U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 50-470/85-01

FACILITY DOCKET NO. 50-470

LICENSEE: Combustion Engineering
1000 Prospect Hill Road
Windsor, Connecticut 06095

FACILITY: Combustion Engineering Training Center

EXAMINATION DATES: June 18-20, 1985

CHIEF EXAMINER: N Dudley 8-19-85
N. Dudley, Reactor Engineer Examiner Date

REVIEWED BY: for D Johnson 8/19/85
Chief, Project Section 1C Date

APPROVED BY: HB Kister 8/22/85
Chief, Project Branch No. 1 Date

SUMMARY: Examinations were administered to twelve Instructor Certification candidates, and nine Instructor Certificates were issued. Continuing deficiencies in training material were identified.

## REPORT DETAILS

TYPE OF EXAMS: Initial ___ Replacement _X_ Requalification ____

EXAM RESULTS:

| | Inst. Cert Pass/Fail |
|---|---|
| Written Exam | 10/2 |
| Oral Exam | 10/0 |
| Simulator Exam | 7/3 |
| Overall | 9/3 |

1. CHIEF EXAMINER AT SITE: J. W. Upton, Jr. PNL

2. OTHER EXAMINERS: J. D. Smith, PNL
R. G. Clark, PNL

3. Summary of Training Deficiencies Observed in Simulator or Written Examinations:

   There is an inconsistency between normal plant parameters as presented in Emergency Operating Procedures, the lesson plan on Reactor Coolant System, and the lesson plan on Reactor Theory. None of the parameters correspond to the values for the core model used for the simulator as presented in CEN-128, vol 1. For example, the value for delta T across the core varies by 5% between all references. This results in many different "correct" values for exact parameters which would be unacceptable at an operating facility.

   The simulator malfunction sheet provided for preparation of the simulator examinations was inadequate. The single line for a malfunction does not provide sufficient detail to an examiner to determine what type of failure is simulated, the expected plant response, nor the expected operator response.

   Some candidates did not understand the effect that the Flow Dependent Setpoint Selector Switch has on the Reactor Protection System setpoints.

   Some candidates were unfamiliar with the technical specifications relating to the spent fuel building ventilation system.

4. Personnel Present at Exit Interview:

   <u>NRC Contractor Personnel</u>

   J. W. Upton, Jr., PNL
   J. D. Smith, PNL
   R. G. Clark, PNL

   <u>Facility Personnel</u>

   W. E. Burchill
   R. E. Price
   P. J. Dellarco

5. Summary of NRC Comments made at exit interview:

The individuals who had clearly passed the oral and simulator examinations were identified. The examiners made the following observations:

a. Knowledge and understanding by the candidates in the areas of radiation limits, protection and instrumentation was weak.

b. Verbal communication between team members during the simulator examinations was noticeably inadequate.

c. Knowledge and understanding by the candidates in the areas of neutron detectors and the excore/incore safety and control systems was good.

d. The Center staff assigned to run the simulator examinations were competent and helpful.

6. CHANGES MADE TO WRITTEN EXAM DURING EXAMINATION REVIEW:

SEE ATTACHED SHEETS (6)

Attachments:

1. Written Examination and Answer Key (SRO)
2. Facility Comments on Written Examinations made after Exam Review

# FACILITY REVIEW OF THE WRITTEN EXAMINATION

C-E TRAINING CENTER
WINDSOR, CT.
JUNE 18, 1985

Attendees

J. W. Upton, Jr. Chief Examiner, PNL
R. G. Clark PNL
J. D. Smith PNL
R. E. Price C-E Section 8
R. S. Rescorl C-E Section 7
J. A. Magennis C-E Section 6
P. J. Dellarco C-E Section 5

Facility Comment to QUESTION 5.04

The reactor operator would never calculate KVA. One could expect him to determine MW and MVARS from the Plant Physics Book, but not KVA.

Resolution

Any training with respect to power and reactive power would teach the relation between watts, vars and voltamps. No change to the answer key.

Facility Comment to QUESTION 5.07b

We have removed in our training all reference to "required" vs "actual" NPSH. The candidates may just comment that the curve in the suppliers manual would show an increase in NPSH.

Resolution

Any phraseology that says that the NPSH would increase with flowrate will be acceptable.

Facility Comment to QUESTIONS 5.01 and 5.05b.

Some spread in the numerical answer should be accepted because the candidates may use the simulator values for $T_C$ and $T_{ave}$ at 100% power which are lower than the published values.

Resolution

A spread in the numerical values will be accepted.

Facility Comment to QUESTION 6.01

QUESTION 6.01 was modified prior to the taking of the examination by the candidates such that the second sentence read, "Figure 6.01 shows the 'FLOW DEMAND' signal being routed to the main feedwater valve controller." The facility reviewers provided a copy of the flow diagram during the examination review.

Resolution

The answer will be based on the flow diagram provided by the reviewers. In a list format, the answer is now:

- level
- feedwater flowrate
- steam flowrate
- level setpoint
- the level signal is passed through a lead/lag circuit and then compared to the level setpoint
- the feedwater flowrate is passed through a lead/lag circuit and then compared to the steam flowrate
- these two signals are added and used to control the main feedwater control valve

(+0.5 for each bullet, +3.0 max)

Facility Comment to QUESTION 6.02

The answer to part "a." is incomplete and should include as "5.", "use the other excore channels". The answer to "b." should be corrected to read, "Pzr pressure and T_H". Part "c." is also incomplete and should be modified to include a "4." which would cover the fact that there are 2 channels of the RRS and the operator could choose the other channel.

Resolution

The answer key will include the additions to parts "a." and "c." and athe change to part "b.". The answer will read in the following manner:

a. 1. _T power
   2. feedwater flowrate, feedwater temperature and S/G pressure (secondary calorimetric)
   3. incore neutron detectors
   4. T_H, T_C and primary-coolant flowrate (primary calorimetric)
   5. other excore neutron detectors

   (+0.5 each, +1.0 max)

b. Pressurizer pressure and T_H (+0.5)

c. 1. T_H and T_C
   2. S/G pressure
   3. incore T/Cs
   4. other channel of the RRS
   (+0.5 each, +1.0 max)

Facility Comment to QUESTION 6.08

The answer to part "b." should include, "Select the other RRS channel after taking local control of the Pressurizer-level set point."

Resolution

The statement above waill be accepted as an alternate answer.

Facility Comment to QUESTION 7.01

The answer to part "d." should accept "(8.)" as an alternate answer because during the blowdown of S/G A the RTDs would behave irratically.

Resolution

Either "(6.)" or "(8.)" will be accepted as an answer to "d.".

Facility Comment to QUESTION 7.02

The reviewers showed documentation that supports "increases" as the answer to parts "a." and to "b.".

Resolution

The answer to parts "a." and to "b." will be "increased".

Facility Comment to QUESTION 7.04

The reference is correct, but it is not the reference that was sent. It is an old EOP and hence the answer is wrong. There are 4 success paths and the statements should refer to greater than 1 CEA not inserted. This question is not a good question as it requires more memorization than should be required.

Resolution

The question will be retained but modified to be consistent with the latest version of the EOP. The corrected answer is the following:

1. CEA insertion (+0.7) used when there is greater than 1 rod bottom light not lit (+0.2) or the equivalent on the metrascope (+0.1)
2. borate via the CVCS (+0.7) used when there is greater than 1 CEA not inserted and the reactor power is >1% or increasing (+0.1) and the RCS pressure is >1250 psia (+0.2)
3. borate via the HPSI (+0.7) used as in "2." above (+0.1) but with the RCS pressure <1250 psia (+0.2)
4. CEA drive-down if path 3 fails (+1.0)

(+3.0 max)

Facility Comment to QUESTION 7.05

This is not an appropiate question as it is beyond what an operator should know by memory. An operator would review the procedure.

Resolution

The question will be retained; but in the grading, a general description of the situation will be accepted for (+1.4).

Facility Comment to QUESTION 7.06

The RCP-seal high-temperature alarm setpoint is 200_F.

Resolution

The alarm setpoint is not required for full credit.

Facility Comment to QUESTION 7.08

The answer is essentially correct as far as AOP-3 is concerned. However, an operator would also consider the possibility that the float valve on the head tank was stuck open.

Resolution

The answer key will be modified to accept as an answer operator actions to verify the status of the valves and to restore CCW. Reference to the concern about a RCS leak will be required for full credit. The answer key is the following:

1. Locate and isolate the leak from the RCS.
2. Verify the status of the valves to the head tank and restore the CCW. Check for indications of RCS leakage into the CCW.

(+1.5 for either answer)

Facility Comment to QUESTION 7.09

QUESTION 7.09 is not a good question as it requires an unnecessary amout of memorization.

Resolution

There is no change to QUESTION 7.09.

Facility Comment to QUESTION 8.01

An answer to part "b." that says, "2 independent HPSI pumps are required to be operable" is indeed the correct answer, but it is often assumed that "2 HPSI pumps" means that they are being powered off of seperate buses. Hence the distributuin of credit should be different. Part "c." is not complete as Tech-Specs also specifies that the plant should be in Cold Shutdown within the next 24 hours.

Resolution

The point credit for part "b." will be (+1.0) for the first sentence and (+0.5) for the second sentence. The sentence, "Be in Cold Shutdown wilthin the next 24 hours. (+0.25)" will be added to the answer to part "c.". The maximum credit for part "c." will remain as (+1.5 max).

Facility Comment to QUESTION 8.02

QUESTION 8.02 asks only for "the maximum time" and hence an answer of 2 hours should be sufficient.

Resolution

Full credit of (+1.0) will be given for an answer of 2 hours.

Facility Comment to QUESTION 8.05

"We take exception to QUESTION 8.05c. 72-hour action statements should not need to be committed to memory. All that an operator needs to know is that there is an action statement that pertains to this situation."

Resolution

No change to QUESTION 8.05.

Facility Comment to QUESTION 8.06

The answer to part "c." is correct according to Tech-Specs, but in the Administrative Procedures it specifies that the Shift Supervisor should approve of the change.

Resolution

No change to QUESTION 8.06.

Facility Comment to QUESTION 8.07

"We fail to see the intent of this question and feel that it is not appropiate. The Shift Supervisor's approval is not required in any of these situations. According to AOP-8 the HP Supervisor and HP Department Head are the same person.

Resolution

HP Supervisor and HP Department Head will be accepted interchangeably in the answer.

Facility Comment to QUESTION 8.09

Same as 8.07 in that there is no need for the Shift Supervisor to know this.

Resolution

No change to QUESTION 8.09.

YRS

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

**Facility:** C-E TRAINING CENTER (470)

**Reactor Type:** PWR-CE

**Date Administered:** June 18, 1985

**Examiner:** Joe Upton et al., PNL

**Candidate:** MASTER

INSTRUCTIONS TO APPLICANT:

Print your name on the line above marked "Candidate." The grade points available for each question are indicated within parentheses after each question. The passing grade is at least 70% in each of the four (4) categories and is at least 80% for the total grade. Use separate paper for your answers and write on only one (1) side of the paper, unless a specific question instructs you otherwise. Staple this question package to your answer sheets. The examination questions and answers will be picked up six (6) hours after the examination was started. Read the statement at the bottom of this page. When you have finished this examination, affirm the statement by signing your name.

| Category Value | % of Total | Applicant's Score | % of Cat. Value | Category |
|---|---|---|---|---|
| 25 | 25 | | | 5. Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics |
| 25 | 25 | | | 6. Plant System Design, Control and Instrumentation |
| 25 | 25 | | | 7. Procedures - Normal, Abnormal, Emergency and Radiological Control |
| 25 | 25 | | | 8. Administrative Procedures, Conditions and Limitations |
| 100 | 100 | | | TOTALS |

Total Grade ________ %

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

Candidate's Signature

5.0 THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS AND THERMODYNAMICS (25)

Points Available

**The following statements apply to SECTIONS 5.01, 5.02, 5.03, 5.04 and 5.05.**

The C-E Training Facility "power plant" has been operating continuously at a steady 100% of full power for 10 days. All control rods (CEAs) are fully withdrawn from the nuclear reactor core (ARO). All controlled parameters are equal to their respective programmed values. The fuel-burnup status is that the core has reached 12K MWD/T in cycle 6. The present boron concentration is 200 ppm. The generator is operating with a pf of 0.95 lagging. Use any of the provided figures and tables. Show your work and your procedures for arriving at your answers.

QUESTION 5.01

a. What is the expected value for the $\Delta T$ across the core? (0.5)

b. What is the expected axial neutron-flux shape; i.e., sketch (qualitatively) the thermal neutron flux as a function of axial distance. Explain your sketch; i.e., provide the rationale for the shape of your sketch. (2.5)

c. If the flowrate of the primary coolant through the core was reduced, explain the effect this would have on the $\Delta T$ across the core. (1.0)

d. If the flowrate of the primary coolant through the core was reduced, explain the effect this would have on the axial neutron-flux shape. (1.0)

- Section 5 continued on next page -

Points Available

QUESTION 5.02

a. What is the temperature in the Pressurizer? (0.5)

b. What is the margin of subcooling in °F for the coolant in the hot leg and for the coolant in the cold leg? (1.0)

c. Plot the point on the figure, "Reactor Coolant System Pressure Temperature Limitations," on page 2 of the Technical Data Book that corresponds to the present operating conditions, and determine the margin in psi to the 50°F Subcooled margin limit. (1.0)

d. If the power level was reduced from 100% to 0%, does the mass of the primary coolant increase or decrease? Explain your answer. (1.5)

QUESTION 5.03

a. If the control rods (CEAs) were inserted while in the manual sequential (MS) mode until Bank 6 is at 54 inches ~~inserted~~, WITHDRAWN and if the boron concentration was adjusted to maintain the present power level of 100%, what would be the new concentration of boron required for steady operation? Neglect any effect from changes in the zenon or samarium concentrations. (1.5)

b. How many gallons of boric acid or makeup water would have to be added? (1.0)

c. According to the CEA Insertion Limits criteria, is this operation allowable? If not, what insertion would be allowable? (1.0)

- Section 5 continued on next page -

Points Available

QUESTION 5.04

The "dispatcher" calls and informs you that your power plant is going to have to carry a lagging pf of 0.85 due to changes in the other generators feeding the large (infinite) grid. (The initial plant conditions are stated prior to QUESTION 5.01.)

a. List the adjustments that must be made in the control room for the power plant to meet this new requirement. (1.5)

b. After the adjustments have been made, what will be the real or true output power, the reactive output power and the apparent output power? (1.5)

QUESTION 5.05

The power plant is to be taken from 100% to 80% of full power. Assume $T_{ave} = T_{ref}$ at both 100% and at 80%.

a. What would be the magnitude of the change in reactivity (in $\% \Delta\rho$) due to the change in the temperature in the power plant? Specify whether this change would add or take away reactivity from the core. (1.0)

b. What is the change in reactivity (in $\% \Delta\rho$) due to the change in the moderator temperature? Specify magnitude and direction. (1.5)

c. If the core were a fresh core; i.e., zero MWD/T on cycle 6, how would your answer to "b." above change? Explain. (1.5)

d. Assuming that this maneuver takes 20 minutes, make a sketch on the provided graph (Figure 5.05) of the xenon worth as a function of time. Show time from the point in time of decreasing power and for the next 50 hours. (1.5)

- Section 5 continued on next page -

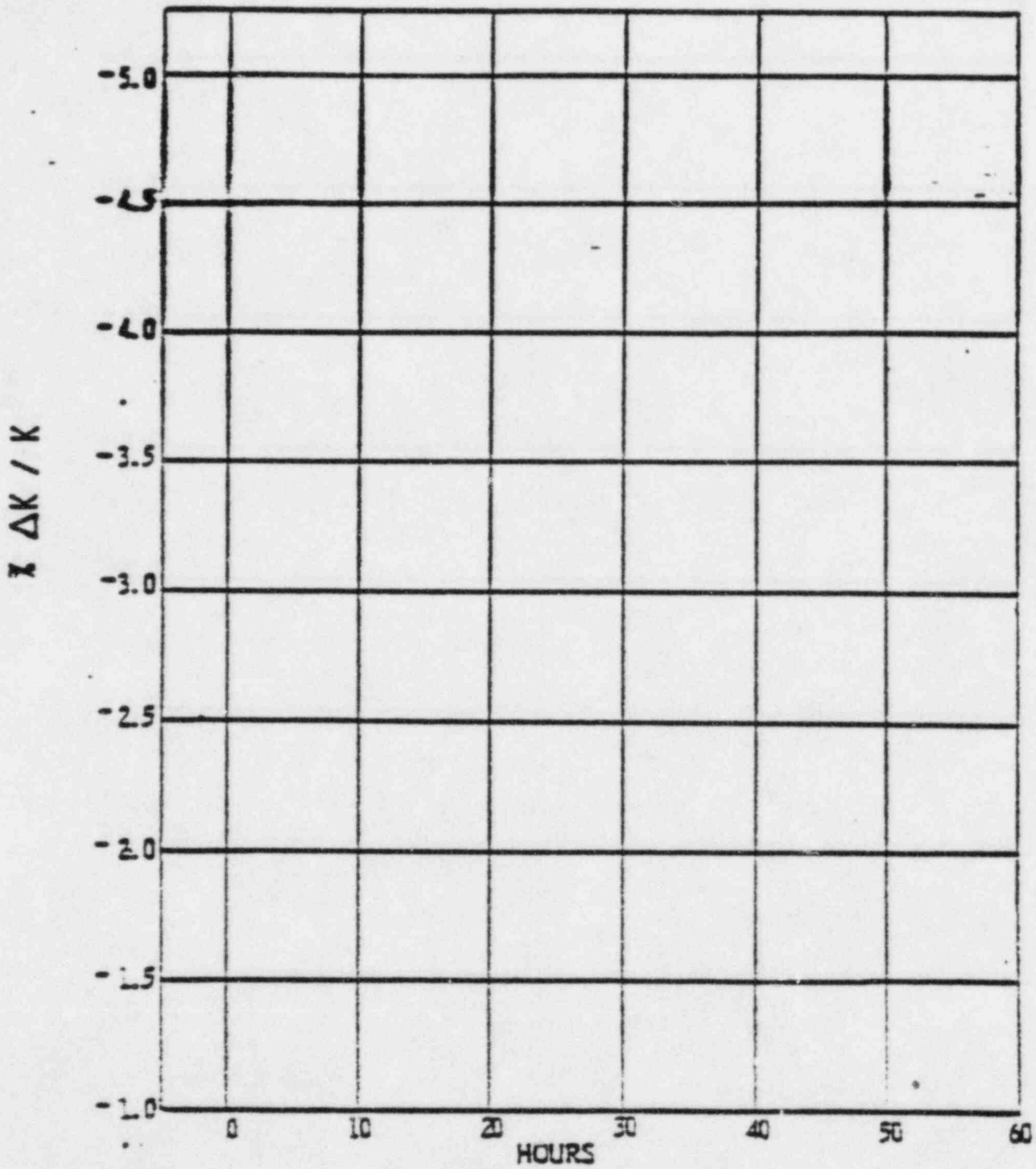


FIGURE 5.05 (QUESTION)

- Section 5 continued on next page -

Points Available

QUESTION 5.06

For QUESTION 5.06, refer to Figure 5.06 which shows a condenser hotwell and condensate pump. Recall that 1 atm = 29.92 in. Hg = 33.9 ft $H_2O$.

Select the letter designation below that most accurately gives the absolute pressure in psia in the condenser.

(a.) 1.93 psia
(b.) 2.31 psia
(c.) 12.77 psia
(d.) 16.91 psia (1.0)

QUESTION 5.07

a. Why is "NPSH" important; i.e., what concern does it address? (1.0)

b. Why is the following statement true: "If the flowrate through a centrifugal pump was increased, then the 'required NPSH' would be increased."? (1.0)

QUESTION 5.08

While cooling the power plant on natural circulation, how can the cooldown rate be controlled? How can the cooldown rate be increased? (1.0)

- Section 5 continued on next page -

7

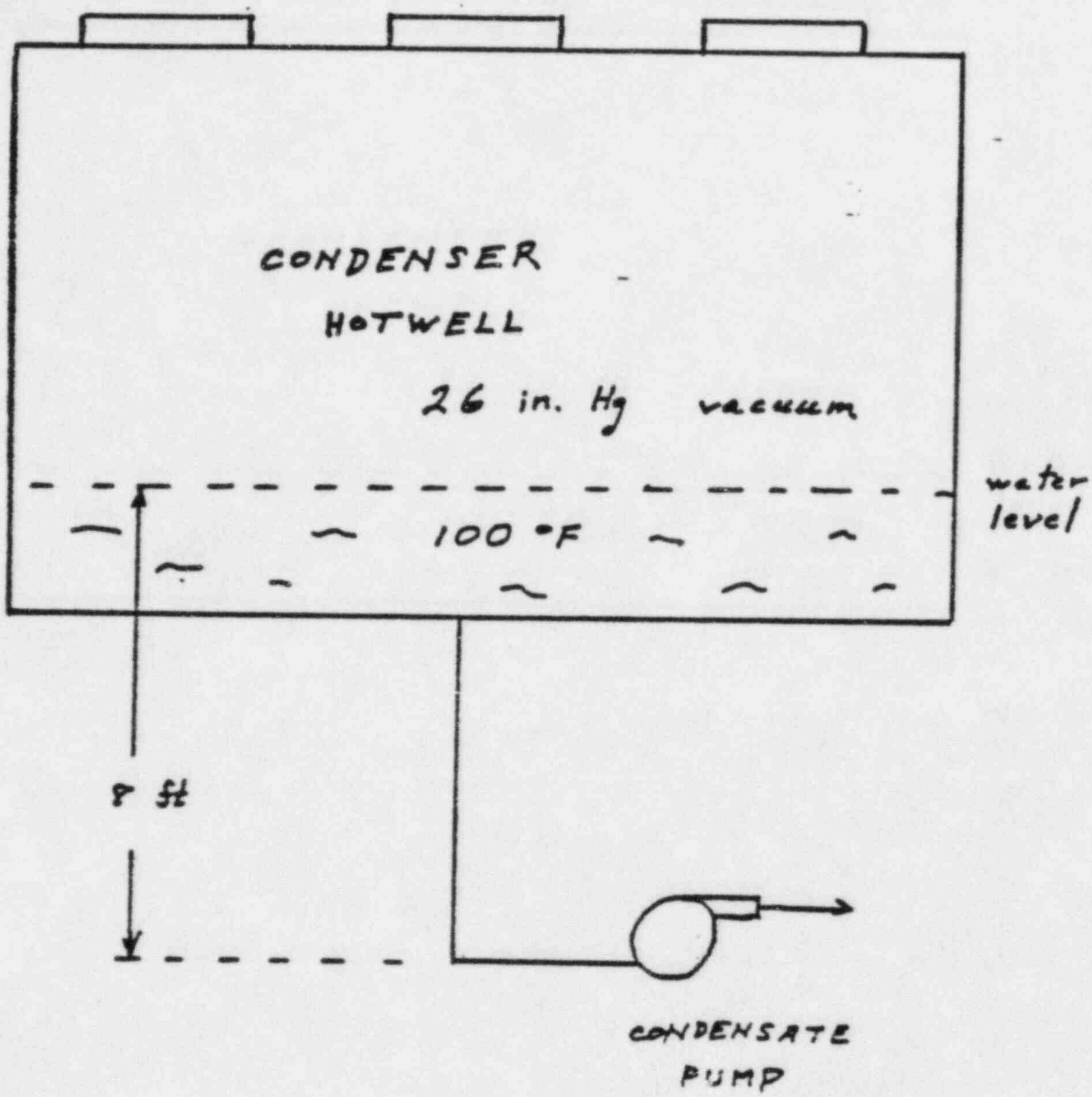


FIGURE 5.06 (QUESTION)

- End of Section 5 -

## 6.0 PLANT SYSTEM DESIGN, CONTROL AND INSTRUMENTATION (25)

Points Available

QUESTION 6.01

Figure 6.01 shows part of the Feedwater Control System (FWCS).

Figure 6.01 shows the "FLOW DEMAND" signal being routed to MAIN FEEDWATER VALVES ~~three (3) "programmers" which feed three (3) "M/As."~~ Describe how this FLOW DEMAND signal is generated. (Assume that the master controller is in AUTO.) In particular, list the sensor signals that are used as inputs, the inputs that are manually set, and specify how these inputs are processed and combined. Do not describe test or calibration functions and operations. (3.0)

QUESTION 6.02

What alternate sets of instrumentation can be used to verify each of the following indications?

a. nuclear excore instrumentation (TWO sets required) (1.0)
b. subcooled-margin meter (ONE set required) (0.5)
c. $T_{avg}$ (TWO sets required) (1.0)

- Section 6 continued on next page -

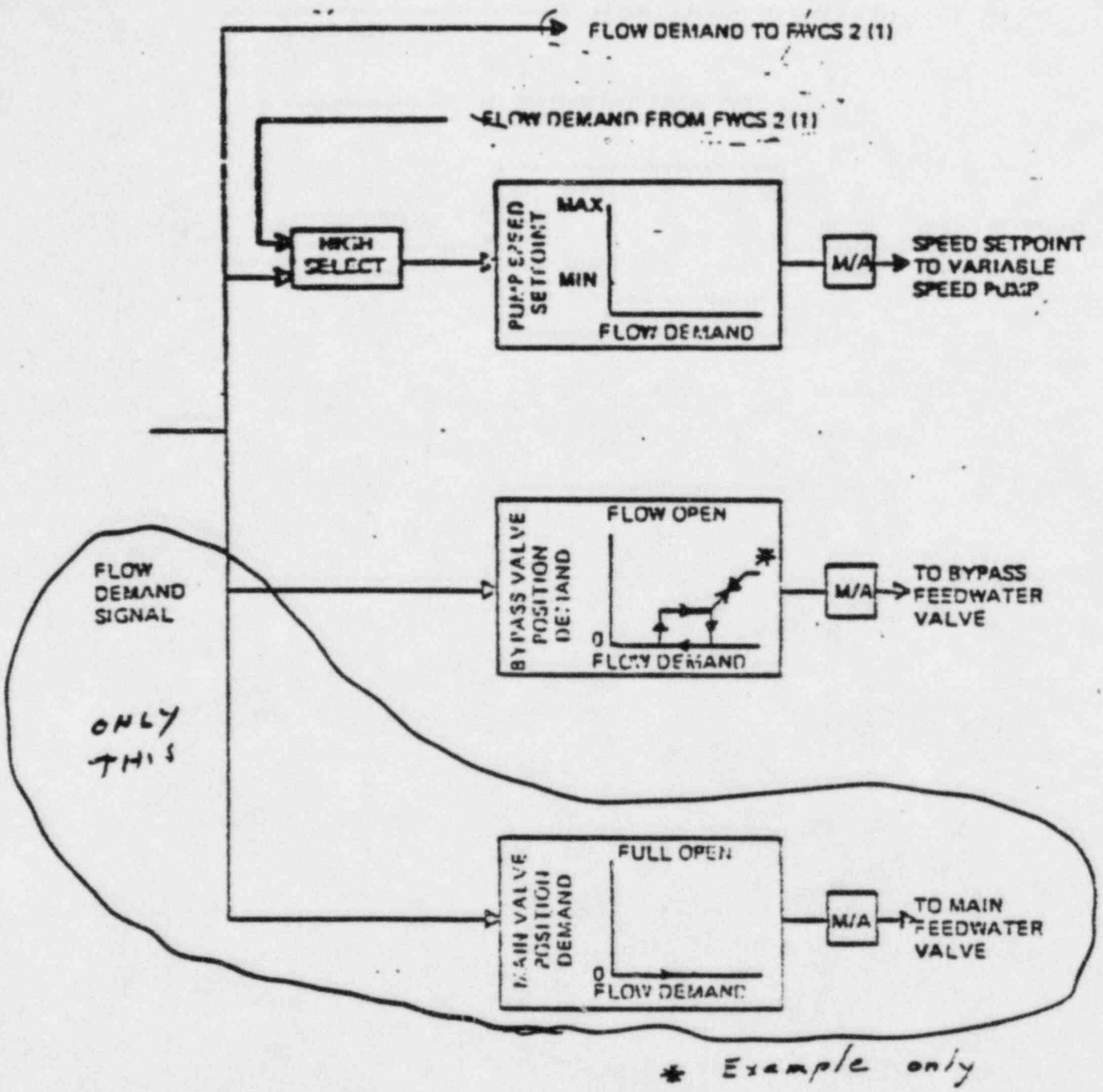


FIGURE 6.01

- Section 6 continued on next page -

Points Available

QUESTION 6.03

Sketch the flow paths associated with Safety Injection Tank (SIT) 11A as directed below. Indicate valve types but not valve number designations.

a. The sketch should show the SIT and the flow path(s) to the Loop 11A cold leg, including all valves. (1.0)

b. Include on the sketch the flow path(s) that connect the AUX HPSI, MAIN HPSI and LPSI to the Loop 11A cold leg. Show all valves and flow sensors associated with this portion. (1.0)

c. Include on the sketch the flow path(s) that allow for a measurement of leakage through the loop-entry check valves. Show all valves and flow sensors. (1.0)

d. If the level indicator for the SIT initiates a LO alarm, specify two (2) additional Control Room indications that you would check as verification that the SIT level is indeed low. (1.0)

- Section 6 continued on next page -

Points Available

QUESTION 6.04

Answer TRUE or FALSE to each of the following statements concerning the Containment Iodine Removal System.

a. Each filter unit consists of a moisture seperator, a high-efficiency particulate air filter and a fan. (0.5)

b. The filter-unit fan starts on either a CIS or a SIAS. (0.5)

c. With two (2) filter units operating and with the other postulated, conservative assumptions associated with a LOCA, the two (2) hour dose at the site boundary and the total dose at the outer perimeter of the low-population zone are within the limits of 25 rem to the whole body and 300 rem to the thyroid. (0.5)

d. Number 12 and 13 filter units can be lined up to receive electric power from either vital bus through the use of remote-operated disconnects. (0.5)

- Section 6 continued on next page -

Points
Available

QUESTION 6.05

Specify the letter designation of the most correct statement from those given below. (1.0)

(A.) The purpose of the Auxiliary Feedwater System is to provide feedwater for the removal of heat from the primary coolant after a Steam-Generator tube-rupture event has been diagnosed.

(B.) The auxiliary feedwater pump has a capacity of 5% of rated feedwater flow at rated head.

(C.) At a level of -40 inches or less in a Steam Generator, the steam inlet valve automatically opens providing steam to the Auxiliary Feedwater Pump Turbine and automatically opens the auxiliary feedwater inlet valve.

(D.) During plant cooldown, the preset (for the Auxiliary Feedwater Pump) turbine speed is greater than that required to supply feedwater for decay-heat removal and cooldown and the speed must be lowered by means of the local manual speed control.

- Section 6 continued on next page -

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Points Available

QUESTION 6.06

Describe the construction and operation of the letdown process radiation monitor. In particular:

a. How and from where is the sample obtained that is to be measured for its radioactive level? (1.0)

b. What type of detectors are used and what type of circuitry is used? What information or output is the result of the design of these detectors and associated circuitry? (1.0)

c. Describe the impact that ABNORMAL flowrates of the Component Cooling Water would have on the operation of the process radiation monitor. Give reasons for any stated impacts. (1.0)

d. How would you use the process radiation monitor information available in the Control Room to determine that you have had a sudden "crud-burst?" (1.0)

QUESTION 6.07

If, during reactor plant operations at 95% power, a feedline rupture were to occur inside the containment, what are ALL the Engineering Safety Features (ESFs) that could possibly be actuated and what signals will cause these actuations? Include setpoints and logic. (3.1)

- Section 6 continued on next page -

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Points Available

QUESTION 6.08

The power plant is in equilibrium (steady) operation at 100% of full power. RRS Channel-Y is selected to calculate $T_{ave}$ from Loop-2. RRS Channel-X is selected to calculate $T_{ave}$ from Loop-1. The Pressurizer level control is selected to Channel-X. Temperature element TE-111X (Loop-1 hot-leg temperature element) fails and reads 150°F low.

a. Explain how and why the charging pumps and letdown valves should respond to the failed TE-111X. (2.0)

b. What actions should the operator take to compensate for the failed temperature indication? (1.0)

QUESTION 6.09

Will the plant trip as a result of the following situations. Explain your answers. Consider each situation separately.

a. 120 volt vital bus A is de-energized and channel B pressurizer pressure indication fails high. (0.8)

b. The Flow Dependent Setpoint Selector Switch is placed in the 3 pump position while at 70% power and Axial Shape Index is -0.3. (0.8)

c. SG 1 pressure channel A fails to 400 psia and SG 2 pressure channel B vails to 450 psia while at 60% power. (0.8)

- End of Section 6 -

## 7.0 PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL (25)

Points Available

### QUESTION 7.01

Assume that the scenario described below has occurred for each of the four (4) conditions (a, b, c, and d). Designate the most probable cause by choosing from the list of possible causes. Consider each of the four conditions separately.

Scenario: The power plant has been operating at 100% of full power for two (2) months. All four (4) RCPs trip, and after 15 minutes your operators inform you that natural circulation flow has NOT developed.

a. The wide-range levels for both Steam Generators are indicating less than 0% and $T_C$ is increasing above the $T_{sat}$ for the Steam Generators. (0.75)

b. Both $T_C$ and the pressures in the Steam Generators are increasing. (0.75)

c. The RCS loop $\Delta T$ is increasing, and the Pressurizer level is erratic. (0.75)

d. The Pressurizer level is low, and there is a mismatch between the loop RTDs and the core exit thermocouples. (0.75)

Possible causes:

(1.) Condensible voids developing in RCS flow path.

(2.) Failure of both Main Feedwater Pumps.

(3.) The Auxiliary Feedwater Pumps are operating at maximum flowrate.

(4.) Inadequate secondary steam flowrate.

- Section 7 continued on next page -

Points Available

QUESTION 7.01 (contd)

(5.) Pressurizer relief valves stuck closed.

(6.) Inadequate RCS inventory.

(7.) More than one CEA stuck in the out position.

(8.) Steam-line rupture on Steam Generator A.

QUESTION 7.02

At the moment a bubble is formed in the Pressurizer, how should each of the following parameters respond?

| | | |
|---|---|---|
| a. | the letdown flowrate | (0.5) |
| b. | the Pressurizer pressure | (0.5) |
| c. | the Pressurizer temperature | (0.5) |
| d. | the Pressurizer level | (0.5) |

QUESTION 7.03

Specify the letter designation of the most correct phrase with which to complete the following sentence concerning the Pressurizer Pressure Control System.

To increase the heat supplied by the proportional heaters with the manual/automatic station for the Pressurizer heaters and spray in the Manual Mode, you must . . . (0.65)

(A.) increase the controller output.
(B.) adjust the pressure setpoint higher.
(C.) decrease the controller output.
(D.) adjust the pressure setpoint lower.

- Section 7 continued on next page -

Points
Available

QUESTION 7.04

In accordance with EOP 7A, Reactivity Control, what are the THREE (3) reactivity control success paths and WHEN would each be used? (3.0)

QUESTION 7.05

In preparation for reactor plant heat up, RCP 11-A has been started. If both RCP 11-B and RCP 12-A had tripped on their initial start, what should be the pump starting sequence to establish three running pumps in the shortest time if all subsequent RCP starts are successful? Explain your reasoning. (2.0)

QUESTION 7.06

In each of the following situations, specify the conditions which would require that the power plant be tripped.

a. decreasing condenser vacuum, while at 80% power (0.7)

b. loss of an operating Component Cooling Water pump while at 50% power (1.1)

c. a reactor coolant system leak which is slowly increasing while at 50% power (0.7)

QUESTION 7.07

Following a reactor trip, what FOUR actions must be taken besides verifying proper automatic functions if all systems operate normally? (2.5)

QUESTION 7.08

In general, what actions need to be taken if Component Cooling Water is lost because of a high-level in the head tank? (1.5)

- Section 7 continued on next page -

Points Available

QUESTION 7.09

Explain why each of the following events would or would not require suspension of movement of irradiated fuel inside the containment.

a. A maintenance person opens the inner door of the air lock to exit the containment. The outer door of the air lock is shut. (0.75)

b. Securing of the operating shutdown cooling loop which leaves both loops secured but operable. (0.75)

c. The damper of a spent fuel ventilation exhaust fan fails shut due to a loss of instrument air to the damper. (0.75)

QUESTION 7.10

Under what conditions would a formal ALARA review be required? (1.0)

QUESTION 7.11

Indicate by title the person who:

a. will act as the Manager of Control Room Operations during an emergency (0.6)

b. is responsibile for the assessment, classification, and declaration of emergencies (0.6)

c. initially assumes the responsibilities of the Director of Station Emergency Operations. (0.6)

- Section 7 continued on next page -

Points Available

QUESTION 7.12

Assume that a Steam Generator tube rupture has been verified at a leak rate of 2 gpm. Emergency Operating Procedure EOP-4 cautions the operator to reduce the RCS $T_h$ to less than 525°F before isolating the affected Steam Generator. What is the reason for this caution? (1.0)

QUESTION 7.13

Following a Steam Generator tube rupture, the operator is instructed to control the RCS pressure, maintaining it below 1000 psia. What three (3) systems are available to the operator to effect this control of the pressure? (1.8)

- End of Section 7 -

## 8.0 ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (25)

Points Available

### QUESTION 8.01

For each of the following situations indicate what REQUIREMENT, if any, applies and what ACTION, if any, should be taken. Consider each situation separately.

a. Diesel Generator A's operability load test, which is required every 31 days, is scheduled for today. The last three tests were completed 36, 68, and 102 days ago, respectively. The plant is at 100% power. (1.5)

b. The plant is at 295°F and heating up at 1°F per minute, when an HPSI pump is found to be inoperable. (1.5)

c. The plant is at 100% power when it is determined that the heat tracing circuits for both boric acid storage tanks are inoperable and cannot be repaired for 4 days. (1.5)

- Section 8 continued on next page -

Points Available

QUESTION 8.02

a. Complete the following table to indicate the minimum shift crew composition in the applicable modes. (2.0)

MINIMUM SHIFT CREW COMPOSITION #

| LICENSE CATEGORY | APPLICABLE MODES 1, 2, 3, & 4 | 5 & 6 |
| --- | --- | --- |
| SOL | | |
| OL | | |
| Non-Licensed | | |
| Shift Technical Advisor | | |

b. If the minimum shift crew composition of Question 8.02 above cannot be met, what is the maximum time allowable to restore the shift crew composition to within the minimum requirements? (1.0)

- Section 8 continued on next page -

Points Available

QUESTION 8.03

a. If a bypass device is to be used on a system and it is determined that using the bypass device WILL cause adverse environmental impact, what form of approval is required to use the bypass device? (1.0)

b. Can safety tags be lifted for any reason other than clearing the tags? EXPLAIN. (1.0)

QUESTION 8.04

Classify the following conditions according to the "Emergency Plan" in EOP-9. Consider each part of the question, each event, as separate and unrelated to the other events. Specify each as Unclassified, Unusual Event, Alert, Site Area Emergency or General Emergency. (3.0)

a. Power: 100%
All Ts: ($T_c$, $T_h$, $T_{ave}$, $T_{feed}$, $T_{steam}$, ...): normal
Pzr level: normal
Pzr pressure: normal
Letdown process monitor: alarms
Chemistry analysis of primary coolant: 256 µCi/gm I '131'
Electrical: normal

b. Power: 100%
All Ts: normal
Pzr level: -5% and decreasing
Pzr pressure: 2200 psia and decreasing
Containment pressure: 1 psig
Containment radiation monitors: $10^4$ mR/hr
Electrical: normal

c. Power: 100%
All Ts: normal
Pzr level: -2% and increasing
Pzr pressure: 2220 and decreasing
Blowdown process monitors: increasing
Condenser air ejector monitors: $5 \times 10^{-3}$ µCi/cc I '131'
Electrical: 6.9 kV and 4.16 kV buses are lost

- Section 8 continued on next page -

Points Available

QUESTION 8.05

The Tech-Specs specify a Limiting Condition for Operation (LCO) with respect to the Auxiliary Feedwater system.

a. What is the LCO for Modes 1-3? (1.5)

b. For what does the OPERABILITY of the Auxiliary Feedwater System provide assurance? (1.0)

c. With one (1) auxiliary feedwater pump inoperable, what action is required? (1.0)

QUESTION 8.06

List the letter designations of those statements chosen from the following statements which are correct. The statements are in response to, "Temporary changes to procedures may be made provided:" (2.0)

(a.) Critical operation of the unit shall not be resumed until authorized by the Commission (U.S. NRC).

(b.) The intent of the original procedure is not altered.

(c.) The change is approved by two (2) members of the plant management staff, at least one of whom holds a SRO License in the affected unit.

(d.) The change is documented, reviewed by the POSRC and approved by the Plant Manager within 21 days of implementation.

- Section 8 continued on next page -

Points Available

QUESTION 8.07

What permission (from whom and at what check points) does a radiation worker need to complete a task which is expected to increase his exposure by 2000 mrem this quarter? The worker is 30 years old, has a completed NRC Form 4 and has a radiation history of 13000 mrem lifetime, 3000 mrem for the year, and 600 mrem for the quarter. (1.5)

QUESTION 8.08

Technical Specification 3/4.1.1.5 states the lowest loop operating temperature for the RCS $T_{ave}$ shall be ≥515°F when the reactor is critical.

a. Explain what four (4) things this specification ensures. (2.0)

b. How often must this be determined when $T_{ave}$ is less than 525°F with the reactor critical? (0.5)

QUESTION 8.09

What is required of personnel before they can be designated as "escorts" in Radiation Work Areas? (1.0)

QUESTION 8.10

When a shift supervisor places his/her signature on an RWP, he/she is verifying that certain conditions have and will exist, and that certain commitments will be kept. List two (2) of these conditions or commitments. (2.0)

- End of Section 8 -

END OF EXAMINATION

## 5.0 THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS AND THERMODYNAMICS (25)

Points Available

The following statements apply to QUESTIONS 5.01, 5.02, 5.03, 5.04 and 5.05.

The C-E Training Facility "power plant" has been operating continuously at a steady 100% of full power for 10 days. All control rods (CEAs) are fully withdrawn from the nuclear reactor core (ARO). All controlled parameters are equal to their respective programmed values. The fuel-burnup status is that the core has reached 12K MWD/T in cycle 6. The present boron concentration is 200 ppm. The generator is operating with a pf of 0.95 lagging. Use any of the provided figures and tables. Show your work and your procedures for arriving at your answers.

QUESTION 5.01

a. What is the expected value for the $\Delta T$ across the core? (0.5)

b. What is the expected axial neutron-flux shape; i.e., sketch (qualitatively) the thermal neutron flux as a function of axial distance. Explain your sketch; i.e., provide the rationale for the shape of your sketch. (2.5)

c. If the flowrate of the primary coolant through the core was reduced, explain the effect this would have on the $\Delta T$ across the core. (1.0)

d. If the flowrate of the primary coolant through the core was reduced, explain the effect this would have on the axial neutron-flux shape. (1.0)

ANSWER 5.01

a. If, for 100% of full power

$T_H = 599°F$

$T_C = 544°F$,

then $\Delta T = 55°F$. (+0.5)

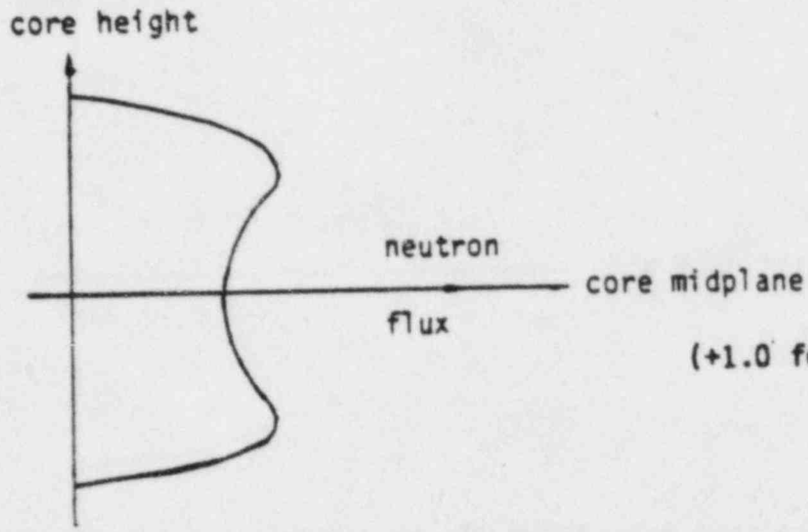
52 to 55°F

- Section 5 continued on next page -

Points
Available

ANSWER 5.01 (contd)

b.




(+1.0 for shape)

With a fuel-burnup status of 12K MWD/T, the core is at End-of-Life (EOL) (+0.5). At EOL, the U-235 fuel has been disproportionally burned-up in the bottom half of the core. This shift in the relative fuel concentration to the top half of the core is responsible for the peak in the top half of the core. At full power the moderator density decreases with core height. The higher moderator density in the bottom half of the core is responsible for the thermal-neutron peak in the bottom half of the core. (+1.0)

c. If the flowrate is reduced, more heat will be added to the coolant as it passes through the core. This will raise the core $\Delta T$ (+1.0). An increase in the temperatures of the core will cause the power produced in the core to try to decrease. However, if the electrical output is to remain the same, the secondary system will adjust and the increase in the core $\Delta T$ will remain to offset the decrease in the flowrate of the primary coolant.

d. If the flowrate of the primary coolant is reduced, the core temperature will rise. The increase in temperature will be greater in the top half of the core than in the bottom half. This will cause the neutron flux to shift to the bottom of the core. (+1.0)

- Section 5 continued on next page -

Points Available

ANSWER 5.01 (contd)

Reference(s)

1. C-E Training Center: "Reactor Theory," Flux Distribution, Section 6.3, Figures 6-24 through 6-29 and Figure 6-38.

QUESTION 5.02

a. What is the temperature in the Pressurizer? (0.5)

b. What is the margin of subcooling in °F for the coolant in the hot leg and for the coolant in the cold leg? (1.0)

c. Plot the point on the figure, "Reactor Coolant System Pressure Temperature Limitations," on page 2 of the Technical Data Book that corresponds to the present operating conditions, and determine the margin in psi to the 50°F Subcooled margin limit. (1.0)

d. If the power level was reduced from 100% to 0%, does the mass of the primary coolant increase or decrease? Explain your answer. (1.5)

ANSWER 5.02

a. The temperature in the Pressurizer is the saturation temperature for 2250 psia and is 653°F **(+0.5 for 651°F to 655°F)**.

b. Margin of subcooling for the hot leg $= T_{sat}$ (2250 psia) $- T_H$

$= 653 - 599$

$= 54$°F **(+0.5 for 52°F to 56°F)**

for the cold leg $= 653 - 544$

$= 109$°F **(+0.5 for 107°F to 111°F)**

- Section 5 continued on next page -

Points
Available

ANSWER 5.02 (contd)

c. See the attached figure. (+0.5)
The psia margin to the 50°F subcooled curve is
2250 - 1500 = 750 psia (+0.5)

d. Considering the dashed curve on the figure on page 6 of the Technical Data Book, the actual operating point is at a higher volume than that corresponding to a constant mass. Hence, in going from 100% to 0% the mass has increased (+0.5). The volume has decreased; the temperature has also decreased which increases the density of the coolant. The increase in density produces a greater effect than that of the decrease in volume, so the mass has increased (+1.0).

Reference(s)

1. C-E Training Center: Technical Data Book, p. 2.
2. Generic: Academic Program for Nuclear Power Plant Personnel," Volume III, pp. 2-45 through 2-56, General Physics Corporation.

- Section 5 continued on next page -

REACTOR COOLANT SYSTEM PRESSURE
TEMPERATURE LIMITATIONS

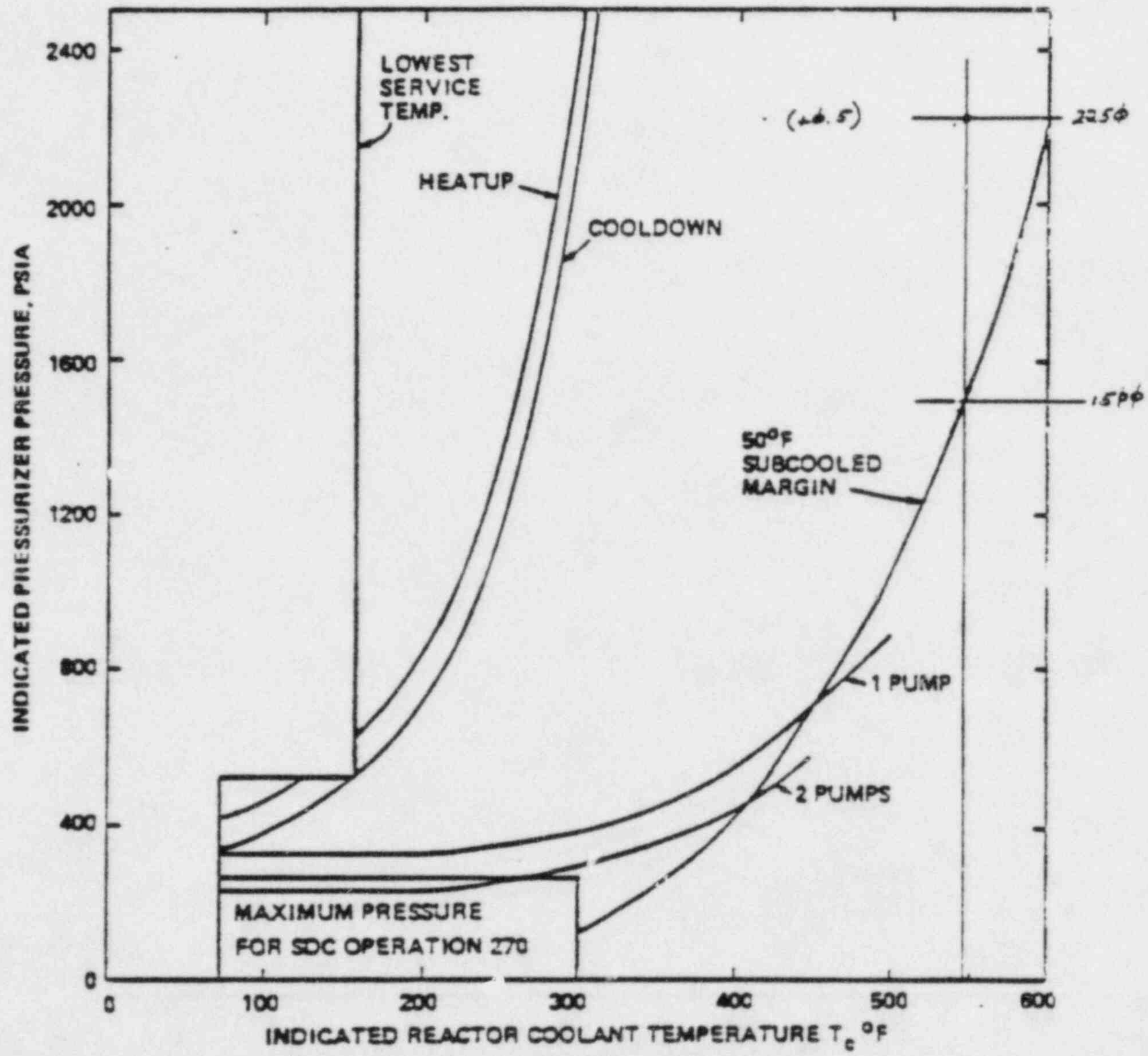



FIGURE 5.02. (ANSWER)

- Section 5 continued on next page -

Points Available

QUESTION 5.03

a. If the control rods (CEAs) were inserted while in the manual sequential (MS) mode until Bank 6 is at 54 inches ~~inserted~~ withdrawn, and if the boron concentration was adjusted to maintain the present power level of 100%, what would be the new concentration of boron required for steady operation? Neglect any effect from changes in the zenon or samarium concentrations. (1.5)

b. How many gallons of boric acid or makeup water would have to be added? (1.0)

c. According to the CEA Insertion Limits criteria, is this operation allowable? If not, what insertion would be allowable? (1.0)

ANSWER 5.03

a. Using the figure on page 11,
$\Delta\rho$ rods = 0.5% $\Delta k/k$. **(+0.5)**

Using the figure on page 17,
0.5% $\Delta k/k$ = 50 $\Delta$ppm-boron.

Or using the data on page 7,
(0.5% $\Delta k/k$) (87 ppm/% $\Delta k/k$) = 43.5 $\Delta$ppm-boron.
**(+0.5 for either $\Delta$ppm procedure)**

Hence, the new boron concentration is 200 - 50 = 150 ppm.
**(+0.5 for 200 - $\Delta$ppm)**

b. Using the equation on page 16,
MWV = 63000 ln 200/150 = 18,124 gal.

Or using page 19, the curve shows $2 \times 10^4$ gal.
**(+1.0 for either result)**

c. No, at 100% power the limit is 80 inches on Group 6. See page 1 of the Technical Data Book. **(+1.0)**

- Section 5 continued on next page -

Points Available

ANSWER 5.03 (contd)

Reference(s)

1. C-E Training Center: Technical Data Book, pp. 1, 11, 16, 17 and 19.

QUESTION 5.04

The "dispatcher" calls and informs you that your power plant is going to have to carry a lagging pf of 0.85 due to changes in the other generators feeding the large (infinite) grid. (The initial plant conditions are stated prior to QUESTION 5.01.)

a. List the adjustments that must be made in the control room for the power plant to meet this new requirement. (1.5)

b. After the adjustments have been made, what will be the real or true output power, the reactive output power and the apparent output power? (1.5)

ANSWER 5.04

a. • increase the generator field current (+0.5)
   • decrease the power level of the plant (consider Fow limit due to $H_2$ PRESSURE)
     - throttle the turbine control valves (+0.5)
     - borate the match $T_{ave}$ to $T_{ref}$ (+0.5)

b. Using the figure on page 5 of the Technical Data Book,
   • 800 MW
   • 500 MVARS
   • 943,000 KVA (+0.5 each)

Reference(s)

1. C-E Training Center: Technical Data Book, p.. 5.
2. C-E Training Center: NSS Program, Turbine Generator, pp. 1, 31 and 32.

- Section 5 continued on next page -

Points Available

QUESTION 5.05

The power plant is to be taken from 100% to 80% of full power. Assume $T_{ave} = T_{ref}$ at both 100% and at 80%.

a. What would be the magnitude of the change in reactivity (in % $\Delta\rho$) due to the change in the temperature in the power plant? Specify whether this change would add or take away reactivity from the core. (1.0)

b. What is the change in reactivity (in % $\Delta\rho$) due to the change in the moderator temperature? Specify magnitude and direction. (1.5)

c. If the core were a fresh core; i.e., zero MWD/T on cycle 6, how would your answer to "b." above change? Explain. (1.5)

d. Assuming that this maneuver takes 20 minutes, make a sketch on the provided graph (Figure 5.05) of the xenon worth as a function of time. Show time from the point in time of decreasing power and for the next 50 hours. (1.5)

ANSWER 5.05

a. Using the EOL curve from the figure on page 21 of the Technical Data Book,

| | |
|---|---|
| 100% | -1.5% |
| 80% | -1.23% |
| | 0.27% has been added. |

**(+0.5 for 0.24% to 0.30% and +0.5 for "added")**

- Section 5 continued on next page -

Points
Available

ANSWER 5.05 (contd)

b. $T_{ave}$ (100%) = 572°F
$T_{ave}$ ( 0%) = 532°F
40°F

$\Delta T_{ave}$ (100% to 80%) = (0.2)(40) = 8°F **(+0.5)**

Using $\alpha_M$ from page 8 of the Technical Data Book,

$$\Delta\rho = (-1.96 \times 10^{-4})\,(-8)$$
$$= 15.7 \times 10^{-4}$$

$\%\Delta\rho = 0.157$ **(+0.5)**

This reactivity is added **(+0.5)** to the core.

c. At BOL (see page 8),

$$\alpha_M = -0.32 \times 10^{-4}$$
$$\%\Delta\rho = 0.0256.$$

Qualitatively, in going from EOL to BOL, the $\alpha_M$ becomes smaller in magnitude, hence the reactivity change due to the moderator-temperature change would be less in magnitude. **(+0.5)**

If the temperature of the moderator were reduced, then the density of the coolant would increase which would increase the amount of boron in the core which in itself would cause a decrease in reactivity. This decrease offsets the increase in reactivity due to the more effective "slowing-down" of the more-dense moderator. Hence, the greater the initial concentration of boron, the smaller in magnitude is the moderator temperature coefficient. There is a higher concentration of boron in the core of BOL. **(+1.0)**

d. See the attached figure.

Reference(s)

1. C-E Training Center: Technical Data Book, p. 8.
2. CE Training Center: "Reactor Theory," pp. 500 (82 J8)/ds-23, Figures 6-34 through 6-37.

- Section 5 continued on next page -

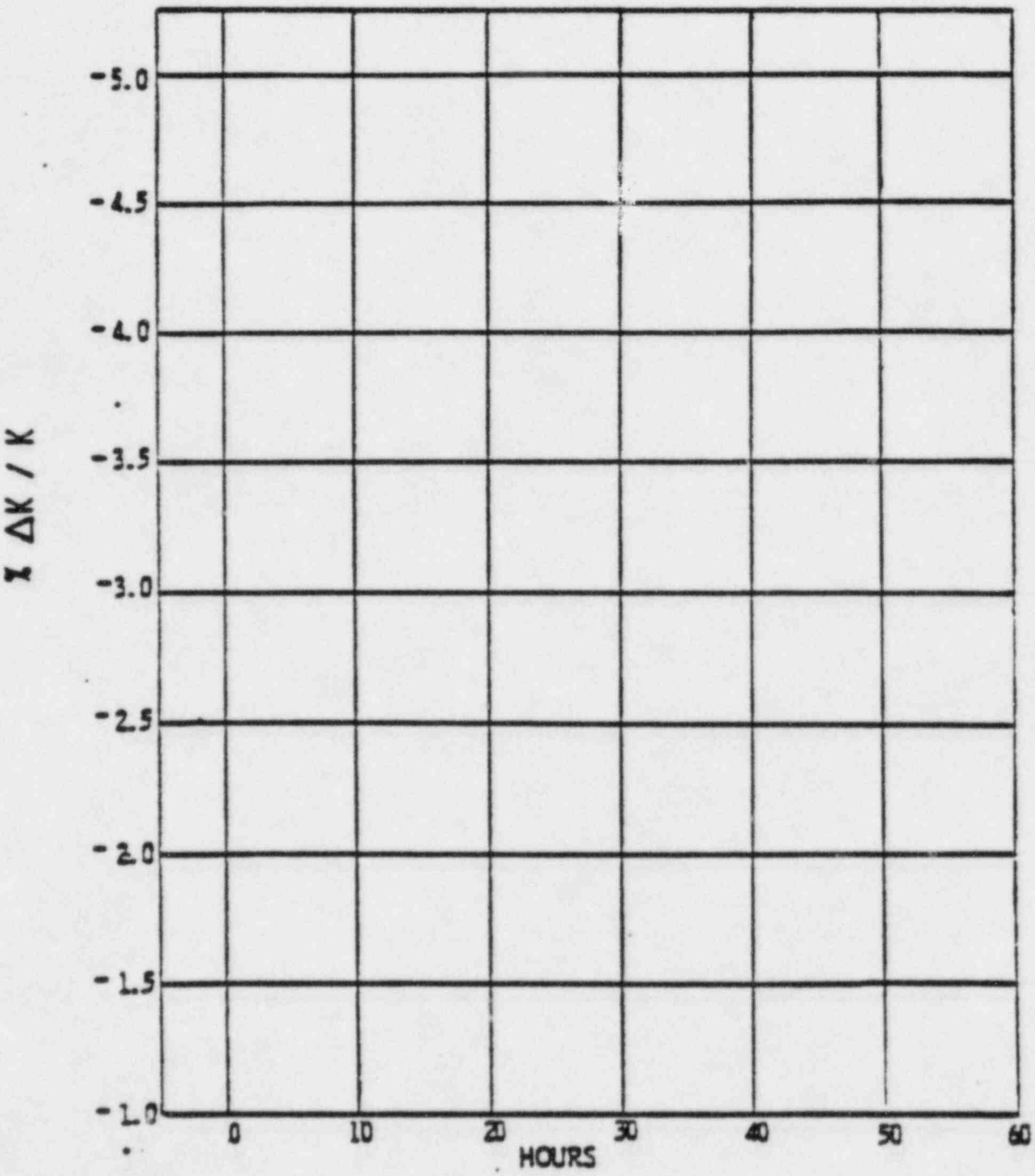
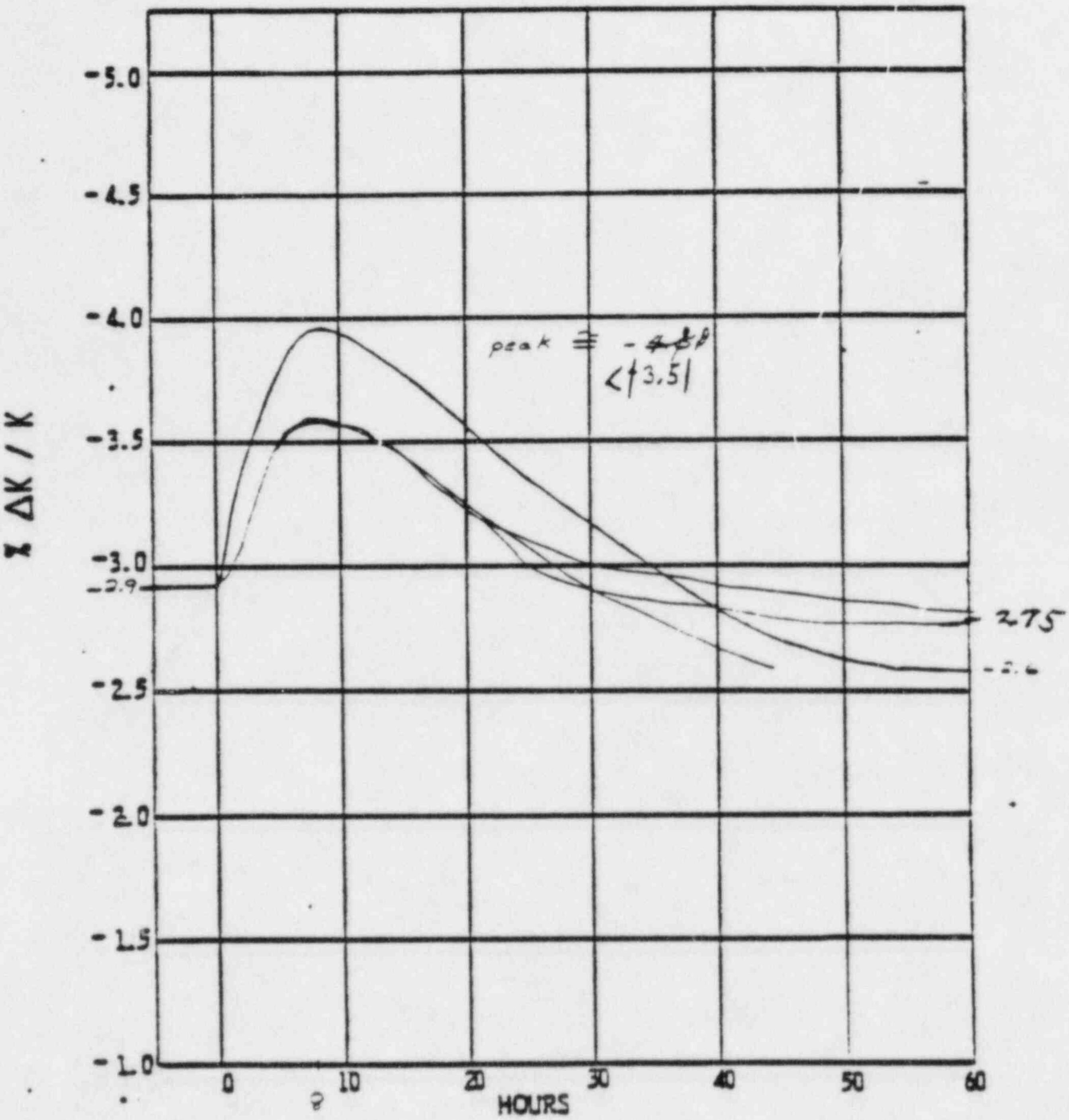



FIGURE 5.05 (QUESTION)

- Section 5 continued on next page -




<u>FIGURE 5.05</u> (ANSWER)

- Section 5 continued on next page -

Points Available

QUESTION 5.06

For QUESTION 5.06, refer to Figure 5.06 which shows a condenser hotwell and condensate pump. Recall that 1 atm = 29.92 in. Hg = 33.9 ft $H_2O$.

Select the letter designation below that most accurately gives the absolute pressure in psia in the condenser.

(a.) 1.93 psia
(b.) 2.31 psia
(c.) 12.77 psia
(d.) 16.91 psia (1.0)

ANSWER 5.06

26 in. Hg vacuum = 29.92 - 26

= 3.92 in. Hg pressure (+0.5)

$$\frac{3.92}{29.92} = \frac{x}{14.7}$$

$x$ = 1.93 psia

The answer is (a.), 1.93 psia. **(+0.5)**

Reference(s)

1. Generic: "Academic Program for Nuclear Power Plant Personnel," Vol. III, "Nuclear Power Plant Technology."
2. St. Lucie 1&2: "Power Plant Thermodynamics," pp. 7-10.

- Section 5 continued on next page -

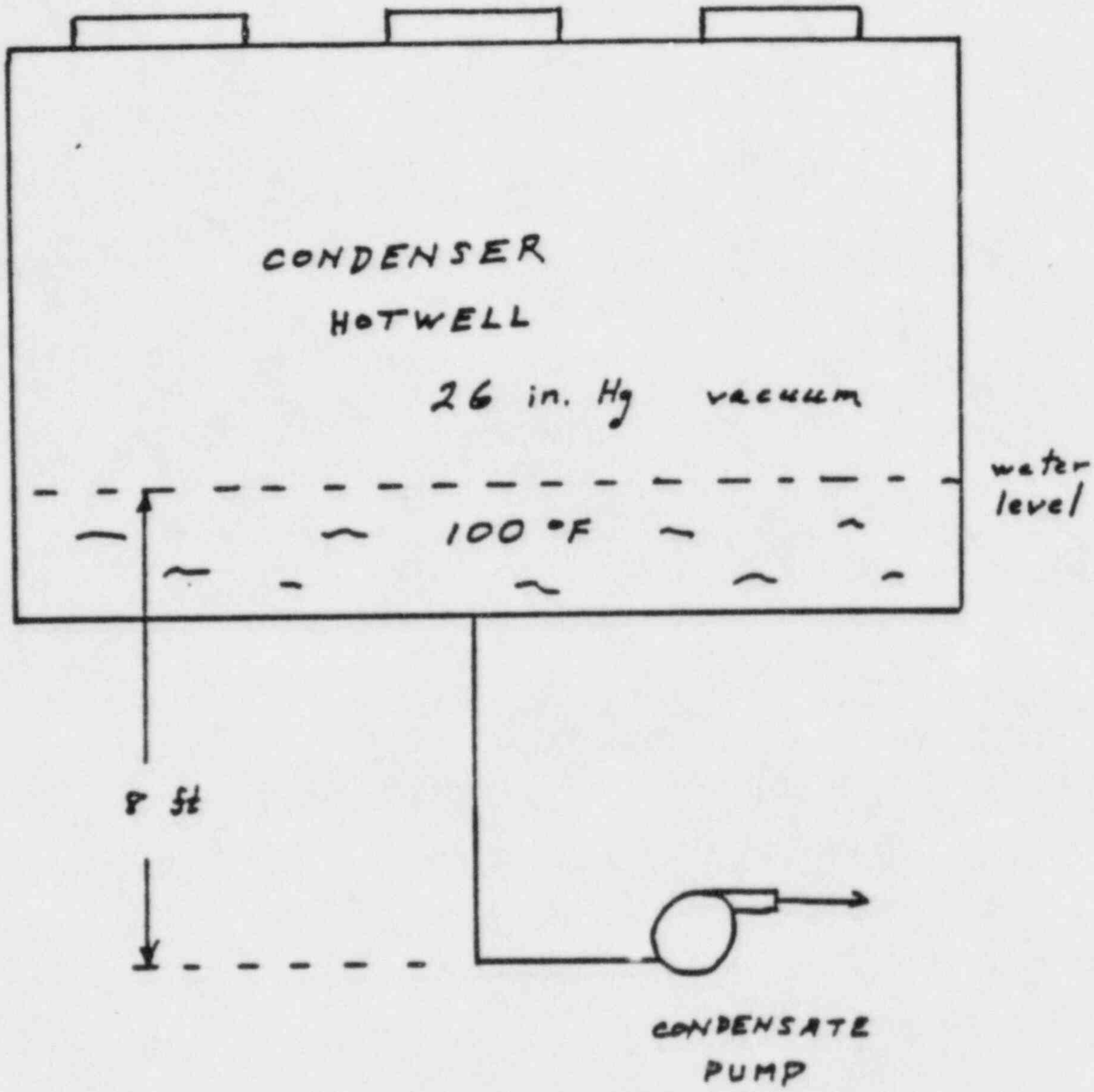



FIGURE 5.06 (QUESTION)

- Section 5 continued on next page -

Points Available

QUESTION 5.07

a. Why is "NPSH" important; i.e., what concern does it address? (1.0)

b. Why is the following statement true: "If the flowrate through a centrifugal pump was increased, then the 'required NPSH' would be increased."? (1.0)

ANSWER 5.07

a. NPSH is important in considering the operation of centrifugal pumps. Sufficient pressure is required at the inlet to the pump in order to prevent the water from flashing to steam in the "eye" of the pump and thereby causing pump cavitation (+1.0). Cavitation causes erratic flowrates and can cause damage to the pump. The damage to the pump comes from the collapse of the vapor bubbles along the impeller vanes (where the pressure is higher than in the "eye") (+0.5).
(+1.0 max)

b. If the flowrate through a centrifugal pump was increased, then the pressure drop from the suction to the "eye" would increase. Hence, to avoid cavitation the suction pressure must be higher (+1.0).

Reference(s)

1. Generic: "Academic Program for Nuclear Power Plant Personnel," Vol. III, "Nuclear Power Plant Technology," pp. 2-232 to 2-236, General Physics Corporation.
2. C-E Training Center: "Thermodynamics Review," p. 20.

- Section 5 continued on next page -

Points Available

QUESTION 5.08

While cooling the power plant on natural circulation, how can the cooldown rate be controlled? How can the cooldown rate be increased? (1.0)

ANSWER 5.08

The cooldown rate can be controlled by controlling the steam and feed flowrate in the Steam Generators (+0.5). If the steam and feed flowrate were increased, then the cooldown rate would be increased (+0.5).

Reference(s)

1. C-E Training Center: Natural Circulation-Loss of Forced Coolant Flow," p. 7.

- End of Section 5 -

## 6.0 PLANT SYSTEM DESIGN, CONTROL AND INSTRUMENTATION (25)

Points Available

### QUESTION 6.01

Figure 6.01 shows part of the Feedwater Control System (FWCS).

Figure 6.01 shows the "FLOW DEMAND" signal being routed to the main feedwater va ~~three (3) "programmers" which feed three (3) "M/As~~." Describe how this FLOW DEMAND signal is generated. (Assume that the master controller is in AUTO.) In particular, list the sensor signals that are used as inputs, the inputs that are manually set, and specify how these inputs are processed and combined. Do not describe test or calibration functions and operations. (3.0)

### ANSWER 6.01

See the attached figure which shows the input signals:

- level (2)
- feedwater flowrate
- steam flowrate
- level setpoint

and how they are processed and combined:

- ~~level signal passed through a lead/lag circuit~~
- ~~flowrates produce a rate of change of Δ flowrate~~
- ~~lead/lag level + flowrate rate + level setpoint summed~~ and ~~passed through a PI unit.~~

- THE LEVEL SIGNAL IS PASSED THROUGH A LEAD/LAG CIRCUIT AND COMPARED TO LEVEL SETPO
- THE FEEDWATER FLOWRATE IS PASSED THROUGH A LEAD/LAG CIRCUIT AND COMPARED TO THE STEAM FLOW RATE
- THESE TWO SIGNALS ARE ADDED AND US TO CONTROL FWR

(+0.5 for each bullet, +3.0 max)

- Section 6 continued on next page -

Points Available

ANSWER 6.01 (contd)

Reference(s)

1. C-E Training Center: "System Description, Feedwater Control System, San Onofre Nuclear Generating Station Units 2 and 3," Description No. 1370-ICE-6425, Revision 00, pp. 6-13 of 85 and 56-60 of 85, Combustion Engineering, Inc., Windsor, CT.

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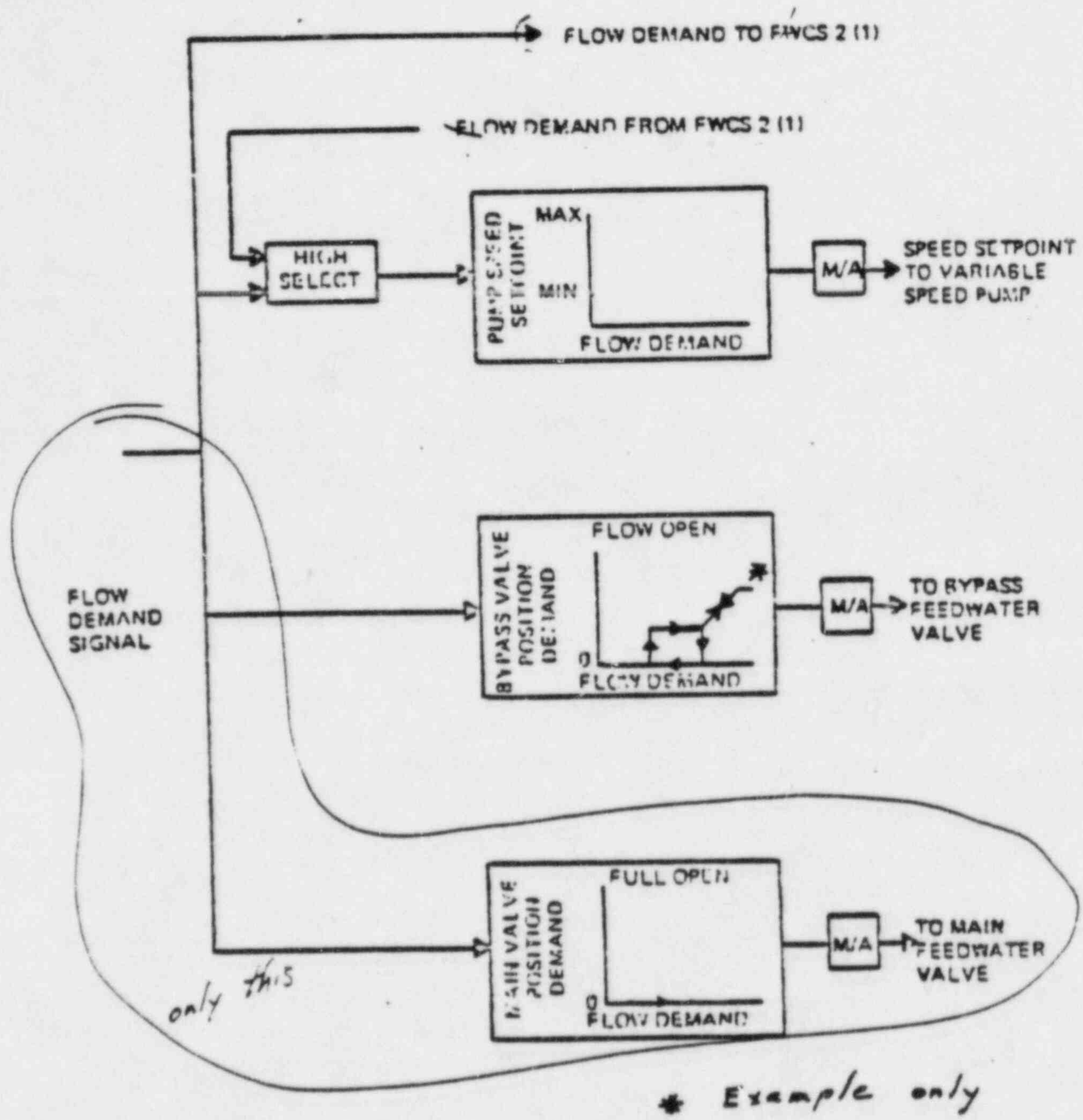
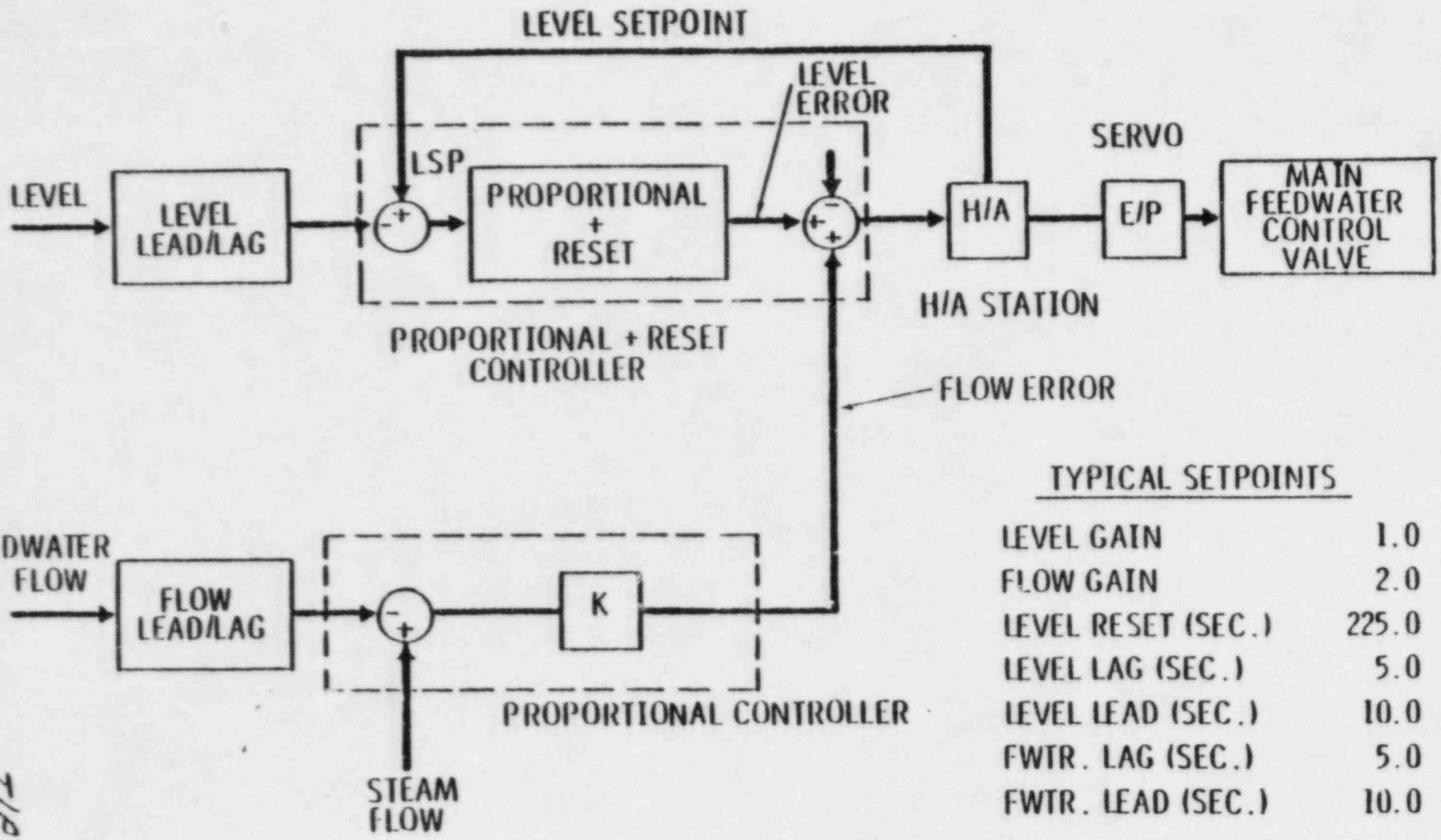



FIGURE 6.01

- Section 6 continued on next page -

Ans. G.01

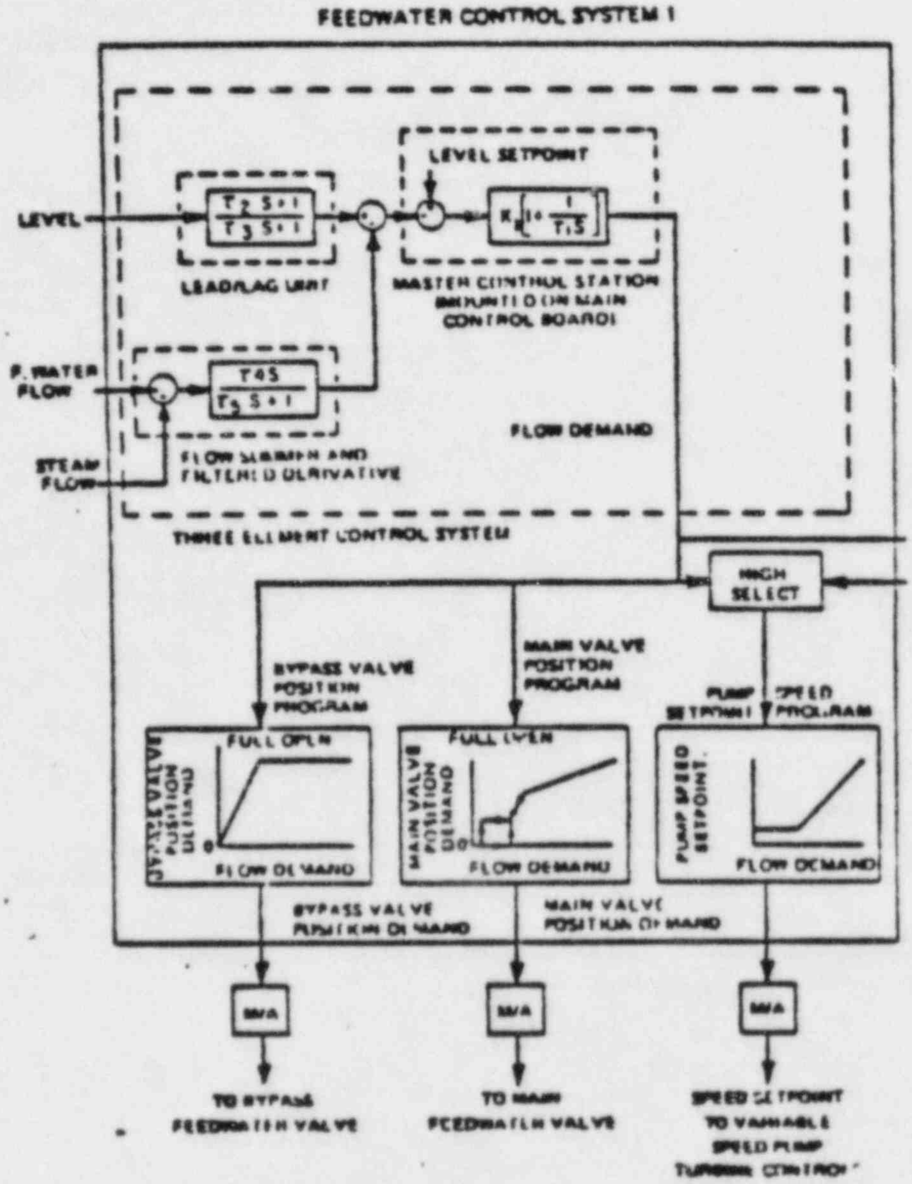
# THREE ELEMENT CONTROL SYSTEM




**TYPICAL SETPOINTS**

| | |
|---|---|
| LEVEL GAIN | 1.0 |
| FLOW GAIN | 2.0 |
| LEVEL RESET (SEC.) | 225.0 |
| LEVEL LAG (SEC.) | 5.0 |
| LEVEL LEAD (SEC.) | 10.0 |
| FWTR. LAG (SEC.) | 5.0 |
| FWTR. LEAD (SEC.) | 10.0 |

T/P #1




ANSWER 6.01 (FIGURE)

- Section 6 continued on next page -

Points Available

QUESTION 6.02

What alternate sets of instrumentation can be used to verify each of the following indications?

a. nuclear excore instrumentation (TWO sets required) (1.0)
b. subcooled-margin meter (ONE set required) (0.5)
c. $T_{avg}$ (TWO sets required) (1.0)

ANSWER 6.02

a. 1. ΔT power
   2. Feed flowrate, feed temperature, SG press (secondary calorimetric)
   3. Incores
   4. $T_H$, $T_C$, primary flowrate (primary calorimetric)
   5. USE OTHER EXCORES
   **(+0.5 each, +1.0 max)**

b. PZR pressure, $T_{\cancel{avg}}$ H **(+0.5)**

c. 1. $T_H$, $T_C$
   2. SG pressure
   3. Incore TCs **(+0.5 each, +1.0 max)**
   4. OTHER RRS CHANNELS

- Section 6 continued on next page -

Points Available

QUESTION 6.03

Sketch the flow paths associated with Safety Injection Tank (SIT) 11A as directed below. Indicate valve types but not valve number designations.

a. The sketch should show the SIT and the flow path(s) to the Loop 11A cold leg, including all valves. (1.0)

b. Include on the sketch the flow path(s) that connect the AUX HPSI, MAIN HPSI and LPSI to the Loop 11A cold leg. Show all valves and flow sensors associated with this portion. (1.0)

c. Include on the sketch the flow path(s) that allow for a measurement of leakage through the loop-entry check valves. Show all valves and flow sensors. (1.0)

d. If the level indicator for the SIT initiates a LO alarm, specify two (2) additional Control Room indications that you would check as verification that the SIT level is indeed low. (1.0)

ANSWER 6.03

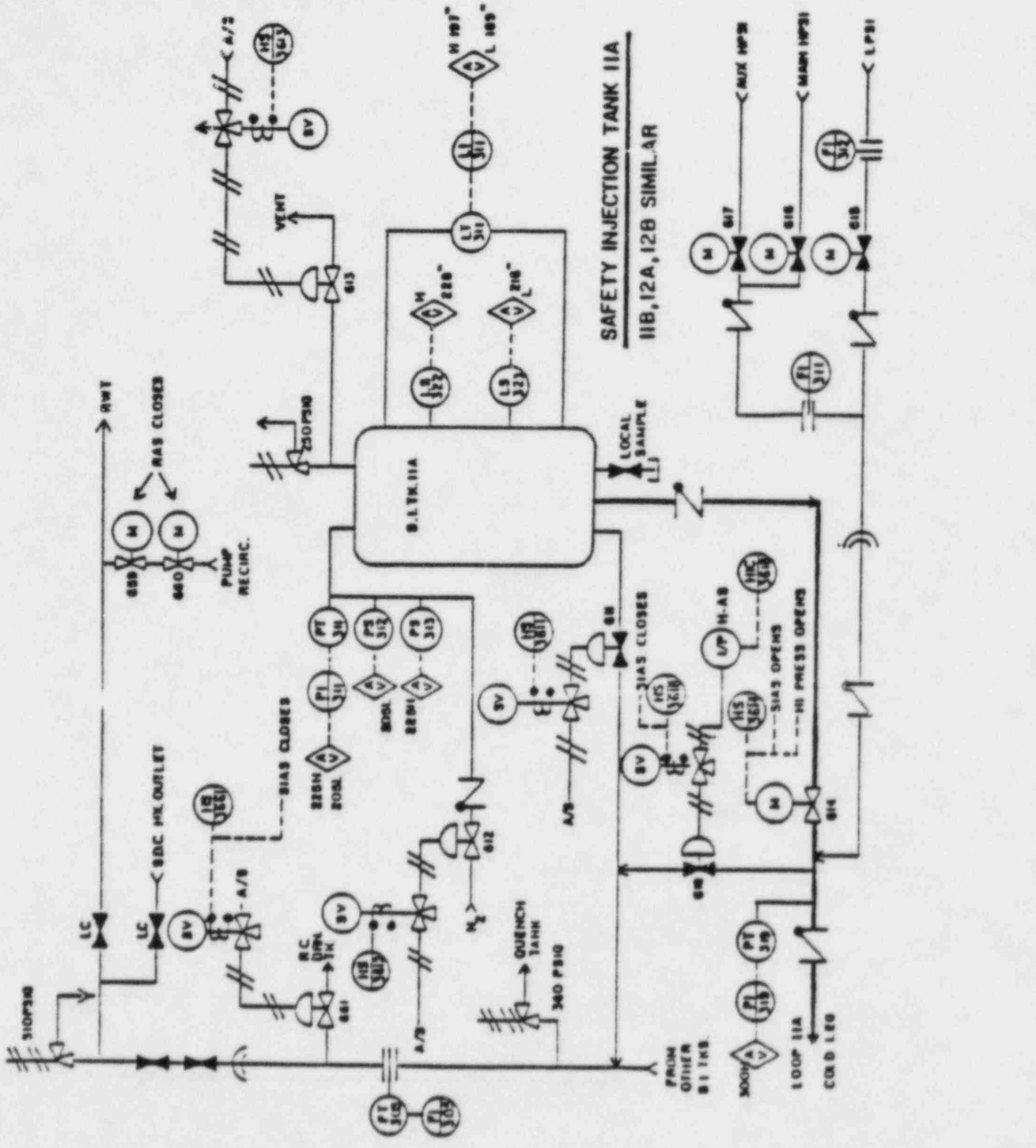
a-c. See the attached sketch which is Figure 1 of the section "Engineered Safety Features." (+1.0 each)

d. Check the redundant LO alarm which is set for 187 inches, the SIT pressure which would alarm LO at 205 psig and verify the positions of valves 611 and 618. (+0.5 each, +1.0 max)

Reference(s)

1. C-E Training Center: NSSS Program, Simulator Training Manual, "Engineered Safety Features," pp. 1-4.

- Section 6 continued on next page -




<u>ANSWER 6.03</u> (FIGURE)

- Section 6 continued on next page -

Points Available

QUESTION 6.04

Answer TRUE or FALSE to each of the following statements concerning the Containment Iodine Removal System.

a. Each filter unit consists of a moisture seperator, a high-efficiency particulate air filter and a fan. (0.5)

b. The filter-unit fan starts on either a CIS or a SIAS. (0.5)

c. With two (2) filter units operating and with the other postulated, conservative assumptions associated with a LOCA, the two (2) hour dose at the site boundary and the total dose at the outer perimeter of the low-population zone are within the limits of 25 rem to the whole body and 300 rem to the thyroid. (0.5)

d. Number 12 and 13 filter units can be lined up to receive electric power from either vital bus through the use of remote-operated disconnects. (0.5)

ANSWER 6.04

a. True
b. False
c. True
d. False **(+0.5 each)**

Reference(s)

1. C-E Training Center: Simulator Training Manual, "Engineered Safety Features, pp. 16-18.

- Section 6 continued on next page -

Points Available

QUESTION 6.05

Specify the letter designation of the most correct statement from those given below. (1.0)

(A.) The purpose of the Auxiliary Feedwater System is to provide feedwater for the removal of heat from the primary coolant after a Steam-Generator tube-rupture event has been diagnosed.

(B.) The auxiliary feedwater pump has a capacity of 5% of rated feedwater flow at rated head.

(C.) At a level of -40 inches or less in a Steam Generator, the steam inlet valve automatically opens providing steam to the Auxiliary Feedwater Pump Turbine and automatically opens the auxiliary feedwater inlet valve.

(D.) During plant cooldown, the preset (for the Auxiliary Feedwater Pump) turbine speed is greater than that required to supply feedwater for decay-heat removal and cooldown and the speed must be lowered by means of the local manual speed control.

ANSWER 6.05

(B.) (+1.0)

Reference(s)

1. C-E Training Center: Simulator Training Manual, "Steam, Feed and Condensate Systems," pp. 44 and 45.

- Section 6 continued on next page -

Points
Available

QUESTION 6.06

Describe the construction and operation of the letdown process radiation monitor. In particular:

a. How and from where is the sample obtained that is to be measured for its radioactive level? (1.0)

b. What type of detectors are used and what type of circuitry is used? What information or output is the result of the design of these detectors and associated circuitry? (1.0)

c. Describe the impact that ABNORMAL flowrates of the Component Cooling Water would have on the operation of the process radiation monitor. Give reasons for any stated impacts. (1.0)

d. How would you use the process radiation monitor information available in the Control Room to determine that you have had a sudden "crud-burst?" (1.0)

ANSWER 6.06

a. A portion of the letdown flow is taken (via a 1/2-inch line) downstream of the letdown flow-control valves and upstream of the flowrate sensor and the purification filters. The pressure drop across the purification filters is used as the driving force for the flow through the process radiation monitor. In the 1/2-inch line is a sample volume which is "viewed" by the detector. (+1.0)

b. The detector is a -scintillation detector whose output feeds a log ratemeter channel and a linear ratemeter channel. The signal is processed to measure gross activity and also I-135 activity. The latter is obtained by the use of a discriminator circuit that is set to count only those pulses that would come from a certain energy -ray associated with I-135. A two-pen recorder for gross and I-~~132~~ 135 activity, selected range and HI alarm are available on IC07. (+1.0)

- Section 6 continued on next page -

Points
Available

ANSWER 6.06 (contd)

c. If the flowrate of the CCW decreases sufficiently that the water temperature of the primary coolant exciting the letdown heat-exchanger equals or exceeds 145°F, then the process radiation monitor would be isolated. This location is designed to protect the process radiation monitor and the boronometer from damage from high temperatures. (+1.0)

d. In order to determine "sudden" versus "gradual," the trending information of the strip-chart recorder would be used (+0.5). A crud-burst would probably not cause an increase in the I-135 indication, but would cause an increase in the gross radiation signal (+0.5).

Reference(s)

1. C-E Training Center: Simulator Training Manual, "Chemical and Volume Control System," pp. 9 and 12.

QUESTION 6.07

If, during reactor plant operations at 95% power, a feedline rupture were to occur inside the containment, what are ALL the Engineering Safety Features (ESFs) that could possibly be actuated and what signals will cause these actuations? Include setpoints and logic. (3.1)

ANSWER 6.07

SIAS (+0.4) and CIAS (+0.4) - High containment pressure (+0.2), 4 psia (+0.2) 2/4 (+0.1).
CSAS (+0.4) - High containment pressure (+0.2) 4 psig (+0.2) 2/4 (+0.1)
SGIS (+0.4) - Low S/G pressure (+0.2) 500 psia (+0.2) 2/4 (0.1).
(3.1 max)

Reference(s)

1. ESF System Description, pp. 22-27.
2. Steam, Feed and Cond. System Description, p. 44.

- Section 6 continued on next page -

Points Available

## QUESTION 6.08

The power plant is inn equilibrium (steady) operation at 100% of full power. RRS Chaannel-Y is selected to calculate $T_{ave}$ from Loop-2. RRS Channel--X is selected to calculate $T_{ave}$ from Loop-1. The Pressurizer level control is selected to Channel-X. Temperature element TE-111X (Loop-1 hot-leg temperature element) fails and reads 150°F low.

a. Explain how and why the charging pumps and letdown valves should respond to the failed TE-111X. (2.0)

b. What actions shoould the operator take to compensate for the failed temperrature indication? (1.0)

## ANSWER 6.08

a. The controlling $T_{ave}$ will drop (+0.25) that will cause the RRS to produce a minimum Pressurizer level setpoint (+0.75) that will cause minimum charging (+0.5) and maximum letdown (+0.5).

b. Change RRS Channel-X to calculate $T_{ave}$ from Loop-2 (+1.0) or change Pressurizer level control to Channel-Y (+0.5). (+1.0 max) OR SELECT OTHER RRS ~~AND~~ CHANNEL AFTER TAKING LOCAL CONTROL OF THE PZR-LEVEL SET POINT

Reference(s)

1. Millstone II: Book 9, System Description #5, "RRS."
2. Millstone II: Book 9, System Description #6, "Pressurizer Control," System 5, Figure 1.5, System 6, pp. 1-3.
3. C-E Training Center: NSSS Program, "Reactor Regulating System."
4. C-E Training Center: NSSS Program, "Pressurizer Level and Pressure Control Systems," pp. 6-14.

- Section 6 continued on next page -

Points Available

QUESTION 6.09

Will the plant trip as a result of the following situations. Explain your answers. Consider each situation separately.

a. 120 volt vital bus A is de-energized and channel B pressurizer pressure indication fails high. (0.8)

b. The Flow Dependent Setpoint Selector Switch is placed in the 3 pump position while at 70% power and Axial Shape Index is -0.3. (0.8)

c. SG 1 pressure channel A fails to 400 psia and SG 2 pressure channel B vails to 450 psia while at 60% power. (0.8)

ANSWER 6.09

a. Yes (+0.3) de-energizing channels provide a trip signal (+0.5).

b. No (+0.3) flow, power, TM/LP, and APD values are all less than the reduced setpoints (+0.5).

c. Yes (+0.3) each SG pressure channel auctioneers lowest pressure from the SGs (+0.5).

Reference(s)

1. SD: RPS, pp. 5, 13, 14-16, 36-37.

- End of Section 6 -

## 7.0 PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL (25)

Points Available

### QUESTION 7.01

Assume that the scenario described below has occurred for each of the four (4) conditions (a, b, c, and d). Designate the most probable cause by choosing from the list of possible causes. Consider each of the four conditions separately.

Scenario: The power plant has been operating at 100% of full power for two (2) months. All four (4) RCPs trip, and after 15 minutes your operators inform you that natural circulation flow has NOT developed.

a. The wide-range levels for both Steam Generators are indicating less than 0% and $T_C$ is increasing above the $T_{sat}$ for the Steam Generators. (0.75)

b. Both $T_C$ and the pressures in the Steam Generators are increasing. (0.75)

c. The RCS loop $\Delta T$ is increasing, and the Pressurizer level is erratic. (0.75)

d. The Pressurizer level is low, and there is a mismatch between the loop RTDs and the core exit thermocouples. (0.75)

Possible causes:

(1.) Condensible voids developing in RCS flow path.

(2.) Failure of both Main Feedwater Pumps.

(3.) The Auxiliary Feedwater Pumps are operating at maximum flowrate.

(4.) Inadequate secondary steam flowrate.

- Section 7 continued on next page -

Points
Available

QUESTION 7.01 (contd)

(5.) Pressurizer relief valves stuck closed.

(6.) Inadequate RCS inventory.

(7.) More than one CEA stuck in the out position.

(8.) Steam-line rupture on Steam Generator A.

ANSWER 7.01

a. (2.)
b. (4.)
c. (1.)
d. (6.) ^ (8.) (+0.75 each)

Reference(s)

1. C-E Training Center: AOP-7.

- Section 7 continued on next page -

e Pressurizer temperature (0.5)
e Pressurizer level (0.5)
(0.5)
7.02 (0.5)

~~s same~~ INCREASE
~~lee~~ INCREASE
le
eases (+0.5 each)

ence(s)

-E Training Center: SOP-C5, g. 3, Rev. 1, p. 4.

- Section 7 continued on next page -

Points Available

QUESTION 7.02

At the moment a bubble is formed in the Pressurizer, how should each of the following parameters respond?

a. the letdown flowrate (0.5)
b. the Pressurizer pressure (0.5)
c. the Pressurizer temperature (0.5)
d. the Pressurizer level (0.5)

ANSWER 7.02

a. ~~stays same.~~ INCREASE
b. ~~stable.~~ INCREASE
c. stable
d. decreases **(+0.5 each)**

Reference(s)

1. C-E Training Center: SOP-C5, g. 3, Rev. 1, p. 4.

- Section 7 continued on next page -

Points
Available

QUESTION 7.03

Specify the letter designation of the most correct phrase with which to complete the following sentence concerning the Pressurizer Pressure Control System.

To increase the heat supplied by the proportional heaters with the manual/automatic station for the Pressurizer heaters and spray in the Manual Mode, you must . . . (0.65)

(A.) increase the controller output.
(B.) adjust the pressure setpoint higher.
(C.) decrease the controller output.
(D.) adjust the pressure setpoint lower.

ANSWER 7.03

(C.) **(+0.65)**

Reference(s)

1. C-E Training Center: SOP-C6, a.(4), p. 2.

- Section 7 continued on next page -

Points Available

QUESTION 7.04

In accordance with EOP 7A, Reactivity Control, what are the THREE (3) reactivity control success paths and WHEN would each be used? (3.0)

ANSWER 7.04

1. CEA insertion (+0.7) used when greater than 2 rod bottom lights not lighted (+0.2) and rods indicate out on metrascope (+0.1).

2. Borate via CVCS (+0.7) when greater than 2 CEA not inserted reactor power >1% or increasing (+0.1) RCS pressure >1250 psia (+0.2).

3. Borate via HPSI (+0.7) when greater than 2 CEA not inserted reactor power >1% or decreasing (+0.1) and RCS pressure <1250 psia (+0.2).

4. CEA DRIVE DOWN IF PATH 3 FAILS [1.0]

Reference(s)

1. C-E Training Center: EOP 7A, Figure 5.5.

- Section 7 continued on next page -

Points Available

QUESTION 7.05

In preparation for reactor plant heat up, RCP 11-A has been started. If both RCP 11-B and RCP 12-A had tripped on their initial start, what should be the pump starting sequence to establish three running pumps in the shortest time if all subsequent RCP starts are successful? Explain your reasoning. (2.0)

ANSWER 7.05

1. Start RCP 12-A then RCP 12-B. **(+0.6)**

2. A cold RCP can be started three consecutive times if no other pump is operating in the loop. **(+0.7)**

3. An RCP must be idled for 60 minutes between starts if there is a pump operating in the loop. **(+0.7)**

**(+2.0 max)**

Reference(s)

1. C-E Training Center: SOP-C3, p. 2.

Points Available

QUESTION 7.06

In each of the following situations, specify the conditions which would require that the power plant be tripped.

a. decreasing condenser vacuum, while at 80% power (0.7)

b. loss of an operating Component Cooling Water pump while at 50% power (1.1)

c. a reactor coolant system leak which is slowly increasing while at 50% power (0.7)

ANSWER 7.06

a. low Vacuum Alarm (20" Hg) **(+0.7)**

b. loss exceeding 10 min **(+0.5)** or RCP seal high temperature alarm (250°F) **(+0.3)** or RCP-bearing high temperature alarm (195°F) **(+0.3)**

c. leak greater than charging pump capacity (132 gpm) **(+0.7)**

Reference(s)

1. C-E Training Center: AOP 1, p. 1.
2. C-E Training Center: AOP 8, p. 2.
3. C-E Training Center: AOP 3, p. 2.

- Section 7 continued on next page -

Points Available

QUESTION 7.07

Following a reactor trip, what FOUR actions must be taken besides verifying proper automatic functions if all systems operate normally? (2.5)

ANSWER 7.07

1. Depress both reactor trip pushbuttons. **(+0.7)**
2. Manually trip turbine. **(+0.7)**
3. Open generator exciter breaker. **(+0.6)**
4. Announce Reactor Trip over the public address system. **(+0.5)**

Reference(s)

1. EOP 1, pp. 2-7.

QUESTION 7.08

In general, what actions need to be taken if Component Cooling Water is lost because of a high-level in the head tank? (1.5)

ANSWER 7.08

Locate and isolate leak from an RCS component. **(+1.5)**
OR VERIFY THE STATUS OF THE VALVES TO THE HEAD TANK AND RESTORE CCW. CHECK FOR INDICATIONS OF RCS LEAKAGE INTO CCW. (+1.5)

Reference(s)

1. AOP-3, p. 1.

- Section 7 continued on next page -

Points Available

QUESTION 7.09

Explain why each of the following events would or would not require suspension of movement of irradiated fuel inside the containment.

a. A maintenance person opens the inner door of the air lock to exit the containment. The outer door of the air lock is shut. (0.75)

b. Securing of the operating shutdown cooling loop which leaves both loops secured but operable. (0.75)

c. The damper of a spent fuel ventilation exhaust fan fails shut due to a loss of instrument air to the damper. (0.75)

ANSWER 7.09

a. Would not (+0.35) only one door of the air lock is required to be closed (+0.4). (+0.75)

b. Would not (+0.35) 1 hr without SDC is allowed by Technical Specifications (+0.4). (+0.75)

c. Would not (+0.35) one exhaust fan would still be operable and that is all that is required (+0.4). (+0.75)

Reference(s)

1. T.S., 3/4 9-12, 3/4 9-13
2. SD: Containment System, p. 30.

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Points Available

QUESTION 7.10

Under what conditions would a formal ALARA review be required? (1.0)

ANSWER 7.10

All jobs and tasks that would involve radiation exposure greater than 1 person-rem. **(+1.0)**

Reference(s)

1. C-E Training Center: Radiation Protection Training Manual, p. ds-35.

QUESTION 7.11

Indicate by title the person who:

a. will act as the Manager of Control Room Operations during an emergency (0.6)

b. is responsibile for the assessment, classification, and declaration of emergencies (0.6)

c. initially assumes the responsibilities of the Director of Station Emergency Operations. (0.6)

ANSWER 7.11

a. shift supervisor
b. shift supervisor or duty officer
c. shift supervisor **(+0.6 each)**

Reference(s)

1. C-E Training Center: Emergency Plan, pp. se-3 and se-26.

- Section 7 continued on next page -

Points Available

QUESTION 7.12

Assume that a Steam Generator tube rupture has been verified at a leak rate of 2 gpm. Emergency Operating Procedure EOP-4 cautions the operator to reduce the RCS $T_h$ to less than 525°F before isolating the affected Steam Generator. What is the reason for this caution? (1.0)

ANSWER 7.12

To minimize the possible lifting of the Steam Generator safety valves. **(+1.0)**

Reference(s)

1. C-E Training Center: EOP-4, p. 3.

QUESTION 7.13

Following a Steam Generator tube rupture, the operator is instructed to control the RCS pressure, maintaining it below 1000 psia. What three (3) systems are available to the operator to effect this control of the pressure? (1.8)

ANSWER 7.13

- Main spray depressurization
- Auxiliary spray depressurization
- Throttling of the HPSI pumps **(+0.6 each)**

Reference(s)

1. C-E Training Center: EOP-4, p. 8.

**End of Section 7**

## 8.0 ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (25)

Points Available

### QUESTION 8.01

For each of the following situations indicate what REQUIREMENT, if any, applies and what ACTION, if any, should be taken. Consider each situation separately.

a. Diesel Generator A's operability load test, which is required every 31 days, is scheduled for today. The last three tests were completed 36, 68, and 102 days ago, respectively. The plant is at 100% power. (1.5)

b. The plant is at 295°F and heating up at 1°F per minute, when an HPSI pump is found to be inoperable. (1.5)

c. The plant is at 100% power when it is determined that the heat tracing circuits for both boric acid storage tanks are inoperable and cannot be repaired for 4 days. (1.5)

### ANSWER 8.01

a. Each test should be conducted within 25% of the required time (+0.35) and each three consecutive time intervals should be within 3.25 of the required time interval (+0.4).

Declare DG A inoperable. (+0.25) Prove operability of DG B within 1 hr. (+0.3) Conduct load test on DG A. (+0.2)

b. Two HPSI pumps are required to be operable. (+1.0) ~~(+0.75)~~ Ensure operable pumps off separate power supplies. ~~(+0.75)~~ (+0.5)

c. Unable to comply with LCO of Action Statement. (+0.75) (General Statement T.S. 3.03.)

Start shutdown within 1 hour. (+0.25) Hot standby within next 6 hr. (+0.25) Hot Shutdown within following 6 hours. (+0.25)
LOC SHUT DOWN WITHIN 24 hr (+0.25)
(1.5 MAX)

Reference(s)

1. C-E Training Center: Safety Technical Specifications, pp. 3/4 0-1, 3/4 0-2, 3/4 5-3, 3/4 1-16, 3/4 8-1.

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Points Available

QUESTION 8.02

a. Complete the following table to indicate the minimum shift crew composition in the applicable modes. (2.0)

MINIMUM SHIFT CREW COMPOSITION #

| LICENSE CATEGORY | APPLICABLE MODES | |
|---|---|---|
| | 1, 2, 3, & 4 | 5 & 6 |
| SOL | | |
| OL | | |
| Non-Licensed | | |
| Shift Technical Advisor | | |

b. If the minimum shift crew composition of Question 8.02 above cannot be met, what is the maximum time allowable to restore the shift crew composition to within the minimum requirements? (1.0)

- Section 8 continued on next page -

Points
Available

ANSWER 8.02

a. MINIMUM SHIFT CREW COMPOSITION #

| LICENSE CATEGORY | APPLICABLE MODES 1, 2, 3, & 4 | APPLICABLE MODES 5 & 6 |
|---|---|---|
| SOL | 2 | 1* |
| OL | 2 | 1 |
| Non-Licensed | 3 | 3 |
| Shift Technical Advisor | 1 | 0 |

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS during fuel reloading.

(+0.25 each)

b. Number of shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours ~~(+0.8)~~ (+1.0) in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew ~~(+0.2)~~ composition to within the minimum requirements of Table 6.2.1.

Reference(s)

1. C-E Training Center: Technical Specifications, Table 6.2.1, p. 6-3.

- Section 8 continued on next page -

Points Available

QUESTION 8.03

a. If a bypass device is to be used on a system and it is determined that using the bypass device WILL cause adverse environmental impact, what form of approval is required to use the bypass device? (1.0)

b. Can safety tags be lifted for any reason other than clearing the tags? EXPLAIN. (1.0)

ANSWER 8.03

a. POSRC approved procedure. (+1.0)

b. Yes (+0.5), to allow testing when it is expected the tags may be re-hung (+0.5). (+1.0)

Reference(s)

1. AP-6, pp. 9-10.

- Section 8 continued on next page -

Points Available

QUESTION 8.04

Classify the following conditions according to the "Emergency Plan" in EOP-9. Consider each part of the question, each event, as separate and unrelated to the other events. Specify each as Unclassified, Unusual Event, Alert, Site Area Emergency or General Emergency. (3.0)

a. Power: 100%
   All Ts: ($T_c$, $T_h$, $T_{ave}$, $T_{feed}$, $T_{steam}$, ...): normal
   Pzr level: normal
   Pzr pressure: normal
   Letdown process monitor: alarms
   Chemistry analysis of primary coolant: 256 $\mu$Ci/gm $I^{131}$
   Electrical: normal

b. Power: 100%
   All Ts: normal
   Pzr level: -5% and decreasing
   Pzr pressure: 2200 psia and decreasing
   Containment pressure: 1 psig
   Containment radiation monitors: $10^4$ mR/hr
   Electrical: normal

c. Power: 100%
   All Ts: normal
   Pzr level: -2% and increasing
   Pzr pressure: 2220 and decreasing
   Blowdown process monitors: increasing
   Condenser air ejector monitors: $5 \times 10^{-3}$ Ci/cc $I^{131}$
   Electrical: 6.9 kV and 4.16 kV buses are lost

ANSWER 8.04

a. Unusual Event (+1.0) fuel element failure
b. Site Area Emergency (+1.0) LOCA
c. Alert (+1.0) S/G TR

Reference(s)

1. C-E Training Center: EOP-9, Table 1.

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Points Available

QUESTION 8.05

The Tech-Specs specify a Limiting Condition for Operation (LCO) with respect to the Auxiliary Feedwater system.

a. What is the LCO for Modes 1-3? (1.5)

b. For what does the OPERABILITY of the Auxiliary Feedwater System provide assurance? (1.0)

c. With one (1) auxiliary feedwater pump inoperable, what action is required? (1.0)

ANSWER 8.05

a. 3.7.1.2. At least two (2) steam turbine-driven steam generator auxiliary feedwater pumps and associated flow paths shall be operable (+1.0) and capable of automatically initiating flow, within the limits of acceptable operation to each steam generator (+0.5).

b. The operability of the auxiliary feedwater system ensures that the reactor coolant system can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. (+1.0)

c. With one (1) auxiliary feedwater pump inoperable, restore at least two (2) auxiliary feedwater pumps to operable status within 72 hours or be in hot shutdown within the next 12 hours. (+1.0)

Reference(s)

1. C-E Training Center: Technical Specifications, "Auxiliary Feedwater System," 3.7.1.2, pp. 3/4 7-4.
2. C-E Training Center: Technical Specifications, "Basis," 3/4.7.1.2, pp. B 3/4 7-2.

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Points Available

QUESTION 8.06

List the letter designations of those statements chosen from the following statements which are correct. The statements are in response to, "Temporary changes to procedures may be made provided:" (2.0)

(a.) Critical operation of the unit shall not be resumed until authorized by the Commission (U.S. NRC).

(b.) The intent of the original procedure is not altered.

(c.) The change is approved by two (2) members of the plant management staff, at least one of whom holds a SRO License in the affected unit.

(d.) The change is documented, reviewed by the POSRC and approved by the Plant Manager within 21 days of implementation.

ANSWER 8.06

(b.) and (c.) - correct
(a.) and (d.) - incorrect **(+0.5 each)**

Reference(s)

1. C-E Training Center: AP-10, Rev. 0, p. 6 of 7, Appendix 10-3.

- Section 8 continued on next page -

Points
Available

## QUESTION 8.07

What permission (from whom and at what check points) does a radiation worker need to complete a task which is expected to increase his exposure by 2000 mrem this quarter? The worker is 30 years old, has a completed NRC Form 4 and has a radiation history of 13000 mrem lifetime, 3000 mrem for the year, and 600 mrem for the quarter. (1.5)

## ANSWER 8.07

HP Supervisor's approval for exceeding 1000 mr/qt. (+0.5)
HP Department Head approval for exceeding 2000 mr/qt. (+0.5)
HP Supervisor and Station Superintendent approval for exceeding 5000 mr/yr. (+0.5)

Reference(s)

1. Radiation Science, p. ds-16.

- Section 8 continued on next page -

Points Available

QUESTION 8.08

Technical Specification 3/4.1.1.5 states the lowest loop operating temperature for the RCS $T_{ave}$ shall be ≥515°F when the reactor is critical.

a. Explain what four (4) things this specification ensures. (2.0)

b. How often must this be determined when $T_{ave}$ is less than 525°F with the reactor critical? (0.5)

ANSWER 8.08

a. 3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range (+0.5), (2) the protective instrumentation is within its normal operating range (+0.5), (3) the Pressurizer is capable of being in an OPERABLE status with steam bubble (+0.5), and (4) the reactor pressure vessel is above its minimum $RT_{NDT}$ temperature (+0.5).

b. At least once per 30 minutes. **(+0.5)**

Reference(s)

1. C-E Training Center: Technical Specifications, "Basis," 3/4.1.1.5, p. B3/4 1-2.
2. C-E Training Center: Technical Specifications, 3/4.1.1.5, Minimum Temperature for Criticality, p. 3/4 1-7.

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Points Available

QUESTION 8.09

What is required of personnel before they can be designated as "escorts" in Radiation Work Areas? (1.0)

ANSWER 8.09

An escort must have been trained as a radiation worker. (+1.0)

Reference(s)

1. C-E Training Center: Radiation Protection Training Manual, p. ds-14.

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Points Available

QUESTION 8.10

When a shift supervisor places his/her signature on an RWP, he/she is verifying that certain conditions have and will exist, and that certain commitments will be kept. List two (2) of these conditions or commitments. (2.0)

ANSWER 8.10

- No plant evolutions are planned which could change the radiological conditions stated in the RWP.
- Operations (Supervisor) will notify HP whenever any plant evolution has taken place or is to take place that would change the radiological conditions in the area listed in the RWP.
- The plant is not and would not be jeopardized by the work indicated on the RWP.

(+1.0 each bullet, +2.0 max)

Reference(s)

1. C-E Training Center: Radiation Protection Training Manual, p. ds-43.

- End of Section 8 -

END OF EXAMINATION