probability (p) (0.5)

b. Resonance integral (total area under the resonance peak). (0.5)

c. Amount of fuel self-shielded. (0.5)

1-2 ~~Because of lengthly delays after a fuel loading outage, the old primary neutron source was replaced by a new primary source~~. After completion of fuel and new source loadings, the Source Range Monitors (SRMs) read an average of 100 cps. (K_{eff} = 0.85 calculated).

a. What portion of the average SRM reading is due to the source neutrons only? (0.75)

b. Where would the SRM readings level off if the same fuel remained in the core, but no ~~neutron sources~~ source neutrons were available? Explain. (0.75)

1-3 Following a reactor trip from 100% power, how long would it take for the source range instrumentation to be energized. Show by calculation assuming that Po = 1×10^{-6} amps after the prompt drop. State all valid assumption(s) neccessary to solve the problem. (1.5)

1-4 Indicate whether you agree or disagree with the following statements, and justify your answers.

a. The thermal neutron flux levels within the fuel pellets are higher than those in the coolant channel between fuel rods. (1.0)

b. For a given power level, the neutron flux level impinging the excore detectors will be the same at EOL as at BOL. (1.0)

1-5 Name three (3) properties that are desirable for a good reactor moderator to possess. (1.5)

(continued next page)

1-6 A startup is being performed 15 hours after a trip from extended full power operation using a calculated estimated critical position for the time the startup commenced. How would each of the following events or conditions affect the actual critical rod position compared to the estimated critical position?

(1. higher than estimated, 2. lower than estimated, or 3. no significant difference.) Consider each item separately.

a. A steam generator's level is increased by 25%. (0.4)
b. The startup is delayed for appx. two (2) hours. (0.4)
c. The steam dump pressure setpoint is increased by 200 psi. (0.4)
d. A new boron sample is ten (10) ppm lower than the previous sample. (0.4)
e. Condenser vacuum decreases by two (2) inches H_g from 29 inches H_g. (0.4)

1-7 Give three (3) reasons for having Rod Insertion limits. (1.5)

1-8 After a reactor startup from refueling, reactor power is maintained at 50% power.

a. How long (approximately) does it take the reactor to reach equilibrium Xenon? (0.5)

b. How long (approximately) does it take the reactor to reach equilibrium Samarium? (0.5)

c. If power is increased to 100% after three (3) months at 50% power operation, will the Samarium concentration immediately increase, decrease, or remain the same? (0.5)

1-9 The reactor is critical at 10^{-8} amps, BOL. By adjusting steam dump set pressure, Tavg is decreased by 7°F. This results in a stable SUR of 0.2 DPM. (1.5)

Compute the moderator coefficient of reactivity. Show your work. Assume that the average decay constant of the delayed neutron precursors is 0.08 sec^{-1}.

(continued on next page)

1-10 Indicate how the following changes in plant conditions would individually affect DNBR. (1. increase, 2. decrease, 3. has no effect) (2.0)

a. Pressurizer pressure decreases
b. Tc decreases
c. Reactor coolant flow decreases
d. The operator withdraws control rods without changing turbine load
e. Reactor power is increased

1-11 a. From the list below, which type of RCS heat transfer method has an increase in the difference in temperature between the heat transfer surface and the bulk fluid temperature (ΔT) accompanied by a decrease in the heat transfer per unit area ($\dot{Q}/A$). (0.5)

1. natural convection
2. nucleate boiling
3. complete film boiling
4. partial film boiling

b. Explain how nucleate boiling influences the heat transferred from the fuel clad to the reactor coolant. (0.75)

1-12 a. Find the enthalpy change in an isentropic expansion of steam through a turbine into a condenser (Note: Pstm = 825 psia, saturated, Pcond = 2 psia). (0.75)

b. How would the change in enthalpy in part (a) be affected by a less than ideal turbine (i.e., some degree of inefficiency)? Select one of the following answer. (0.5)

1. Higher
2. No Change
3. Lower

1-13 a. Using the RCS loop design and operating conditions following a reactor trip and loss of RCPs, state how the three (3) basic requirements for developing a natural circulation driving head are satisfied. 1.2 (~~1.5~~)

b. Why should the steaming rate from the steam generators be limited during natural circulation cooldown? 0.4 (~~0.5~~)

c. With regard to the mechanical performance of power operated relief valves (PORV's) why must an operator prevent RCS pressure from rising to the PORV's setpoint during natural circulation? How could this operational concern affect RCS natural circulation? 0.4 (~~0.5~~)

1-14 Answer the following questions regarding operation of centrifugal pumps either true or false.

a. One way to increase a pump's net positive suction head (NPSH) is to raise the level in its suction tank. (0.5)

b. Centrifugal pumps are operated in series to significantly increase system flow rate. (0.5)

c. When identical pumps are operated in parallel, the combined pump curve (h_p vs. V) head vs. flow capacity can be derived by adding individual flow rates at constant pump head. (0.5)

d. The typical system flow will increase by a factor of 2.0 (double) if a second identical parallel pump is started. Refer to Figure 1.1 (0.5)

1-15 Figure 1-2 represents a typical K(Z) vs. core height plot for a typical Westinghouse PWR.

a. Explain why the Fq(Z) is limited from a core height of 6.0 ft. to 11.0 ft. Include the type of accident considered in this restriction. (1.0)

b. Explain why the Fq(Z) limit at 11.0 ft. to 12.0 ft is even further restricted as shown. Include the type of accident considered in this restriction and also state any difference in RCS heat removal conditions considered for this portion of the core. (1.0)

END OF CATEGORY 1

(Write end of category 1 on your paper)

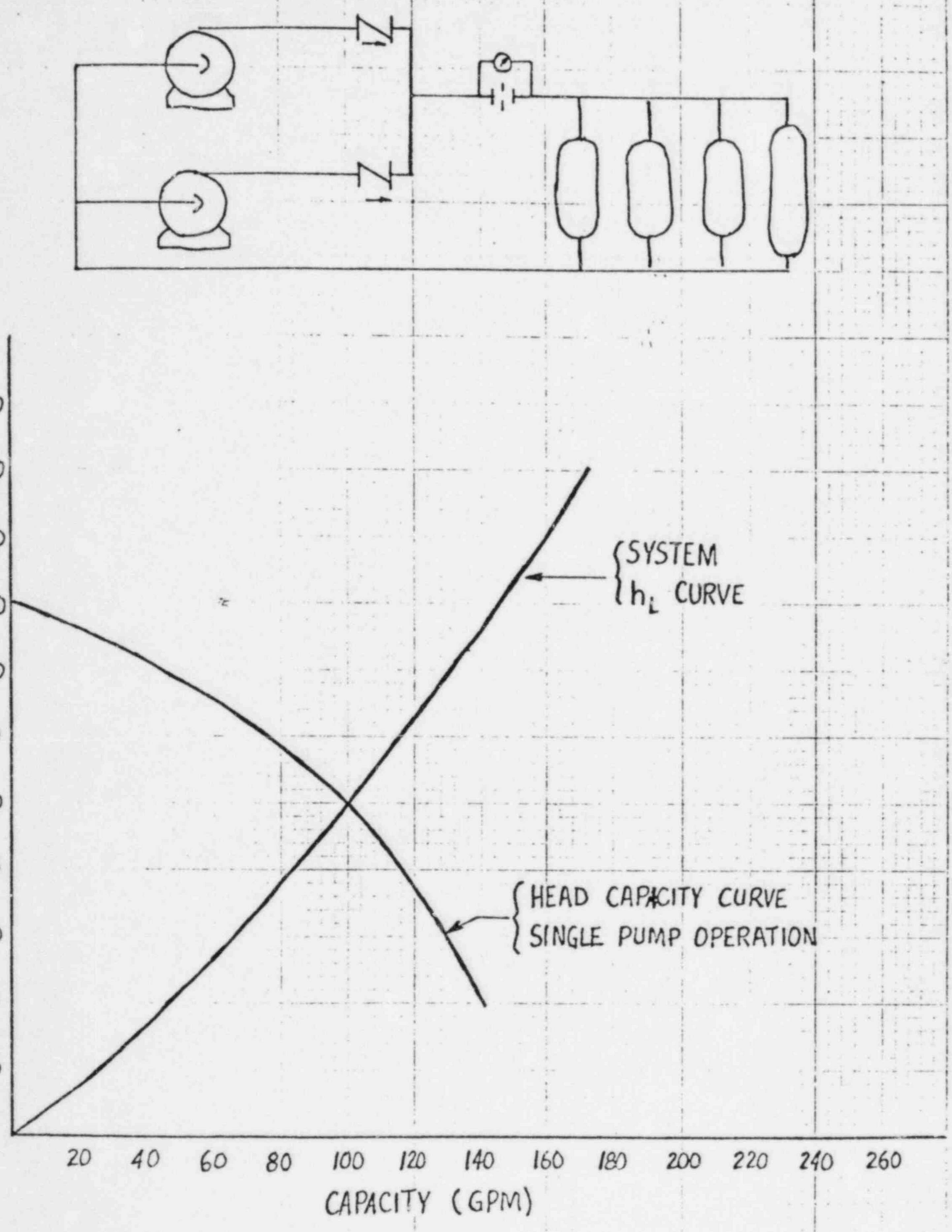
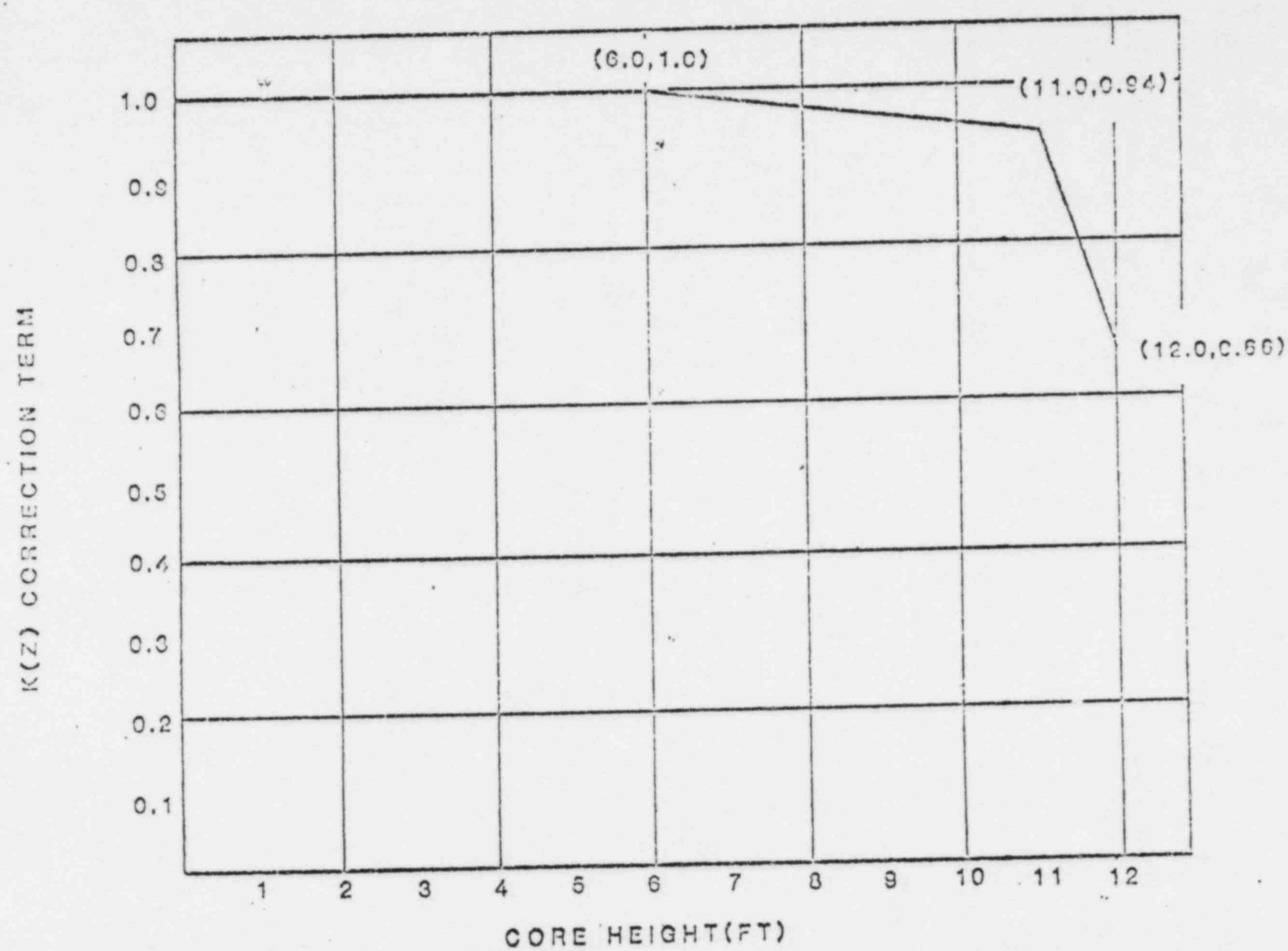

220
200
180
160
140
120
100
80
60
40
20
HEAD (FT.)
SYSTEM
h_L CURVE
HEAD CAPACITY CURVE
SINGLE PUMP OPERATION
20
40
60
80
100
120
140
160
180
200
220
240
260
CAPACITY (GPM)

FIGURE 1-1

FIGURE 2.1-2

K(Z) CORRECTION TERM AS A FUNCTION OF CORE HEIGHT



2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS (25.0)

CANDIDATE STATED PLANT MAY BE DIFFERENT [illegible]

2-1 a) What systems and pumps provide normal and backup makeup water to the Component Cooling Surge Tank? (1.0)

b) What thermal loads are auto-isolated from the Component Cooling subsystem when a High-High Containment (Phase 3) Isolation occurs? List eight (8) different types of loads. ? (2.0)

2-2 What design provision(s) has been made to ensure NPSH will be sufficient during all conditions at a LOCA for:

a) Outside Recirculation Pumps (2 provisions)? (1.5)

b) Inside Recirculation Pumps (1 provision)? (0.75)

2-3 a. List two (2) services provided by the Diesel 125V DC supply system. (1.5)

b. If both electric motors are inoperative for the diesel air start compressors, what prompt action can be performed to restore air compressor operability? (0.75)

2-4 Match the list of Unit 1 loads in the first column with the appropriate "normal" power supply/bus in the second column. (~~0.3/ea.~~ RHT (0.4 ea)

RHT delete — a)	Battery charge 1A1	4160 V Emerg. Bus 1J
b)	4160 v. switchgear (CONTROL POWER)	480 V DC 1A
c)	INVERTER 1-III	4160 V 1H STUB BUS
d)	RHR pump 1A	480 V Emer. Bus 1H
e)	Charging pump 1B	125 V DC 1A
f)	Recirc. Spray Pump 1A	
g)	Containment Recirc. Fan 1A	
RHT delete — h)	Emergency Gen. Fuel Oil Pump 1A	

2-5 a) List the input drain sources (name & number of each type component) which enter the High Pressure drain tank (receiver). (0.8)

b) Between which two (2) components of the feedwater system does the HP Heater Drain Pumps discharge? (0.4)

c) What operating conditions (control actions and permissives) are necessary to get an auto start of the second HP Heater Drain Pump. Name four (4). (1.2)

(continued on next page)

2-6 a) Name the three (3) functions of the extraction steam system. (0.9)

Relative to other components

b) Describe where non-return valves (NRV's) are located ^ in the extraction steam system. Be specific as to the NRV and component(s) being supplied (6 valves total). (1.2)

c) What two (2) conditions will cause extraction line drain valves to open automatically? (0.8)

2-7 Since the Component Cooling System operates at approximately 150 psia, how is this system protected against overpressurization and rupture due to a leak in the reactor coolant pump thermal barrier heat exchanger. Include all design feature and auto-actions considered in the protection design (5 features). (2.0)

2-8 a) Between which two (2) components does excess letdown flow originate in the reactor coolant system? (0.75)

b) List two (2) problems that result from remaining on excess letdown instead of using the normal letdown line during power operation. (1.5)

Individually clarified

2-9 List five (5) protective trips for the steam generator main feed pumps. (2.0)

2-10 Excluding automatic valve opening or closing signals, state seven (7) other automatic actions that result from a safety injection system initiation. (2.8)

2-11 What pumps provide fluid for the following functions? ~~(0.75)~~ 0.5 RH

a. Exhaust hood spray

omit → b. BIT recirculation

c. Makeup to cold leg accumulators

END OF CATEGORY 2

(Write end of category 2 on your paper)

3-1 a) Explain how the inputs in the core cooling monitor system are used to determine the margin to saturation indicated on the meter. (1.0)

b) Explain what the three (3) regions of the margin to saturation meter indicate and include the units of measure where applicable. (0.9)

3-2 Assume the reactor is operating at 80% power when a N.R.T_H RTD fails high. Explain how the following parameters would initially be affected, (increase, decrease, remain the same) considering the T_H failure either a control or protection channel as applicable.

a. Steam dump valve position (0.5)
b. Charging flow rate, initially (0.5)
c. OPΔT setpoint (0.5)
d. Rod insertion limit setpoint (0.5)
e. Control rod bank position (0.5)

3-3 What is the function of each of the following control interlocks (permissives)? MANY QUESTIONS - CANDIDATES SAY THEY DON'T HAVE "C's"

delete RHJ

a. C-2 (0.5)
b. C-5 (0.5)
c. C-7 (0.5)
d. C-9 (0.5)

3-4 Name the equipment reset by the Start-Up Pushbutton mounted on the Control Board prior to plant startup. (2.4)

3-5 Which of the following signals is not used an an input to the steam generator level control system to modulate the main feedwater regulating valves? (0.5)

a. Actual steam generator level
b. Feedwater flow
c. steam flow
d. T (avg)

(continued on next page)

3-6 What position (fully open, fully closed, or throttled) will the main feedwater flow control valve assume under the following conditions? (1.5)

a. Reactor Trip with Low Tavg of 543°F
b. Increase in load from 50% to 60%
c. Safety Injection
d. S.G. level of 75%
e. Loss of Air
f. Loss of Instrument Electrical Power

3-7 After a Safety Injection signal occurs, which component of the Emergency Core Cooling System is designed to act first to initially reflood the core after a design basis Loss of Coolant Accident? (1.0)

a. High head safety injection system (centrifugal charging pumps) through the boron injection tank.

b. The high volume-low head SI pumps

c. The cold leg safety Injection Accumulators.

3-8 Fill in the blank's concerning the pressurizer level control and protection system.

a. Pressurizer level control system compares actual level to level reference. Level reference is calculated from ______________. (0.25)

b. The level error signal is used to control ____________ if necessary to vary PZR level. (0.25)

c. High level reactor trip occurs at ____________________% level. (0.25)

d. Letdown isolation and ________________ occur at 14.4% level. (0.25)

e. Heaters come on at +5% level deviation due to ________________. (0.25)

f. The program Pressurizer levels corresponding to 0-100% power is ________________% to __________________% level respectively. (0.5)

(continued next page)

3-9 a. The function of P-7 is to block six (6) reactor trips under certain signal conditions, one (1) signal being reactor power less than 10% (P-10). Name the six (6) reactor trips P-7 blocks. (2.4)

b. What other signal besides the P-10 input is required to block these six (6) trips. (0.5)

3-10 a. Name each of the conditions that cause a turbine runback. Include setpoint, interlocks, and coincidence where applicable. (2.0)

b. Explain how the amount of turbine runback for each of the above conditions is controlled and limited. (1.0)

3-11 What six (6) events occur when the "LATCH" signal is activated on the Main Turbine EHC Control Panel? (1.8)

3-12 List three (3) control systems which use the turbine 1st Stage Pressure input. State how the parameter is used as a control input/interlock. (Exclude P-2 control input and setpoint data.) (0.75)

3-13 What direction of change takes place in the OTΔT trip setpoint for the changes listed below (increase, decreases, remains the same)? Consider each change separately with reactor power at 90%. (1.5)

a. Pressurizer pressure decreases 5 psig
b. Loop Tavg increases 2°F
c. F (Δq) changes from + 1% to -6% on NI41

3-14 Explain what auto functions occur if the following levels in the VCT are reached? (1.5)

a) 85%
b) 17%
c) 3%

END OF CATEGORY 3

(Write end of category 3 on your paper)

4. PROCEDURES -- NORMAL, ABNORMAL, EMERGENCY & RADIOLOGICAL CONTROL (25.0)

4-1 Assuming the CVCS system is operating in auto makeup mode, name the five (5) basic steps (control manipulations only) required to perform a "dilute" operation without returning to Auto. Limit your answer to reactor makeup controls only, using specific names for each control device. (2.5)

[Auto refers to a switch position on one of the items to be manipulated.]

4-2 a. In addition to the Daily Reactor Coolant System Leak Test, what observed changes or indications warrant another RCS leak test to be performed? (List 5) (1.5)

b. Excluding time data what five (5) other parameters are recorded in order to calculate the RCS leak rate? (1.5)

4-3 a. Procedure 1-OP-1C states the amounts of reactivity by which actual critical rod position may differ from the ECP. What are the allowable deviations (pcm) above and below the ECP? (0.5)

b. What operator actions are required to take the reactor critical if it appears that criticality will be achieved at a point below the allowable limit, but still above the minimum insertion limit. (Assume no precedure mistakes have been made.) (1.0)

c. During the reactor startup, what four (4) operator actions are performed concerning excore detectors after the P6 status lights comes "on"? (1.0)

4-4 a) List two (2) symptoms of a #1 seal failure for a RCP. (1.2)
b) List two (2) symptoms of a #2 seal failure for a RCP. (1.2)
c) List one (1) symptom of a #3 seal failure for a RCP. (0.6)

4-5 Assume that your plant is operating at 100% power when a Circulating Water Pump trips. What are your three (3) immediate operator actions? Include the numerical limitations that are part of the actions. (1.8)

4-6 List six (6) indications of a dropped RCC~~S~~A event. (2.1)
RKJ

4-7 a. How is a ~~faulted~~ RUPTURED S/G identified during a S/G tube rupture? (List at least three (3) methods.) (0.9)

b. How is the ~~faulted~~ RUPTURED S/G isolation ensured? (5 actions) (1.5)

c. What two (2) steps would be taken to reduce RCS temperature to the no-load condition if the ~~faulted~~ Ruptured S/G MSIV and NRV failed to completely shut? (0.6)

(continued on next page)

4-8 Assume that a LOCA and Safety Injection have occurred on your plant:

a. What are the safety injection termination criteria? (5 items) (2.0)

b. What are the criteria for which SI must be manually reinitiated? (3 items) (1.2)

4-9 Answer the following questions as they relate to you as an operator at Surry Power Station.

a. What are your quarterly administrative limits for radiation doses to the the whole body, hands, and skin? (1.2)

b. What is the maximum quarterly whole body administrative limit that can be approved by station management? (0.4)

c. According to the HP Radiation Protection Manual & GET, a person must meet three (4) requirements in order to wear a respirator. List these three (4) requirements. Include time frame if applicable. (1.0)

4-10 Does the biological effect of 100 rem dose depend on whether the exposure was neutron or gamma radiation? (YES or NO) (0.4)

4-11 Fill in the blanks in the Surry Area Designation Criteria for Radiation Exposure Control chart in Figure 4.1 using the appropriate values and units as applicable. (0.9)

END OF CATEGORY 4

(Write end of category 4 on your paper)

TABLE 1

AREA DESIGNATION CRITERIA FOR RADIATION EXPOSURE CONTROL

	Radiation Levels	Smearable Activity β-γ dpm/100 cm^2	Airborne β-γ Particulate $\mu Ci/ml$ *
Unrestricted			**
Restricted Clean			
Restricted Control Radiation		-	-
High Radiation		-	-
Airborne	-	-	
Contaminated	-		-

* Airborne radioactivity is expressed in terms of maximum permissible concentrations (MPC) as defined in 10CFR20, Appendix B. If an isotopic analysis is not performed, 1×10^{-10} $\mu ci/ml$ for unrestricted areas and 3×10^{-9} $\mu ci/ml$ for restricted areas may be used in lieu of the MPC's for airborne β-γ gross particulate activity.

** The airborne MPC's for unrestricted areas are specified in 10CFR20, Appendix B, Table 2.

FIGURE 4-1

Equation Sheet

$$\dot{Q} = \dot{m}\,\Delta h$$

$$\dot{Q} = UA\,(T_{avg} - T_{stm})$$

$$\dot{Q} = \dot{m}\,c_p\,\Delta T$$

$$h_L = k\,\dot{V}^2$$

$$DNBR = \frac{Q_c}{Q_x}$$

$$P = P_0\,10^{SUR(t)}$$

$$P = P_0\,e^{t/\tau}$$

$$SUR = \frac{26.06}{\tau}$$

$$\tau = \frac{\beta - \rho}{\lambda \rho}$$

$$\tau = \frac{\ell^*}{\rho} + \frac{\beta - \rho}{\bar{\lambda}\,\rho}$$

$$\rho = \frac{K_{eff} - 1}{K_{eff}}$$

$$\Delta\rho = \frac{K_2 - K_1}{K_2 K_1}$$

$$\frac{CR_1}{CR_2} = \frac{1 - K_{eff_2}}{1 - K_{eff_1}}$$

$$RR = \Sigma_f\,\phi_{th}$$

$$SCR = \frac{S}{1 - K_{eff}}$$

$$M = \frac{CR_1}{CR_0}$$

$$\frac{1}{M} = 1 - K_{eff}$$

$$A = \lambda N$$

$$\lambda = \frac{\ln 2}{t_{1/2}}$$

$$N = N_0\,e^{-\lambda t}$$

WRITTEN EXAMINATIONS
INSTRUCTIONS

1. Compromising this examination must be avoided.

   a. Restroom trips are to be limited and only one candidate at a time may leave (check with proctor). Avoid all contact with anyone except examiners (including facility staff). If contact is unavoidable, Do Not Talk.

   b. Paper for your exam will be provided. All other paper is to be removed from this room.

   c. Eating lunch/snacks is permissible, but trips outside of the examination area are prohibited. Please obtain food/drinks before exam is passed out.

   d. Cheating on the examination means an automatic denial of your applications, and could result in more severe penalties.

2. Instruction for taking the exam.

   a. Use only black ink or dark pencil.

   b. Use paper provided.

   c. Print your name in the upper right hand corner of each page of the answer sheet.

   d. Consecutively number each answer sheet.

   e. Write only on one side of each sheet.

   f. As you complete each answer sheet, place it face down on the table.

   g. Number each answer as to category and question number as they appear on the exam.

   h. Allow at least three lines between each answer. (Not required for multiple choice).

   i. Use acronyms only if commonly used in facility literature.

   j. Partial credit will be given where applicable. Likewise points will be deducted for any incorrect information. Show all calculations, methods or assumptions used to obtain answers.

k. If any question is not clear to your or seems ambiguous, raise your hand and the proctor will come to you. Ask only an examiner for clarification. We are pleased to help.

l. If you don't know the entire answer to a problem but can answer it from some point; make an assumption, state the assumption and answer what your can.

m. To pass the examination, an overall grade of 80% and greater than 79% in each category is required.

n. You are allowed six hours to complete the examination.

3. When the exam is completed:

   a. Review the instruction sheet at the end of the exam.

   b. Staple the exam questions on top of the answer sheets. Turn in all unused and scrap paper to the proctor.

   c. Sign the statement on the cover sheet.

   d. Turn in your exam, then pick up your belongings.

   e. Leave the examination area. Being found in this area after you have completed the exam could result in your license being denied.

1-1 a. decrease (0.5)
b. remains the same (0.5)
c. decrease (0.5)

REF.: Surry, Reactor Physics (Westinghouse), PWR Core Physics page I-5.16-I-5.21

1-2 a. $\frac{1}{m} = \frac{Co}{Ci}$ $\frac{1}{m} = 1 - Keff = 1 - 0.85 = 0.15$ (0.75)

$0.15 = \frac{Co}{100 \text{ cps}}$

Co = <u>15 cps</u>

b. SRMs would level off at <u>0</u> cps. (0.75)
With the reactor subcritical (Keff $\cong$ 0.85) and the sources removed, neutron population begins to decrease by a factor of 0.85 from one generation to the next. Eventually all neutrons disappear in the core and SRM counts are proportional to core neutrons so that SRMs = 0

REF.: Surry Reactor Physics (Westinghouse) Neutron Sources & Subcritical Multiplication, p. I 4-10 thru p.I.4-28

1-3 Assumption: (1) $\tau \cong -80$ (due to longest delayed neutron half-life, Br-87) (0.5)

(2) P-6 is energized at ~~10^{-10}~~ 5×10^{-11} amps (0.5)

Calculation: $P = Poe^{t/\tau}$ (0.5)
$10^{-10} = 10^{-6}$ e

$t = -80 \text{ sec} \times \ln(\cancel{10^{-10}}\ 5 \times 10^{-11}/10^{-6})$ = -80 sec $\cdot \ln(5 \times 10^{-5})$ RKJ
t = ~~737~~ 792 sec. or ~~12.3~~ 13.2 min.

REF.: Surry, Reactor Physics (Westinghouse), Neutron Kinetics p. I-I-3.6, p. I-3.10, 1-PT-1.1 p.17.

1-4 a. Disagree. Thermal neutron flux levels within the fuel are lower because of self-shielding and because neutrons are thermalized in the moderator, not the fuel. (1.0)

b. Disagree. Excore detectors will see more at EOL due to increased leakage (because of reduced boron concentration and radial flux shift to the outer portions of the core.) (1.0)

REF.: Surry, Reactor Physics (Westinghouse), Neutron Physics, p. I 2, 31

(continued on next page)

1-5 a. a large scattering cross section (0.5)
b. a very small absorption cross section (0.5)
c. a large neutron energy decrease per collision (0.5)

REF.: Surry, Reactor Physics (Westinghouse), Neutron Physics, p. I-2.19

1-6 a. 2 (lower than) (0.4)
b. 2 (lower than) (0.4)
c. 1 (higher than) (0.4)
d. 2 (lower than) (0.4)
e. 3 (no significant) (0.4)

REF.: CAF

1-7 a. Provide hot shutdown by reactor trip at any time, assuming highest worth rod remains fully withdrawn. (0.5)
RNJ OR ACCEPT : Provide adequate shutdown margin
b. Limit reactivity insertion for rod ejection accident. (0.5)

c. Provide for acceptable nuclear peaking factors. (0.5)

REF.: Surry, Tech. Specs. p. 3-12-11 and 3.12-12 and PLS book, p. 2, Item C (RNJ)

1-8 a. ~~30-40 hours~~ 40 ± 10 hours (Ref. Surry Curve Book, unit 1) (0.5)
b. 30-40 days (0.5)
c. decrease (0.5)

REF.: Surry, Reactor Physics (Westinghoue). PWR Core Physics, p. I-5.65 & p. I-5.787-5.79, Part a. - Surry Unit 1 Cycle 7 Curve Book

1-9 $\rho = \frac{\beta eff}{1 + \lambda\tau}$; $\tau = \frac{26}{SUR} = \frac{26}{0.2} = 130$ sec. (0.25/e

Assume $\beta eff = .0058$ (0.25)
RNJ [Accept βeff = .006 to .005]

$\rho = \frac{.0058}{1 \div 0.08\ (130)} = \frac{.0058}{11.4} = .0005 = 50$ pcm (0.25)

$MTC = \frac{50\ pcm}{-7°F} = -7.14\ \frac{pcm}{°F}$ (0.5)

1-10 a. 2 (decrease) (0.4)
b. 1 (increase) (0.4)
c. 2 (decrease) (0.4)
d. 2 (decrease) (0.4)
e. 2 (decrease) (0.4)

1-11 a. Partial film boiling (0.5)

b. Nucleate boiling creates turbulence in the boundary layer at the clad surface by bubbles breaking through and increasing heat transfer rate. Also collapse of bubbles adds latent heat to the coolant to enhance heat transfer. (0.75)

REF.: General Physics Corp., Heat Transfer, Thermo, Fluid Flow Fund. p. 13

1-12 a. 378 BTU/LBM (note: use Mollier Diagram) ± 5 Btu/lbm [Steam table Calculation Also acceptable method] (0.75)
b. 3 (lower) (0.5)

1-13 a. 1) Reactor decay heat provides a heat source following a reactor trip increasing RCS temperature in upper head and hot legs. (Heat source). (~~0.3~~ 0.4/ea.)

2) The steam generator steaming causes RCS temperature to decrease in the cold legs. (Heat sink).

3) The steam generator is located higher than the reactor so that bouyant forces will cause RCS flow. Hotter RCS hot leg fluid travels upward while cooler RCS cold leg fluid travels downward (Heat Sink higher than heat source.)

b. Over-steaming the steam generators during natural circulation cooldown can reduce the effective heights difference between the heat source & heat sink by cooling both the T_h and T_c sides of the U-tubes in the steam generators thus inhibiting natural circulation flow rate. Also: Excessive steaming rate could cause cold-water block. RNJ (0.4)

c. These valves have been known to fail to reseat. A sudden drop in RCS pressure due to a stuck open relief valve could cause flashing in the reactor coolant loop hot legs and a loss of natural circulation. OR - flashing in the flow path which will cause a loss or reduction of natural circulation flow rate. (0.4)

REF.: General Physics Corp., Heat Transfer, Thermo. Fluid Flow Fund, p. 355-357. Westinghouse background document to EP 2.02 Post LOCA Cooldown and Depress. Step 6, 10 & 11.

1-14 (0.5/ea)
a. True
b. False
c. True
d. False

REF.: General Physics Corp., Heat Transfer, Thermo, Fluid Flow Fund, p. 324, 328

(continued on next page)

1-15 a) Design basis LOCA or large LOCA. Following a design basis LOCA, the upper portions of the core will be reflooded last and therefore remain uncovered for a longer period and more fuel damage could occur. (The penalty ensures that the LOCA will not start with unacceptably higher power densities in the upper half.)* (1.0)

b) Small break LOCA. For certain small break LOCA's, the reflood rate from the injection systems is very slow (< 1 inch/sec.)* causing the top core region to be cooled by steam only heat transfer resulting in stored energy build-up in this region. (The more restrictive penalty additionally protects this region of the core from small break LOCA's.)* (1.0)

*Information in parenthesis not required.

REF.: General Physics Corp. Heat Transfer, Thermo, Fluid Flow Fund. p. 249

2-1 a) ~~Normal~~ - Condensate via main condenser pumps. (BC-P-2) RNJ (0.5)
RNJ ~~Backup~~ - Bearing cooling via bearing cooling water makeup pump. (0.5)

b) Any eight (8) of the following @ 0.25 pts each

1) RCP thermal barriers
2) RCP bearing oil coolers
3) RCP motor ~~starters~~ stator cooling
4) Excess letdown Hx
5) RHR pumps, seal cooler ~~and stuffing box jackets~~ RNJ
6) RHR Hxs
7) Primary shield penetration cooling coils
8) Primary shield water wall coolers
9) Primary drain coolers
10) Neutron shield tank coolers
11) Reactor containment air recirculation coolers

REF.: Surry System Description, Component Cooling, pps. 1,2 and 7

Any (2) Time delay of 5 min. after HiHi CLS initiation and

2-2 a) Outside Recirc. pumps have limiting flow orifice in discharge lines (3000 gpm) and containment spray water (300 gpm max) is diverted to pump suction sump to ensure subcooling. (1.5)

Any (1)
b) Inside Recirc. pump suctions are supplied by bleed-off flow after each spray cooler (350 gpm max)*. AND Time delay of 2 min. after HiHi CLS initiation. (0.75)

REF.: Surry System Description, Recirc. Spray Subsystems, page 3
* Not Required

2-3 a. 1) Provide excitation for the generator (0.75)
2) Provide DC control for local and remote switchboards (0.75)
3) ~~Governor Motor~~ 4 Gov. motor booster pumps
3) Annunciator Panel (5) Various Controls & Alarms (6) Start Circuits
b) Manual drive-belt change over from the electric motor to diesel engine for one of the compressors. (0.75)

REF.: Surry System Descriptions, Emergency Power and Distribution, p.4-14, and 4-37 Also part (a) Surry Diesel Generator Lesson Plan

2-4 delete ~~a) 480V MCC 1H1~~
b) 125V DC 1A
c) 125 V DC 1A
d) 4160V 1H Stub Bus
e) 4160V Emer. Bus 1J
f) 480V Emer Bus 1H
g) 480V Emer. Bus 1H
delete ~~h) 480V MCC 1H-1~~
(0.4)/ea

REF.: Surry System Description, Electrical System, pps. 4-20, 4-69, 4-70, 6-1A, 5-3

(continued on next page)

2-5 a) (4) MSR moisture removal drains (0.4)
(2) drains from feedwater heaters (2A and 2B)

b) Between condensate/feedwater heaters no. 2 and main feed pumps suction. (0.4)

c) 1) HP heater drain pump control switch (CS-15B6 or CS-15C6) in "auto" ~~(0.3/ea.)~~ 0.4/ea.
omit — ~~2)~~ HP heater drain tank level "normal"
3) Motor electrical protection reset
4) HP heater drain tank level high

REF.: Surry System Description Vol.1, High Pressure Drains System, p. 7&8

2-6 a) 1) Preheat feedwater (0.3/ea.)
2) Heat source for the Flash Evaporator
3) Steam for auxiliary steam system during normal operation

b) Non Return Valves are located on the extraction supply lines as follows: (0.2/ea.)

HP NRV-~~ES-101~~ to point 1A and 1B feedwater heaters
HP NRV-~~ES-102~~ to point 2A and 2B feedwater heaters
RNJ LP NRV-~~ES-103A~~ to feedwater 3A
LP NRV-~~ES-103B~~ to feedwater 3B
LP NRV-~~ES-104A~~ to feedwater 4A
LP NRV-~~ES-104B~~ to feedwater 4B

c) 1) Turbine trip (0.4)
2) High level in heater shell

REF.: Surry System Descriptions Vol. 1, Extraction Steam p. 1 & 2

2-7 5 items at 0.4 pts. each

a) A check valve at the inlet side of the thermal barrier HX prevents backflow into the CCS.

b) Flow instrumentation on the discharge side of the HX detects high flow and transmits an alarm signal (50 gpm).

c) A manual trip valve at the outlet of the HX isolates the CCS to RCP thermal barrier.

d) The piping from the check valve at inlet to the trip valve on outlet is high pressure piping.

e) A safety valve is provided in the high pressure piping to relieve excessive pressure that may be caused by heating.

REF.: Surry System Description Vol. 1, Reactor Coolant Pump, p. 8, Annuciator Procedure 1C-2, and C.A.F.

2-8 a. Each or all RCS loop cold legs between RCP and Steam Generator (also loop drain header, or RCP suction). (0.75)

b. Any two (2) of the following at 0.75 pts/ea.

1) The reactor coolant does not pass through the demineralizers and filters. The activity of the coolant and impurities will increase much faster using the excess letdown line only.

2) Since the letdown flow rate is not as great as when using the normal letdown line, borations and dilutions will take much longer, limiting the rate of change of power at certain times.

3 ~~2~~) In addition there will be no hydrogen addition to the coolant due to bypassing of the volume control tank during normal system lineup.

REF.: Surry, System Descriptions Vol. 1, Chemical and Volume Control System, p. 77, C.A.F.

2-9 Any 4 ~~5~~ of the following at 0.5 ~~0.4~~ pts. each

1) Low lube oil pressure (<4 psig)*
2) Insufficient suction pressure beyond a preset time delay (<55 psig for >7 sec.)*
3) Bus undervoltage
4) Motor protection trip due to phase or neutral overcurrent
delete ~~5) Safety Injection System trip~~
delete ~~6) Hi-Hi Steam Generator Level~~

*Setpoints not required

REF,: Surry System Descriptions, Main Feed System, page 9

2-10 Any 7 of the following at 0.4 pts. each:

a) Reactor trip, (I) turbine trip
b) All charging pumps start
c) Low head SI pumps start
d) Main feed pumps trip
e) Containment vacuum pumps trip
f) Both motor driven auxiliary feedwater pumps start
g) The emergency diesel generators start and come up to speed
h) The control room supply and exhaust ducts close

Also: J) H_2 Analyzer switches

(K) electrical Bus transfer (Station service to Reserve Station Service)

REF.: Surry System Descriptions, Vol. 1 ~~7~~, Safety Injection System p. 27
Surry EP 1.0

(continued on next page)

2-11 a. Running condensate pumps (0.25/e
b. Boric acid transfer pumps
c. PD hydrotest pump
not required

REF.: Surry System Description Vol. 7, Condensate System p. 10, Safety Inj. System p.3 and p. 6

3-1 a) The margin to saturation is the auctioneered lowest system pressure input used to calculate saturation Temperature ~~to the monitor minus the calculated saturation pressure. The saturation pressure is calculated from either~~ The highest RTD (T_H or TC) if selected (RTD position) or highest incore thermocouple if the selector switch is in the T/C position, is subtracted from the calculated saturation temperature to provide margin to saturation. (1.0)

b) 1) Extreme left - invalid results due to unit failure or invalid inputs (0.3)

2) Center region - margin to saturation in ~~psi~~ °F (subcooled) (0.3)

3) Right region - Superheat, °F (0.3)

REF.: ~~Surry System Descriptions Vol. 1, RCS p. 67C~~
Revised Surry lesson Plan, per Tng. Staff
Also V.C. Summer Sys. Descrip. Core Cooling Monitor, p. 8 (section 18 C)

3-2
- a. No change (because no arming signal) (0.5)
- b. Initially increases (0.5)
- c. Decrease (0.5)
- d. Increase (0.5)
- e. Decrease (0.5)

REF.: Surry Instrumentation Manual, Chapt. 6, p.4, Chapt. 10 (CAF) Chapt. 9, Chapt. 4

3-3 delete
- a. Blocks auto and manual rod withdrawal (0.5)
- b. Blocks auto rod withdrawal (0.5)
- c. Arms steam dump for either tripping or modulation (0.5)
- d. steam dump condenser permissive (0.5)

REF.: Surry CAF

3-4
- a. All group step counters on the Control Board (0.4)
- b. The master cycler counter (0.4)
- c. All slave cycler counters (0.4)
- d. The bank overlap counter (0.4)
- e. All internal memory and alarm circuits (0.4)
- f. All pulse-to-analog converters in the Rod Position Indication System (0.4)

REF.: Surry Instrumentation Manual, Rod Control System, p. 25

(continued on next page)

3-5 d. (Tavg)

REF.: Surry Instrumentation Manual, Steam Generator Level Control, p. 8.8

3-6 (0.25/ea

a. closed
b. throttled
c. closed
d. closed
e. closed
f. closed

REF.: Surry Instrumentation Manual, Steam Generator Water Level Control and <u>CAF</u> (some reference material in manual deleted).

3-7 c.

REF.: Surry System Description Vol. 1 Safety Injection System

3-8 (0.25/

a. Auct. Tave
b. ~~Makeup flow control valve~~ Charging flow Control valve (FCV-X122)
c. 88%
d. Turns off heaters
e. Insurge of cold water
f. 22.2% of 57.7%

REF.: Surry Instrument Manual, chapt. 10 and chapt. 11

3-9 (0.4/ea)

a. 1) Low flow or (3) RCP Bkrs. open
2) RCP Bus Undervoltage
~~3) RCP Bus Underfrequency~~
4) Turbine trip
5) Pressurizer Low Pressure
6) Pressurizer High ~~Pressure~~ level

b. Turbine Impulse Pressure $\leq$ 10% (P-13) (0.5)

REF.: Surry Instrumentation Manual, chapt. 11
SRO/RO Retraining Control & Protection Inst. lecture Series

(continued on next page)

3-10 a) 1) 2/3 OTΔT channels within 3% below Trip Setpoint
2) 2/3 OPΔT channels within 3% below Trip Setpoint
3) 1/4 Power Range detectors sense power reduction of (-) 5%/2 seconds (dropped Rod signal) ~~with power >70%~~
4) Rod bottom signal (RPI Dropped Rod) with power >70% (Any Control Rod · 1/48 within ~~bottom bistable~~ 20 steps ~~below bank @ 35 steps (control banks B.C.D) at other Westinghouse plants~~. of bottom
EXAMINER'S NOTE: CAF for setpoint information not available (2.0)

b) 1) & 2) High ΔT Runbacks: (cyclic on 1.5 sec off 28.5 sec. intervals) at rate of 200%/min or 10%/min. equivalent turbine power until signal is clear

3) Dropped Rod Signals: ~~C.A.F. for runback data information not available~~. ~~EXAMINER'S NOTE:~~ ~~Appx. 3.3%/sec. timed at appx.~~ Load reference at 9 sec. ~~this is same~~ 200%/min. rate). ~~(Rate reduction is timed to limit runback to 30% in event of circuit failure probably.~~
and load limit : 120%/min. until 1/2 ch. First Stage Press. < 70% power.

(4) REF.: Surry Instrumentation Manual, Chapt. 9 Temp. Ind. Sys. p. IV5.7 for parts a(1), a(2), Chapt. 7 and 14 (partical information). Surry Systems Descriptions Turbine Control Systems no information available.
Also Logic diagrams
→ RPI Dropped Rod : load limit, 120%/min until 1/2 ch. First Stage Press. < 70% power

RHT
3-11 (Any 6 of following) (0.3/ea.)
a. Turbine vacuum trip will latch
b. Auto-stop oil dump will reset
c. Main Stop Valves open
d. Governor valves close
e. Reheat stop valves open
f. Intercept valves open
(G) Close Emergency Trip Valve (interface)
(H) De-energize 20 ET (after ASO > 45psig 2/3)
(I) NRV Turbine Extract. Steam open

REF.: Surry Systems Description Vol. 2, Turbine Control p. 6-2 and Turbine Protection p. 7-2 and p. 7-6 (RHJ)

3-12 a) Rod Control - T ref (0.25/ea)
b) Steam Dump - Tref and sudden loss of load interlock
c) S.G. Level Program - Ref. Level

REF.: Surry Instrumentation Manual, Chapt. 11, Process Protection Instrumentation p.4

(continued on next page)

3-13 a. decreases (0.5 pts/ea)
RHJ
b. decreases
c. remains the same (assume not outside normal band)

REF.: Surry Instrumentation Manual, Chapt. 9, Temperature Indicating System p. IV-5.4

3-14 a) Letdown is diverted to the BRS by the three-way diversion valve. (0.5/ea.)
b) Automatic makeup is initiated.
c) Charging pump suction transfers from the VCT to the RWST.

REF.: Surry, System Description, Primary Systems, CVCS, p.17

4-1 a. Place the makeup mode controller "start-stop" switch to the "Stop" position. (0.5)

b. Set the primary grade flow controller (HFC FC-114A) for the desired flow rate. (0.5)

c. Set the primary grade water integrator (YIC-114A) to desired quantity. (0.5)

d. Place the makeup mode selector switch to "Dilute". (0.5)

e. Place the makeup mode controller "Start-Stop" switch to "Start". (0.5)

REF.: Surry, I-O-P 8.3.6 p. 4

4-2 Any five of the following, 0.3 pts. ea.

a. 1) Increase make-up water required to maintain normal PRZ level.
2) Temp. increase in reactor head flange leak-off piping.
3) Containment sump level increasing and sump pump operation greater than normal.
4) Containment pressure, temp. and humidity rising above normal.
5) Indication increases for either containment gas or particulate, condenser air ejector, component cooling water, steam generator blowdown or aux. bldg. area Radiation Monitors.
6) Primary drain tank level increases above normal rate.
RH7 - - - 7) Hi Temp. down stream of PRZ relief or safeties (8) Hi leak off flow indication #1 seal

b. 1) VCT level (0.3/ea
2) PDTT level
3) PRT level
4) PZR level
5) Tave

REF.: Surry, OP-5.3.1, pgs. 1 and 9

4-3 a. 400 pcm above and 250 pcm below ECP (0.5)

b. 1) Insert all control group rods (0.25/e
2) Re-evaluate the ECP
3) Obtain Operations Superintendent's permission to approach criticality
4) Use 1/M plot (per OP-1.4 App. A)

c. 1) Verify IR and SR have 1 decade overlap (0.25/e
2) Block SR trips
3) Verify SR high voltage off
4) Monitor second IR on NR-45

REF.: Surry, I-OP-1.4 p. 8, 1-OP-1C p. 5

(continued on next page)

4-4 RHT a. 1. RCP seal leakoff high flow (4) RCP Vibration Alarm (0.6/ea.)
2. RCP seal water return low ΔP (5) Abnormal increase in PDTT level
3. RCP Thermal Barrier CC Hi Temp. (195°F)

RHT b. 1. RCP Vapor seal tank high level (0.6/ea.)
2. RCP seal leakoff low flow
3. Abnormal increase in PDTT level (4) RCP vibration Alarm

c. Vapor seal tank low level (0.6)

REF.: Surry, Annunciator Procedures (1C-4, 1C-5, 1B-8, 1C44 and 1B-32)
RHT 7C-3, AP-9, AP-16

4-5 1. Throttle condenser outlet valves manually to maintain normal level of 22 feet, do not exceed 15°F ΔT. (0.8)

2. Reduce generator load to allow condenser vacuum to be maintained above the trip point (18"-22") and temp. at the groin not to exceed 98°F. (0.8)

3. Notify the System Operator of the reason for reduced load. (0.2)

REF.: Surry, AP-13

4-6 Any seven (~~7~~ 6) of the following: (0.3/ea.)

a. Rod Bottom Lights
b. Rod Deviation Alarm
c. Rod Bottom Rod Drop Alarm
d. NIS Rod Drop Alarm
e. Flux Deviation Alarm
f. Rapid drop in Tavg and Power Level
g. Rapid drop in Pressuizer Pressure and Level
h. Turbine Runback
i. Steam Dump Arming

RHT j. Rods stepping in after Runback - (K) Tave-Tref deviation Alarm

REF.: Surry, Dropped RCCA, AP 1.4

(continued on next page)

4-7 a. Any three (3) of the following (0.3 pts/ea.)

1. Unexpected rise in any S/G narrow range level
2. High radiation from any S/G blowdonw line
3. High radiation from any MS line monitor
4. High radiation as determined by sampling and analysis

b. 1. Shut AFW to ruptured S/G if NR level >50% (0.3/ea.)
2. Shut ruptured S/G MSTV
3. Verify that ruptured S/~~B~~G PORV is shut
4. Shut ruptured S/G steam supply to the TDAFW pump
5. Verify ruptured S/G blowdown TV's are shut

c. 1. Shut non-ruptured S/G MSTV's (0.3/ea.)
2. Use non-ruptured S/G PORV's for steam dump

REF.: Surry EP-4.0 pgs. 3 and 4

4-8 a. 1. Containment conditions — Normal (0.4/ea.)
2. RCS Pressure > 2000 psig
3. RCS Subcooling $\geq$ 50°F
4. PZR Level >50%
5. Heat Sink: S/G level >65% WR or 17% NR or AFW flow > 540 gpm

b. 1. RCS Pressure < 1715 psig (0.4/ea.)
2. RCS Subcooling < 50°F
3. PZR Level < 20%

REF.: Surry, EP-2.00

4-9 a. WB - 0.75 rem/qtr. (0.4/ea.)
Hands - 18.75 rem/qtr.
Skin - 5.00 rem/qtr.

b. 1.75 rem/qtr.

c. 1. WB count ~~(0.3)~~ 0.2
2. Satisfactory Respirator Fit Test ~~(0.3)~~ 0.2
3. Satisfactory Pulmonary Function Test ~~(0.3)~~ 0.
4. Sat. Respiratory Protection Training 0.2
[All above within 12 months] ~~(0.1)~~ 0.

REF.: Surry, HP Manual-Section I pgs. 1.2-3 and 4
General Employee Training Manual

(continued next page)

4-10 No

REF.: Surry, H.P. Rad. Prot. Manual page 1.2-2 (0.4)

4-11 See Figure 4.1

ANSWER
4-11

TABLE 1

AREA DESIGNATION CRITERIA FOR RADIATION EXPOSURE CONTROL

	Radiation Levels	Smearable Activity β-γ dpm/100 cm^2	Airborne β-γ Particulate μCi/ml *
Unrestricted	< 500 mrem/yr.	< 1000	**
Restricted Clean	< 2.5 mrem/hr.	< 1000.	< 25% MPC
Restricted Control Radiation	> 2.5 mrem/hr	-	-
High Radiation	> 100 mrem/hr	-	-
Airborne	-	-	> 25% MPC
Contaminated	-	> 1000	-

* Airborne radioactivity is expressed in terms of maximum permissible concentrations (MPC) as defined in 10CFR20, Appendix B. If an isotopic analysis is not performed, 1×10^{-10} μci/ml for unrestricted areas and 3×10^{-9} μci/ml for restricted areas may be used in lieu of the MPC's for airborne β-γ gross particulate activity.

** The airborne MPC's for unrestricted areas are specified in 10CFR20, Appendix B, Table 2.

FIGURE 4-1