ENCLOSURE 1

EXAMINATION REPORT - 50-338, 339/OL-84-01

Facility Licensee: Virginia Electric and Power Company

Facility Name: North Anna Power Station Units 1 & 2

Facility Docket Nos. 50-338 and 50-339

Written and oral examinations were administered at North Anna Power Station near Mineral, Virginia

Chief Examiner: Sandy Lawyer — 1/8/85
Sandy Lawyer — Date Signed

Approved by: Bruce A. Wilson — 1/11/85
Bruce A. Wilson, Section Chief — Date Signed

Summary:

Examinations on October 29, 1984

Examinations were administered to nine RO candidates; two of whom passed.

Examinations were administered to four SRO candidates; two of whom passed.

REPORT DETAILS

1. Persons Examined

SRO Candidates:

Michael Crist
David Critchfield
Kenneth DeVor
David Heacock

RO Candidates:

Mark Bittmann
William Carlin
Robert Counts
Gary Dowell
Paul Fleisher
Randy Hummell
Robert Hutsell
Robert Peters
Kevin Tucker

Other Facility Employees Contacted:

Curtis G. Meyer, Supervisor - Training - Pwr Stat. OPs
R. O. Enfinger, Supt. Operations
Leatrice Kaplan, Nuclear Training Coordinator
Walter Shura, Sr. Nuclear Training Instructor
L. Edmonds, Superintendent Nuclear Training
Roger D. Garner, Supervisor Training-Simulator

2. Examiners:

*Sandy Lawyer
Tom Rogers
Mark E. Baldwin
William E. Eldridge

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with Roger Garner, Curtis Meyer, Larry Edmonds, and Tom Porter, to review the written examination and answer key. The following comments were made by the facility reviewers:

a. SRO Exam

(1) Question 5-1

Facility Comment: This question on fuel cycle # is not in listed scope of ES-402 pg. 1 of 4. This matter is a question of effectiveness of our "control Document" procedures, not operator knowledge.

NRC Resolution: This question was not intended to measure nor did it measure the candidates knowledge of fuel cycle #. ES-402 pg. 1 of 4 does state "This category includes questions to determine the candidates understanding and use of curves depicting reactor behavior ..." The question is a valuable measure of the candidates understanding and use of the reactivity curves for Unit 1 and Unit 2. No change is required.

(2) Question 6-3

Facility Comment: This question should be "deleted". The Residual Heat Removal System (RHR) at North Anna is not used as part of the ESF systems.

NRC Resolution: A review of the material provided demonstrated this to be the case. The examiner inappropriately assumed the additional function for the RHR system. This question was deleted.

(3) Question 6-14

Facility Comment: This question should be "deleted". The Westinghouse rod control system at North Anna will allow "D" control bank rods to continue withdrawal in "manual" past the 220 "rod withdrawal limit". The rods will continue out to 232 steps where the rods will "slip lands", the board demand step counter indication will continue on! This was demonstrated to Mr. Thomas Rogers (NRC) on the simulator on 10-30-84.

NRC Resolution: The referenced demonstration was adequate to convince the examiners that the proper choice was 228 steps. The answer key was changed accordingly.

(4) Questions - all sections 7 & 8

Facility Comment: In general, questions in Sections 7 & 8 required too much memorization of procedures which normally the operators are not required to memorize. Examples are: 7-9, 7-12, 7-18, 7-20, 7-21; 8-3, 8-4, 8-6, 8-13, 8-14, 8-15, 8-18, 8-19, and 8-22.

NRC Resolution: Each of the cited questions was individually reviewed in light of this comment. In no case is memorization per se required. The question format is such that recognition of a correct choice from among four choices must be made. This is quite different from memorization. It was noted that in most cases the correct choice was a direct quote from the referenced document which further aids in the recognition process. (See paragraph 4. of this report).

b. RO Exam

(1) Question 1-5 c

Facility Comment: The question should be reworded on future examination to avoid confusion between the choices "increase" and "remain essentially the same."

NRC Resolution: This wording will be carefully reviewed prior to future utilization of this question.

(2) Question 1-6 a

Facility Comment: Choice a should be reworded to reflect that the increase is rapid. This would make the distinction between choices clearer on future examinations.

NRC Resoluation: If this question is utilized on future exams, this choice will be reviewed for clarity.

(3) Question 1-12

Facility Comment: The question is poorly worded in that the word "immediately" is vague. If this were interpreted to mean at time zero, the decay heat would be as much as 20% of full power.

NRC Resolution: As used here, "immediately" is clearly referring to time zero. The decay heat at this time is 5.89% of full power. No change is required.

(4) Question 2-8

Facility Comment: This question relates to the operation of the reactor trip and bypass breaker trip functions. This question may cause two classes of answers due to Design Change 84-05 being installed in unit 2 this month, and the unit being in startup from refueling. Answered as unit 1 and 2 the same, or reflecting this design change should be acceptable due to trainees not being informed of DCP Completion, but being aware of work in progress.

NRC Resolution: It is possible that the Candidate may make either of the assumptions referred to but the most appropriate would be to assume the question and answer were based upon the information transmitted to the NRC and not upon a proposed change to the facility. Despite this, the alternate answer will be accepted based upon material presented by the utility.

(5) Question 2-15

Facility Comment: This question should be "deleted" due to terminology conflict. We do not have a "steam generator high level." We have "level error" at ± 5% from reference, and a High-High level trip.

Ref: a) Westinghouse logic 5655D33 sheets 7, 13 and 15
b) 11715-ESK-10F
c) 11715-LSK-1-2c and 5-8c

NRC Resolution: Review of the facility references provided show that the material submitted for examination preparation was inaccurate. The question was deleted.

(6) Question 2-16

Facility Comment: The answer key shows MOV-1275A, B, and C and MOV-1373 as receiving a "Auto" close signal on SI. The auto close signals were removed. Reference Attachment I-2.

NRC Resolution: Review of the facility reference provided showed that the material submitted for examination preparation was inaccurate. The answer key was changed accordingly.

(7) Question 3-16

Facility Comment: The candidate may use the new value for Tave Program which is 587.8 °F. This will be the new value used when the unit starts up after this refueling.

NRC Resolution: Review of the facility reference provided showed that the material submitted for examination preparation was outdated. The new values were utilized in calculating an alternate answer which was added to the answer key.

(8) Question 4-6

Facility Comment: In part a, change part of answer referring to skin dose to 7.5 Rem/Quarter.

NRC Resolution: GET booklet, pg. 10, attachment I-3 supports the proposed answer. The answer key was changed accordingly. In addition, post examination review revealed that the requirements referred to in 4-6c are located in an administrative procedure and not in the HP manual. Therefore, the answer key to 4-6a was changed and 4-6c was deleted.

(9) Question 4-8

Facility Comment: Change the answer to "true". See attachment I-4.

NRC Resolution: Procedure 1-OP-6.4 requires both that the prelube pump be run for two minutes and that a precaution not to operate the prelube pump for more than two minutes be observed. This combination renders the question, as written, ineffective as a measurement item. The question and answer were deleted.

4. Post-Examination Review

The examinations, facility comments and the results were reviewed in Region II. Following this review, the following changes were made to the SRO examination and answer key:

a. Question 7.9

Although the concept of the question was valid, there was insufficient clarification of and distinction between the distractors. Question was deleted.

b. Question 7.20

Choice (c) is also accepted. On reviewing the objectives of the two procedures we find that 3.2 is very similar to 3.1. therefore, the correct answer is (b) or (c).

c. Question 8.14

We believe that the material asked for is within the scope of knowledge of an SRO; however, the procedure appears to be poorly constructed and therefore the question is ambiguous. Question was deleted.

5 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 was no generic weakness noted during the oral examination.

Enclosure 3 (1 of 2) MASTER

U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: North Anna
Reactor Type: Westinghouse, 3 Loop
Date Administered: October 29, 1984
Examiners: S. Lawyer , T. Rogers
Applicant: ______________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheets 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	Total	Score	Value	Category
25	25	____	____	5. Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics
25	25	____	____	6. Plant Systems: Design, Control & Instrumentation
25	25	____	____	7. Procedures-Normal, Abnormal, Emergency Radiological Control
25	25	____	____	8. Administrative Procedures, Conditions and Limitations
100	100	____	____	TOTALS

Final Grade ________%

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

____
Applicant Signature

5.0 Theory of Nuclear Power Plant Operations, Fluids, and Thermodynamics (25.0)

✓5.1 The reactivity curves currently used at North Anna should specify (1.0)

(a) Cycle 4 for Unit 1 and
Cycle 3 for Unit 2.

(b) Cycle 3 for Unit 1 and
Cycle 2 for Unit 2.

(c) Cycle 3 for Unit 1 and
Cycle 3 for Unit 2.

(d) Cycle 4 for Unit 1 and
Cycle 2 for Unit 2.

✓5.2 Which of the following will <u>NOT</u> change over core life? (1.0)

(a) The acceptable AFD target band.
(b) The minimum acceptable shutdown margin.
(c) The control rod reactivity worth.
(d) The power defect reactivity worth.

✓5.3 The change in reactivity associated with a change in Keff from 0.920 to 1.004 is approximately (1.0)

(a) 0.091
(b) 0.084
(c) 0.087
(d) 0.080

✓5.4 Which of the following is <u>NOT</u> a characteristic of subcritical multiplication (1.0)

(a) If the reactor is shutdown long enough, the source range instruments will lose their ability to determine the subcritical multiplication level even though the core may still be at MOL.

(b) Doubling the indicated count rate by reactivity additions will reduce the margin to critical by approximately one half.

(c) For equal reactivity additions, it takes longer for the equilibrium subcritical multiplication level to be reached as Keff approaches unity.

(d) If ten steps of rod withdrawal increases the subcritical multiplication level by 10cps, then twenty steps of rod withdrawal will increase the subcritical multiplication level by approximately 20 cps.

5.5 Which of the following demonstrates the effects of the delayed neutron fraction changing over core life? (1.0)

(a) A lower boron concentration.
(b) A higher rod bite.
(c) A higher startup rate for equal reactivity additions.
(d) A larger (more negative) moderator temperature coefficient.

5.6 An estimated critical position has been calculated for a reactor startup that is to be performed 15 hours after a trip following a 60-day full power run. Which of the following actions will contribute to a higher actual rod positions than the calculated ECP? (1.0)

(a) Feeding the steam generators to increase level by 15%.
(b) Delaying the startup six hours longer than anticipated.
(c) Increasing the steam dump pressure setpoint by 100 psig.
(d) Increasing the pressurizer level using the dilute mode of boron concentration control.

5.7 Which of the following will contribute to a smaller (less negative) doppler coefficient over core life? (1.0)

(a) Fuel densification.
(b) Clad creep.
(c) Crud buildup on the fuel cladding.
(d) Fission gases released to the gap between the fuel and cladding .

5.8 Which of the following is a true statement concerning the moderator temperature coefficient? (1.0)

(a) The MTC tends to drive the neutron flux toward the top of the core over core life.

(b) The MTC increases (more negative) as the boron concentration increases.

(c) The MTC effects the axial neutron flux distribution more than the radial neutron flux distribution.

(d) The MTC will not change with a change in temperature if the boron concentration is maintained constant.

5.9 Which of the following statements describes the behavior of Xenon and Samrium? (1.0)

(a) After a reactor trip occurs, xenon concentration initially increases and samarium initially decreases.

(b) After a reactor trip occurs xenon will eventually decay to a xenon free condition, but a samarium free condition will not occur until after the next refueling outage.

(c) The xenon and samarium peak concentration following a trip occurs at a time independent of the previous power level.

(d) Xenon concentrations may increase or decrease when taking the plant from Mode 3 to full power but samarium will always decrease during this transient after the core's equilibrium samarium has been reached.

5.10 Which of the following radioactive isotopes found in the reactor coolant would NOT indicate a leak through the fuel cladding? (1.0)

(a) I-131
(b) Xe-133
(c) Co-60
(d) Kr-85

5.11 Which of the following is a true statement concerning radioactive decay? Remember the atomic number is the number of protons and the mass number is the number of neutrons plus protons. (1.0)

(a) When an element decays by beta emission, the new element will have increased in atomic number by one and the mass number will remain the same as the original element.

(b) When an element decays by alpha emission, the new element will have decreased in atomic number and mass number by two, from the original element.

(c) When an element decays by neutron emission, the new element will have increased in atomic number by one and decreased in mass number by one, from the original element.

(d) When an element decays by gamma emission, the new element will have increased in atomic number by one and the mass number will remain the same as the original element.

5.12 The highest internal stresses placed on a pressurized system boundary such as the reactor vessel is (1.0)

(a) on the thickest components during a heatup.
(b) on the thinest components during a heatup.
(c) on the thickest components during a cooldown.
(d) on the thinest components during a cooldown.

5.13 The need to change the RTNDT of the reactor vessel over the life of the plant is a result of: (1.0)

(a) thermal cycles (heatup and cooldown transients).
(b) pressure cycles (changes in pressure).
(c) gamma irradiation.
(d) neutron irradiation.

5.14 Which of the following actions will increase the DNBR? Assume Mode 1 and no reactor trip occurs. (1.0)

(a) Tripping a reactor coolant pump.
(b) Closing reactor coolant loop stop valves.
(c) Closing reactor coolant loop stop valves in a loop with a nonoperating reactor coolant pump.
(d) Closing a mainsteam stop valve.

5.15 The rod bow penalty used in calculating the nuclear enthalpy rise hot channel factor is a function of (1.0)

(a) total core flow.
(b) fuel burnup.
(c) nuclear power.
(d) reactor coolant system pressure.

5.16 Which of the following is a true statement when adjusting the power range channels to 100% based on a calculated calorimetric? (1.0)

(a) If the feedwater temperature used in the calorimetric calculation was lower than actual feedwater temperature, actual power will be higher than indicated power.

(b) If the reactor coolant pump heat input used in the calorimetric calculation was neglected, actual power will be less than indicated power.

(c) If the steam flow used in the calorimetric calculation was lower than actual steam flow, actual power will be less than indicated power.

(d) Caution must be taken in adjusting the power range channel gamma compensating voltage because overcompensating will cause actual power to be higher than indicated power and is not an input to the calorimetric calculation.

5.17 The largest contribution of hydrogen released to containment due to an accident involving inadequate core cooling and reactor vessel void formation is from (1.0)

(a) a zirconium - steam reaction.
(b) an aluminum - steam reaction.
(c) the release of dissolved hydrogen in the coolant from the hydrogen overpressure on the volume control tank.
(d) radiolysis of the coolant.

5.18 With the main steam temperature and pressure at 552°F and 1062 psia respectively, a main steam relief valve seat begins to leak to atmospheric pressure. The temperature of the steam three feet out of the relief valve is approximately (1.0)

(a) 552°F.
(b) 483°F.
(c) 296°F.
(d) 212°F.

5.19 The quality of steam from the steam generators refers to (1.0)

(a) the ratio of the liquid mass to the vapor mass.
(b) the ratio of the vapor mass to the liquid mass.
(c) the ratio of the liquid mass to the sum of the liquid and vapor masses.
(d) the ratio of the vapor mass to the sum of the liquid and vapor masses.

5.20 The thermal energy addition from the primary plant to the secondary plant in the steam power cycle T-S diagram, shown as figure 5.20, is represented by the path ____________. The cycle shown consists of SG, a turbine, a condenser, and feed pumps. (1.0)

(a) 5 to 1 to 2 to 3
(b) 1 to 2 to 3
(c) 2 to 3
(d) 2 to 3 to 4

5.21 The reactor coolant system is subcooled by approximately ____________ during Mode 3 when Tave is 400°F and the pressurizer pressure is 1000 psia. (1.0)

(a) 145°F.
(b) 125°F.
(c) 100°F.
(d) 75°F.

5.22 The mode of heat transfer used to transfer heat from the core to the steam generators during natural circulation is (1.0)

(a) conduction.
(b) convection.
(c) black body radiation.
(d) white body radiation.

5.23 When cooling down on natural circulation, the procedure cautions the operator not to depressurize the plant below 400 psig before the entire RCS is below 200°F. The reasons for this is (1.0)

(a) to reduce the combined thermal and pressure stresses on the reactor vessel.

(b) to prevent void formation in the reactor vessel.

(c) to be within the design transients specified in section 5 of Tech Specs.

(d) to prevent the reactor vessel from entering a condition susceptible to brittle fracture.

5.24 When applying the flow energy equation given below to a heat exchanger, which of the terms will drop out because they do not play a role in the heat exchange process? (1.0)

KE1 + h1 + q12 = KE2 + h2 + W12

(a) KE1 and KE2 only.
(b) KE1, q12, and KE2 only.
(c) KE1, KE2, and W12 only.
(d) h1, h2, and W12 only.

5.25 If a centrifugal pump is operating at 1800 rpm to give 400 gal/min at a discharge head of 20 psi, what would be the discharge head if the speed is increased in order to deliver 1600 gpm? (1.0)

(a) 40 psi
(b) 80 psi
(c) 160 psi
(d) 320 psi

Write "end of section 5.0" on your answer sheet.

6.0 - PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION (25.0)

6.1 The turbine driven AFW pump is powered by steam from: (1.0)

(a) all three steam generators.

(b) steam generators one and three.

(c) steam generators one and two.

(d) steam generators two and three.

6.2 The chemical added to the Quench Spray Subsystem Water is: (1.0)

(a) boron.

(b) sodium hydroxide.

(c) hydrazine.

(d) lithium hydroxide.

~~6.3 The RHR pumps are NOT used to transfer water from:~~ ~~(1.0)~~

DELETED

~~(a) the reactor cavity to the RWST during refueling operations.~~

~~(b) the RWST to the reactor cavity during refueling operations.~~

~~(c) the RWST to the cold leg accumulators during CLA filling operations.~~

~~(d) the containment sump to the RCS hot legs during emergency operations.~~

6.4 The main steam trip valves are: (1.0)

(a) located in the reactor containment.

(b) shut during phase A containment isolation.

(c) air and spring operated check valves.

(d) motor operated gate valves.

6.5 The cooling water normally supplied to the head of each waste gas compressor is supplied by the: (1.0)

(a) component cooling water system.

(b) chilled water system.

(c) circulating water system.

(d) service water system.

6.6 Which of the following components provides the most significant reduction in cesium from the reactor coolant when placed in service? (1.0)

(a) the reactor coolant filter upstream of the VCT.

(b) the cation bed demineralizer.

(c) the mixed bed demineralizer.

(d) the deborating demineralizer.

6.7 The pressurizer surge line connects to the: (1.0)

(a) hot leg of loop 1 and the spray lines are supplied from loop 2 and 3 cold legs.

(b) hot leg of loop 3 and the spray lines are supplied from loop 1 and 3 cold legs.

(c) cold leg of loop 3 and the spray lines are supplied from loop 1 and 2 cold legs.

(d) cold leg of loop 1 and the spray lines are supplied from loop 2 and 3 hot legs.

6.8 Which of the following is a load on a 4160V non-vital bus? (1.0)

(a) Pressurizer Heaters.

(b) Condensate pump.

(c) Charging pump.

(d) Rod drive MG set.

6.9 The primary fire protection system serving the emergency diesel generators is the: (1.0)

(a) CO_2 system.

(b) halon system.

(c) sprinkler system.

(d) deluge system.

6.10 Which of the following statements is true concerning the component cooling water system? (1.0)

(a) The control power for the 1H, 1J and 2H, 2J supply breakers is supplied by the 120 VAC vital buses.

(b) Component cooling water passes through the tube side of the component cooling water to service water heat exchanger.

(c) The component cooling water flow is automatically controlled to each load by a separate pressure regulating valve for each load.

(d) Makeup to the component cooling water system is normally provided automatically from the condensate system.

6.11 The motor driven AFW pumps will auto start on: (1.0)

(a) 1/2 breakers open on 2/3 main feedwater pumps.

(b) 1/3 Lo/Lo levels in 2/3 steam generators.

(c) low feed water pump discharge pressure.

(d) 2/3 high containment pressure signals.

6.12 Which of the following describes a response of the safety injection system due to the given operator action? (1.0)

(a) Closing the reactor trip breakers will re-initiate SI following reset and termination if the initiating conditions still exist.

(b) Manually initiating spray actuation and containment isolation phase B will initiate SI.

(c) Inadvertantly energizing two Hi-Hi containment pressure bistables during testing will initiate SI.

(d) Depressing Train A and Train B SI reset switches will secure all emergency core cooling water systems.

6.13 Which of the following will <u>NOT</u> <u>automatically</u> occur during phase A isolation? (1.0)

(a) BIT injection isolation valves open.

(b) BIT recirc isolation valve shuts.

(c) Containment vacuum pumps trip.

(d) Waste gas release isolation valve shuts.

6.14 Control Bank D is at 191 steps when a reactor trip occurs. Following the trip, the operator re-establishes all conditions to perform a unit startup except he does not reset the step counters. He then manually drives rods out until the rod control system prohibits any further rod motion. Control Bank D rods will then be at _________ when rod motion stops. (1.0)

(a) 0 steps.

(b) 29 steps.

(c) 37 steps.

(d) 228 steps.

6.15 Which of the following is NOT a fuel transfer system interlock? (1.0)

(a) The control panel in the containment building and the control panel in spent fuel building must give permission to move the transfer car to move toward or away from the containment building up-ender.

(b) The transfer tube gate valve must be fully open to allow transfer car operation.

(c) The spent fuel lifting arm cannot be operated unless the movable platform is over the spent fuel racks or the hoist is in the fully retracted position.

(d) The spent fuel building and containment building lifting arms must be down to allow transfer car operation.

6.16 Which of the following emergency diesel generator trips is NOT bypassed during an emergency start? (1.0)

(a) High jacket coolant temperature trip.

(b) Overexitation trip.

(c) Generator differential trip.

(d) Low lube-oil pressure trip.

6.17 P-14, Steam generator high water level interlock, will NOT generate a signal to: (1.0)

(a) trip the turbine.

(b) trip the reactor.

(c) trip the main feedwater pumps.

(d) shut all feedwater control valves.

6.18 An auto start signal to charging pump B will be generated when: (1.0)

(a) the 15H6 and 15H7 breakers are open. (Charging pump A and C normal supply breakers).

(b) the discharge pressure in the charging pump header is 2000 psig or less.

(c) an undervoltage condition exists on the J emergency bus.

(d) a safety injection signal from Train A or Train B is present.

6.19 Which of the following is true concerning reactor coolant pump trips? (1.0)

(a) The reactor coolant pump breaker trip coil de-energizes to stop the pump.

(b) If the 4Kv switchgear has an undervoltage condition for 5 seconds the associated reactor coolant pump will trip.

(c) If the bearing oil lift pump is tripped while the reactor coolant pump is running, the associated reactor coolant pump will trip.

(d) If the loop bypass valve drifts off the open seat to a midposition when the associated hot leg stop valve is closed, the associated reactor coolant pump will trip.

6.20 Which of the following is NOT a true statement concerning the reactor vessel level instrument system? (1.0)

(a) Each train consists of three d/p cells, all of which are located outside of the containment.

(b) A common line is shared between the upper range and the dynamic range d/p cells that senses the reactor coolant system pressure.

(c) The dynamic range, the full range, and the upper range level displays are effected by the number of reactor coolant pumps running.

(d) Temperature elements on the impulse lines, the wide range hot leg temperatures, and the wide range reactor coolant system pressures are inputs to the microprocessor for density compensation between the reactor coolant system side and the reference leg side at the d/p cell.

6.21 Which of the following is NOT a true statement concerning inputs to the primary coolant saturation meters? (1.0)

(a) All three loops provide inputs from the hot leg RTDs.

(b) All three loops provide inputs from the cold leg RTDs.

(c) All three loops provide inputs from the wide range pressure sensors.

(d) The low pressure lamp indicates the lowest pressure input when illuminated.

6.22 Which of the following reactor protection system inputs is provided by a power range detector (as opposed to a power range channel)? (1.0)

(a) The power range input to the OPΔT trip.

(b) The power range input to the high-high power trip.

(c) The power range input to the Low-high power trip.

(d) The power range input to the positive rate trip.

6.23 Which of the following detectors operate under the lowest applied voltage? (1.0)

(a) Source range detector.

(b) Intermediate range detector.

(c) Condenser air ejector radiation detector.

(d) Manipulator Crane radiation detector.

6.24 Which of the following is a post-accident monitoring instrument? (1.0)

(a) Feedwater flow rate.

(b) Axial Power Distribution monitoring system.

(c) Loose parts monitoring system.

(d) Pressurizer PORV block valve position indicator.

6.25 Which of the following is true concerning the turbine? (1.0)

(a) The turbine is rotated at low speed when shutdown in order to prevent distortion of the turbine casing.

(b) Turbine eccentricity is the measure of turbine speed.

(c) The turbine blades are cooled by hydrogen gas.

(d) Tech Specs require at least one turbine overspeed protection system be operable in Mode 2.

WRITE "END OF SECTION 6.0" ON YOUR ANSWER SHEET

7.0 PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL (25.0)

✓ 7.1 Procedure 1-AP-1.1 specifies that in the event of a failure of the Rod Control System to respond in automatic to a Tavg-Tref mismatch of 1.5°F or greater, certain immediate actions should be taken by the reactor operator. Which one of the following is a correct operator immediate action? (1.0)

a. Tavg error from program with no Rod Control "Urgent Failure" - Shift rod control to "MANUAL" but do not attempt any rod motion.

b. Tavg error from program with Rod Control "Urgent Failure" - Verify Rod Control in "Auto."

c. If unable to control Tavg, manually trip the reactor and turbine and go to 1-EP-0, Reactor Trip or Safety Injection.

d. Tavg error from program with no Rod Control "Urgent Failure" - Stabilize reactor power and control Tavg by use of operator control of the Turbine Generator and/or boration and dilution.

✓ 7.2 Which of the following is an indication of a malfunction in the Reactor Makeup Control System according to procedure 1-AP-2.0 "Reactor Makeup Control Malfunction." (1.0)

a. Rod insertion limit low or low-low alarms.

b. Tavg-Tref deviation of >1.5°F with no rod motion.

c. Decreasing pressurizer pressure.

d. Increased charging flow when pressurizer level <21.4%.

✓ 7.3 Procedure 1-AP-3, "Loss of Vital Instrumentation" provides the indications of, probable causes for, and the immediate and long term operator actions to be taken upon losing which of the following instrumentation? (1.0)

a. DC battery voltage

b. Feed flow

c. Power level

d. Air Ejector Radiation level

7.4 Which of the following correctly completes the statement from the Immediate Operator Actions of procedure 1-AP-9, "Reactor Coolant Pump Vibration"? (1.0)

"If vibration exceeds:

a. 1.5 mils seismic, increase frequency of vibration, seal leakoff flow and bearing temperature readings."

b. 7 mils proximity, increase frequency of vibration, seal leakoff flow and bearing temperature readings."

c. 3 mils seismic, trip the reactor and the affected pump."

d. 20 mils proximity, trip the reactor and the affected pump."

7.5 The attached figure 1-AP-10.1 displays the North Anna Power Station electrical distribution system. Point A on that drawing must be connected to a source of voltage. Which of the four numbered points is the proper connection for Point A? (1.0)

a. 1

b. 2

c. 3

d. 4

7.6 Upon loss of electrical power, diagnostic procedure 1-AP-10.1 requires operability testing of an Emergency Diesel Generator. Which of the following is sufficient for a satisfactory test of the Unit 1 diesel but not for the Unit 2 diesel? (1.0)

a. Diesel has successfully carried 1800 kw for one hour.

b. Diesel has accelerated to 900 RPM from ambient within 10 seconds.

c. Generator voltage has reached 3740 - 4580 volts within 10 seconds.

d. Frequency has reached 58.8 - 61.2 Hz within 10 seconds.

7.7 Which of the following is true of procedure 1-AP-11, "Loss of Residual Heat Removal System"? (1.0)

a. This procedure provides the necessary actions to be taken in the event of a loss of either train of the RHR system.

b. "Residual Heat Removal High Temperature Alarm" is listed as an indication in the procedure.

c. A Probable Cause listed in 1-AP-11 is "Closure of either RHR inlet valve caused by a spurious actuation signal."

d. One of the Immediate Operator Actions is "attempt to start the other Residual Heat Removal pump."

7.8 Which one of the following is an indication of loss of service water system according to 1-AP-12, "Loss of Service Water System." (1.0)

a. Low Service Water Return Header Flow (<6000 gpm) 1J-B3.

b. Charging pump seal/Gearbox Cooler inlet low flow (<3 gpm) 1B-B7, C8, E8; 2B-B7, C8, E8.

c. Low Service Water reservoir level (<252') 1J-E5.

d. Charging pump lube oil high temperature alarm (>260°F) 1C-B6, 2C-B6.

7.9 The following statements are based upon North Anna Unit 1 procedure "Low Condenser Vacuum," 1-AP-14. Which one is a true statement? (1.0)

a. <u>Purpose</u>: This procedure provides indications of, probable causes for and immediate and long term actions to be taken during a complete or partial loss of condenser vacuum.

b. <u>Indication</u>: Increasing exhaust hood temperature.

c. <u>Probable Cause</u>: Low Condenser hotwell level.

d. <u>Immediate Operator Action</u>: If Condenser pressure decreases below 9.5" Hg abs. and the turbine has not tripped automatically, manually trip the turbine. Trip the reactor if power >10%.

7.10 Which one of the following is an indication of excessive primary plant leakage according to 1-AP-16, "Excessive Primary Plant Leakage"? (1.0)

a. High Radiation on Containment High Range RMS-161.

b. RCP standpipe low level.

c. Increased auto operation of the gas strippers.

d. Decreasing level in the SI accumulators.

7.11 Which one of the following is not an immediate operator action required by "Main Control Room Uninhabitable with Operation at the Auxiliary Shutdown Panel," 1-AP-20. (1.0)

a. If unable to control Tavg, manually trip the turbine and go to 1-EP-0, reactor trip or safety injection.

b. If possible, trip the reactors before evacuating the control room.

c. If it is not possible to trip the reactor from the control room, trip the turbine at the front pedestal, then manually open the reactor trip breakers or the rod drive M-G output breakers.

d. Initiate EPIP-1.01 (classify as an alert) and 1-AP-50.

7.12 Which of the following major functions is accomplished by North Anna procedure 1-AP-20, "Main Control Room Uninhabitable with Operation at the Auxiliary Shutdown Panel." (1.0)

a. Maintain hot shutdown conditions (keff<0.99; 350°F > Tavg > 200°F).

b. Establish and maintain natural circulation.

c. If the emergency busses are not energized, start the emergency diesels from the auxiliary shutdown panel.

d. Maintain Tavg so that steam generator header pressure is between 985 psig and 1005 psig.

7.13 In the North Anna Unit 1 abnormal procedures is a series of eight procedures all grouped under 1-AP-22, entitled, "Steam Generator Auxiliary Feedwater System Alternate Lineups." From the following statements, choose the one which is true of those eight attached procedures. (1.0)

a. None require immediate operator actions.

b. The purpose of this series of procedures is to provide the indications of, probable causes for, and the immediate and long term actions to be taken in the event of a complete or partial loss of feedwater flow.

c. In each of the eight procedures, a probable cause listed is "Secondary Coolant Breaks on the Secondary Side of a Steam Generator."

d. Each of the eight procedures lists "abnormal steam generator levels" as an indication.

7.14 Which of the following statements is true of 1-AP-28.1, "Loss of Instrument Air Outside Containment"? (1.0)

a. Indication: at 88 psig: Instrument air low pressure annunciator, audible, and visual alarms on the main control board.

b. Indication: at 75 psig: Instrument air compressor trouble annunciator, audible, and visual alarms on the main control board, due to auto start of back-up compressors.

c. One of the immediate operator actions under certain conditions is "immediately secure all reactor coolant pumps and the charging pump."

d. Valves that will close on loss of air are listed in an Appendix attached to 1-AP-28.1.

7.15 Which one of the following statements regarding North Anna abnormal procedures (APs) is correct? (1.0)

a. "Inadvertent opening of a feedwater heate ›'pass valve" is a probable cause for both 1-AP-3/, 'Excessive Heat Removal due to Feedwater System Malfunctions" and 1-AP-38 "Excessive Load Increase."

b. 1-AP-39, "Flooding of Turbine Building," provides the indications of, probable causes for, and the immediate and long term actions to be taken following a rupture of a main condenser circulation water expansion joint (worst case flooding) as well as following more minor flooding.

c. The operator immediate actions as listed in 1-AP-43, "Vibration and Loose Parts Monitoring Panel Alarm" must be done by Unit 1 or 2 CRO.

d. 1-AP-45, "Generator Core Monitor Alarm," procedure states that a possible indication of generator overheating is "a step increase or a gradual increase in the strip chart recorder reading."

7.16 Which of the following is a probable cause of a Unit 1 annunciator light 1-A-4 entitled "Rod Control Urgent Failure"? (1.0)

a. Internal failure of power cabinet from slave cycler failure.

b. Internal failure of power cabinet from printed card removed.

c. Internal failure of logic cabinet from phase failure detector.

d. Normal alarm during pickup of a dropped rod procedure.

7.17 Which of the following combinations of setpoints correspond to those which will initiate Unit 1 annunciator 1B-7 (1B-G1) "PRZ RELIEF TANK HI-LOW LEVEL"? (1.0)

a. Hi 92% - Lo 16%

b. Hi 85% - Lo 21%

c. Hi 81% - Lo 32%

d. Hi 78% - Lo 66%

7.18 Termination of safety injection (SI) is governed by five separate procedures at North Anna Unit 1. One of these procedures states the SI termination criteria as: (1.0)

RCS pressure - stable or increasing
RCS subcooling - 50°F
Pressurizer level - increasing
SG level - >10% or AFW flow > 730 gpm

For which of the following procedures is the above termination criteria appropriate?

a. 1-ES-0.2, SI Termination Following Spurious SI.

b. 1-ES-1.1, SI Termination Following Loss of Reactor Coolant.

c. 1-ES-2.1, SI Termination Following Loss of Secondary Coolant.

d. 1-ES-2.2, SI Termination Following Excessive RCS Cooldown.

7.19 Which of the following statements about 1-ES-0.1, "Reactor Trip Response" is correctly stated? (1.0)

a. The purpose of this procedure is to provide the necessary instructions to stabilize and control the plant following a reactor trip with a safety injection.

b. This procedure is entered from 1-EP-0, reactor trip or safety injection, step 5 after verification of SI actuation.

c. RCP trip criteria - trip any RCP if component cooling water to that pump is lost. Trip all RCPs if either of the conditions listed below is met:

(1) SI is on
(2) RCS pressure - ≤1230 [1680] psig

d. SI reinitiation criteria following spurious SI - Reinitiate SI if any one of the parameters listed below occurs:

(1) RCS pressure - <1765 psig
(2) RCS subcooling - <50°F
(3) pressurizer level - <10%

7.20 Which of the following procedures states as its entry conditions "This procedure is entered from 1-ES-3.4, SI termination following steam generator tube rupture"? (1.0)

a. 1-EP-3, Steam Generator Tube Rupture

b. 1-ES-3.1, SGTR Alternate Cooldown by Backfilling RCS

c. 1-ES-3.2, Post SGTR Cooldown and Depressurization

d. 1-ES-3.3, SGTR with Secondary Depressurization

7.21 The Function Restoration Procedures C.1 through C.4 are entitled as follows: (1.0)

1-FRP-C.1, Response to Inadequate Core Cooling
1-FRP-C.2, Response to Degraded Core Cooling
1-FRP-C.3, Response to Potential Loss of Core Cooling
1-FRP-C.4, Response to Saturated Core Conditions

A copy of F-0.2 is attached. It directs the user to the above function restoration procedures under certain conditions. From the following list, choose the condition which is correct for its referenced letter on F-0.2.

a. RVLIS Narrow Range >84%

b. RVLIS Narrow Range >48%

c. RVLIS Wide Range > 1RCP 42%
2RCPs 61%
3RCPs 100%

d. Core exit TCs <1200°F but >700°F

7.22 You need to direct an operator to the proper location in the North Anna procedures to find the "containment integrity checklist." Choose the <u>one</u> correct statement from the following: (1.0)

a. It can be found as an attachment to 1-OP-18.0 entitled "Containment Access."

b. It can be found as a separate procedure, 1-OP-18.3 entitled "Containment Integrity Checklist" which is in the 1-OP-18.0 "Containment Access" series.

c. It can be found as an attachment to 1-OP-1.0 entitled "Unit Startup Operation."

d. It can be found as a separate procedure, 1-OP-1E entitled "Containment Integrity Checklist" which is in the 1-OP-1.0 "Unit Startup Operation" series.

7.23 Which of the following is a 10 CFR 20 occupational dose limit that does not require form NRC-4 to be kept on record? (1.0)

a. 3 rems per quarter - whole body

b. 1250 mrems per quarter - whole body

c. 5 rems per year - whole body

d. 7500 mrems per quarter - hands and forearms.

7.24 According to the North Anna Health Physics Manual, which one of the following is equal to one rem? (1.0)

a. a dose of 1 R due to X, gamma radiation, or beta radiation

b. a dose of 1 rad due to X or gamma radiation or 0.1 rad due to beta radiation

c. a dose of 0.1 rad due to alpha particles or fast neutrons

d. a dose of 0.1 rad due to alpha particles, 0.2 rad due to beta radiation, and 0.33 rad due to thermal neutrons.

7.25 Which one of the following statements is a true statement from the North Anna Health Physics Manual Section F "General Rules Concerning Activities While in Restricted Areas." (1.0)

a. During routine station operations, dose control will not be in effect.

b. RWPs must be co-signed by the dose control technician on duty.

c. Personnel issued self-reading dosimeters shall always enter the appropriate exposure data on form HP-3.1.2.1 after reading the dosimeter.

d. Dose control shall be established only by Health Physics and only in the vicinity of the clean change/locker room.

WRITE "END OF SECTION 7.0" ON YOUR ANSWER SHEET.

8.0 ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS (25.0)

8.1 Which of the following is true of AFD limits? (1.0)

a. The limits on axial flux difference assure that the $F_Q(z)$ upper bound envelope is not exceeded during either normal operation or Condition II events.

b. Target flux difference is determined at equilibrium Xenon conditions.

c. The periodic updating of the target flux difference value is necessary to reflect long term changes in peak Xenon concentrations.

d. During rapid plant thermal power reductions, resulting changes in Tavg will cause the AFD to deviate outside the target band at reduced thermal power levels.

8.2 If the shutdown margin is less than 1.77% Δ K/K, then the required action would be to ____________________, during mode 1 operations: (1.0)

a. immediately initiate a shutdown and emergency borate at greater than or equal to 10 gpm using the boric acid tank until a new shutdown margin calculation has been done to verify the shutdown margin is 1.77% Δ K/K.

b. immediately initiate and continue boration at greater than or equal to 10 gpm using the refueling water storage tank to borate until the shutdown margin is 1.77% Δ K/K.

c. immediately initiate and continue boration at greater than or equal to 10 gpm using the boric acid tank to borate until the shutdown margin is 1.77% Δ K/K.

d. immediately initiate a shutdown and initiate and continue boration of greater than or equal to 10 gpm using the boric acid tank to borate until the rod insertion limits are satisfied.

8.3 The pressurizer pressure-high reactor trip channels have a channel functional test to be performed monthly. A review of the logs indicates that the test is normally due on the tenth of each month. The logs show, however, that it was done on July 29, August 31, and October 3. What would be the date of the maximum allowable extension of the surveillance test this month without declaring the channels inoperable? (1.0)

a. November 4

b. November 6

c. November 10

d. November 11

8.4 The Technical Specifications require that the Unit 2 Boron Injection Tank be operable. With the Boron Injection Tank inoperable, in certain modes, action must be taken within one hour to restore the tank to an operable status or further action must be taken. Which one (1) of the following describes the applicable modes and the further action that must be taken? (1.0)

	Applicable Modes	Further Action
a.	1, 2, 3, and 4	Be in cold shutdown and borated to a shutdown margin equivalent to 1.77% delta k/k at 200°F within the next 20 hours.
b.	1, 2, and 3	Be in hot shutdown and borated to a shutdown margin equivalent to 1.77% delta k/k at 200°F within the next 12 hours.
c.	1, 2, 3, and 4	Be in hot standby and borated to a shutdown margin equivalent to 1.77% delta k/k at 200°F within the next 12 hours.
d.	1, 2, and 3	Be in hot standby and borated to a shutdown margin equivalent to 1.77% delta k/k at 200°F within the next six hours.

8.5 Which one (1) of the following is not a "Limiting Safety System Setting" trip setpoint? (1.0)

a. Reactor Coolant Pump underfrequency $\geq$56.1 Hz - each bus.

b. Turbine trip - low trip system pressure $\geq$45 psig.

c. Power Range, Neutron flux, high negative rate $\leq$5% of rated thermal power with a time constant $\geq$2 seconds.

d. Pressurizer water level - low $\geq$18% of instrument span.

8.6 North Anna Unit 2 Technical Specifications LCO 3.8.1.1 requires two separate and independent diesel generators. It further specifies certain diesel support equipment as a limiting condition for operation. Which of the following is not part of that LCO? (1.0)

a. Each with a separate 125-volt battery bank and charger.

b. Each with a separate day tank containing a minimum of 750 gallons of fuel.

c. A fuel storage system containing a minimum of 45,000 gallons of fuel.

d. A separate fuel transfer pump.

8.7 Which of the following is a North Anna Unit 1 Refueling Operations Technical Specification LCO? (1.0)

a. With a reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

(1) Either a K_{eff} of 0.95 or less, which includes a 1.77% delta k/k conservative allowance for uncertainties, or

(2) A boron concentration of greater than or equal to 20,000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

b. The reactor shall be subcritical for at least 72 hours prior to movement of irradiated fuel in the reactor pressure vessel.

c. The containment building penetrations shall be in the following status:

(1) The equipment door closed and held in place by a minimum of four bolts,

(2) A minimum of one door in each airlock is closed, and

(3) Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

(a) Closed by an isolation valve, blind flange, or manual valve, or

(b) Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

d. Direct communications shall be maintained between the control room and personnel in the containment building.

8.8 ADM-5.7 entitled "Correcting Data on Completed Procedures" makes which of the following provisions. (1.0)

a. It is permissible to make corrections to data which has been recorded. This must be performed in the following manner: draw a line through the erroneous data and record correct data. Initial and date correction.

b. The cognizant supervisor shall list any available substantiating information.

c. The individual who conducted the test and the cognizant supervisor shall attest to the corrected data by affixing their signature and the date.

d. The Station Nuclear Safety and Operating Committee shall review the circumstances and the Chairman shall sign and date the form when their review is completed.

8.9 Which of the following is true of ADM-5.16, "Emergency Usage Procedure Writer's Guide"? (1.0)

a. This writer's guide provides guidance applicable to procedures intended for use during normal as well as abnormal conditions.

b. Emergency Plan Implementing Procedures are defined as station procedures which offer guidance to address specific events beyond design basis conditions.

c. Spaces for entering checkmarks, notations, or data are to be avoided in the body of Emergency Plan Implementing Procedures but may be used as necessary in normal operating procedures.

d. This writer's guide addresses format, writing instructional steps, mechanics of style, and such miscellaneous items as status trees and foldouts.

✓ 8.10 According to ADM-11.7, "Cycle Core Exposure Calculations," Attachment 3.2, which of the following limitations or instructions apply during the initial return to power following a refueling shutdown or following a cold shutdown where fuel assemblies have been handled? (1.0)

a. The rate of reactor power increase shall be limited to 1% of full power in an hour between 25% and 100% of full power.

b. The rate of reactor power increase shall be limited to 1% of full power in an hour between 50% and 100% of full power.

c. The rate of reactor power increase shall be limited to 3% of full power in an hour between 25% and 100% of full power.

d. The rate of reactor power increase shall be limited to 3% of full power in an hour between 50% and 100% of full power.

✓ 8.11 Which of the following is true of tagging of systems and/or components at North Anna according to ADM-14.0? (1.0)

a. Special Order Tags, form 625.6 are grey tags.

b. The operator shall be responsible for placing all tags.

c. Maintenance personnel to whom the tagging record is issued will independently review the tag selection and authorize the operator to place the tags.

d. When more than one department is to work on the same equipment, a separate tag will be placed for each department unless a fifteen minute headway tag has been authorized.

8.12 Which one of the following statements is true of jumpers as controlled at North Anna by administrative procedure ADM-14.1, "Jumpers (temporary modification)"? (1.0)

a. Temporary hose connections necessary to perfrom a test are defined as constituting a jumper if their use is described in the text or drawings of the FSAR.

b. Jumpers not controlled by any approved procedure shall only be used with the shift supervisor's prior knowledge and approval.

c. In order to control their use, ADM-14.1 only permits the use of jumpers as described in either an MOP or an MMP.

d. For those jumpers that are installed by an approved procedure, a jumper log form shall be initiated and all pertinent data associated with the installation recorded.

8.13 The following statements refer to "Transportation of Contaminated Injured Personnel," EPIP-5.01. Which one correctly states a fact about that procedure? (1.0)

a. An entry condition is "any time deemed necessary by the Radiological Assessment Director or Shift Supervisor."

b. Provision is made to assure notification of Medical College of Virginia (MCV) prior to the ambulance (transporting contaminated injured victims) leaving the site.

c. The data that must be reported to the MCV when they are notified includes the number of neutron irradiated victims but does not include the number of gamma irradiated victims.

d. This procedure only requires notification of MCV if there are ten victims or greater.

8.14 You are the Shift Supervisor. The plant is in mode 1 when there is a failure of a PORV to close after pressure reduction. You believe this may affect the health and safety of the public. You have PORV flow as indicated by accoustical monitoring equipment and RCS presssure is less than 1600 psig. In accordance with the attached procedure, step 3.a(1), determine the event category. The proper category is: (1.0)

a. Safety, Shutdown, or Assessment System Event.

b. Reactor Coolant System Event.

c. Radioactivity Event.

d. Hazard to Station Operation.

8.15 Response to the four event classifications (Notification of Unusual Event, Alert, Site Area emergency, and General Emergency) ____________________. (1.0)

a. is controlled by two separate procedures depending upon severity.

b. is always governed by a procedure whose entry condition states, "Entry, from EPIP-1.01, Emergency Manager Controlling Procedure."

c. requires evaluation of issuance of Radioiodine blocking agent to onsite personnel on recommendation of Radiological Assessment Director for two of the four classifications (Site area emergency and General Emergency).

d. does not require Damage Control Assessment for "Notification of Unusual Event."

8.16 During implementation of the North Anna Emergency Plan Implementing Procedures, survey results may indicate 10 CFR 20, quarterly limits will be exceeded. North Anna procedure EPIP-4.04, "Emergency personnel radiation exposure" provides the Emergency Manager with guidance. Which of the following is true of a properly authorized dose received while performing volunteer emergency duties? (1.0)

a. It need not be included as part of the worker's current quarterly occupational exposure record and need not be added to the previous accumulated occupational exposure record of the worker.

b. It must be included as part of the worker's current quarterly occupational exposure record but need not be added to the previous accumulated occupational exposure record of the worker.

c. It need not be included as part of the worker's current quarterly occupational exposure record but must be added to the previous accumulated occupational exposure record of the worker.

d. It must be included as part of the worker's current quarterly occupational exposure record and must be added to the previous accumulated occupational exposure record of the worker.

8.17 Assume your reactors are in cold shutdown for refueling and maintenance. Incore instruments have been withdrawn to their storage position and the incore instrumentation thimble retraction has just been completed. All access doors to the reactor cavity are locked. Flooding of the refueling cavity in preparation for refueling was started six hours ago. It has just been determined that the water level in the refueling cavity is decreasing slowly. Assume you are the senior licensed SRO on shift under the above conditions. You have decided to send an auxiliary operator into the reactor cavity in an effort to locate the leakage source. Which of the following would be the anticipated dose rate range in the cavity? (1.0)

a. <30 mr/hr.

b. 30-300 mr/hr.

c. 3-30 R/hr.

d. 300-3,000 R/hr.

8.18 Which of the following is true of a site evacuation conducted in accordance with EPIP-5.05, "Site Evacuation." (1.0)

a. All personnel not responding to the emergency should turn in their security badge as they evacuate the station but should retain their pocket dosimeter and TLD.

b. All personnel not responding to the emergency should turn in their security badge, pocket dosimeter, and TLD as they evaucate the station.

c. If the wind is from the north, use the primary remote assembly area.

d. The key for the primary remote assembly area can only be obtained from the on-shift reservoir operator at the North Anna Dam or from the control room.

8.19 As Station Emergency Director, it is anticipated that you may need to initiate some Emergency Plan Implementing Procedures. This is reflected by listing "direction of the Station Emergency Manager" as an entry condition to the EPIP. Which of the following EPIPs have such a listing? (1.0)

a. EPIP-6.01, "Re-entry/recovery Guideline."

b. EPIP-5.08, "Damage Control Guideline."

c. EPIP-5.07, "Administration of Radioprotective Drugs."

d. EPIP-4.26, "High Level Activity Sample Analysis."

8.20 Which one of the following is a condition requiring stopping of all work and immediate evacuation of containment according to the precautions and limitations in 1-OP-4.1, "Controlling Procedure for Refueling"? (1.0)

a. The "hi flux at shutdown" alarm is actuated during movement of fuel.

b. Loss of audible neutron count rate (less than two tones per minute) after offloading 3/4 of the reactor core.

c. The station evacuation alarm sounds.

d. Declaration of an "alert."

8.21 Which of the following conditions for refueling must be met if refueling is to continue? (1.0)

a. The flow rate of reactor coolant through the reactor coolant system shall be $\geq$3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

b. Valve 1-CH-217 (PG to blender isolation) shall be locked in the open position to assure an adequate supply of borated water in the event emergency boration is required.

c. The following boron injection flow paths shall be operable:

(1) a flow path from the boric acid tanks via a boric acid transfer pump through a charging pump to the RCS, and

(2) the flow path from the refueling water storage tank via a charging pump to the RCS.

d. Two boric acid transfer pumps shall be operable if neither charging pump is operable.

8.22 Which one of the following evolutions is <u>not</u> covered by 1-OP-4.2, "Receipt and Storage of New Fuel"? (1.0)

a. Unloading containers from truck to container storage area.

b. Removing the new fuel from the containers to the storage area.

c. Movement of fuel from new fuel storage area to the spent fuel pit.

d. Movement of fuel from the spent fuel pit to the new fuel elevator.

8.23 ADM-19.3 (shift conduct, relieving the shift) states, "as a minimum, the shift supervisor and control room operator will perform the following prior to assuming the watch:" (Supply the nine items listed in the procedure.) (3.0)

Write "End of Section 8.0" on your answer sheet.

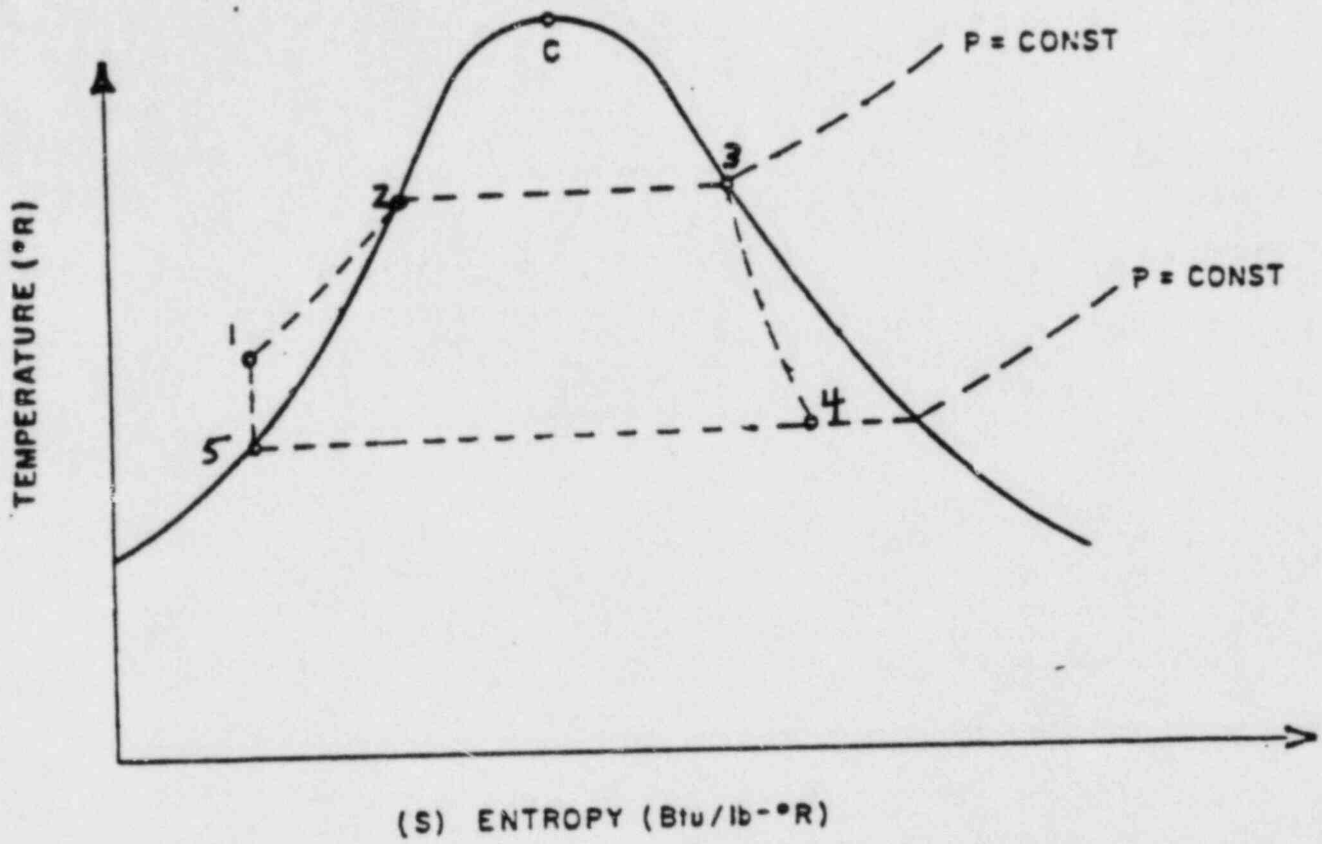


FIGURE 5.20

TABLE D-1b
Properties of Dry Saturated Steam *(continued)*
Temperature

Temp., °F	Abs press., psia	Specific volume: Sat. liquid	Specific volume: Sat. vapor	Enthalpy: Sat. liquid	Enthalpy: Evap.	Enthalpy: Sat. vapor	Entropy: Sat. liquid	Entropy: Evap.	Entropy: Sat. vapor
t	p	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
32	0.08854	0.01602	3306	0.00	1075.8	1075.8	0.0000	2.1877	2.1877
35	0.09995	0.01602	2947	3.02	1074.1	1077.1	0.0061	2.1709	2.1770
40	0.12170	0.01602	2444	8.05	1071.3	1079.3	0.0162	2.1435	2.1597
45	0.14752	0.01602	2036.4	13.06	1068.4	1081.5	0.0262	2.1167	2.1429
50	0.17811	0.01603	1703.2	18.07	1065.6	1083.7	0.0361	2.0903	2.1264
60	0.2563	0.01604	1206.7	28.06	1059.9	1088.0	0.0555	2.0393	2.0948
70	0.3631	0.01606	867.9	38.04	1054.3	1092.3	0.0745	1.9902	2.0647
80	0.5069	0.01608	633.1	48.02	1048.6	1096.6	0.0932	1.9428	2.0360
90	0.6982	0.01610	468.0	57.99	1042.9	1100.9	0.1115	1.8972	2.0087
100	0.9492	0.01613	350.4	67.97	1037.2	1105.2	0.1295	1.8531	1.9826
110	1.2748	0.01617	265.4	77.94	1031.6	1109.5	0.1417	1.8106	1.9577
120	1.6924	0.01620	203.27	87.92	1025.8	1113.7	0.1645	1.7694	1.9339
130	2.2225	0.01625	157.34	97.90	1020.0	1117.9	0.1816	1.7296	1.9112
140	2.8886	0.01629	123.01	107.89	1014.1	1122.0	0.1984	1.6910	1.8894
150	3.718	0.01634	97.07	117.89	1008.2	1126.1	0.2149	1.6537	1.8685
160	4.741	0.01639	77.29	127.89	1002.3	1130.2	0.2311	1.6174	1.8485
170	5.992	0.01645	62.06	137.90	996.3	1134.2	0.2472	1.5822	1.8293
180	7.510	0.01651	50.23	147.92	990.2	1138.1	0.2630	1.5480	1.8109
190	9.339	0.01657	40.96	157.95	984.1	1142.0	0.2785	1.5147	1.7932
200	11.526	0.01663	33.64	167.99	977.9	1145.9	0.2938	1.4824	1.7762
210	14.123	0.01670	27.82	178.05	971.6	1149.7	0.3090	1.4508	1.7598
212	14.696	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566
220	17.186	0.01677	23.15	188.13	965.2	1153.4	0.3239	1.4201	1.7440
230	20.780	0.01684	19.382	198.23	958.8	1157.0	0.3387	1.3901	1.7288
240	24.969	0.01692	16.323	208.34	952.2	1160.5	0.3531	1.3609	1.7140
250	29.825	0.01700	13.821	216.48	945.5	1164.0	0.3675	1.3323	1.6998
260	35.429	0.01709	11.763	228.64	938.7	1167.3	0.3817	1.3043	1.6860
270	41.858	0.01717	10.061	238.84	931.8	1170.6	0.3958	1.2769	1.6727
280	49.203	0.01726	8.645	249.06	924.7	1173.8	0.4096	1.2501	1.6597
290	57.556	0.01735	7.461	259.31	917.5	1176.8	0.4234	1.2238	1.6472
300	67.013	0.01745	6.466	269.59	910.1	1179.7	0.4369	1.1980	1.6350
310	77.68	0.01755	5.626	279.92	902.6	1182.5	0.4504	1.1727	1.6231
320	89.66	0.01765	4.914	290.28	894.9	1185.2	0.4637	1.1478	1.6115
3[illegible]0	103.06	0.01776	4.307	300.68	887.0	1187.7	0.4769	1.1233	1.6002
3[illegible]0	118.01	0.01787	3.788	311.13	879.0	1190.1	0.4900	1.0992	1.5891

TABLE D-1b
Properties of Dry Saturated Steam *(continued)*
Temperature

Temp., °F	Abs. press., psia	Specific volume		Enthalpy			Entropy		
		Sat. liquid	Sat. vapor	Sat. liquid	Evap	Sat. vapor	Sat. liquid	Evap	Sat. vapor
t	p	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
350	134.63	0.01799	3.342	321.63	870.7	1192.3	0.5029	1.0754	1.5783
360	153.04	0.01811	2.957	332.18	852.2	1194.4	0.5158	1.0519	1.5677
370	173.37	0.01823	2.625	342.79	853.5	1196.3	0.5286	1.0287	1.5573
380	195.77	0.01836	2.335	353.45	844.6	1198.1	0.5413	1.0059	1.5471
390	220.37	0.01850	2.0836	364.17	835.4	1199.6	0.5539	0.9832	1.5371
400	247.31	0.01864	1.8633	374.97	826.0	1201.0	0.5664	0.9608	1.5272
410	276.75	0.01878	1.6700	385.83	816.3	1202.1	0.5788	0.9386	1.5174
420	308.83	0.01894	1.5000	396.77	806.3	1203.1	0.5912	0.9166	1.5078
430	343.72	0.01910	1.3499	407.79	796.0	1203.8	0.6035	0.8947	1.4982
440	381.59	0.01926	1.2171	418.90	785.4	1204.3	0.6158	0.8730	1.4887
450	422.6	0.0194	1.0993	430.1	774.5	1204.6	0.6280	0.8513	1.4793
460	466.9	0.0196	0.9944	441.4	763.2	1204.6	0.6402	0.8298	1.4700
470	514.7	0.0198	0.9009	452.8	751.5	1204.3	0.6523	0.8083	1.4606
480	566.1	0.0200	0.8172	464.4	739.4	1203.7	0.6645	0.7868	1.4513
490	621.4	0.0202	0.7423	476.0	726.8	1202.8	0.6766	0.7653	1.4419
500	680.8	0.0204	0.6749	487.8	713.9	1201.7	0.6887	0.7438	1.4325
520	812.4	0.0209	0.5594	511.9	686.4	1198.2	0.7130	0.7006	1.4136
540	962.5	0.0215	0.4649	536.6	656.6	1193.2	0.7374	0.6568	1.3942
560	1133.1	0.0221	0.3868	562.2	624.2	1186.4	0.7621	0.6121	1.3742
580	1325.8	0.0228	0.3217	588.9	588.4	1177.3	0.7872	0.5659	1.3532
600	1542.9	0.0236	0.2668	610.0	548.5	1165.5	0.8131	0.5176	1.3307
620	1786.6	0.0247	0.2201	646.7	503.6	1150.3	0.8398	0.4664	1.3062
640	2059.7	0.0260	0.1798	678.6	452.0	1130.5	0.8679	0.4110	1.2789
660	2365.4	0.0278	0.1442	714.2	390.2	1104.4	0.8987	0.3485	1.2472
680	2708.1	0.0305	0.1115	757.3	309.9	1067.2	0.9351	0.2719	1.2071
700	3093.7	0.0369	0.0761	823.3	172.1	995.4	0.9905	0.1484	1.1389
705.4	3206.2	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580

TABLE D-1a*
Properties of Dry Saturated Steam +
Pressure

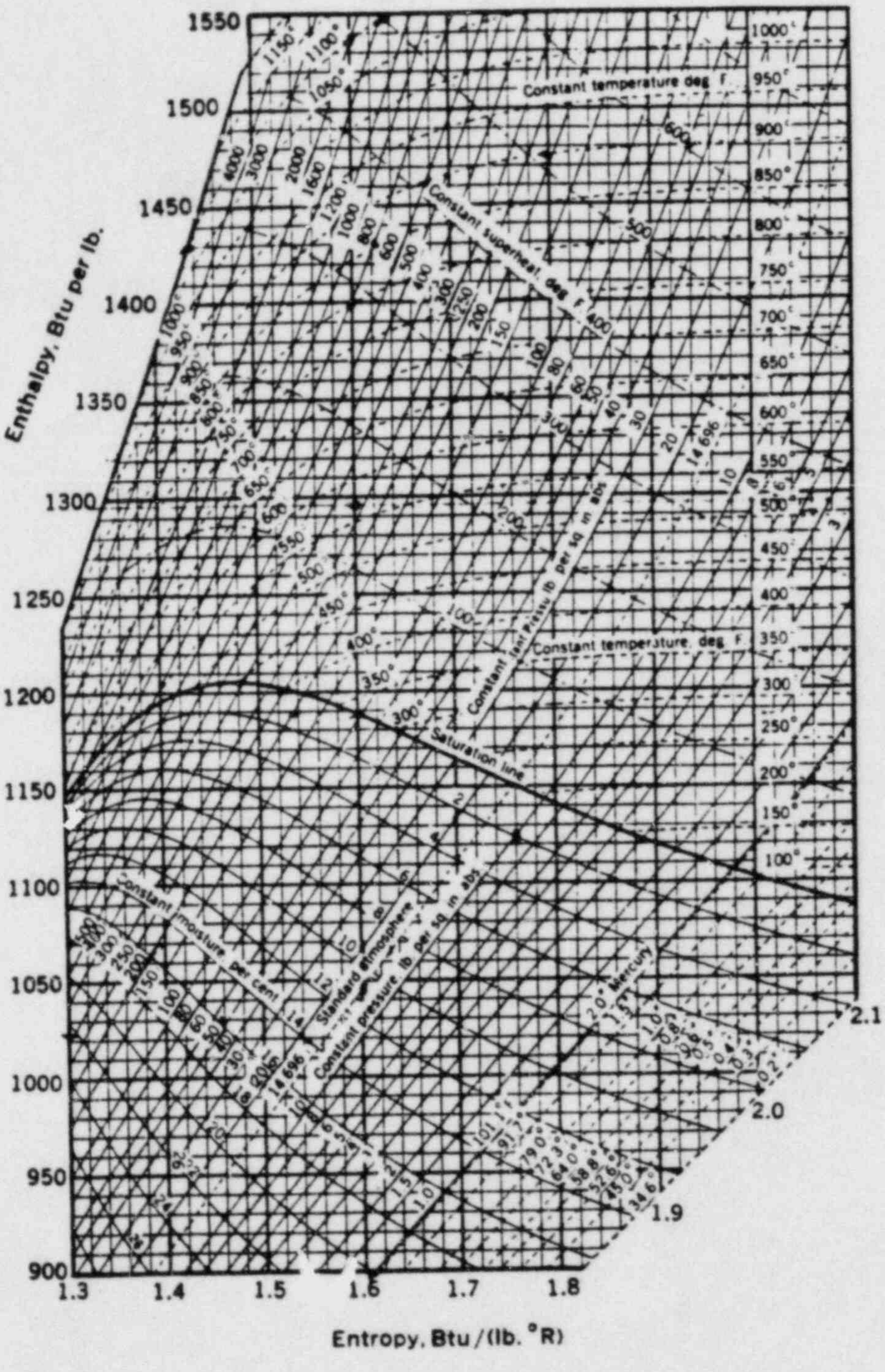
Abs. press., psia	Temp., °F	Specific volume		Enthalpy			Entropy		
		Sat. liquid	Sat. vapor	Sat. liquid	Evap.	Sat. vapor	Sat. liquid	Evap.	Sat. vapor
p	t	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
1.0	101.74	0.01614	333.6	69.70	1036.3	1106.0	0.1326	1.8456	1.9782
2.0	126.08	0.01623	173.73	93.99	1022.2	1116.2	0.1749	1.7451	1.9200
3.0	141.48	0.01630	118.71	109.37	1013.2	1122.6	0.2008	1.6855	1.8863
4.0	152.97	0.01636	90.63	120.86	1006.4	1127.3	0.2198	1.6427	1.8625
5.0	162.24	0.01640	73.52	130.13	1001.0	1131.1	0.2347	2.6094	1.8441
6.0	170.06	0.01645	61.98	137.96	996.2	1134.2	0.2472	1.5820	1.8292
7.0	176.85	0.01649	53.64	144.76	992.1	1136.9	0.2581	1.5586	1.8167
8.0	182.86	0.01653	47.34	150.79	988.5	1139.3	0.2674	1.5383	1.8057
9.0	188.28	0.01656	42.40	156.22	985.2	1141.4	0.2759	1.5203	1.7962
10	193.21	0.01659	38.42	161.17	982.1	1143.3	0.2835	1.5041	1.7876
14.696	212.00	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566
15	213.03	0.01672	26.29	181.11	969.7	1150.8	0.3135	1.4415	1.7549
20	227.96	0.01683	20.089	196.16	960.1	1156.3	0.3356	1.3962	1.7319
25	240.07	0.01692	16.303	208.42	952.1	1160.6	0.3533	1.3606	1.7139
30	250.33	0.01701	13.746	218.82	945.3	1164.1	0.3680	1.3313	1.6993
35	259.28	0.01708	11.898	227.91	939.2	1167.1	0.3807	1.3063	1.6870
40	267.25	0.01715	10.498	236.03	933.7	1169.7	0.3919	1.2844	1.6763
45	274.44	0.01721	9.401	243.36	928.6	1172.0	0.4019	1.2650	1.6669
50	281.01	0.01727	8.515	250.09	924.0	1174.1	0.4110	1.2474	1.6585
55	287.07	0.01732	7.787	256.30	919.6	1175.9	0.4193	1.2316	1.6509
60	292.71	0.01738	7.175	262.09	915.5	1177.6	0.4270	1.2168	1.6438
65	297.97	0.01743	6.655	267.50	911.6	1179.1	0.4342	1.2032	1.6374
70	302.92	0.01748	6.206	272.61	907.9	1180.6	0.4409	1.1906	1.6315
75	307.60	0.01753	5.816	277.43	904.5	1181.9	0.4472	1.1787	1.6259
80	312.03	0.01757	5.472	282.02	901.1	1183.1	0.4531	1.1676	1.6207
85	316.25	0.01761	5.168	286.39	897.8	1184.2	0.4587	1.1571	1.6158
90	320.27	0.01766	4.896	290.56	894.7	1185.3	0.4641	1.1471	1.6112
95	324.12	0.01770	4.652	294.56	891.7	1186.2	0.4692	1.1376	1.6068
100	327.81	0.01774	4.432	298.40	888.8	1187.2	0.4740	1.1286	1.6026
110	334.77	0.01782	4.049	305.66	883.2	1188.9	0.4832	1.1117	1.5948

TABLE D-1a
Properties of Dry Saturated Steam *(continued)*
Pressure

Abs press., psia	Temp., °F	Specific volume		Enthalpy			Entropy		
		Sat liquid	Sat vapor	Sat liquid	Evap	Sat vapor	Sat liquid	Evap	Sat vapor
p	t	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
120	341.25	0.01789	3.728	312.44	877.9	1190.4	0.4916	1.0962	1.5878
130	347.32	0.01796	3.455	318.81	872.9	1191.7	0.4995	1.0817	1.5812
140	353.02	0.01802	3.220	324.82	868.2	1193.0	0.5069	1.0682	1.5751
150	358.42	0.01809	3.015	330.51	863.6	1194.1	0.5138	1.0556	1.5694
160	363.53	0.01815	2.834	335.93	859.2	1195.1	0.5204	1.0436	1.5640
170	368.41	0.01822	2.675	341.09	854.9	1196.0	0.5266	1.0324	1.5590
180	373.06	0.01827	2.532	346.03	850.8	1196.9	0.5325	1.0217	1.5542
190	377.51	0.01833	2.404	350.79	846.8	1197.6	0.5381	1.0116	1.5497
200	381.79	0.01839	2.288	355.36	843.0	1198.4	0.5435	1.0018	1.5453
250	400.95	0.01865	1.8438	376.00	825.1	1201.1	0.5675	0.9588	1.5263
300	417.33	0.01890	1.5433	393.84	809.0	1202.8	0.5879	0.9225	1.5104
350	431.72	0.01913	1.3260	409.69	794.2	1203.9	0.6056	0.8910	1.4966
400	444.59	0.0193	1.1613	424.0	780.5	1204.5	0.6214	0.8630	1.4844
450	456.28	0.0195	1.0320	437.2	767.4	1204.6	0.6356	0.8378	1.4734
500	467.01	0.0197	0.9278	449.4	755.0	1204.4	0.6487	0.8147	1.4634
550	476.94	0.0199	0.8424	460.8	743.1	1203.9	0.6608	0.7934	1.4542
600	486.21	0.0201	0.7698	471.6	731.6	1203.2	0.6720	0.7734	1.4454
650	494.90	0.0203	0.7083	481.8	720.5	1202.3	0.6826	0.7548	1.4374
700	503.10	0.0205	0.6554	491.5	709.7	1201.2	0.6925	0.7371	1.4296
750	510.86	0.0207	0.6092	500.8	699.2	1200.0	0.7019	0.7204	1.4223
800	518.23	0.0209	0.5687	509.7	688.9	1198.6	0.7108	0.7045	1.4153
850	525.26	0.0210	0.5327	518.3	678.8	1197.1	0.7194	0.6891	1.4085
900	531.98	0.0212	0.5006	526.6	668.8	1195.4	0.7275	0.6744	1.4020
950	538.43	0.0214	0.4717	534.6	659.1	1193.7	0.7355	0.6602	1.3957
1000	544.61	0.0216	0.4456	542.4	649.4	1191.8	0.7430	0.6467	1.3897
1100	556.31	0.0220	0.4001	557.4	630.4	1187.7	0.7575	0.6205	1.3780
1200	567.22	0.0223	0.3619	571.7	611.7	1183.4	0.7711	0.5956	1.3667
1300	577.46	0.0227	0.3293	585.4	593.2	1178.6	0.7840	0.5719	1.3559
1400	587.10	0.0231	0.3012	598.7	574.7	1173.4	0.7963	0.5491	1.3454
1500	596.23	0.0235	0.2765	611.6	556.3	1167.9	0.8082	0.5269	1.3351
2000	635.82	0.0257	0.1878	671.7	463.4	1135.1	0.8619	0.4230	1.2849
2500	668.13	0.0287	0.1307	730.6	360.5	1091.1	0.9126	0.3197	1.2322
3000	695.36	0.0346	0.0858	802.5	217.8	1020.3	0.9731	0.1885	1.1615
3206.2	705.40	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580

Properties of Superheated Steam*

Abs. press., psia (Sat. temp., °F)		Temperature, °F 200	300	400	500	600	700	800	900	1000	1100	1200	1400
	v	392.6	452.3	512.0	571.6	631.2	690.8	750.4	809.9	869.5	929.1	988.7	1107.8
1	h	1150.4	1195.8	1241.7	1288.3	1335.7	1383.8	1432.8	1482.7	1533.5	1585.2	1637.7	1745.7
(101.74)	s	2.0512	2.1153	2.1720	2.2233	2.2702	2.3137	2.3542	2.3923	2.4283	2.4625	2.4952	2.5566
	v	78.16	90.25	102.26	114.22	126.16	138.10	150.03	161.95	173.87	185.79	197.71	221.6
5	h	1148.8	1195.0	1241.2	1288.0	1335.4	1383.6	1432.7	1482.6	1533.4	1585.1	1637.7	1745.7
(162.24)	s	1.8718	1.9370	1.9942	2.0456	2.0927	2.1361	2.1767	2.2148	2.2509	2.2851	2.3178	2.3792
	v	38.85	45.00	51.04	57.05	63.03	69.01	74.98	80.95	86.92	92.88	98.84	110.77
10	h	1146.6	1193.9	1240.6	1287.5	1335.1	1383.4	1432.5	1482.4	1533.2	1585.0	1637.6	1745.6
(193.21)	s	1.7927	1.8595	1.9172	1.9689	2.0160	2.0596	2.1002	2.1383	2.1744	2.2068	2.2413	2.3028
	v		30.53	34.68	38.78	42.86	46.94	51.00	55.07	59.13	63.19	67.25	75.37
14.696	h		1192.8	1239.9	1287.1	1334.8	1383.2	1432.3	1482.3	1533.1	1584.8	1637.5	1745.5
(212.00)	s		1.8160	1.8743	1.9261	1.9734	2.0170	2.0576	2.0958	2.1319	2.1662	2.1989	2.2603
	v		22.36	25.43	28.46	31.47	34.47	37.46	40.45	43.44	46.42	49.41	55.37
20	h		1191.6	1239.2	1286.6	1334.4	1382.9	1432.1	1482.1	1533.0	1584.7	1637.4	1745.4
(227.96)	s		1.7808	1.8396	1.8918	1.9392	1.9829	2.0235	2.0618	2.0978	2.1321	2.1648	2.2263
	v		11.040	12.628	14.168	15.688	17.198	18.702	20.20	21.70	23.20	24.69	27.68
40	h		1186.8	1236.5	1284.8	1333.1	1381.9	1431.3	1481.4	1532.4	1584.3	1637.0	1745.1
(267.25)	S		1.6994	1.7608	1.8140	1.8619	1.9058	1.9467	1.9850	2.0212	2.0555	2.0883	2.1498
	v		7.259	8.357	9.403	10.427	11.441	12.449	13.452	14.454	15.453	16.451	18.446
60	h		1181.6	1233.6	1283.0	1331.8	1380.9	1430.5	1480.8	1531.9	1583.8	1636.6	1744.8
(292.71)	s		1.6492	1.7135	1.7678	1.8162	1.8605	1.9015	1.9400	1.9762	2.0106	2.0434	2.1049
	v			6.220	7.020	7.797	8.562	9.322	10.077	10.830	11.582	12.332	13.830
80	h			1230.7	1281.1	1330.5	1379.9	1429.7	1480.1	1531.3	1583.4	1636.2	1744.5
(312.03)	s			1.6791	1.7346	1.7836	1.8281	1.8694	1.9079	1.9442	1.9787	2.0115	2.0731
	v			4.937	5.589	6.218	6.835	7.446	8.052	8.656	9.259	9.860	11.060
100	h			1227.6	1279.1	1329.1	1378.9	1428.9	1479.5	1530.8	1582.9	1635.7	1744.2
(327.81)	s			1.6518	1.7085	1.7581	1.8029	1.8443	1.8829	1.9193	1.9538	1.9867	2.0484
	v			4.081	4.636	5.165	5.683	5.195	6.702	7.207	7.710	8.212	9.214
120	h			1224.4	1277.2	1327.7	1377.8	1428.1	1478.8	1530.2	1582.4	1635.3	1743.9
(341.25)	s			1.6287	1.6869	1.7370	1.7822	1.8237	2.8625	1.8990	1.9335	1.9664	2.0281
	v			3.468	3.954	4.413	4.861	5.301	5.738	6.172	6.604	7.035	7.895
140	h			1221.1	1275.2	1326.4	1376.8	1427.3	1478.2	1529.7	1581.9	1634.9	1743.5
(353.02)	s			1.6087	1.6683	1.7190	1.7645	1.8063	1.8451	1.8817	1.9163	1.9493	2.0110
	v			3.008	3.443	3.849	4.244	4.631	5.015	5.396	5.775	6.152	6.906
160	h			1217.6	1273.1	1325.0	1375.7	1426.4	1477.5	1529.1	1581.4	1634.5	1743.2
(363.53)	s			1.5908	1.6519	1.7033	1.7491	1.7911	1.8301	1.8667	1.9014	1.9344	1.9962
	v			2.649	3.044	3.411	3.764	4.110	4.452	4.792	5.129	5.466	6.136
180	h			1214.0	1271.0	1323.5	1374.7	1425.6	1476.8	1528.6	1581.0	1634.1	1742.9
(373.06)	s			1.5745	1.6373	1.6894	1.7355	1.7776	1.8167	1.8534	1.8882	1.9212	1.9831
	v			2.361	2.726	3.060	3.380	3.693	4.002	4.309	4.613	4.917	5.521
200	h			1210.3	1268.9	1322.1	1373.6	1424.8	1476.2	1528.0	1580.5	1633.7	1742.6
(381.79)	s			1.5594	1.6240	1.6767	1.7232	1.7655	1.8048	1.8415	1.8763	1.9094	1.9713
	v			2.125	2.465	2.772	3.066	3.352	3.634	3.913	4.191	4.467	5.017
220	h			1206.5	1266.7	1320.7	1372.6	1424.0	1475.5	1527.5	1580.0	1633.3	1742.3
(389-86)	s			1.5453	1.6117	1.6652	1.7120	1.7545	1.7939	1.8308	1.8656	1.8987	1.9607
	v			1.9276	2.247	2.533	2.804	3.068	3.327	3.584	3.839	4.093	4.597
240	h			1202.5	1264.5	1319.2	1371.5	1423.2	1474.8	1526.9	1579.6	1632.9	1742.0
(397.37)	s			1.5319	1.6003	1.6546	1.7017	1.7444	1.7839	1.8209	1.8558	1.8889	1.9510
	v				2.063	2.330	2.582	2.827	3.067	3.305	3.541	3.776	4.242
260	h				1262.3	1317.7	1370.4	1422.3	1474.2	1526.3	1579.1	1632.5	1741.7
(404.42)	s				1.5897	1.6447	1.6922	1.7352	1.7748	1.8118	1.8467	1.8799	1.9420
	v				1.9047	2.156	2.392	2.621	2.845	3.066	3.286	3.504	3.938
280	h				1260.0	1316.2	1369.4	1421.5	1473.5	1525.8	1578.6	1632.1	1741.4
(411.05)	s				1.5796	1.6354	1.6834	1.7265	1.7662	1.8033	1.8383	1.8716	1.9337
	v				1.7675	2.005	2.227	2.442	2.652	2.859	3.065	3.269	3.674
300	h				1260.0	1316.2	1368.3	1420.6	1472.8	1525.2	1578.1	1631.7	1741.0
(417.33)	s				1.5701	1.6268	1.6751	1.7184	1.7582	1.7954	1.8305	1.8638	1.9260
	v				1.4923	1.7036	1.8980	2.084	2.266	2.445	2.622	2.798	3.147
350	h				1251.5	1310.9	1365.5	1418.5	1471.1	1523.8	1577.0	1630.7	1740.3
(431.72)	s				1.5481	1.6070	1.6563	1.7002	1.7403	1.7777	1.8130	1.8463	1.9086
	v				1.2851	1.4770	1.6508	1.8161	1.9767	2.134	2.290	2.445	2.751
400	h				1245.1	1306.9	1362.7	1416.4	1469.4	1522.4	1575.8	1629.6	1739.5
(444.59)	s				1.5281	1.5894	1.6398	1.6842	1.7247	1.7623	1.7977	1.8311	1.8936



Mollier diagram for steam

EQUATION SHEET

$Q = m\Delta h$

$Q = UA\Delta T$

$x^3{}_p = xw$

$Q = mcp\Delta T$

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

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

$P = P_o e^{t/T}$

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

$$T = \frac{\beta - p}{\lambda p}$$

$$T = \frac{l^*}{p} + \frac{\beta - p}{\lambda p}$$

$$p = \frac{Keff-1}{Keff}$$

$$p = \frac{K_2 - K_1}{K_2 K_1}$$

$$\frac{CR1}{CR2} = \frac{1 - Keff_2}{1 - Keff_1}$$

$RR = \Sigma f\phi th$

$$SCR = \frac{S}{1-Keff}$$

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

$x^2(ft) = xw$

$\ell^* = 10^{-5}$ sec

$A = \lambda N$

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

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

$$t_{1/2} = \frac{0.693}{\lambda}$$

$$R/_{hr} = \frac{6CEn}{d^2}$$

$\lambda = 0.1\ sec^{-1}$

$q\ 1\text{-}2 = h_2 - h_1$

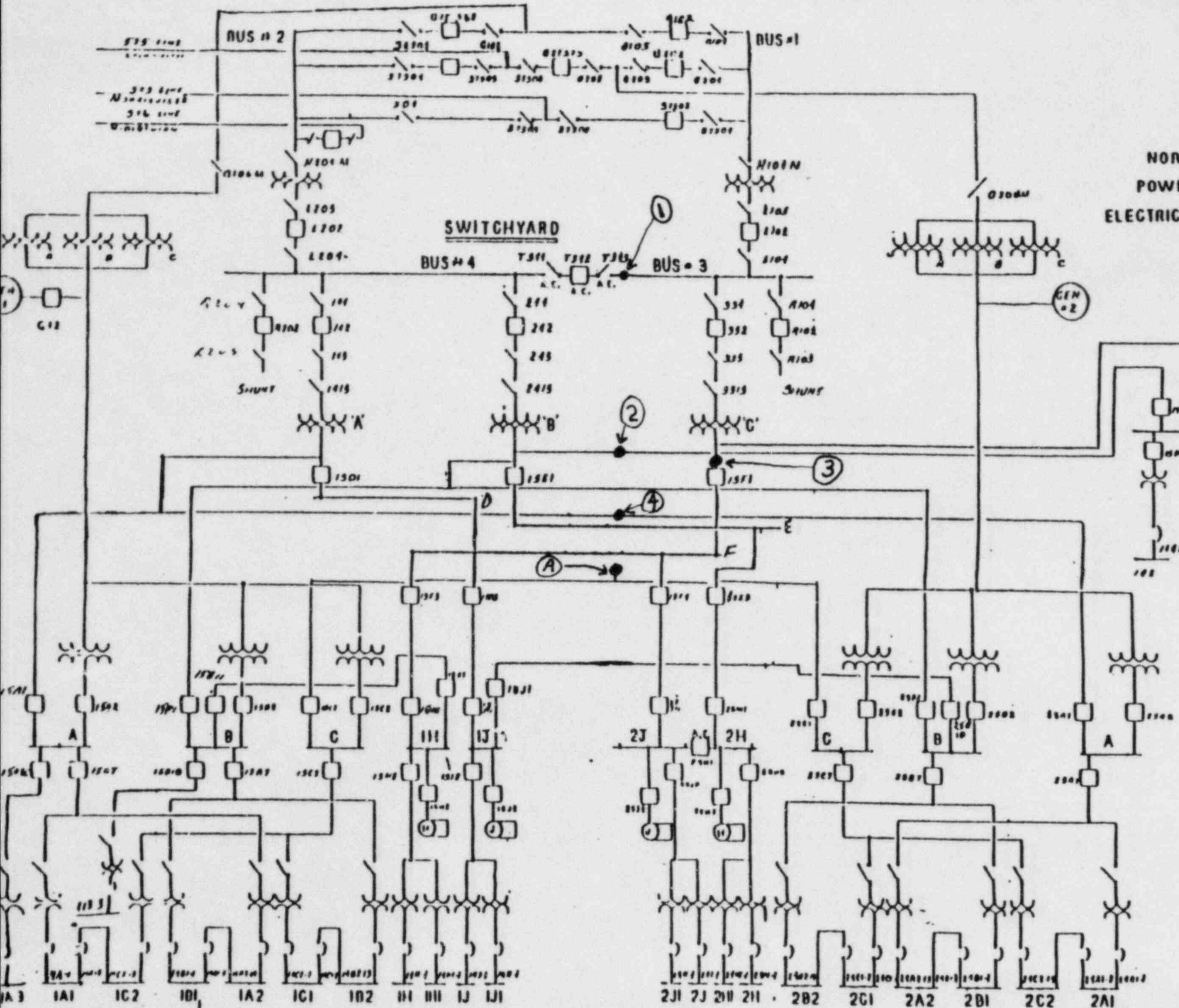
$q = h\, a \Delta t$

$x\dot{m} = xw$

$KE_1 + h_1 + q_{12} =$
$KE_2 + h_2 + w_{12}$

Where:
1) KE is Kenetic energy
2) w is work done
3) q is the heat transferred
4) h is the enthalpy

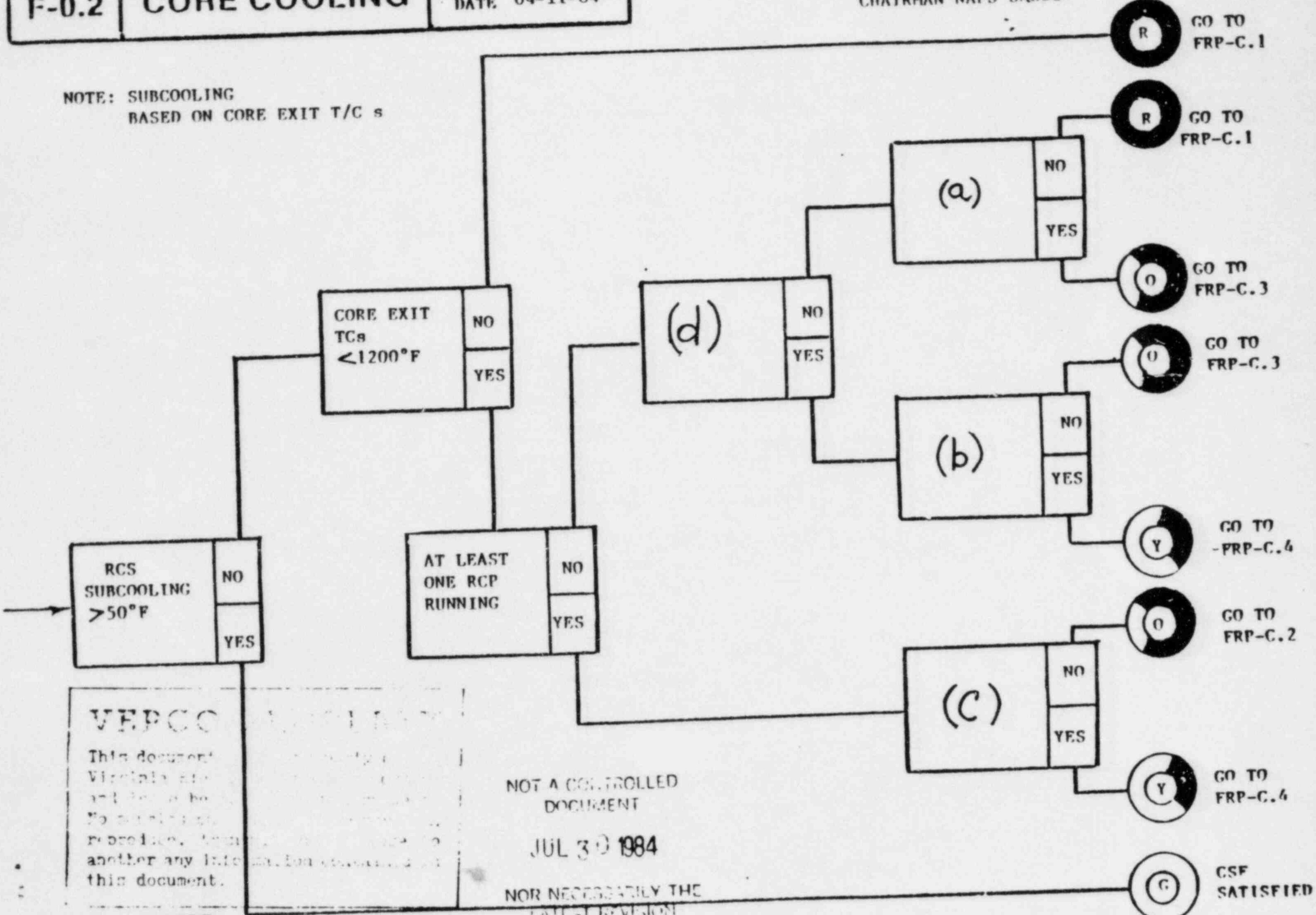
NORTH ANNA POWER STATION ELECTRICAL DISTRIBUTION



1-AP-10.1
Attachment 1
Page 1 of 6
09-09-83

Number	Symptom/Title	Revision/Date
F-0.2	CORE COOLING	REV-00 DATE 04-11-84

APPROVED: [signature]
CHAIRMAN NAPS SNSOC



SAFETY RELATED

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	INITIATE PROCEDURE:	
	a) BY: ___________	
	DATE: ___________	
	TIME: ___________	

NOTE: Continue through this and all further instructions unless otherwise directed to hold.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	IDENTIFY EVENT:	
	a) Event-TRANSPORT OF CONTAMINATED INJURED PERSONNEL	a) GO TO Step 2.b of this instruction.
	1) Initiate EPIP-5.01 Transport of Contaminated Injured Personnel	
	2) Verify initiation of EPIP-4.20, H.P. Actions for Transport of Injured Contaminated Personnel	
	3) Continue this instruction	
	b) Event-Any of the following:	b) GO TO Step 3 of this instruction.
	Radiation Release OR Fuel Handling Incident OR Secondary Release OR S/G Tube Rupture OR LOCA	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	(CONTINUED)	
	1) Request Health Physics initiate EPIP-4.01, Radiological Assessment Director Controlling Procedure, and continue this instruction	

* *

CAUTION: Declaration of the highest emergency class for which an Emergency Action Level is exceeded shall be made.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	ASSESSMENT AND CLASSIFICATION:	
	a) Refer to Index EPIP 1.01, of Attachment 1, Emergency Action Levels	
	AND	
	1) Using index, determine event category AND GO TO proper EAL tab	
	AND	
	2) Evaluate event, determine classification, AND Go To Step 4 of this procedure	
4.	NOTIFICATION AND VERIFICATION:	
	a) EOF- NOT ACTIVATED	a) If EOF activated, announce to staff the transfer of command from TSC to EOF and proceed to 4.b.
	b) TSC - NOT ACTIVATED	b) IF TSC activated, GO TO Step 6.

No. 97887220

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE	04
ATTACHMENT	INDEX	PAGE
1		1 of 38

****

CAUTION: Declaration of the highest emergency class for which an EAL is exceeded shall be made.

****

MASTER

5.0 THEORY OF NUCLEAR POWER PLANT OPERATIONS, FLUIDS, AND THERMODYNAMICS (ANSWERS) (25.0)

5.1 (a) (1.0)
REF: NAPS station curves 1/2 - 3.4 - 3.13

5.2 (b) (1.0)
REF: NAPS Tech Specs ¶3.1.1.1 and 3.2.1 and Station Curves 3.5-3.8

5.3 (a) (1.0)

$$P = \frac{K2 - K1}{K2K1} = \frac{1.004 - 0.92}{(1.004)(.92)} = \frac{.084}{.924} = .091$$

REF: Nuclear Energy Training, Reactor Operations, NUS Corp. ¶6.1

5.4 (d) (1.0)
REF: Nuclear Energy Training Reactor Operations, NUS Corp., ¶12.1

5.5 (c) (1.0)

REF: Nuclear Energy Training, Reactor Operations, NUS corp, ¶6.4

5.6 (c) (1.0)
REF: NAPS, 1-OP-1C, pg.3

5.7 (b) (1.0)
REF: NRC I&E Tech Manual, PWR, ¶1.1.7.1

5.8 (c) (1.0)
REF: NAPS Physics Training Lesson Plan, Dynamics of Flux Distribution. ¶5

5.9 (d) (1.0)
REF: Nuclear Energy Training Reactor Operations, NUS CORP, ¶10.4 and 10.3

5.10 (c) (1.0)
REF: NAPS UFSAR, Table 15.1-5

5.11 (a) (1.0)
REF: Nuclear Reactor Analysis, Duderstadt and Hamilton, 1976, pg. 13

5.12 (c) (1.0)
REF: NRC I&E Tech Manual, PWR, ¶2.2.3.2

5.13 (d) (1.0)
REF: NAPS Tech Spec Bases 3/4.4.9

5.14 (c) (Stops core bypass flow in loop) (1.0)
REF: NAPS UFSAR, ¶15.2.6.1.1

5.15 (b) (1.0)
REF: NAPS Tech Specs, ¶3.2.3

5.16 (b) (1.0)
REF: NAPS Lesson Plans, Thermo Exam, p.3.

5.17 (a) (1.0)
REF: TMI, Report to Commissioners and Public, Vol. II, part 2 p. 527

5.18 (c) by steam tables or Mollier Diagram (1.0)

Solving h1 = ’1190 BTU/lb

h1 = h2 ’1190 BTU/lb

1190 BTU/lb @ 14.7psia } 296°F

REF: Themo Fluid Flow, Heat Xfer for NPP, DPC p. 87-6

5.19 (d) (1.0)
REF: NAPS Thermo Lesson Plans p.4

5.20 (b) (1.0)
REF: Thermo, Fluid Flow and Heat Xfer for NPP, DPC p.97

5.21 (a) (1.0)
Tsat @ 1000psia = 544.61
Subcooling = 544.61 - 400°F = 144.61°F

REF: Thermo, Fluid Flow, & Heat Xfer for NPP, DPC, p. 184

5.22 (b) (1.0)
REF: NAPS Thermo Lesson Plan, Heat Transfer Methods p.1

5.23 (b) (1.0)
REF: NAPS, 1-ES-0.5B, p.6

5.24 (c) (1.0)
REF: Thermo, Fluid Flow, Heat Xfer for NPP, DPC, APP. D PJS, p.15

5.25 (d) (1.0)
400gpm to 1600 gpmn } increase speed 4X.

head = $4^2(20) = 16(20) = 320$psi

REF: Thermo, Fluid Flow, heat Xfer for NPP, DPC p.161.

6.0 PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION (ANSWERS) (25.0)

6.1 (a) (1.0)

Ref: NAPS Lesson Plans, AFW, Para. B.

6.2 (b) (1.0)

Ref: NAPS Lesson Plans, QS, RS, Casing Cooling, p.3.

~~6.3 (c)~~ DELETED ~~(1.0)~~

~~Ref: NAPS 1 OP4.1, Para. 4.3.10, 1 OP 14.0, 1 ES 1.4, and 1 OP 7.3~~

6.4 (c) (1.0)

Ref: NAPS Plant Manual, Vol. 3, Group 8, Para. B.1., C.1, C.2

6.5 (a) (1.0)

Ref: NAPS Plant Manual, Vol. 3, Group 30, p. 30-22-1.

6.6 (b) (1.0)

Ref: NRC I&E Tech Manual, PWR, Para. 3.1.2.1 &
NAPS P&ID CVCS, DN# 11715-FM-95A-9.

6.7 (b) (1.0)

Ref: NAPS RCS P&ID #11715-EM-93B-16.

6.8 (b) (1.0)

Ref: NAPS OFSAR, Figure 8.2-14.

6.9 (a) (1.0)

Ref: NAPS Plant Manual, Vol. 2, Para. 20-2-8.

6.10 (d) (1.0)

Ref: NAPS Plant Manual, Vol. 1, Group 12-7-1.

6.11 (d) (1.0)

Ref: NAPS Lesson Plans AFW Para. D & Tech Spec Table 3.3-3.

6.12 (a) (1.0)

Ref: NAPS Safeguards Actuation Signal Diagram, Sheet 8.

6.13 (d) (1.0)

Ref: NAPS 1-EP-0, Attachment 2, pp. 2-3.

6.14 ~~(b)~~ (d) ~~(BOU thinks they are at 191 steps & since the 200 step bank D rod block does not prohibit manual rod withdrawal~~, ~~they will stop at 228-191=37 steps)~~. (1.0)

Ref: I&E Tech Manual Para. 7.1.2.6 and NAPS Lesson Plans, Rod Control System.

6.15 (c) (1.0)

Ref: NAPS UFSAR Para. 9.1.4.6.1.2.

6.16 (c) (1.0)

Ref: NAPS UFSAR p. 8.3-17.

6.17 (b) (1.0)

Ref: NAPS UFSAR, Table 7.3-3.

6.18 (a) (1.0)

Ref: NAPS Plant Manual, Vol. 2, Group 26, Para. 3.2.

6.19 (d) (1.0)

Ref: NAPS Plant Manual, Vol. 2, Group 25, Para. C.3.

6.20 (b) (1.0)

Ref: NAPS RVUS Lesson Plan, Para. C.1, C.4, & Fig. 2.

6.21 (c) (1.0)

Ref: NAPS AP-46, Para. 5.7 & 5.8.

6.22 (a) (1.0)

Ref: NAPS Lesson Plans - Nuclear Instrumentation System Power Range Diagram.

6.23 (b) (only ion chamber) (1.0)

Ref: NAPS, Rad. Monitoirng Lesson Plan, Table I & II.
Nuclear Instrumentation Lesson Plan.

6.24 (d) (1.0)

Ref: NAPS Tech Specs, Table 3.3-10.

6.25 (d) (1.0)

Ref: NAPS Tech Specs 3.7.1.7 & NAPS Plant Manual,
Vol. 1, Group 11.

SECTION 7

Answers

7.1 Answer: c. (1.0)

Reference: North Anna Unit 1, "RCCA Deviation from Tavg Control," 1-AP-1.1, page 3 of 5.

7.2 Answer: a. (1.0)

Reference: North Anna Unit 1, "Reactor Makeup Control Malfunction," 1-AP-2.0, page 2 of 5.

7.3 Answer: b. (1.0)

Reference: North Anna Unit 1, "Loss of Vital Instrumentation," 1-AP-3, page 2 of 105.

7.4 Answer: d. (1.0)

Reference: North Anna Unit 1, "Reactor Coolant Pump Vibration," 1-AP-9, page 5 of 6.

7.5 Answer: c. (1.0)

Reference: 1. North Anna Unit 1, "Loss of Electrical Power," 1-AP-10, Attachment 1, page 1 of 6.

2. VEPCO drawing for Switchyard G1726-OL.

7.6 Answer: b. (1.0)

Reference: North Anna Unit 1, "Loss of Electrical Power Diagnostic," 1-AP-10.1, Attachment 3, page 3 of 5.

7.7 Answer: d. (1.0)

Reference: North Anna Unit 1, "Loss of Residual Heat Removal System," 1-AP-11, page 3 of 10.

7.8 Answer: a. (1.0)

Reference: North Anna Power Station, Unit 1 and 2, "Loss of Service Water System," 1-AP-12, page 2 of 8.

7.9 Answer: b. (1.0)

Reference: North Anna Power Station, Unit 1, "Low Condenser Vacuum," 1-AP-14, page 2 of 5.

7.10 Answer: c. (1.0)

Reference: North Anna Power Station, Unit 1, "Excessive Primary Plant Leakage," 1-AP-16, page 3 of 6.

7.11 Answer: a. (1.0)

Reference: North Anna, Unit 1, "Main Control Room Uninhabitable with Operation at the Auxiliary Shutdown Panel," 1-AP-20, page 3 of 8.

7.12 Answer: d. (1.0)

Reference: North Anna, Unit 1, "Main Control Room Uninhabitable with Operation at the Auxiliary Shutdown Panel, 1-AP-20, page 6 of 8.

7.13 Answer: a. (1.0)

Reference: North Anna, Unit 1, "Steam Generator Auxiliary Feedwater System Alternate Lineups," 1-AP-22 series.

7.14 Answer: c. (1.0)

Reference: North Anna, Unit 1, "Loss of Instrument Air - Outside of the Containment," 1-AP-28.1, page 2 of 6.

7.15 Answer: b. (1.0)

Reference: 1-AP-37, 39, 43, and 45.

7.16 Answer: d. (1.0)

Reference: North Anna, Unit 1, "Panel 1A - Main Control Board," 1-AR-1, Annunciator 1A-4; 1A-D1.

7.17 Answer: d. (1.0)

Reference: North Anna, Unit 1, "Panel 1B - Main Control Board," 1-AR-2, 1B-G1.

7.18 Answer: a. (1.0)

Reference: North Anna, Unit 1, "SI Termination Following Spurious SI," 1-ES-0.2, foldout for E-0 and ES-0 Guidelines.

7.19 Answer: d. (1.0)

Reference: North Anna, Unit 1, "Reactor Trip Response," 1-ES-0.1, page 1 and foldout.

7.20 Answer: b. (1.0)

Reference: North Anna Emergency Procedure "SGTR Alternate Cooldown by Backfilling RCS," 1-ES-3.1, page 1 of 8.

7.21 Answer: c. (1.0)

Reference: North Anna, Unit 1, F-0.2.

7.22 Answer: d. (1.0)

Reference: North Anna, Unit 1, "Containment Integrity Checklist," 1-OP-1E.

7.23 Answer: b. (1.0)

Reference: 10 CFR 20.101.

7.24 Answer: c. (1.0)

Reference: North Anna Health Physics Manual, page 1.2-2.

7.25 Answer: a. (1.0)

Reference: North Anna Health Physics Manual, page 1.3-6.

SECTION 8

Answers

8.1 Answer: b. (1.0)

Reference: N.A. TS B3/4 2-1 & 2

8.2 Answer: c. (1.0)

Reference: NA TS 3/4 1-1

8.3 Answer: b. (1.0)

Reference: NA TS 3/4 0-2

8.4 Answer: d. (1.0)

Reference: NA U2 TS 3/4 5-8

8.5 Answer: d. (1.0)

Reference: NA U2 TS Table 2.2-1, page 2-6

8.6 Answer: a. (1.0)

Reference: North Anna U2 TS 3/4 8-5

8.7 Answer: c. (1.0)

Reference: NA U2 TS 3/4 9-4

8.8 Answer: a. (1.0)

Reference: North Anna Administrative Procedure ADM-5.7, page 1 of 1.

8.9 Answer: d. (1.0)

Reference: North Anna Administrative Procedure ADM-5.16, page 5 of 90.

8.10 Answer: c. (1.0)

Reference: North Anna ADM-11.7, Attachment 3.2, page 3 of 8.

8.11 Answer: b. (1.0)

Reference: NA ADM-14.0, page 1 of 12.

8.12 Answer: b. (1.0)

Reference: NA ADM-14.1, page 3 of 6.

8.13 Answer: c. (1.0)

Reference: NA, EPIP-5.01, "Transportation of Contaminated Injured Personnel," page 3 of 4.

8.14 Answer: a. (1.0)

Reference: NA EPIP-1.01, "Emergency Manager Controlling Procedure," Attachment 1, page 2 of 38.

8.15 Answer: b. (1.0)

Reference: NA EPIP-1.02, "Response to Notification of Unusual Event," page 1 of 7.

8.16 Answer: d. (1.0)

Reference: 1. IE Information Notice No. 84-40.

2. North Anna EPIP-4.04, page 6 of 6.

8.17 Answer: d. (1.0)

Reference: 1. IE Information Notice No. 82-51, "Overexposures in PWR Cavities."

2. Surry 2 Event, April 1979.

8.18 Answer: a. (1.0)

Reference: North Anna EPIP 5.05, "Site Evacuation," page 2 of 5.

8.19 Answer: b. (1.0)

Reference: NA EPIP-5.08, "Damage Control Guidelines."

8.20 Answer: c. (1.0)

Reference: NA 1-OP-4.1, "Controlling Procedure for Refueling," page 15 of 47.

8.21 Answer: a. (1.0)

Reference: NA 1-OP-4.1, "Controlling Procedure for Refueling," page 5 of 47

8.22 Answer: d. (1.0)

Reference: NA 1-OP-4.2, "Receipt and Storage of New Fuel," page 2 of 26.

8.23 Answer: (0.33/ea)

1. Review all lighted annunciators in the Control Room

2. Read all entries, in the log for which he is responsible that were made since his own last entry and initial the log.

3. Review the "Abnormal System Status" board.

4. Review the Action Statement Status Log.

5. Discuss processes that are in progress and any known conditions that could create problems or constitute a safety hazard (e.g., maintenance, H.P. problems, etc.).

6. Persons returning to Shift from vacation, retraining, etc. should read and initial the logs for the previous seven days.

7. Review the Chemistry Status Board.

8. Complete SRO/CRO Shift Turnover Checklist (Attachments 1 and 2).

9. Review and initial the "REMARKS" section of all operator logs of the preceding shift.

Reference: ADM-19.3, article 1.1, page 1 of 5.

U. S. NUCLEAR REGULATORY COMMISSION (Baldwin)

REACTOR OPERATOR LICENSE EXAMINATION

Enclosure 3 (2 of 2)

Facility: NORTH ANNA

Reactor Type: WESTINGHOUSE

Date Administered: 10-29-84

Examiner: MARK 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	________	________	1. Principles of Nuclear Power Plant Operations, Thermodynamics, Heat Transfer and Fluid Flow
24.0 ~~25.0~~	25	________	________	2. Plant Design Including Safety and Emergency Systems
25.0	25	________	________	3. Instruments and Controls
23.25 ~~25.0~~	25	________	________	4. Procedures -- Normal, Abnormal, Emergency & Radiological Control
97.25 ~~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 The equilibrium xenon reactivity worth at 50% power is approximately (1.0)

a. equal to the equilibrium xenon reactivity worth at 100% power.

b. one-half the equilibrium xenon reactivity worth at 100% power.

c. two-thirds the equilibrium xenon reactivity worth at 100% power.

d. three-fourths the equilibrium xenon reactivity worth at 100% power.

1-2 Which of the following most closely describes the effects of a short unintentional emergency boration on your reactor at 75% power? Assume that rods are in manual. (1.0)

a. Reactor power initially decreases, T_{ave} increases, then reactor power increases to approximately the initial value.

b. T_{ave} initially decreases, reactor power decreases, then T_{ave} increases to approximately the initial value.

c. Reactor power initially decreases, T_{ave} decreases, then reactor power increases to approximately the initial value.

d. T_{ave} initially increases, reactor power decreases, then T_{ave} decreases to approximately the initial value.

1-3 The equilibrium samarium reactivity worth at 100% power is approximately (1.0)

a. four times the equilibrium samarium reactivity worth at 25% power.

b. three times the equilibrium samarium reactivity worth at 25% power.

c. twice the equilibrium samarium reactivity worth at 25% power.

d. equal to the equilibrium samarium reactivity worth at 25% power.

1-4 If reactor power doubles every 30 seconds the stable startup rate is: (1.0)

a. 1.39 DPM
b. 0.5 DPM
c. 0.6 DPM
d. 0.01 DPM

(continued on next page)

1-5 a. Would the integral control rod worth (increase, decrease or remain essentially the same) when T_{avg} is increased from 150° to 500°F? (0.5)

b. Would the differential control rod worth of bank D at, or near, 215 steps (increase, decrease, or remain essentially the same) as the core goes from BOL to EOL? (Assume HFP for both cases.) (0.5)

c. Would the integral control rod worth of bank D (increase, decrease, or remain essentially the same) when power is changed from 25 to 75%? (Consider the two steady states and not the transient between them, and assume no rod motion was used for the power increase.) (0.5)

1-6 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)

1-7 In the North Anna 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.

(continued on next page)

1-8 Which of the following most closely describes the effects of a short unintentional emergency boration on your reactor at 10^{-8} amps? (1.0)

a. Reactor power decreases, then T_{ave} decreases.

b. Reactor power decreases and T_{ave} remains the same.

c. T_{ave} decreases, then reactor power decreases.

d. T_{ave} increases, then reactor power decreases.

1-9 A reactor has the following characteristics:

Boron worth = 8.5 pcm/ppm
Burnup = 12,500 MWD/MTU
Critical boron concentration = 200 ppm
Equilibrium xenon = 2800 pcm
Peak xenon = 4400 pcm
Power defect = 2200 pcm
Shutdown bank rod worth = 3200 pcm
Total rod worth = 7800 pcm

The reactor has been operating at steady-state for three weeks when a trip from 100% power occurs. The shutdown rods are pulled two hours after the trip and the boron concentration is changed to 700 ppm two days after the trip.

a. By what amount (pcm) is the reactor subcritical immediately following the trip? (1.0)

b. By what amount (pcm) is the reactor subcritical eight hours after the trip? (1.0)

c. By what amount (pcm) is the reactor subcritical three days after the trip? (1.0)

NOTE: SHOW ALL CALCULATIONS

1-10 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. (1.25)

b. What is the new value of K_{eff} at this condition? (1.25)

(continued on next page)

1-11 A reactor is critical at 10^4 CPS when two (2) steam generator PORV's fail 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 new steady state. (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

1-12 The highest rate of production of decay heat immediately following shutdown from full power (100%) is about ______. (1.0)

a. 1%
b. 2%
c. 6%
d. 10%

1-13 State whether the following statements are true or false.

a. At steady-state, full power operations, values of T_H indicated by the loop-mounted RTDs should always read slightly lower than the thermocouples mounted at the core exit. (0.5)

b. Less than half of the power generated by the core is eventually rejected to the main condenser circulating water. (0.5)

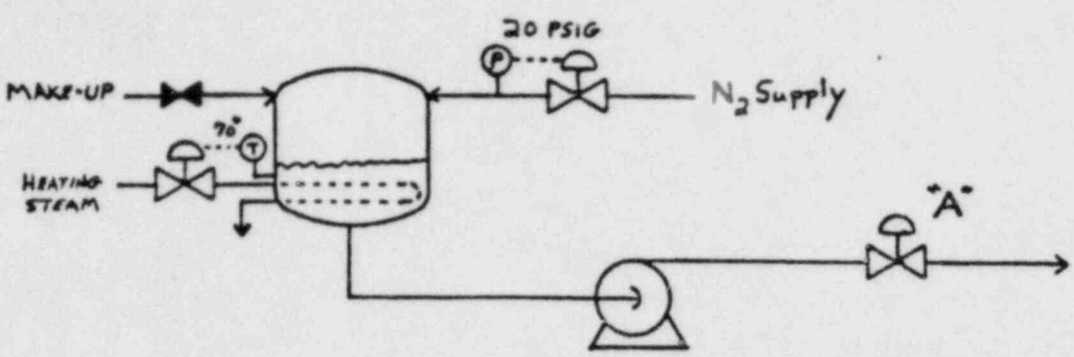
c. The condition of the fluid exiting a small T_H leg break (while the plant is at steady state full power) is superheated steam. (0.5)

d. If a pressurizer PORV lifts to relieve pressure, the temperature downstream will be equal to the pressurizer steam space temperature. (0.5)

1-14 Assume that your unit has experienced a small LOCA due to the failure of a pressurizer cold-calibrated level transmitter sensing line. The failure of the upper sensing line (as compared to the failure of the lower sensing line) would require a (greater, lesser, or equal) makeup flow rate to maintain pressurizer level at setpoint. (Assume that both sensing lines are the same size.) (0.5)

(continued on next page)

1-15



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 (0.5)

b. double the level of the water in the tank (0.5)

c. double the flow through valve "A" (pump discharge-throttle valve) (0.5)

d. double the pressure on the tank (0.5)

1-16 Refer to Figure 1-1 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. (0.75)

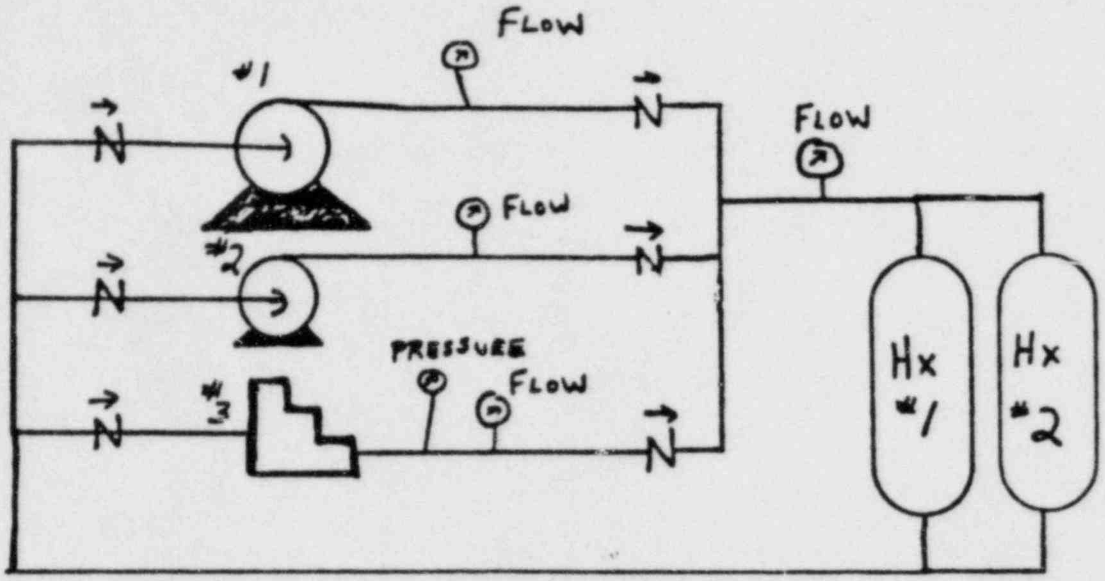
b. Assume that pump 3 is running (by itself) at a certain speed (the speed that correlates to the graph given). What would the pump discharge pressure be if the pump speed was increased 30%? (0.75)

1-17 State the value that represents the amount of subcooling that exists in the Unit 1 reactor at steady state ten percent (10%) power. Include all assumptions and calculations. (1.0)

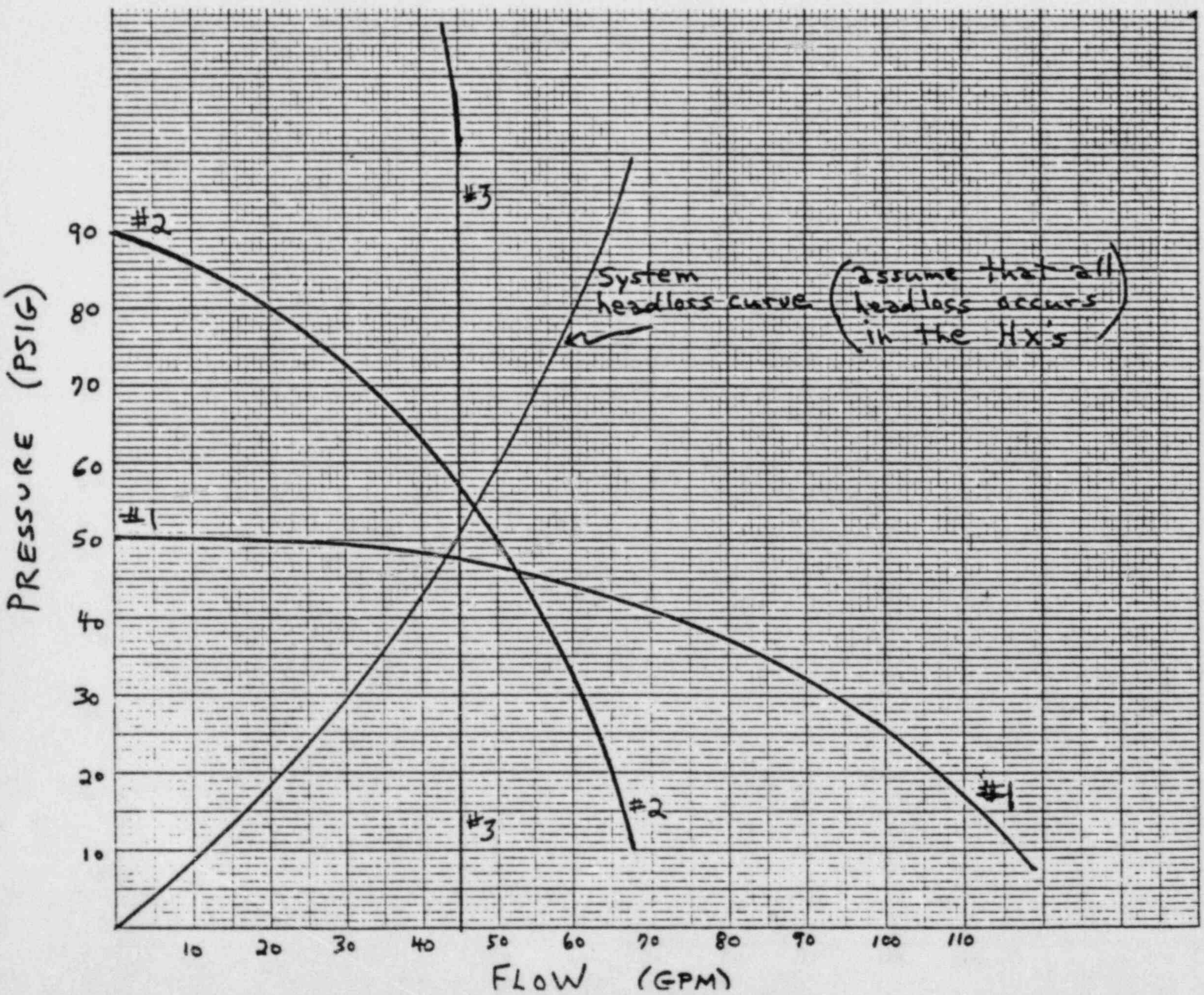
1-18 Your reactor has been operating at full power for three months when a manual reactor trip occurs. All systems are operational and the steam dumps are immediately placed in the steam pressure control mode. Ten (10) minutes after the reactor trip, all RCPs are tripped. Twenty (20) minutes after the reactor trip, Loop 1 RCP is jogged momentarily. Which set of traces (a - d) on Figure 1-2 most closely represents the previously described events? (1.0)

END OF SECTION 1

FIG. 1-1



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



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

a.

b.

c.

d.

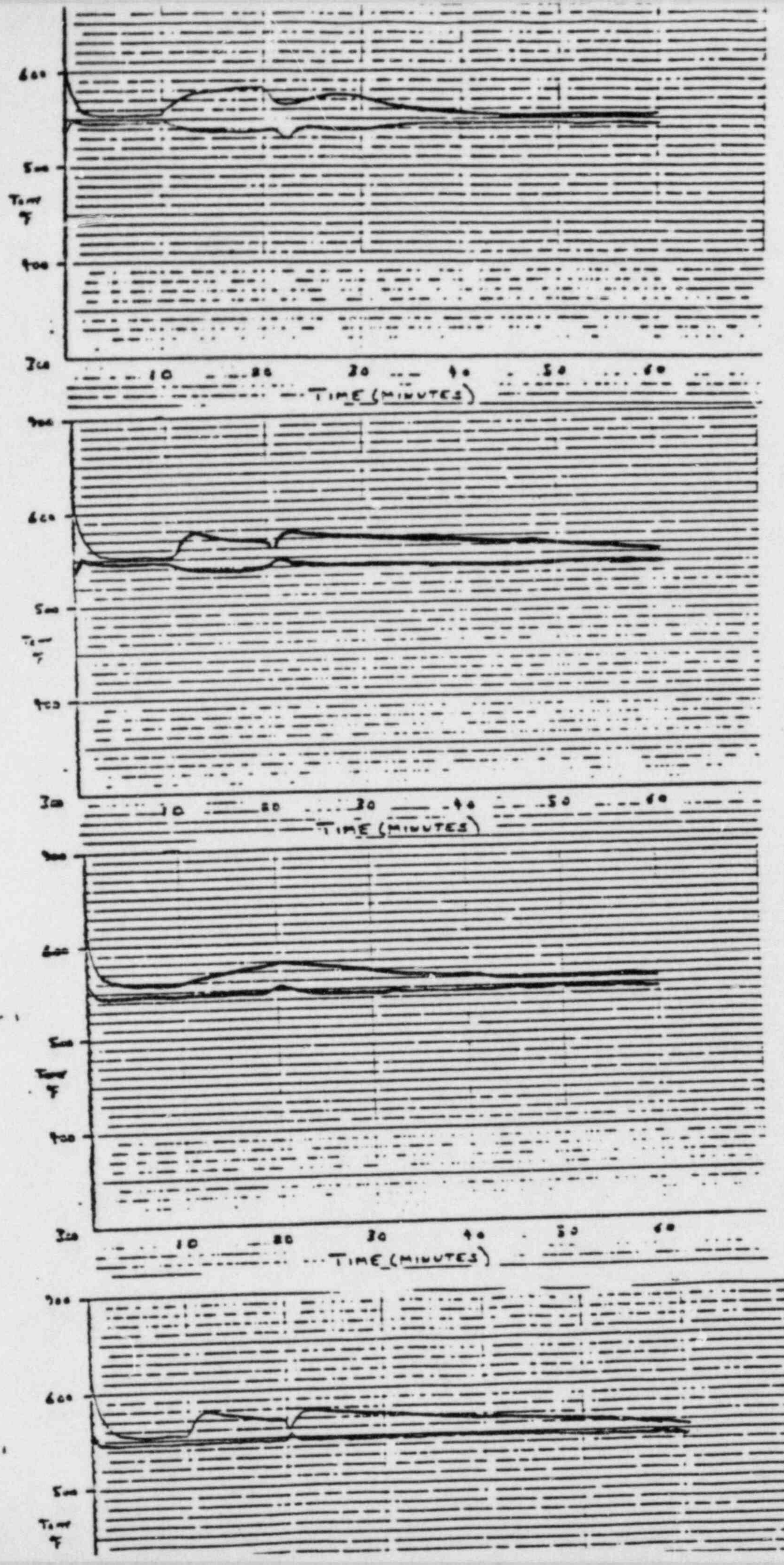


FIG. 1-2

Avg. core exit thermocouple temp and Loop 1 Tc WR temp vs Time (Minutes)

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

2-1 Place an "X" in the appropriate boxes of the table on Fig. 2-1 to indicate where the systems connect to the Reactor Coolant System (RCS). (2.2)

NOTE: Refer to Fig. 2-2 (Reactor Makeup System) when answering questions 2-2 thru 2-4.

2-2 During a makeup with the makeup system mode selector switch in "auto" the system control valves will be as follows: (select one a-d) (1.0)

a.	113A - modulated	114A - open	113B - open	114B - closed
b.	113A - modulated	114A - modulated	113B - open	114B - open
c.	113A - modulated	114A - modulated	113B - open	114B - closed
d.	113A - modulated	114A - modulated	113B - closed	114B - open

2-3 During a makeup with the makeup system mode selector switch in "dilute" the system control valves will be as follows: (select one a-d) (1.0)

a.	113A - closed	114A - open	113B - closed	114B - open
b.	113A - closed	114A - modulated	113B - open	114B - closed
c.	113A - closed	114A - open	113B - open	114B - open
d.	113A - closed	114A - modulated	113B - closed	114B - open

2-4 During a makeup with the makeup system mode selector switch in "alternate dilute" the system control valves will be as follows: (select one a-d) (1.0)

a.	113A - closed	114A - modulated	113B - open	114B - closed
b.	113A - closed	114A - modulated	113B - closed	114B - open
c.	113A - closed	114A - open	113B - open	114B - open
d.	113A - closed	114A - modulated	113B - open	114B - open

2-5 Which of the following areas is NOT served by a fire protection device (such as an automatic deluge water spray or automatic CO_2 system)? (1.0)

a. Station service transformers
b. Reserve station service transformers
c. Main transformer
d. Unit 1 switchgear

(continued on next page)

2-6 Assume that a large load is placed on the fire main system such that the level and pressure in the hydropneumatic tank are steadily decreasing (also assume that the system was initially in a normal steady state condition). Which of the following most closely represents the actions of the fire protection system equipment? (1.0)

a. The pressure maintenance pump (FP-P-6) starts, the air compressor (FP-C-1) starts, the motor driven pump (FP-P-1) starts, and then the diesel driven pump (FP-P-2) starts.

b. The air compressor (FP-C-1) starts, the pressure maintenance pump (FP-P-6) starts, the motor driven pump (FP-P-1) starts, and then the diesel driven pump (FP-P-2) starts.

c. The air compressor (FP-C-1) starts, the pressure maintenance pump (FP-P-6) starts and the air compressor (FP-C-1) stops, the motor driven pump (FP-P-1) starts, and then the diesel driven pump (FP-P-2) starts.

d. The air compressor (FP-C-1) starts, the pressure maintenance pump (FP-P-6) starts, the standby pressure maintenance pump (FP-P-5) starts, the motor driven pump (FP-P-1) starts, and then the diesel driven pump (FP-P-2) starts.

2-7 Which of the following most closely represents the way in which the fire pumps in question 2-6 are stopped when the load on the system is removed? (1.0)

a. The motor driven and diesel driven pumps are stopped locally and the pressure maintenance pumps(s) is/are stopped automatically when the level in the hydropneumatic tank reaches its setpoint.

b. The motor driven and diesel driven pumps are stopped locally and the pressure maintenance pump(s) is/are stopped automatically when the pressure in the hydropneumatic tank reaches its setpoint.

c. The motor driven and diesel driven pumps are stopped from the control room and the pressure maintenance pump(s) is/are stopped automatically when the level in the hydropneumatic tank reaches its setpoint.

d. The motor driven and diesel driven pumps are stopped from the control room and the pressure maintenance pumps(s) is/are stopped automatically when the pressure in the hydropneumatic tank reaches its setpoint.

(continued on next page)

2-8 Refer to Fig. 2-3 (Reactor Trip Breaker and Bypass Breakers). For each of the twelve (12) boxes enter either an "E" for energized, "D" for de-energized, or "----" for no signal to indicate what kind of signal (if any) is sent to the various coils of the trip breakers during the two (2) situations listed on the figure. (2.4)

(if there is a difference between units then answer for both units, labeling each unit)

2-9 Which of the following is a true statement concerning the reactor protection system's response for protecting DNBR? (1.0)

a. If "loop flow-channel 1" on loop 1 indicates 80% flow, "loop flow-channel 2" on loop 2 indicates 75% flow, and $\geq$ 2/4 PR nuclear instruments indicate $\geq$30% power then a reactor trip will occur.

b. If 2/2 UV sensors on RCP busses 1A and 1C indicate a sustained voltage of 2800 volts when 1/4 PR nuclear instruments indicates >10% and 2/2 1st stage impulse pressure instruments indicate <10% then a reactor trip will occur.

c. If $\geq$2/4 PR nuclear instruments indicate power dropping at a rate of $\overline{2}$% per second then a reactor trip will occur.

d. If $\geq$3/4 PR nuclear instruments indicate 15% when RCP busses 1A and 1B drop to 55 Hz (for greater than 1 second) then all three (3) RCP breakers will open and a reactor trip will occur.

2-10 A "high containment pressure" Automatic Safety Injection signal will: (1.0

a. cause main steamline isolation

b. be initiated by 2/4 containment pressure instruments greater than 17 psig

c. be blocked whenever the reactor trip breakers are open

d. cause a feedwater isolation and a phase "A" isolation, but still allow the containment recirculation air coolers to operate

2-11 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, etc. Use actual alphanumeric designations where applicable. (3.

(continued on next page)

2-12 Which of the following would not cause a diesel stop signal assuming the diesel had started on an automatic emergency start signal and all switches were in their normal lineup positions? (1.0)

a. lube oil pressure low trip signal
b. phase A differential current high trip signal
c. both local emergency stop pushbuttons depressed
d. diesel overspeed trip signal

2-13 Refer to Fig. 2-5 (partial AFW system diagram) and complete the chart to indicate the normal position of the valves following an automatic AFW initiation. (2.5)

2-14 For each of the following radiation monitors listed below briefly describe the automatic action(s), if any, that is/are initiated upon receipt of a Hi/Hi Alarm.

a. Condenser Air Ejector Monitor (0.5)
b. Vent Stack "A" High Range Monitor (0.5)
c. Clarifier Effluent Monitor (0.5
d. Manipulator Crane Monitor (0.5

2-15 Which of the following will not trip the main feedwater pump? (1.0

Deleted

a. Load Shed trip signal
b. ~~Steam Generator Hi level~~
c. Low lube oil pressure
d. Low suction header pressure

2-16 Refer to Fig. 2-6 and circle the valves that receive an automatic signal to operate when a Safety Injection Actuation occurs. Also, place an "O" or "S" next to the circled valves to indicate whether the automatic signal is an "open" or "shut" signal. (2.

END OF SECTION 2

FIG. 2-1

	LOOP 1					LOOP 2					LOOP 3				
	Hot Leg		Int. Leg	Cold Leg		Hot Leg		Int. Leg	Cold Leg		Hot Leg		Int. Leg	Cold Leg	
	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve
SI Accum.															
Pzr Surge															
Pzr Spray															
RHR Suction															
RHR Return															
Normal Charging															
Loop Fill Conn.'s															
Normal Letdown															
Excess Letdown															
Process Sampling System															

FIG. 2-1

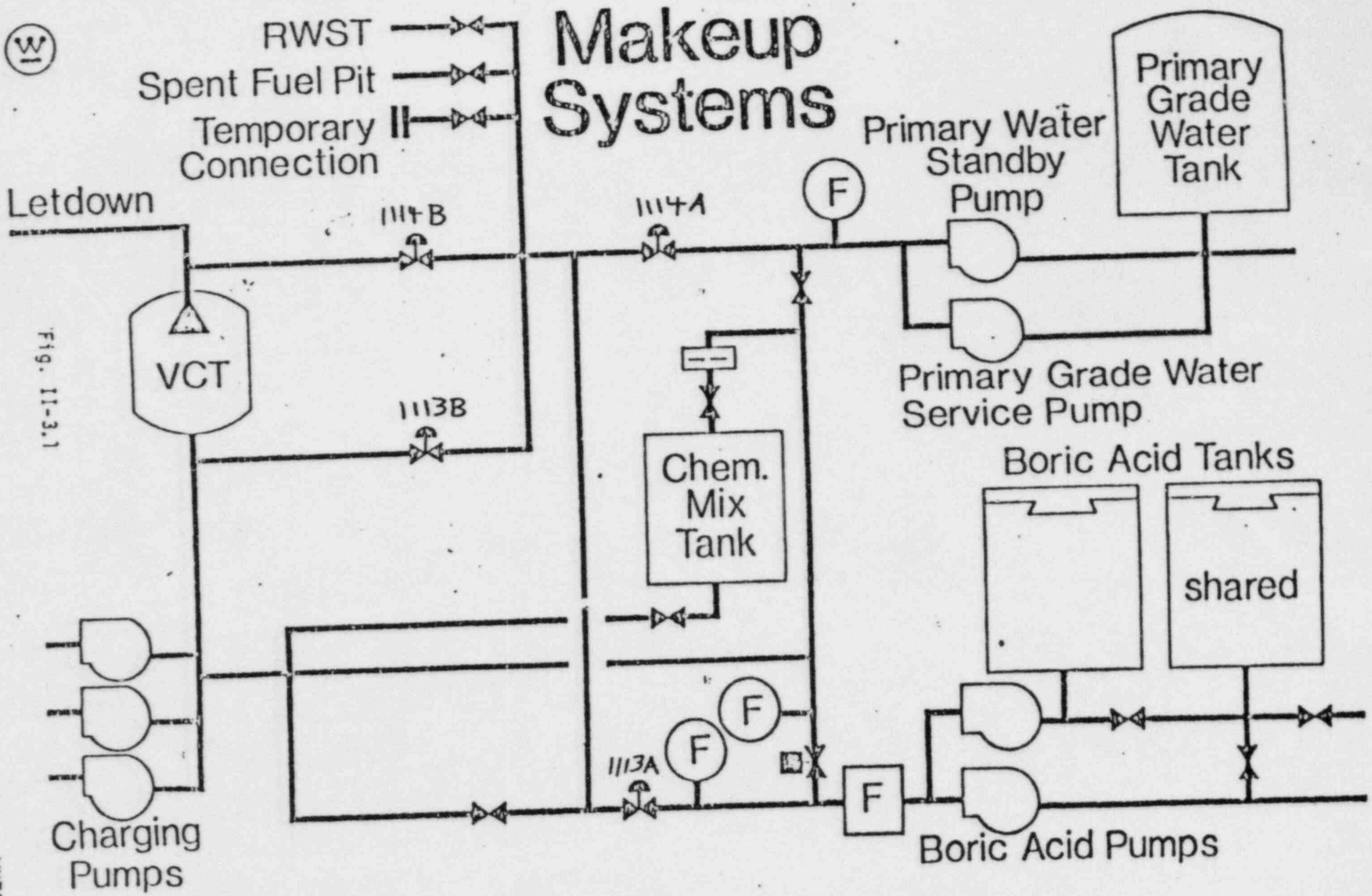


Fig. II-3.1

II-3.10

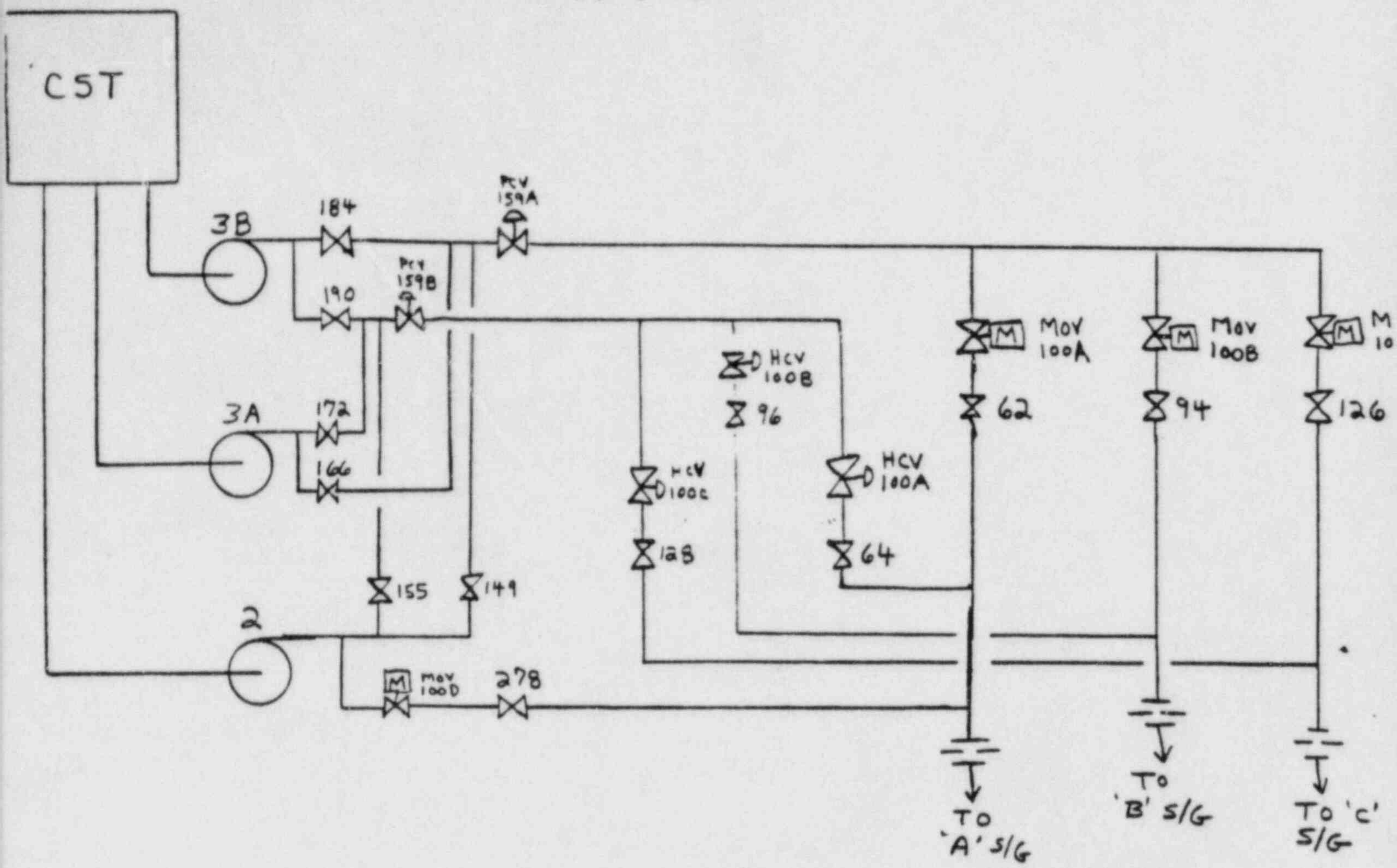
WNTO-7905

FIGURE 2-3

		RTA		BYA		BYB	
		SHUNT	UV	SHUNT	UV	SHUNT	UV
Automatic reactor trip signal is present on logic train A	Unit 1						
	Unit 2						
Manual reactor trip signal is present on logic train A	Unit 1						
	Unit 2						

Applicant's Signature

FIG. 2-5



VALVE NUMBER	OPEN	CLOSED
184		
190		
166		
172		
149		
155		
MOV-100D		
278		
HCV-100A		
HCV-100B		
HCV-100C		
64		
96		
128		
MOV-100A		
MOV-100B		
MOV-100C		
62		
94		
126		

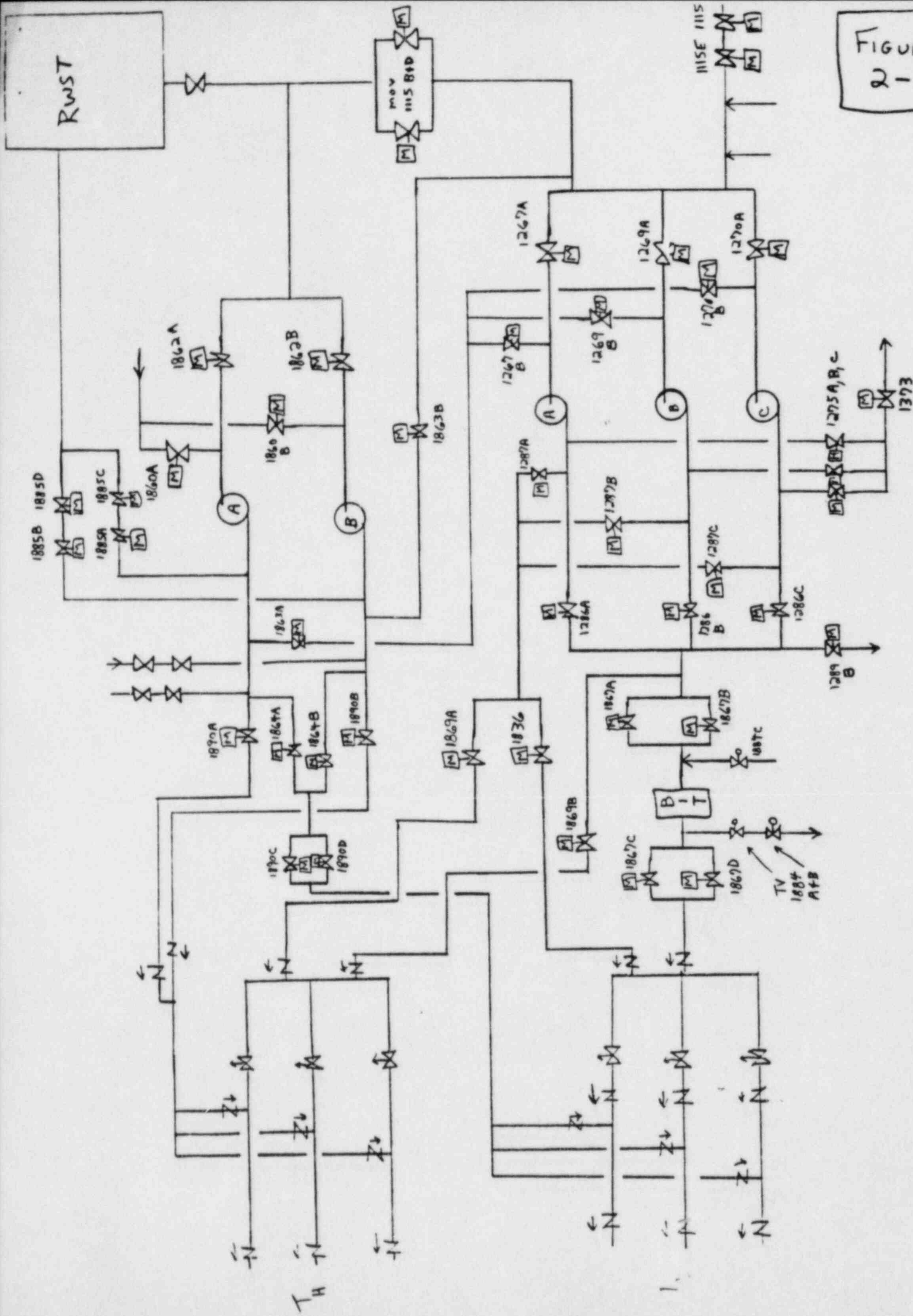

RWST
MOV 1115 B&D
1115E
1115
1267A
1269A
1270A
1267B
1269B
1270B
1862A
1862B
1860A
1860B
1885D
1885C
1885B
1885A
1863A
1863B
1287A
1287B
1287C
1286A
1286B
1286C
1275A,B,C
1373
1289B
1890A
1864A
1864B
1890B
1869A
1836
1867A
1867B
1887C
B I T
1869B
1890C
1890D
1867C
1867D
TV 1884 A+B
A
B
C
T_H

FIGURE 2-6

3. INSTRUMENTS AND CONTROLS (25.0)

3-1 Recently the Intermediate Range trip setpoint was found to be non-conservative. The reason for this problem was: (1.0)

a. A gas leak had developed in one of the detectors.
b. Current outputs (from the detectors) had changed with age.
c. The trip setpoint had been entered incorrectly during a surveillance test.
d. Core loading changes had changed the flux shapes.

3-2 A break in the reference leg in a pressurizer level indicator will cause the indicated level to be higher than the actual level. (0.5)
TRUE OR FALSE

Refer to Figure 3-1 (core cooling monitor front panel).

3-3 Which two (2) of the calculations listed below are made when the "ΔT loop 1" button is pushed? (2.0)

1) Highest thermocouple minus Th
2) Highest thermocouple minus Tc
3) Lowest thermocouple minus Th
4) Lowest thermocouple minus Tc
5) Average thermocouple minus Th
6) Average thermocouple minus Tc
7) Th minus Tc
8) Highest thermocouple minus lowest thermocouple

3-4 Match the correct control rod position indicating instrumentation

(1--Individual Rod Position Indication Circuitry
OR
2--Group Demand Circuitry)

with the functions listed below:

a. Provides an input to the step counters on the main control board (0.5)
b. Measures actual rod position (0.5)
c. Actuates rod bottom lights (0.5)
d. Provides an input to the rod insertion limit alarm circuit (0.5)

3-5 Which of the following will cause the OPΔT setpoint calculator to reduce its setpoint? (1.0)

a. Tavg above rated
b. Pressure below 2235 psig
c. Rate of change of Tavg in a decreasing direction
d. Delta flux exceeding the deadband

(continued on next page)

3-6 Assume that during normal full power operation Loop A T_h RTD (control channel) fails high. Which of the following alarms would not annunciate due to the failure? (1.0)

a) Loop A-B-C Hi-Lo Tavg Deviation
b) Tavg ≷ Tref Dev.
c) Rx Loop A-B-C Hi Tavg
d) Loop A-B-C Hi-Lo ΔT Deviation

3-7 For each of the following situations indicate the initial direction of travel for the feedwater regulator valve (answer either "open" or "close").

a. Channel III steam generator level transmitter fails low (0.5)
b. Controlling feed flow transmitter fails high (0.5)
c. Controlling steam generator pressure transmitter fails low (0.5)
d. Controlling first stage pressure transmitter fails low (0.5)

3-8 List ten (10) of the Post-Accident monitoring instruments as defined in the North Anna Technical Specifications. (2.0)

3-9 List six (6) of the Auxiliary Shutdown Panel monitoring instruments as defined in the North Anna Technical Specifications. (1.2)

3-10 Assume 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. (1.8)

3-11 Pulling the control power fuses when the Source Range level trip switch is in "Bypass" will cause a trip signal to occur. TRUE or FALSE. (0.5)

3-12 The load shedding feature is actuated when the diesel output breaker closes. TRUE or FALSE (0.5)

3-13 The pressure controllers for the atmospheric steam dumps incorporate a potentiometer with a range of 600-1400 psig. What would be the required potentiometer setting to obtain a setpoint of 825 psig? (1.0)

3-14 Identify the nine (9) Reactor Trips (other than Safety Injection) for which there are no associated blocks or permissives. (1.8)

(continued on next page)

3-15 If pressurizer level channel select switch (1/LM-459) is set in its normally selected position (position #2) when pressurizer level detector LT-460 fails low, which of the following will NOT occur? (1.0)

a. Letdown isolation valve LCV-460B will shut
b. A -5% "low level" alarm will occur
c. Letdown orifice isolation valves HCV-1200 A, B, & C will shut
d. Pressurizer heater groups A, B, C, D, & E will de-energize

3-16 Assume you are operating the Unit #2 reactor at a steady state 40% power level when the simultaneous failure of the following instruments occurs: PT-447 (1st Stage Turbine Pressure) failed "low" (assume an output of 0 psig), and TE 411C (Loop 1 T_C Control Channel RTD) failed high (assume a output of 615°F). Assume that the channel select switch (PM-446) is selected to the PT-446 position.

Answer the questions below assuming NO operator actions.

a. Calculate the value that represents Loop 1 Tavg before the instrument failures. (0.6)

b. Calculate the value that represents Loop 1 Tavg immediately after the instruments failures. (0.6)

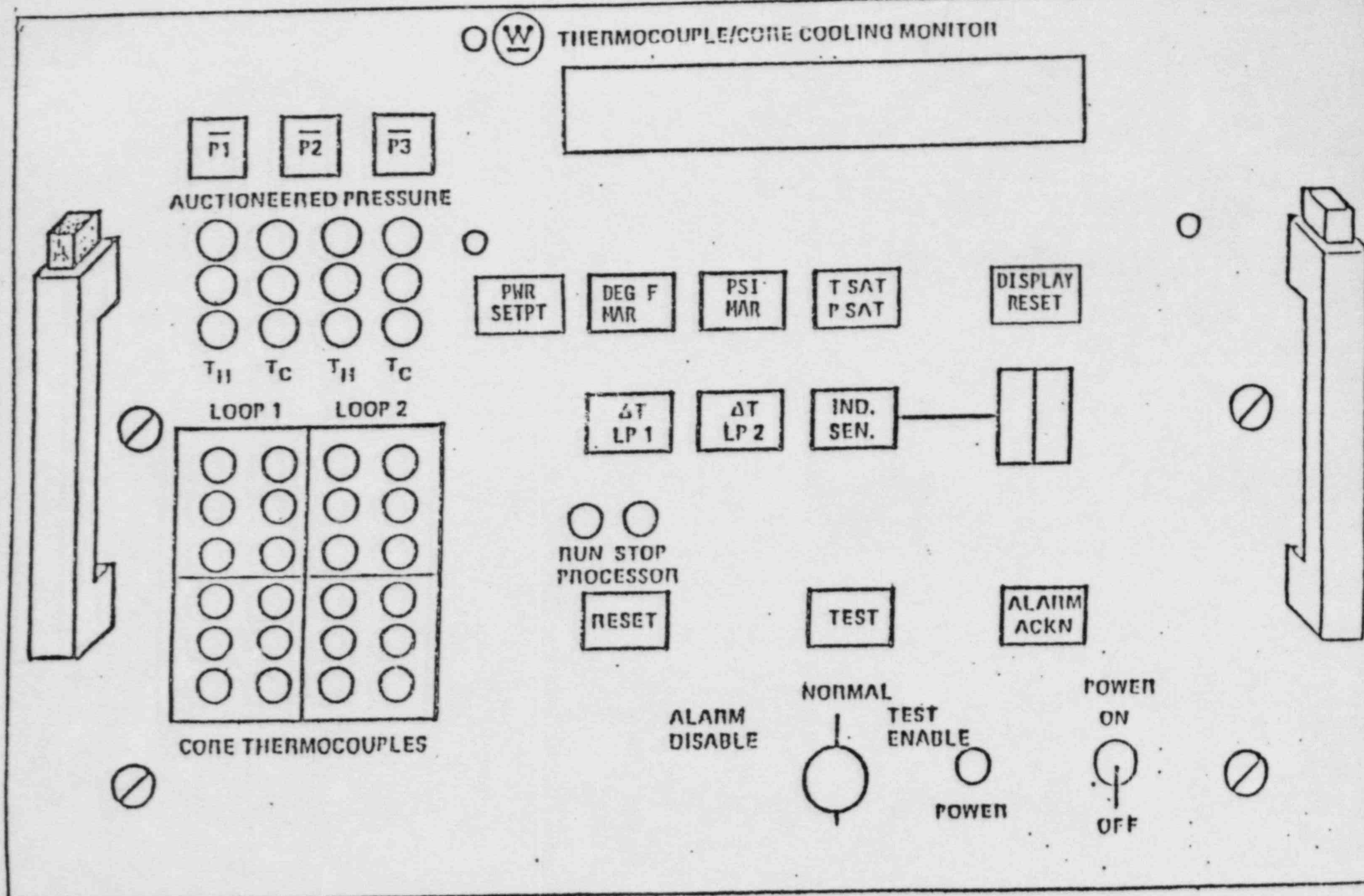
c. Describe how your plant would respond to these failures. Include all affected control systems (primary and secondary). Limit your description to the first minute following the failures, and do not include protection system actions. (2.0)

3-17 During plant operation the drive mechanisms hold the control rods withdrawn from the core in a static position with the use of the stationary gripper. When a signal is sent from the rod control system, the rod will withdraw or insert into the core. Arrange the following steps for the right sequence during a Control Rod Withdrawal. (1.5)

a. stationary gripper deenergizes
b. lift coil energizes
c. movable gripper energizes
d. lift coil deenergizes
e. stationary gripper energizes
f. movable gripper deenergizes

3-18 List the interlock(s) associated with the RHR suction valve(s) from the RCS. (1.0)

END OF SECTION 3



THERMOCOUPLE/CORE COOLING MONITOR
P1
P2
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

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

4-1 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. Outline the major actions required to take the reactor critical if criticality is not achieved before the upper limit of the ECP. Limit your answer to the actions required of a Reactor Operator to bring the reactor critical under the stated condition. (1.5)

4-2 A 100 rem neutron dose produces more biological damage than a 100 rem gamma dose. TRUE or FALSE (0.5)

4-3 a. List (in detail) the nine (9) 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. (2.25)

b. List (by broad subheading or subtitle) the nine (9) immediate operator action steps required for a Safety Injection Actuation (i.e., steps 5-13). Assume that the reactor trip responses (steps 1-3) and "check if SI is actuated" (step4) are completed. Also assume that all actions and expected responses are completed without problems. (2.25)

4-4 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%? Show all work. (3.0)

DATE	TIME (leaving band)	TIME (re-entering band)	POWER(%)
7-19-84	0300	0308	95
7-19-84	1747	1833	55
7-19-84	2238	2400	10
7-20-84	0148	0300	30

4-5 ADM-20.9 indicates that entry into the reactor containment during reactor operation exposes personnel to four (4) distinct hazards. List these hazards. (1.0)

(continued on next page)

4-6 Answer the following questions as they relate to you as an operator at the North Anna Power Station.

a. What are your quarterl[illegible]inistrative limits for radiation doses to the whole body, han[illegible]nd skin? (1.2)

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

Deleted c. According to the HP Radiation Protection Manual ~~and General Employee Training (G.E.T)~~ a person must meet ~~four (4)~~ 3 requirements in order to wear a respirator. List these ~~four (4)~~ (3) requirements. Include time frame, if applicable. (1.25)

4-7 Upon a loss of component cooling water and seal injection flow to a RCP the component cooling water flow to the thermal barrier must be restored prior to restarting seal injection flow. TRUE or FALSE (0.5)

4-8 Deleted For local starts of the emergency diesel generators the pre-lube pump should be operated for a minimum of two (2) minutes to ensure proper lubrication of the upper pistons. TRUE or FALSE (0.5)

4-9 If all steam generators are ruptured, then the steam generator with the lowest level should not be isolated. TRUE or FALSE. (0.5)

4-10 Either a control room operator or control room operator trainee will perform the duties of Interim Emergency Communicator upon initiation of the Emergency Plan. TRUE or FALSE (0.5)

Applicant's Signature

4-11 From the following symptoms list the ones (by number) that could possibly apply to each of the conditions a-c.

Symptoms:

1. High Containment Pressure
2. High Containment Temperature
3. High Containment Sump Level
4. High Containment Radiation
5. High S/G Blowdown Radiation
6. High Condenser Air Ejector Radiation
7. Low S/G Pressure
8. High Steam Flow
9. Steam Flow-Feed Flow Mismatch

a. Loss of Reactor Coolant (other than SGTR) (1.0)
b. Loss of Secondary Coolant (inside containment) (1.0)
c. Steam Generator Tube Rupture (1.0)

4-12 List the immediate operator actions required for a loss of Bearing Cooling Water (assuming the plant is at full power). Note: Procedure AP-19 groups these actions into three (3) steps. (1.2)

4-13 List the immediate actions required for a loss of component cooling water. In addition to the three (3) major steps also list the actions or verifications required under each major step (total of nine (9) actions or verifications). Assume that all initial actions/expected responses are obtained. (2.25)

4-14 List the five (5) immediate operator actions required for a "large steam generator tube leak." (1.5)

4-15 The North Anna Emergency Procedures Book contains Critical Safety Function Status Trees. List the "symptom/title" heading for six (6) of the seven (7) status trees. (1.2)

END OF SECTION 4

Equation Sheet

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$$A = \lambda N$$

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

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

1-1 d. (1.0)

REF: North Anna Curve Book, #1-SC-3.9.

1-2 c. (1.0)

REF: Simulator Training, Rx Theory, Module 1, Section 1.

1-3 d. (1.0)

REF: North Anna Curve Book, #1-SC-3.13 (pp. 1 and 2).

1-4 c.

REF.: North Anna Lesson Plans, Rx Theory, Sec. 4, pg. 6 (1.0)

1-5 a. increase (0.5)

b. increase (0.5)

c. remain essentially the same (0.5)

REF: North Anna Curve Book, #2-SC-3.5, pp. 1 and 5; NUS Module-3, pg. 9.5-1.

1-6 a. lower than estimated (0.5)
b. lower than estimated (0.5)
c. higher than estimated (0.5)
d. lower than estimated (0.5)

REF.: North Anna 1-OP-1C

1-7 b. (1.0)

REF.: Westinghouse Reactor Theory, Section I-5

1-8 b. (1.0)

REF: NUS, Module 3, Unit 13.

(continued on next page)

1-9 a. Total rods = -7800 pcm (1.0)
Power defect = +2200 pcm
-5600 pcm (-5.6% ΔK/K)

b. Control bank rods = -4600 pcm (1.0)
Δ xenon = -1600 pcm
Power defect = +2200 pcm
-4000 pcm (-4.0% ΔK/K)

4 items C. Rods
S.D. Rods
Xe
PD

c. Control bank rods = -4600 pcm (1.0)
Δ xenon from initial value = +2800 pcm
Power defect = +2200 pcm
Δ boron = -4250 pcm
-3850 pcm (-3.8% ΔK/K)

5 items C. Rods
S.D. Rods
Xe
P.D.
Boron

REF: North Anna Curve Book, 3.1, pg. 1; 3.2, pg. 1; 3.4, pg. 1; 3.8, pg. 1; 3.9, pg. 1; 3.11, pg. 1; and 8.1, pg. 3.

1-10 a. $1/M = CR_1/CR_2$ (1.25)

$= 400/1857$

$= 0.215$

b. $1/M = 1\text{-}Keff_2/1\text{-}Keff_1$ (1.25)

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

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

$1-0.0129 = K_{eff_9}$

$0.9871 = K_{eff_9}$

REF.: Westinghouse Reactor Physics, Section I-4

(continued on next page)

1-11 d. (1.0)

REF.: Westinghouse Reactor Physics, Section I-5

1-12 c. (1.0)

REF: General Physics HT&FF, Section II-C, pg. 267.

1-13 a. True (0.5)
b. False (0.5)
c. False (0.5)
d. False (0.5)

REF: General Physics HT&FF, Sections II-A and II-B,
North Anna Lesson Plan - Rx Vessel and Internals, pg. 25.

1-14 lesser (0.5)

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

1-15 a. decrease (0.5)
b. increase (0.5)
c. decrease (0.5)
d. increase (0.5)

REF.: General Physics HT & FF, Section III-B, North Anna Lesson Plan - Centrifugal Pump Characteristics, pg. 3

1-16 See Fig. 1-1
a. essentially the same (0.75)
b. appx. 76 psig (0.75)

REF.: General Physics HT & FF, Section III-B, pg. 326 and 329; North Anna Lesson Plan - Centrifugal Pump Characteristics, pg. 8

(continued on next page)

1-17 (615 - 547) (.10) + 547 = 553.8°F Operating T_H at 10% power. (1.0)

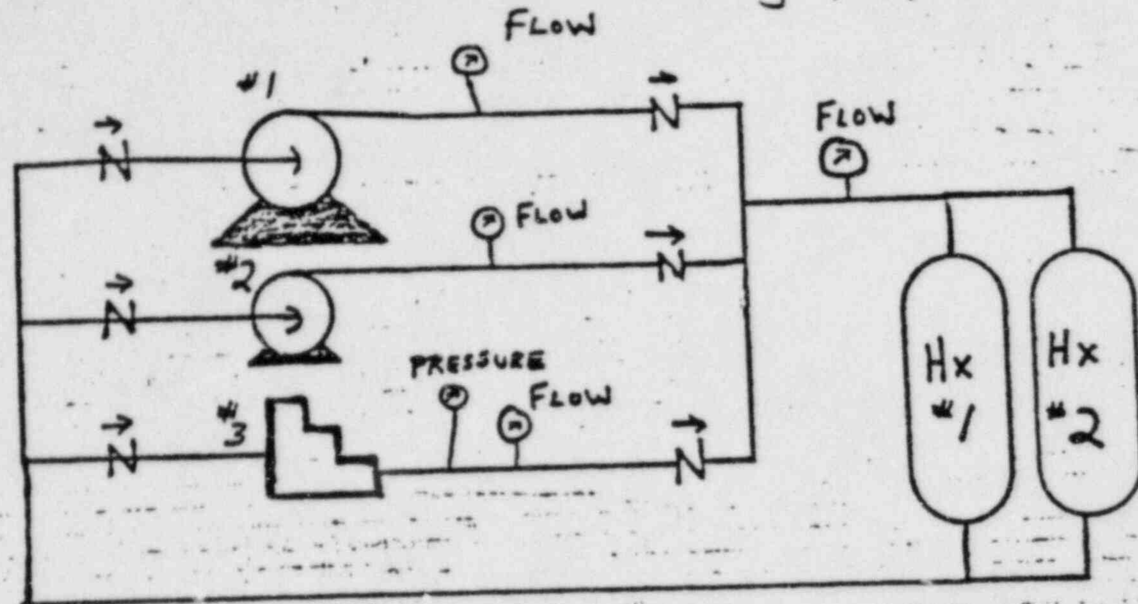
652.0°F	Sat. temp. for 2250 psia
-553.8°F	Operating T_H at 10% power
98.2°F	subcooled at 10% power

REF: North Anna Systems Training - RCS, pg. 30, and T.S. Unit 1 Amendment No. 54.

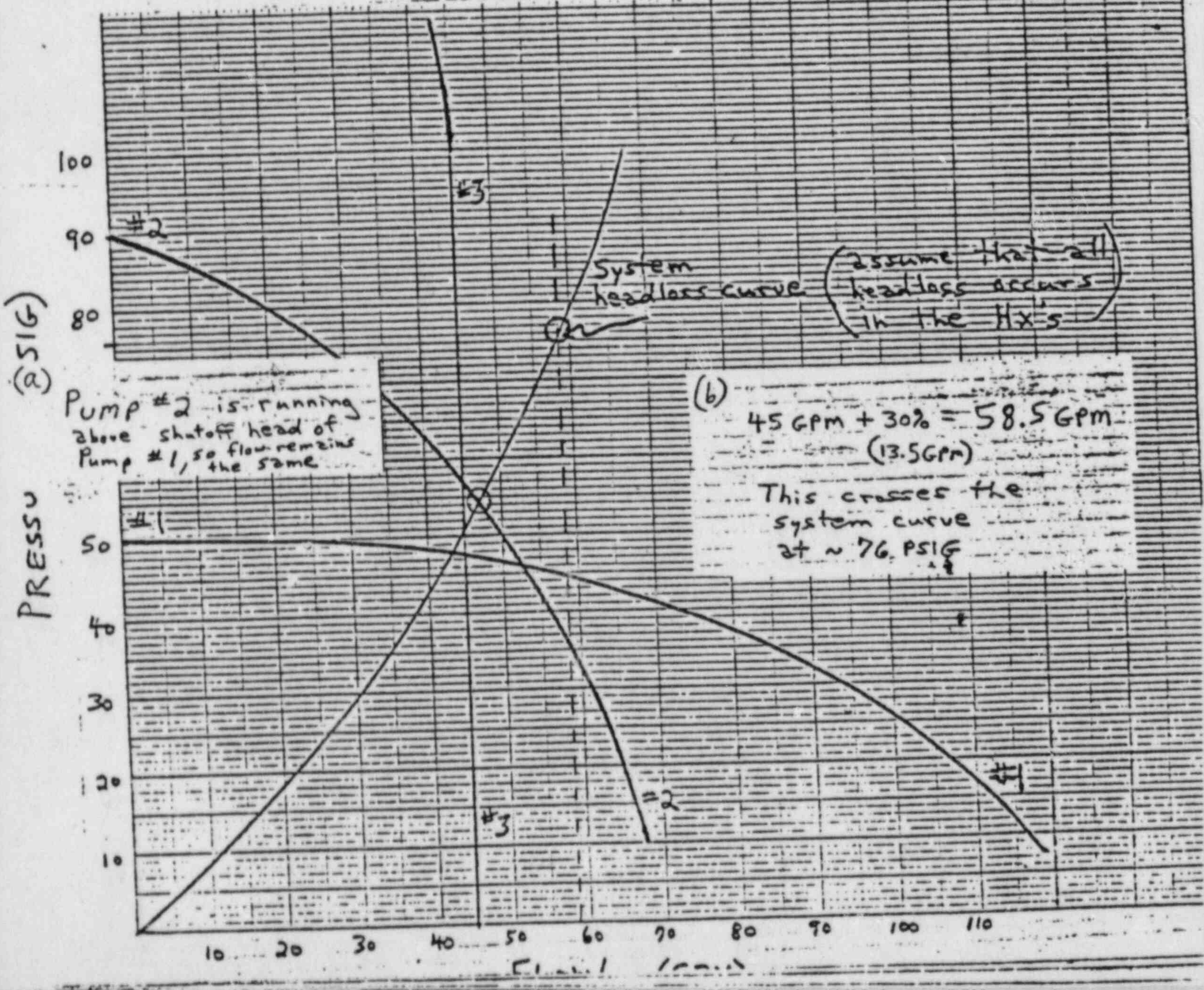
1-18 d. (1.0)

REF: Shearon Harris Simulator Data.

Fig 1-1



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

2-1 See Figure 2-1

REF.: North Anna RHR Lesson Plan pp 3&4,
CVCS Lesson Plan Pg. 4, Pzr Lesson Plan pp 4&5
ECCS Lesson Plan pg. 2, & OP-12.1 pg. 3

2-2 c.

2-3 d.

2-4 d.

REF.: For 2-2 thru 2-4: North Anna Plant Manual, Vol. II pp. 26-10-5 thru 26-10-9

2-5 b.

REF.: North Anna Plant Manual, Vol. 2, page 20-1-7

2-6 c.

REF.: North Anna Plant Manual, Vol. 2, pp 20-4-4 & 5

2-7 a.

REF.: North Anna Plant Manual, Vol. 2, pp 20-4-4 & 4-5

2-8 See Fig. 2-3

REF.: North Anna dwg. 5655D33 sheet 2, and North Anna Basic Systems Training pgs. IV-10.9 and 36

2-9 d.

REF.: North Anna Basic Systems Training, pp IV-9.6, IV-10.3&4

2-10 d.

REF.: North Anna Lesson Plan on RPS, pgs. 18 & 19

(continued on next page)

2-11 See Fig. 2-4

REF.: North Anna Dwg. 11715-FE-1AE-8

2-12 a.

REF.: North Anna Plant Manual, Vol. 2, dwgs 11715-LSK-22-12 J,L,M, & S

2-13 See Fig. 2-5

REF.: North Anna Plant Manual, Vol. 1, pg. 5-13-11; OP-31.2A, and dwg. 11715-FM-74A-15

2-14
a. diverts flow to containment vs atmosphere (0.5)
b. no control actions — clarifier (0.5)
c. shuts effluent and influent ~~discharge~~ valves & trips S/G blowdown pumps — hold up tank (0.5)
d. closes purge exhaust butterfly valves and trips purge supply & exhaust fans. (0.5)

REF.: North Anna Q&A bank question 2,3, & 6-66 (modified) and Lesson Plan on Hi Range Rad. Monitors, pgs. 6,7, & 15

2-15 b. delete if b is not in literature

REF.: North Anna Plant Manual, Vol. 13, dwg. 12050-LSK-5-8C

2-16 Open - 1115B, D, 1867A,B,C,D

Shut - 1115C, E, ~~1371~~, ~~1275A,B,C~~, 1289B, 1884A,B,C,
Design change

Also see Fig. 2-6

REF.: North Anna Lesson Plan on ECCS, pg. 19 (dwg.)

FIG. 2-1

FIG. 2-1

	LOOP 1					LOOP 2					LOOP 3				
	Hot Leg		Int. Leg	Cold Leg		Hot Leg		Int. Leg	Cold Leg		Hot Leg		Int. Leg	Cold Leg	
	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve	Rx side of isol. valve	Dnstrm of isol. valve		Upstrm of isol. vlv.	Rx side of isol. valve
SI Accum.					X					X					X
Pzr Surge											X				
Pzr Spray					X										X
RHR Suction	X														
RHR Return										X					X
Normal Charging										X					
Loop Fill Conn.'s			X					X					X		
Normal Letdown					X										
Excess Letdown			X					X					X		
Process Sampling System		X		X		X	X		X			X		X	

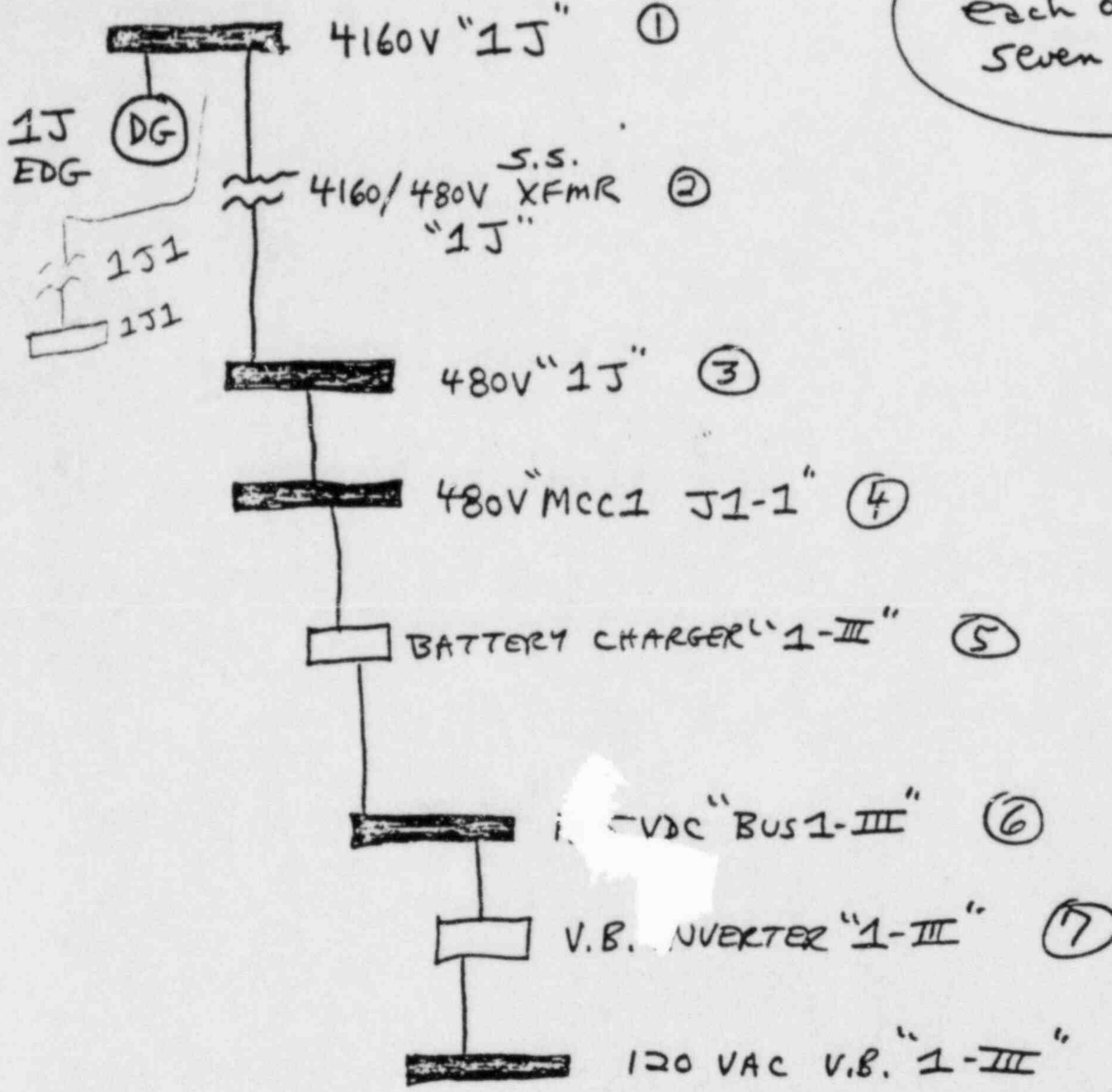
CAF ~~for~~ Unit 1 vs Unit 2

FIGURE 2-3

	RTA		BYA		BYB	
	SHUNT	UV	SHUNT	UV	SHUNT	UV
Unit 1	—	D	—	—	—	D
Automatic reactor trip signal is present on logic train A Unit 2	E	D	---	---	---	D
Manual reactor trip signal is present on logic train A	E	D	E	---	---	D

24 boxes X 0.1 pts each

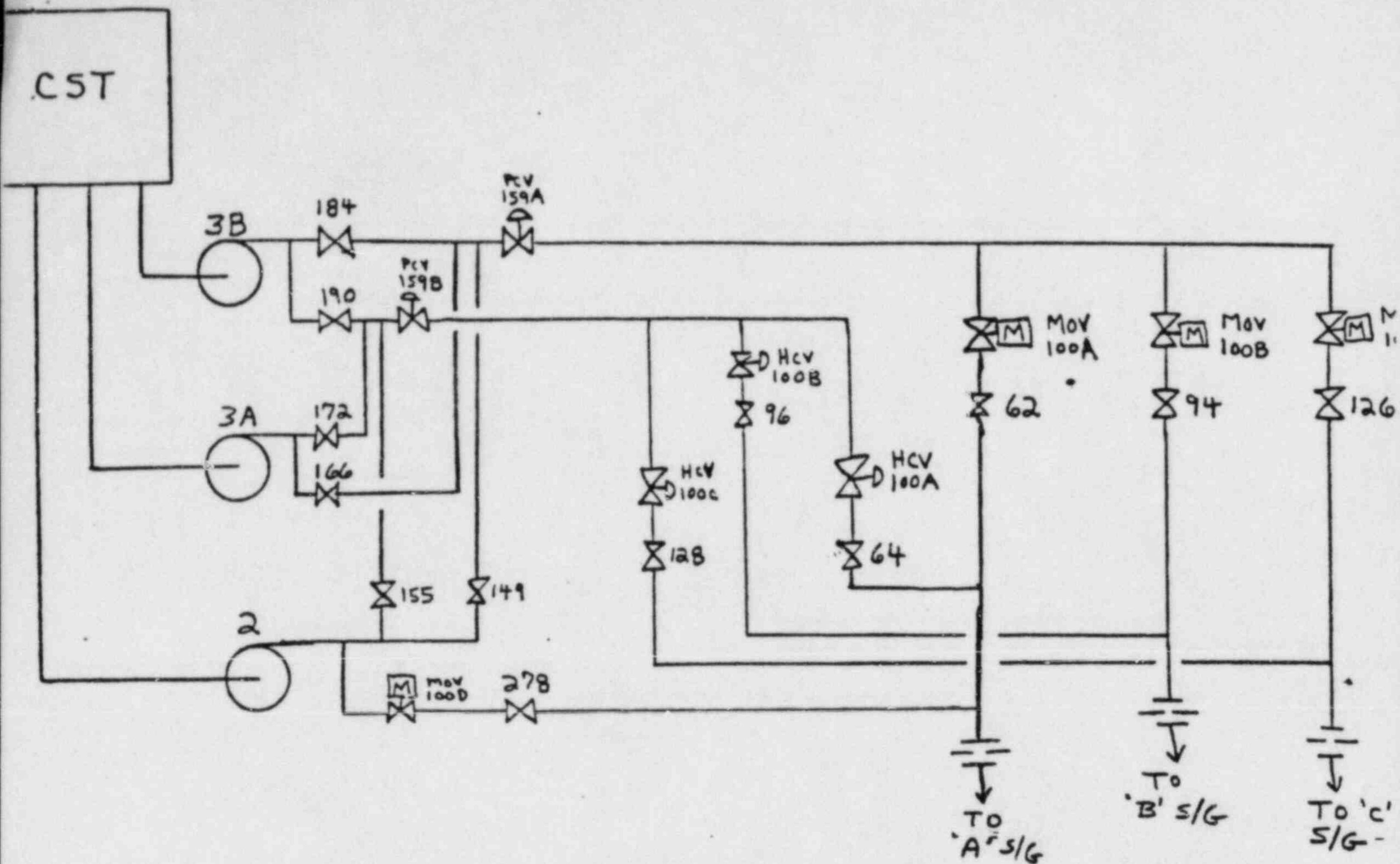
0.5 pts for each of the seven items



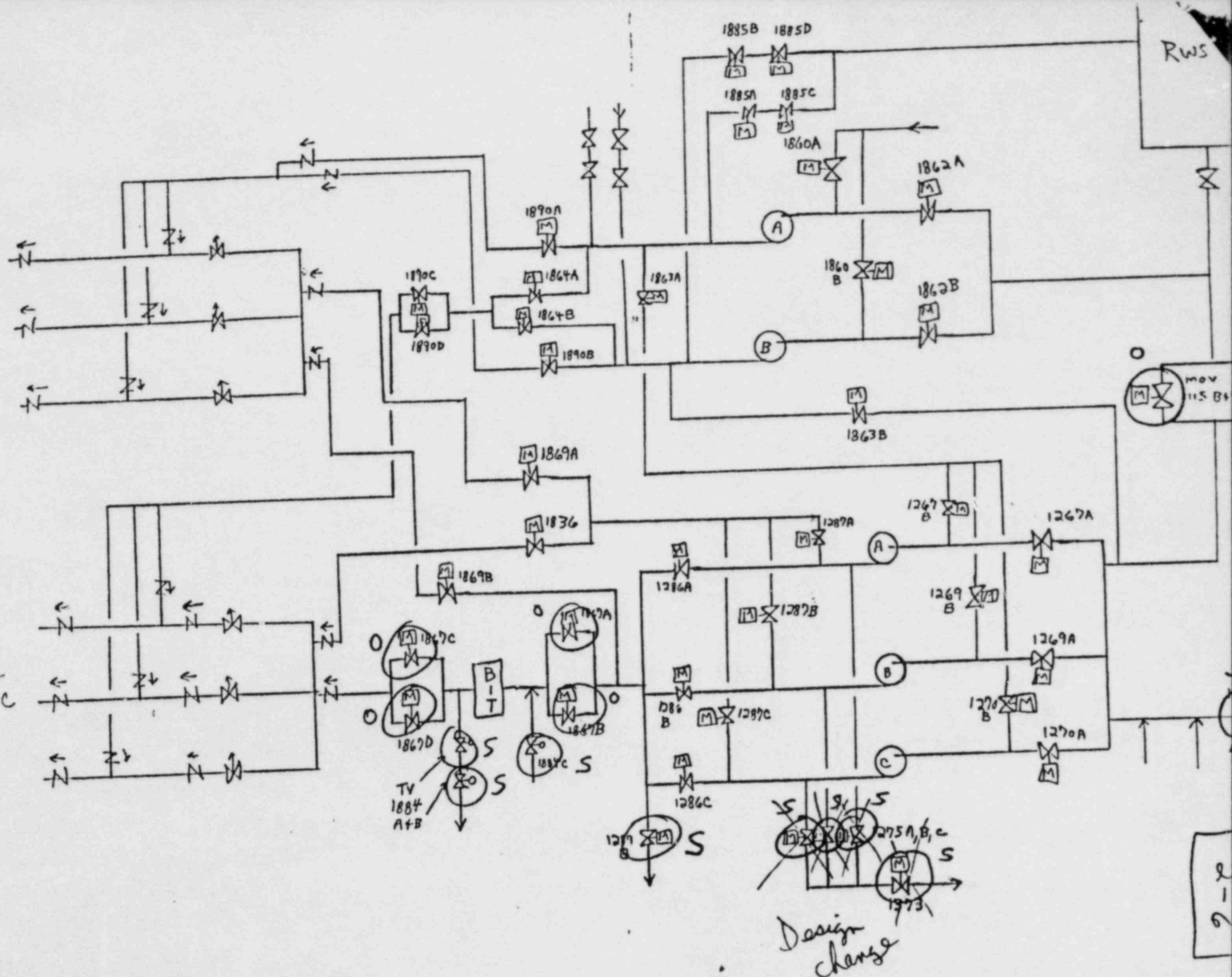
— 0.25 pts if label is missin
— 0.5 pts if label given is actually the label for another piece of equipment

See OP-26B for clarification

FIG. 2-5



VALVE NUMBER	OPEN	CLOSED
184	X	
190		X
166		X
172	X	
149		X
155		X
MOV-100D	X	
278	X	
HCV-100A		X
HCV-100B		X
HCV-100C	X	
64		X
96		X
128	X	
MOV-100A		X
MOV-100B	X	
MOV-100C		X
62		X
94	X	
126		X


RWS
1885B
1885D
1885A
1885C
1860A
1862A
1860 B
1862B
1863A
1863B
MOV 115 B
1890A
1864A
1864B
1890B
1890C
1890D
1869A
1836
1869B
1867A
1867B
1867C
1867D
1884C
TV 1884 A+B
B I T
1286A
1286B
1286C
1287A
1287B
1287C
1267A
1267 B
1269A
1269 B
1270A
1270 B
1275A,B,C
1373
A
B
C
T_H
T_C
O
S
Design change

3-1 d.

REF.: Variation of North Anna Q&A Bank, question 2,3,& 6-18 and LER 83-63

3-2 TRUE

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

3-3 5 & 6 (Avg. therm. minus Th and Avg. therm. minus Tc)

REF.: North Anna Lesson Plan on the Subcooling Margin Monitor System and AP-46 p. 5

3-4
a. 2
b. 1
c. 1
d. 2

REF.: North Anna Lesson Plan for Rod Control System and Rod Position Indication, pg. 24-32

3-5 a.

REF.: North Anna Lesson Plan for RCS Temperature Instruments, pg. 12

3-6 c.

REF.: North Anna dwgs 5655D33(9) and 108D014(1), and Annun. Proc. 1B-B4, 1B-A7, 1B-A8, & 1B-B8

3-7
a. Open
b. Close
c. Close
d. Close

REF.: North Anna/Westinghouse dwgs. 5655D33(13) & 108D014(6)

(continued on next page)

According to phone conv. with Curt Myers on 11-8-84 Stm line press & S/G press are nearly the same / Main stm line press is the press downstream of the NRV's

3-8 Any ten (10) of the following:

Containment Pressure
Th - wide range
Tc - wide range
RCS pressure - wide range
Pzr level
Steam line pressure
S/G level - narrow range
RWST level
BAT level
AFW flow rate
RCS subcooling margin monitor
PORV position indicator
PORV block valve position indicator
Safety Valve position indicator

REF.: North Anna T.S. table 3.3-10 (pg 3/4 3-50) and Q&A Bank question 2,3, & 6-26

3-9 Any six (6) of the following:

Tavg
Pzr Pressure
Pzr Level
AFW pump discharge header pressure
Emerg. CST level
Charging flow
Main Steam line pressure
S/G level (WR)
Relay Room Positive Ventilation

REF.: North Anna T.S. table 3.3-9 (pg. 3/4 3-47)

3-10
1. Pzr Hi Press. Alarm (2310 psig on PT-445) (0.1
2. PCV-456 opens causing pressure to drop (0.4
3. Backup heaters are full on (appx. 2210 psig) (0.1
4. As pressure drops below 2000 psig (as sensed by 2/3 protection channel pressure detectors) an interlock will cause PCV-456 to shut (0.6
5. As pressure increases above 2000 psig PCV-456 will re-open (0.3
6. Pressure should continue cycling around 2000 psig (0.3

REF.: North Anna Lesson Plan on RCS Pressure Instrumentation, pgs. 7-10 and AP-44

(continued on next page)

3-11 TRUE

REF.: North Anna Lesson Plan on Nuclear Instrumentation, pg 14

3-12 FALSE

REF.: North Anna Lesson Plan for Electrical Distribution, Degraded voltage scheme chapter, pg 2

check pot scale

3-13 (825-600)/(1400-600) = 225/800 = 28.1% 0-10 pot scale ∴ setting = 2.81

REF.: North Anna Lesson Plan for Steam Dump Control System, pg. 11

3-14 PR Hi Flux (Hi Stpt)
PR Hi Pos. Flux Rate
PR Hi Neg. Flux Rate
OTΔT
OPΔT
Pzr Hi Press.
S/G low level with stm flow/feed flow mismatch
Manual
General Warning

REF.: North Anna Lesson Plan for Reactor Protection System (attached charts)

3-15 b.

REF.: North Anna/Westinghouse dwgs 108D014(4) and 5655D33 (11&12) and Lesson Plan on Pzr level control, pages 6-8

(continued on next page)

new values

3-16 a. (582.8-547) (.4) + 547 = <u>561.3°F</u> | (587.8-547)(.4) + 547 = <u>563.3°F</u> (0.6)

b. (610-547) (0.4) + 547 = 572.2 (T_H) | (615 - 547)(.4) + 547 = 574.2°F
(572.2 + 615)/2 = <u>593.6°F</u> | (574.2+615)/2 = <u>594.6°F</u> (0.6)

c. 1. Loop one Tavg becomes the Auctioneered High Tavg (0.2)

2. <u>Rods</u>: Rods begin moving in due to the Tavg-Tref deviation (0.6)

3. <u>Pzr</u>: Pzr reference level setpoint will increase to the maximum allowable level. (0.6)

4. <u>Steam dumps</u>: The failure of PT-447 would arm the dumps and the large Tavg-Tref deviation would cause the dump valves to open. (0.6)

REF.: North Anna Lesson Plan on RCS, pg 30; T.S. pg. 2-10; dwg. 108D014 (1 & 2) and 5655D33(9 & 10)

3-17 c,a,b,e,f,d (1.5)

REF.: North Anna Lesson Plan on Rod Control, pg 1

3-18 1. Cannot open when RCS pressure is >418 psig (0.5
2. Automatically closes when RCS pressure increases to >582 psig. (0.5

REF.: North Anna Lesson Plan on RHR, pg. 3

4-1 a. $\pm$400 pcm ($\pm$ 10% accepted) (0.5)

b. 1. partially insert control rods (to bottom of D bank) (0.5)

2. Record power level and dilute until counts have doubled (0.5)

3. Withdraw (D bank) rods to take reactor critical using a 1/M plot. (0.5)

REF.: North Anna OP-1C pg. 4 and OP-1.5 pg 10

4-2 FALSE (0.5)

REF.: North Anna H.P. Training Lesson Plan, pgs. 7&8

4-3 a. 1. Manually trip reactor (0.25)
2. Verify reactor trip and bypass breakers are open (0.25)
3. Verify rod bottom lights are lit (0.25)
4. Verify RPI's are indicating zero (0.25)
5. Verify neutron flux decreasing (0.25)
6. Manually trip turbine (0.25)
7. Verify turbine stop valves are closed (0.25)
8. Close reheater inlet valves (by pushing reset) (0.25)
9. Verify AC Emergency Busses are energized (0.25)

b. 1. Check charging/SI pump (0.25)
2. Verify FW isolation (0.25)
3. Verify AFW flow (0.25)
4. Verify charging/SI flow (0.25)
5. Verify LHSI Pumps running (0.25)
6. Verify Containment Isolation - Phase A (0.25)
7. Verify SW pumps running (0.25)
8. Verify RCS Heat Removal (0.25)
9. Check Containment pressure (0.25)

REF.: North Anna 1-EP-0 pgs. 2-5

(continued on next page)

4-4 8 x 1.0 = 8 penalty minutes (0.5)
46 x 1.0 = 46 penalty minutes (0.5)
82 x 0.0 = 0 penalty minutes (0.5)
72 x 0.5 = 36 penalty minutes (0.5)

90 total penalty minutes (0.5)

@ 0308 on 7-20-84 there would be 82 penalty minutes
@ 1809 on 7-20-84 there would be 60 penalty minutes
and power could be increased to >50% (1.0)

REF.: North Anna T.S. pgs. 3/4 2-1 & 2 and OP-2.1 (Unit Power Operation) pgs. 3&4

4-5
1. ionizing radiation (0.25)
2. heat stress (0.25)
3. differential pressure (0.25)
4. potential oxygen deficiency (0.25)

REF.: North Anna ADM-20.9 pg. 1

4-6 a. WB 0.75 rem/qtr. (0.4)
Hands 18.75 rem/qtr. (0.4)
Skin 5.00 rem/qtr. → 7.5 from GET N.A. pg 10 (0.4)

b. 1.75 rem/qtr. (0.4)

delete, not part of HP Prot Manual

c. 1. WB count ≤ 12 months ago
2. Sat. Resp. Fit Test ≤ 12 months ago
3. Sat. Pulmonary Function Test ≤ 12 months ago
? 4. Sat. Resp. Protection Training ≤ 12 months ago

NOTE: Time frame (~~0.25~~ 0.2 pts.) each of ~~four (4)~~ 3 requirements (~~0.25 pts.~~) (0.35)

REF.: Surry HP Manual, Sect. I, pgs. 1.2-3&4 and G.E.T. Manual pg. 40
CAF for North Anna Variations

4-7 TRUE (0.5)

REF.: North Anna OP-5.2, pg. 6

4-8 DELETE FALSE (Operation for more than two (2) minutes can damage the engine by oil collecting in inverted upper pistons). (0.5)

REF.: North Anna OP-6.5, pg. 3

(continued on next page)

4-9 TRUE (0.5)

REF.: North Anna EP-3, pg. 2

4-10 TRUE (0.5)

REF.: North Anna Emergency Plan

4-11 a. 1,2,3,4 (1.0)
b. 1,2,3,7,8,9 (1.0)
c. 5,6,9 (1.0)

REF.: North Anna Q & A Bank, question 4 & 7-56

4-12 1. Start the standby bearing cooling water pump (0.3)

2. If the standby pump will not start, attempt to restart the tripped pump. (0.3)

3. If neither pump will start, trip the reactor and the turbine and carry out EP-1. (0.6)

4-13 1. Check CC Head Tank level (0.25)

a. level - indicated (0.25)
b. CC pump amps - stable (0.25)
c. attempt to start the standby pump (0.25)

2. Check if CC pumps are running (0.25)

a. check flow - normal (0.25)

3. Monitor RCP temperatures (0.25)

a. motor brg. temp. <195°F (0.25)
b. pump brg. temp. <225°F (0.25)

1. head tank level
2. CC pump amps
3. attempt to start stby pump
4. Check flow
5. RCP mtr brg temp <195
6. RCP pump brg temp <225

0.375 pts each

REF.: North Anna AP-15 pg. 3

(continued on next page)

4-14
1. Start a second charging pump, as required (0.4)
2. Commence manual makeup to the VCT, if necessary (0.4)
3. Commence ramping down the unit (0.4)
4. If a reactor trip occurs proceed to 1-EP-1 (0.1)
5. Notify the Shift Supervisor and Health Physics Dept. (0.2)

REF.: North Anna AP24.1, pg. 2

4-15 Any of the following 6 (six) (1.2)

1. Subcriticality
2. Core cooling
3. Integrity
4. Heat Sink
5. Containment
6. Inventory
7. Inventory (without RVLIS)

REF.: North Anna Critical Safety Function List