ENCLOSURE 1

EXAMINATION REPORT - 50-321/OL-88-01

Facility Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Facility Name: Edwin I. Hatch Nuclear Plant

Facility Docket No.: 50-321 and 50-366

Written examinations and operating tests were administered at Edwin I. Hatch Nuclear Plant near Baxley, Georgia.

Chief Examiner: K. E. Brockman, Chief, Operator Licensing Section 2 — 28 MAR 88, Date Signed

Approved by: for J. F. Munro, Chief, Operator Licensing Section 1 — 29 MAR 88, Date Signed

Summary:

Examinations were administered on February 8-11, 1988.

Written and operating examinations were administered to eleven Senior Reactor Operator (SRO) and three Reactor Operator (RO) candidates. All SROs and all ROs passed the written examination. All SROs and all ROs passed the operating examination.

Based on the results described above, eleven of eleven SROs and three of three ROs passed the overall examination.

Of the nine technical corrections for the examinations, four (44%) were due to inaccurate/incomplete materials provided to the Commission for examination preparation. The facility is encouraged to ensure the accuracy and completeness of facility reference material, especially in light of the numerous changes begin made within the plant procedures.

Section 8 was administered in an "open book" format to 11 SRO candidates. Overall, this section of the exam was well received. Eleven of eleven SROs passed the section. However, the time required to complete this section was much longer than anticipated from the time validation conducted by the Regional staff. The candidates were not adversely penalized as a result of this problem since additional time was allotted by the NRC proctor who closely monitored the pace of the section. A qualitative evaluation will be made of this testing format for lessons learned and the desirability for use in future examinations.

REPORT DETAILS

1. Facility Employees Contacted:

   *E. Morris Howard, Manager, Nuclear Training EP & Security
   *Harvey Nix, Plant Manager
   *Lewis Sumner, Manager of Operations
   *Curtis Coggins, Training & EP Manager
   *Charles T. Moore, General Manager, Quality Assurance
   *O. M. Fraser, QA Site Manager
   *Steve Grantham, Operator Training Manager
   *S. J. Bethay, Acting NSAC Manager
   *Dan F. Moore, Nuclear Training Coordinator
   *S. M. Crosby, Operations Training Coordinator
   *C. L. Tully, Senior Nuclear Engineer
   *T. H. Hunt, Senior Simulator Instructor

   *Attended Exit Meeting

2. Examiners:

   *K. E. Brockman, Region II
   D. C. Payne, Region II
   G. T. Hopper, Region II
   J. M. McGhee, EG&G
   J. F. Hanek, EG&G
   T. L. Morgan, EG&G
   M. A. King, EG&G

   (P. Holmes-Ray, Senior Resident Inspector, attended Exit Meeting)

   *Chief Examiner

3. Examination Review Meeting

   At the conclusion of the written examinations, the examiners provided your training staff with a copy of the written examination and answer key for review. The NRC Resolutions to comments made by the facility reviewers are listed below.

   a. SRO Exam

      (1) Question 6.06a

      Comment accepted. This portion of the question has been deleted from the exam. Section and total points have been adjusted accordingly.

      (2) Question 6.11b

      Comment accepted. Due to the recently installed modification to the plant, this portion of the question has been deleted from the exam. The utility is encouraged to ensure that training material is complete and accurate. Section and total points have been adjusted accordingly.

(3) Question 7.06b

Comment accepted. Scanning the top of the core with the Fuel Grapple lowered to just above the fuel bundles will be added to the key as an acceptable answer. However, both answers are required for full credit. The utility is encouraged to ensure that material is complete and accurate. The point value remains unchanged.

(4) Question 8.03c

Comment accepted. The answer key has been revised to accept One Hour Report per 40AC-REG-002-0S due to the recent amendment to Technical Specifications deleting this requirement. The utility is encouraged to ensure that training material is complete and accurate. The point value remains unchanged.

(5) Question 8.05

Comment not accepted. It is acknowledged that a failure of the TIP valve is an isolation valve failure. However, the LCO for Primary Containment Integrity still requires all automatic valves to be operable with specific exceptions. These exceptions apply to both isolation valve failure and Primary Containment Integrity. Since the question stated that the TIP valve was only closed (but not deactivated), Primary Containment Integrity is NOT satisfied. For Unit 1, due to definition 1.0.T.3, the answer key has been expanded to allow deactivation of the valve as an acceptable answer. The point value remains unchanged.

(6) Question 8.07

Comment partially accepted. The answer key has been modified to reflect a response regarding operability for each part. For Part 1, no additional change was made since the question did not ask for the conservative response to the situation. For Part 3, continued operation is not allowed since, TS 3.0.3 applies. The point value remains unchanged.

(7) Question 8.13

Comment accepted. The question has been deleted from the examination. Section and total points have been adjusted accordingly.

(8) Section 8 - General

Comments acknowledged. A qualitative evaluation of the Section 8 open book format is being made. These comments will be duly assessed and incorporated, as appropriate, in lessons learned.

b. RO Exam

(1) Question 3.04c

Comment accepted. This part of the question has been deleted. Section and total points have been adjusted accordingly.

(2) Question 3.09d

Comment accepted. This part of the question has been deleted. Section and total points have been adjusted accordingly.

4. Exit Meeting

At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the examination.

There were no generic weaknesses noted during the operating examination.

Simulator weaknesses encountered during the operating examinations are noted in Enclosure 4.

The cooperation given to the examiners and the effort to ensure an atmosphere in the Control Room conducive to oral examinations was noted and appreciated.

The licensee did not identify as proprietary any of the material provided to or reviewed by the examiners.

MASTER COPY

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: HATCH 1&2

REACTOR TYPE: BWR-GE4

DATE ADMINSTERED: 88/02/08

EXAMINER: MCGHEE, J.

CANDIDATE ______________________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question 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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
26.50	25.35 ~~25.12~~	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
26.25	25.11 ~~24.88~~	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.75 ~~26.75~~	24.64 ~~25.36~~	________	________	3. INSTRUMENTS AND CONTROLS
26.00	24.88 ~~24.64~~	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
104.5 ~~105.5~~		________ Final Grade	________%	Totals

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

____
Candidate's Signature

MASTER COPY

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil only to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of each section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a new page, write only on one side of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least three lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility literature.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the examiner only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

   a. Assemble your examination as follows:

      (1) Exam questions on top.

      (2) Exam aids - figures, tables, etc.

      (3) Answer pages including figures which are part of the answer.

   b. Turn in your copy of the examination and all pages used to answer the examination questions.

   c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

   d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 1.01 (1.50)

The reactor is taken to CRITICALITY from a cold condition and an 80 second POSITIVE period is attained:

a. From control room nuclear instrumentation, HOW can the operator tell when the heating range has been reached? (Rod position and recirculation flow are held constant.) (0.5)

b. In which ONE of the following intervals was the heating range entered? (1.0)

(1) Interval 1 - reactor power increased by a factor of 6 in 143.3 seconds.

(2) Interval 2 - reactor power increased by a factor of 3 in 99.0 seconds

(3) Interval 3 - reactor power increased by a factor of 5 in 128.8 seconds.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.02 (1.00)

Which ONE (1) of the following thermal limits protects the fuel from clad rupture due to PLASTIC STRAIN (deformation)?

a. APLHGR

b. LHGR

c. MCPR

d. MAPRAT

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.03 (1.00)

Choose the ONE (1) phrase below which BEST completes the following sentence.

Without core orificing, the coolant flow through a high power bundle will be less than the flow through a low power bundle because:

a. the channel quality increases.

b. the two phase flow friction multiplier decreases.

c. the fuel rods expand due to thermal effects.

d. the bypass flow increases.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.04 (1.50)

Saturated steam with 100% quality enters the main condenser at 4.5 psia and with a flow rate of 6E+6 lbm/hr. Condensate exits as a saturated liquid. Circulating water enters the condenser at 62 deg F and exits at 77 deg F.

a. Choose the circulating water flow rate (in lbm/hr) from the list below: (1.0)

   1. 4.01E+8 lbm/hr
   2. 3.90E+8 lbm/hr
   3. 3.65E+8 lbm/hr
   4. 3.03E+8 lbm/hr

b. STATE whether the condenser vacuum will would INCREASE, DECREASE, or remain the same if the circulating water flow rate were DECREASED. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (1.00)

Choose the word(s) in parenthesis which best complete the following statement;

Shaping control rods are (DEEP, INTERMEDIATE, SHALLOW) rods that are used to change the power profile because they (ARE, ARE NOT) affected by shadowing. (Choose one answer in each parenthesis.)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.06 (1.00)

MULTIPLE CHOICE (Select the ONE correct answer.)

The Doppler Coefficient of Reactivity correlates the change in fuel temperature to a reactivity insertion.

Which statement is TRUE concerning Doppler Coefficient?

a. The coefficient becomes less negative with fuel burnup, and more negative with control rod withdrawal.

b. The coefficient becomes more negative with fuel temperature increase and less negative with void fraction increase.

c. The coefficient becomes less negative with control rod withdrawal, and more negative with fuel temperature increase.

d. The coefficient becomes more negative with void fraction increase and less negative with fuel temperature increase.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.07 (1.00)

The reactor is critical at 106 cps. Which ONE (1) of the following BEST describes the behavior of neutron power following a prompt insertion of negative reactivity?

a. Neutron power drops immediately to "Beta" (delayed neutron fraction) times the neutron power prior to the prompt insertion of negative reactivity.

b. Neutron power decreases linearly with time after the initial prompt drop.

c. After the initial prompt drop, neutron power decreases on a constant negative period; the magnitude of the period determined by the amount of negative reactivity inserted.

d. Because only delayed neutrons are left immediately after a negative reactivity insertion, neutron power decreases on an 80-second period regardless of the size of the negative reactivity insertion.

QUESTION 1.08 (1.00)

Reactivity is defined as which of the following?

a. The ratio of the number of neutrons at some point in this generation to the number of neutrons at the same point in the previous generation.

b. The fractional change in neutron population per generation.

c. The factor by which neutron population changes per generation.

d. The rate of change of reactor power in neutrons per second.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.09 (1.50)

State HOW EACH of the below listed conditions will effect control rod worth. (Limit the answer to INCREASE, DECREASE, or REMAINS THE SAME.)

a. Increasing moderator temperature

b. Increasing the percent voids

c. Increasing the fuel temperature

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.10 (3.00)

a. Define the term Critical Power (CP). (1.0)

b. State how Critical Power would change for each of the following events (i.e., INCREASE, DECREASE, or NO CHANGE). Assume that the reactor is at full power. Consider each event separately. (2.0)

1. Loss of a feedwater heater string

2. Reactor pressure increase from 950 psig to 1040 psig

3. Recirc Flow Control system fails to maximum demand

4. Feedwater Control system fails to maximum demand

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.11 (2.00)

Figure 1 contains charts of several key reactor parameters following a Feedwater Controller Failure to Maximum Demand.
For the areas marked, give the cause of each parameter change as stated below.

a. State WHY reactor power rises at Point A then immediately decreases. (0.5)

b. State WHY feedwater flow drops sharply at Point B. (0.5)

c. State WHY core flow drops at Point C. (0.5)

d. EXPLAIN the slight increase in reactor water level at Point D. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.12 (1.00)

The total amount of reactivity that must be added to bring a reactor to a critical condition is known as the:

a. Reactivity Defect
b. Excess Reactivity
c. Subcritical Factor
d. Shutdown Margin

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.13 (2.00)

With regard to PCIOMR, IDENTIFY EACH of the following statements as TRUE or FALSE.

a. PCI failures are dependent on absolute power, increase in power, duration of power increase, previous power history and fuel exposure.

b. Regardless of the stress state, strain rate, temperature, or state of irradiation, zircalloy fuel tubes will be ductile.

c. If power level is reduced prior to completing the 12 hour soak, pre-conditioning is resumed at either the new power level or the highest power level which has soaked for 12 hours before the power decrease, whichever is higher.

d. PCIOMR limitations are not required for barrier fuel because redesigned pellets cannot expand faster than the clad.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.14 (2.00)

For each of the following, indicate whether the available NPSH at the suction of the recirculation pump would INCREASE/DECREASE/ REMAIN THE SAME:

a. The Feedwater Flow is INCREASED

b. The Recirculation Flow is INCREASED

c. The Vessel Pressure is INCREASED from 200 psig to 800 psig

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.15 (1.00)

In a boiling water reactor, non-condensible gases produced in the reactor are carried through the turbine and into the condenser. Non-condensible gases can also leak into the condenser.

Which ONE (1) of the following correctly describes the effect of an increase in the amount of non-condensible gases in a steam turbine condenser?

a. Condenser pressure decreases (vacuum increases)

b. Circulating water outlet temperature decreases

c. Steam cycle efficiency decreases

d. Condensate depression increases

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.16 (2.00)

Indicate whether each of the following statements are TRUE or FALSE:

a. The delayed neutron fraction (beta) is defined as the ratio of thermal neutrons absorbed in the fuel to all thermal neutrons which are absorbed.

b. The effective delayed neutron fraction of the core is greater at BOL than at EOL.

c. When calculating reactor period, the delayed neutron term may be considered insignificant if reactivity added is less than beta.

d. Delayed neutrons are fast neutrons, but are usually born at lower energies than prompt neutrons.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.17 (2.00)

For each of the indicated changes in core parameters (a. through d. below) INDICATE whether the Void Coefficient of Reactivity will:

1. Become more negative.
2. Become less negative.
3. Not Change.

a. Increase in void fraction. (0.5)
b. Buildup of fission product poisons. (0.5)
c. Increase in fuel temperature. (0.5)
d. Decrease in control rod density. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.18 (1.00)

Concerning control rod worth during a reactor startup with 100% peak Xenon versus a startup with Xenon free conditions, WHICH statement (ONE) below is CORRECT?

a. Peripheral control rod worth will be LOWER during the 100% peak Xenon startup than during the Xenon free startup.

b. Central control rod worth will be HIGHER during the 100% peak Xenon startup than during the Xenon free startup.

c. Peripheral control rod worth will be HIGHER during the 100% peak Xenon startup than during the Xenon free startup.

d. Both central and peripheral control rod worths WILL BE THE SAME regardless of core Xenon concentration.

(***** END OF CATEGORY 1 *****)

QUESTION 2.01 (1.50)

With regard to the Unit 1 RHR System, MATCH each of the items in COLUMN A with its ONE (1) associated setpoint OR interlock from COLUMN B.

COLUMN A	COLUMN B
a. RHR HX bypass valve	1. 145 psi
b. LPCI injection valves open	2. 2/3 core coverage permissive
c. Recirculation discharge valves	3. 425 psi close
d. Containment spray valves	4. 105 psig
e. ADS permissive	5. Open for 3 minutes following LPCI initiation signal.
f. Shutdown cooling valves close	6. 325 psig
	7. 422 psig

QUESTION 2.02 (3.00)

Regarding the Standby Gas Treatment System:

a. State FOUR conditions which will automatically initiate the SGTS trains, isolate both unit's refuel floor ventilation systems, and isolate Unit One's reactor building ventilation. INCLUDE ANY SETPOINTS. (2.0)

b. EXPLAIN WHY it may be necessary to provide continued cooldown flow after the train is no longer required (assume the train has been operating for four hours). (1.0)

QUESTION 2.03 (2.50)

a. List FOUR (4) automatic trips/actuations which occur due to a High-High Main Steam Line Radiation signal (3 X Normal Background). (2.0)

b. State the MSL Radiation Monitor High-High setpoint when Hydrogen injection is in progress (assuming it has been changed as required). (0.5)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.04 (2.00)

Determine if EACH of the following statements concerning the Standby Liquid Control System is TRUE or FALSE.

a. In the event a remote (outside control room) reactor shutdown is required, SBLC injection can be actuated by the local pump START switch.

b. The pumps may be operated simultaneously if necessary to shutdown the reactor in an ATWS.

c. When the control room handswitch is placed to "PUMP A RUN", the "A" pump starts and all squib valves fire.

d. Nitrogen-charged accumulators assure adequate suction pressure for the pumps.

QUESTION 2.05 (1.50)

LIST and EXPLAIN HOW the Control Rod Hydraulic system design features, components, and/or interlocks provide the following functions;

a. Constant control rod speed/system flow during normal rod movement. (ONE COMPONENT/INTERLOCK REQUIRED) (0.5)

b. Prevent pump runout while a scram signal is present. (TWO COMPONENTS/INTERLOCKS REQUIRED) (1.0)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.06 (2.00)

RCIC is being used to control reactor level from the Remote Shutdown Panel when level is inadvertently allowed to increase to + 58".

a. EXPLAIN HOW the RCIC system will respond to this level and WHY it responds as it does. (0.75)

b. List FIVE (5) systems in addition to RCIC which can be operated (all or in part) from the Remote Shutdown Panel. (1.25)

QUESTION 2.07 (2.50)

Regarding the Unit 2 SRVs and associated Low Low Set (LLS) Logic System; (SETPOINTS NOT REQUIRED).

a. There are three lights associated with each SRV (RED, GREEN, and AMBER). EXPLAIN what EACH of the three colored lights indicates when energized. (1.5)

b. HOW and WHERE is the amber light reset after the conditions which caused it to be illuminated have cleared? (0.5)

c. List the TWO (2) conditions (signals) needed to arm the LLS logic. (0.5)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.08 (2.00)

State whether EACH of the following statements concerning the MSIV Leakage Control System is TRUE or FALSE.

a. The system is electrically interlocked to prevent initiation until 10 minutes after a LOCA event has occurred.

b. If the inboard system is actuated by the operator with the reactor at normal operating pressure and one MSIV failed open, the reactor will be vented to the Torus and will blowdown via the open steam line.

c. Dilution flow is intended to lower the temperature of the leakage steam to prevent exceeding the operating limits of the blowers.

d. The MSIV Leakage Control System is interlocked to prevent initiation if all MSIV's are not full closed.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.09 (1.00)

Which ONE (1) of the following statements BEST describes the purpose of the Exhaust Hood Spray?

a. Provide additional scrubbing for steam to aid in removal of non-condensibles during startup or low load operations.

b. Provide turbine blade cleaning during startup to remove accumulated silica deposits.

c. Provide a spray path to aid the condenser in maintaining vacuum during high circulating water temperature conditions by condensing steam as it leaves the last stage buckets.

d. Provide cooling for the last stage buckets during low steam flow conditions.

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.10 (2.00)

a. List THREE (3) of the four loads on Unit 2 RBCCW (Reactor Building Component Cooling Water) system which are not included in the Unit 1 system. (1.5)

b. After a temporary loss of essential 600 VAC bus had been corrected, an operator attempted to restart the affected RBCCW pump by taking the pump breaker control switch to OFF and then to ON, but the pump did not start. What additional action(s) must be performed in order to start the pump? (0.5)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.11 (2.75)

a. List THREE (3) of the four parameters which cause isolation of the Containment Atmosphere Monitor Hydrogen/Oxygen sample lines. (1.5)

b. List FIVE (5) containment parameters monitored by the CAMS in addition to monitoring hydrogen and oxygen concentrations in the Drywell. (1.25)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.12 (2.00)

a. If the Standby PSW pump is determined to be inoperable with the system in a normal configuration, STATE the action(s) required to be taken to ensure adequate cooling to components served.

b. LIST the conditions/signals which will automatically close the PSW turbine building isolation valves.
(Setpoints not required.)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.13 (1.50)

While operating RHR Pump 2A (Loop A) in Shutdown Cooling Mode with pump 2C tagged out, a LPCI initiation signal is received. All actions/trips occur in accordance with system design. LIST all actions required to be performed by the operator to use the A Loop of RHR in the LPCI mode. (Valve numbers are not required, but valve descriptions should be as specific as possible.)

(***** END OF CATEGORY 2 *****)

QUESTION 3.01 (3.00)

a. HOW is total core flow indication obtrined?

NOTE: BE SPECIFIC as to:
1. WHAT is measured
2. WHERE measured (sensing point(s)
3. HOW signal is converted and/or conditioned prior to control room indication (1.5)

b. Briefly EXPLAIN how the total core flow instrumentation compensates for a trip of one Recirculation pump (the other is still operating)? (1.0)

c. Briefly EXPLAIN why a different method is used when one loop is operating and one loop is shutdown. (0.5)

QUESTION 3.02 (3.00)

a. State THREE conditions/signals which will cause an APRM INOP trip. (1.5)

b. List THREE actuations or signals which MAY result when an APRM INOP trip signal is generated. (1.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.03 (2.00)

Concerning the Neutron Monitoring System (NMS), answer EACH of the following TRUE or FALSE.

a. Removing the Unit 2 "shorting links" will place all NMS scram signals in a coincidence mode.

b. All IRM trips are bypassed when the mode switch is in run.

c. The Unit 2 APRM flow biased scram is "clamped" at 118% regardless of recirculation flow.

d. The APRM flow biased scram is conditioned through a six second time constant.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (~~1.50~~ 1.0)

What effect would EACH of the following conditions have (INCREASE, DECREASE, or NO CHANGE) on indicated Reactor Vessel level indication?

a. Seat leakage on a level transmitter equalizing valve.

b. Increase in Drywell temperature.

~~c. Recirculation loop operation on wide range instrumentation.~~ delete

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (1.00)

Given: Unit 2 in control of D/G "B"
D/G "B" Mode Switch in TEST (Surveillance being performed)
Electrical distribution NORMAL (Full Power Lineup)

D/G "B" is at rated speed and voltage, but not synchronized, when all off-site power to 4160 volt Bus 2F is lost. Which ONE (1) of the following accurately describes the system operation?

a. Bus 2F can be powered by D/G "B" when the operator takes the Output Breaker Switch to CLOSE and has the SYNC SCOPE activated.

b. Bus 2F will be powered by D/G "B" automatically, after 12 seconds; appropriate loads will be picked up sequentially.

c. Bus 2F can not be powered by D/G "B" while it is in the TEST mode, given these conditions.

d. Bus 2F can be powered by D/G "B" when the operator resets the Lockout Relay, activates the SYNC SCOPE, and takes the Output Breaker to CLOSE.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.06 (2.00)

a. In addition to having power available, what CRITERIA (interlocks) must be satisfied in order to start a Condensate Booster Pump using the control switch? (INCLUDE SETPOINTS) (1.5)

b. List the SPECIFIC condition(s) which will cause a standby Condensate Booster Pump to automatically start if power is available and the control switch is in auto. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.07 (2.50)

List FIVE (5) of the seven Reactor Feed Pump Trip signals (INCLUDE ALL SETPOINTS for full credit).

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.08 (2.25)

List THREE (3) control systems/components outside of the feedwater system which receive inputs from the Reactor Water Level Control system. For EACH signal identify BOTH the INPUTS and what FUNCTION that input is used for (IDENTIFY ANY INTERLOCKS, CALCULATIONS, OR TRIPS).

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.09 (1.5 ~~2.00~~)

LIST the pressure setpoint (including system) where each of the following UNIT 2 Plant Air System isolations or initiations occur.

a. Nonessential Air Header ISOLATES.
b. Service Air Header ISOLATES.
c. Nitrogen backup valves OPEN to Noninterruptable Essential Instrument Air.
~~d. Nitrogen backup valves ISOLATE to Drywell Pneumatic System.~~ delete

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.10 (1.50)

STATE the THREE (3) conditions required for the Main Condenser Low Vacuum Isolation signal to be bypassed.
[Conditions may include operator action.]

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.00)

Choose ti parameter from the following list used by the Rod Sequence Control S tem to determine the automatic bypass power level.

a. APRM indicated power
b. Process Computer calculated thermal power
c. First Stage Turbine Pressure
d. Steam Flow and Feed Flow

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.12 (1.00)

a. Whose permission is required before manually bypassing the Rod Worth Minimizer? (0.5)

b. List the automatic bypasses of the RWM. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.13 (1.50)

Briefly DESCRIBE the consequences of closing the SUT supply breaker to a 4160V bus while the UAT breaker is closed (the control switch is released while both breakers are closed) and the "Cutout Switch" (panel 651) is not in thu Cutout position. (Include final position of both breakers and the status of the electrical bus.)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.14 (2.50)

List FIVE (5) conditions which must be met for the Diesel Generator output breaker to close when the mode switch is in "AUTO".

(***** END OF CATEGORY 3 *****)

QUESTION 4.01 (2.00)

List the immediate operator actions required by 34AB-OPS-055-2S, "Control Room Evacuation-Unit Shut Down", to be performed prior to leaving the control room.

(REACTOR IS IN OPERATIONAL CONDITION 3 AND NO SCRAM SIGNAL IS PRESENT)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.02 (2.00)

STATE which Emergency Classification is appropriate for EACH of the following definitions.

a. Events are in progress or have occurred which involve actual or potential substantial degradation of the level of safety of the plant.

b. Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant.

c. Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with the potential for loss of containment integrity.

d. Events are in progress or have occurred which involve an actual or likely major failure of plant functions needed for protection of the public.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.03 (2.00)

34AB-OPS-038-2, "Control of Sustained Combustion in the Offgas Systems," lists several system parameters which would be expected to change significantly if ignition occurs. List FOUR (4) parameters indicated by system instrumentation which could be expected to change EXCLUDING system and component temperatures.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.04 (2.50)

For each of the following sets of plant conditions, IDENTIFY the EOP flowpath (BY NUMBER) specifically designed to address those conditions.

a. Reactor transients or failure of vital equipment while in hot standby or startup.

b. High radiation, loss of vital power, failure of vital equipment, or stuck open relief valve.

c. High radiation, loss of coolant, and loss of primary containment integrity.

d. Reactor transients or failure of vital equipment while in the RUN mode.

e. Failure of reactivity control systems.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.05 (1.50)

The EOP contingency procedure for Alternate Pressure Control requires the operator to determine if adequate core cooling is assured. State THREE (3) criteria which may be used to verify adequate core cooling.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (1.00)

34SO-C11-005-2S, "Control Rod Drive Hydraulic System," contains a precaution stating the CRD Suction from Condensate Control Valve Bypass Valve, 2N1-F182, must remain LOCKED CLOSED while the Condensate system is in service. State the BASIS for this caution.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (1.00)

During operation with the Condensate and Feedwater Cleanup Recirc. Control valve (2NI-F165) open, limits are placed on how far the control valve may be opened with a Condensate pump or Condensate Booster pump running. EXPLAIN WHY it is necessary to limit the control valve position.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.08 (1.50)

RWCU system operating procedure, 32SO-G31-003-2S, contains a caution statement that states:

During operation of the RWCU system CU & Demin Bypass Vlv 2G31-F044 must not be fully closed for extended periods of time and left unattended.

a. This valve is opened in anticipation of WHAT plant or system transient? (0.5)

b. State the function provided by this valve lineup. (1.0)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.09 (1.00)

State TWO (2) reasons Mechanical Vacuum Pump operation is not permitted above 5% Thermal Power.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.10 (1.00)

Explain WHY it is not necessary to trip the Main Generator following a loss of both the Main and Emergency Seal Oil Pumps.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.11 (1.50)

List THREE (3) radiological conditions which require a Normal Operations Job Specific Radiation Work Permit be issued for work in that area. (SETPOINTS NOT REQUIRED)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.12 (1.00)

Determine if the following statements are TRUE or FALSE.

a. After obtaining a new P1 printout, the operator can use the old P1 for scratch paper or discard it.

b. Chart recorders are required to be checked within one hour of shift change and marked with date, time, and operator initials.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.13 (2.00)

List TWO (2) conditions where removal of Danger Tags may be authorized from an Equipment Clearance Sheet (ECS) when all subclearances on that ECS have not been released.

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.14 (3.50)

a. You are verifying a valve line-up, STATE how you confirm position of EACH of the following:

1. Closed valve. (0.5)

2. Open valve. (0.5)

3. Motor-operated valve. (0.5)

4. Locked throttle valve. (0.5)

b. List the actions required if a LOCKED VALVE is found in the wrong position while performing a valve lineup. (1.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.15 (2.00)

42FH-ENG-010-2, "Control Rod Movement", provides numerous STANDARD PRACTICES which apply when rods are being moved for the purpose of changing power level. Answer EACH of the following in accordance with 42FH-ENG-010-2.

a. With LHGR > 8 Kw/ft, STATE the period of time you must wait between successive notch withdrawals of the same rod. (0.5)

b. With HIGH POWER and HIGH FLOW conditions, STATE which type of rod - SHALLOW, DEEP, CENTRAL, or EDGE - should NOT be withdrawn nor inserted. (1.0)

c. With power < 30%, STATE the verification which is utilized when latching the first rod in any group. (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.16 (0.50)

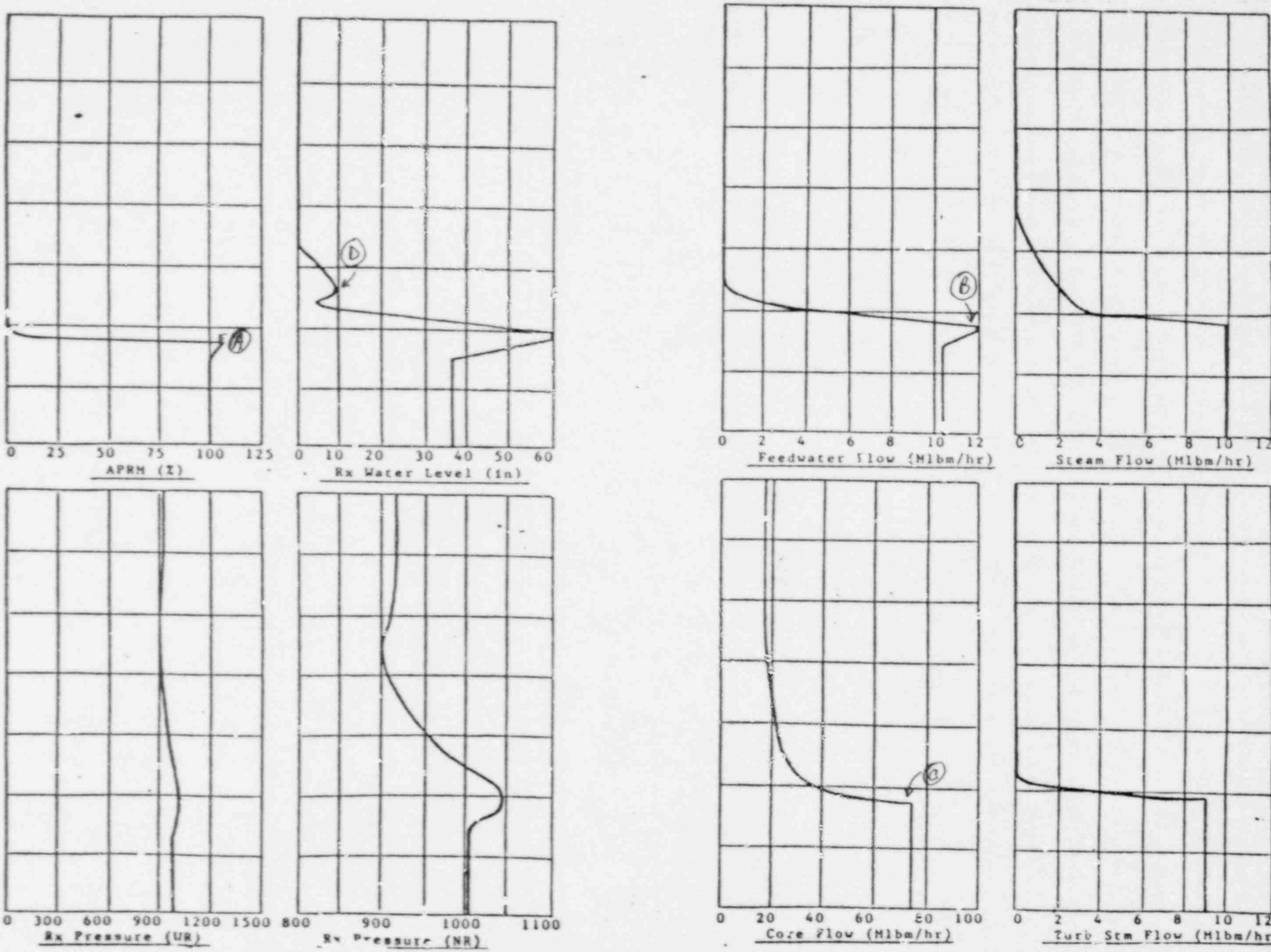
Determine if the following statement concerning a Cable Spreading Room fire and Procedure 34AB-FPX-053-2S if is TRUE or FALSE.

For an exposure fire (in the cable spreading room) involving combustables WITHOUT electrical insulation involvement, the Control Room Operator in the affected unit will initiate a rapid load reduction, will trip the turbine generator, will manually SCRAM the reactor, and will place the Mode Switch in SHUTDOWN.

(***** END OF CATEGORY 4 *****)

(********** END OF EXAMINATION **********)

EXAM FIGURE 1



FEEDWATER CONTROL FAILURE (MAX DEMAND)
(FIGURE 10.4/33)

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = UA\Delta T$

$Pwr = W_f\,\dot{m}$

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

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

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\lambda_{eff}\rho]$

$T = \ell^*/(\rho - \bar{\beta})$

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\rho = [\ell^*/TK_{eff}] + [\bar{\beta}/(1 + \lambda_{eff}T)]$

$P = \Sigma\phi V/(3 \times 10^{10})$

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

Cycle efficiency $= \dfrac{\text{Net Work (out)}}{\text{Energy (in)}}$

$A = \lambda N \qquad A = A_o e^{-\lambda t}$

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

$t_{1/2}(eff) = \dfrac{(t_{1/2})(t_b)}{(t_{1/2} + t_b)}$

$I = I_o e^{-\Sigma x}$

$I = I_o e^{-\mu x}$

$I = I_o\,10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1$ seconds^{-1}

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$R/hr = (0.5\ CE)/d^2$(meters)

$R/hr = 6\ CE/d^2$(feet)

<u>WATER PARAMETERS</u>

1 gal. = 8.345 lbm

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg

1 ft. H_2O = 0.4333 lbf/in^2

<u>MISCELLANEOUS CONVERSIONS</u>

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 BTU/hr

1 Mw = 3.41×10^6 Btu/hr

1 Btu = 778 ft-lbf

1 inch = 2.54 cm

$^oF = 9/5\,^oC + 32$

$^oC = 5/9\,(^oF - 32)$

ANSWER 1.01 (1.50)

Either of the following responses is correct (0.5);

a. Operator can notice that period has become longer ~~(0.25)~~ e *or*
and that power change on IRMs/SRMs is leveling off
(turning around due to power overshoot). ~~(0.25)~~ e

b. (2) (1.0) (From P = Poe(t/T) --> T = t/ln (P/Po), in Interval 2
the period has lengthened from 80 seconds. The other intervals have
80 second periods)

REFERENCE

EIH, GE Reactor Theory, LO 7.4.3, P. 7-10
CPS Introduction to Nuclear Reactor Operations, LO 4.1.1.2
INRO PP. 4-17 & 4-18
(3.8/3.6)

292008K112 292008K113 ..(KA's)

ANSWER 1.02 (1.00)

b. LHGR (1.0)

REFERENCE

EIH, GE Heat Transfer & Fluid Flow, LO 9.3.3, P. 9-15
CPS Nuclear Power Plant Thermal Sciences, LO 12 1.1.1
NPPTS P. 12-2
(3.0)

293009K108 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.03 (1.00)

a. (1.0)

REFERENCE

EIH, GE Heat Transfer & Fluid Flow, LO 8.9.2, P. 8-46
BSEP ,HEAT TRANSFER, CH. 9 Page 9-51 Lesson Objective = Third from top of page 9-1A (no number assigned)
(2.9)

293008K131 ..(KA's)

ANSWER 1.04 (1.50)

a. 1 (4.01E+8) [1.C]

b. Decrease (absolute pressure would increase) [0.5]

REFERENCE

EIH, GE Heat Transfer & Fluid Flow, LO 7.5.2, P. 7-40
(2.8/2.5)

293004K113 293003K123 ..(KA's)

ANSWER 1.05 (1.00)

Shallow (0.5)
ARE (0.5)

REFERENCE

EIH, GE Reactor Theory, LO 5.3.3, P. 5-25
Monticello: Reactor Theory, Chapter 5
(2.8/2.4)

292005K111 292005K110 ..(KA's)

ANSWER 1.06 (1.00)

d [1.0]

REFERENCE

EIH, GE Reactor Theory, LO 4.6.3, PP. 4-38 thru 4-42
WNP-2 REACTOR THEORY TEXT page IV-25
(2.9)

292004K105 ..(KA's)

ANSWER 1.07 (1.00)

c [1.0]

REFERENCE

EIH, GE Reactor Theory, LO 5.5.6 & 5.5.9, PP. 5-42, 43
NMP-2 Operations Technology, Module 1, Part
(3.3/3.7)

292003K106 292003K107 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.08 (1.00)

b

REFERENCE

EIH, GE Reactor Theory, LO 1.6.1, P. 1-38
DPC, Fundamentals of Nuclear Reactor Engineering, p. 96
GGNS: OP-NP-511
(3.2)

292002K111 ..(KA's)

ANSWER 1.09 (1.50)

a. Increase

b. Decreases

c. Remains the Same
(3 @ 0.5 ea)

REFERENCE

EIH, GE Reactor Theory, LO 5.2.5, PP. 5-12a, 5-13a, and 5-14a.
(2.5)

292005K109 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.10 (3.00)

a. Critical Power is the bundle power needed to produce the critical quality (or the bundle power needed to cause OTB to occur somewhere in the bundle. [1.0]

b. 1. (inlet subcooling ^) CP increases [0.5]
2. (pressure ^) CP decreases [0.5]
3. (core flow ^) CP increases [0.5]
4. (inlet subcooling ^) CP increases [0.5]

REFERENCE

EIH, GE Reactor Theory, LO 9.5.1 & 9.5.7, PP. 9-26, 9-36 thru 9-39
SSES SC023 G-3 Specific Objectives 3.3, 3.4, 3.6
(3.3/2.9/2.8/2.7)

293009K124 293009K123 293009K122 293009K117 ..(KA's)

ANSWER 1.11 (2.00)

a. Rx power increases due to increased subcooling (+p) then decreases sharply due to scram from turbine trip (hi level). (0.5)

b. FW flow decreases due to RFPTs trip on high RPV water level (0.5)

c. Core flow decreases due to RPT breaker actuation. (0.5)

d. Momentary increase from reformation of voids as pressure decreases and feed pumps coastdown. (0.5)

REFERENCE

EIH Nuclear Training, 10.4-36 and Figure 10.4(33)
(3.8/3.5/3.8/3.7)

259002K302 259002K301 259002K104 259002K103 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.12 (1.00)

d [1.0]

REFERENCE

EIH, GE Reactor Theory, LO 1.5.1, P. 1-36
EIH Nuclear Training, 10.103-22
(3.2)

292002K110 ..(KA's)

ANSWER 1.13 (2.00)

a. True
b. False
c. True
d. False

(4 @ 0.5 each)

REFERENCE

EIH Nuclear Training, Chap 10.2, PP. 18-21
GE BWR Academic Series, HT/FF LO 7.2, 7.5, and 8.2, PP. 9-49 through 9-52
(2.8/2.9)

293009K132 293009K136 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.14 (2.00)

a. INCREASE (More Subcooling at the pump suction)

b. DECREASE (Reduced pressure at the eye of the pump results in being closer to saturation pressure)

c. DECREASE (Further to saturation temperature and increased density causing less static head)

(3 @ ~~0.50~~ 0.66 each)

REFERENCE

EIH, Heat Transfer & Fluid Flow, LO 6.10.10, P.6-81
(2.7/2.5)

293006K108 293006K110 ..(KA's)

ANSWER 1.15 (1.00)

c [1.0]

REFERENCE

EIH, Heat Transfer & Fluid Flow, LO 5.5.2, P. 5-53
(2.7)

293007K107 ..(KA's)

ANSWER 1.16 (2.00)

a. FALSE [0.5]

b. TRUE [0.5]

c. FALSE [0.5]

d. TRUE [0.5]

REFERENCE

GE Reactor Theory, LO 3.4.4, 3.4.5, 3.5.6, and 3.4.1, PP. 3-29 thru 36
(2.5/3.7/2.4/3.0)

292001K102 292003K103 292003K106 292003K104 ..(KA's)

ANSWER 1.17 (2.00)

a. 1 (more negative) [0.5]
b. 1 (more negative) [0.5]
c. 1 (more negative) [0.5]
d. 2 (less negative) [0.5]

REFERENCE

GE Reactor Theory, LO 4.4.3, pp. 4-19 through 4-23
(2.5/2.2)

292004K112 292004K111 ..(KA's)

ANSWER 1.18 (1.00)

c [1.0]

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH, GE Reactor Theory, LO 5.2.4, P. 5-15 and 6-12
(3.1)

292006K114 ..(KA's)

(***** END OF CATEGORY 1 *****)

ANSWER 2.01 (1.50)

a. 5
b. 7
c. 6
d. 2
e. 4
f. 1

(6 @ 0.25 each)

REFERENCE

EIH LT-IH-00701-00, LO 11,12,&14, TABLE 00701-2, PP. 48, 58-62
Nuclear Training, Main Steam, 5.1-5
Nuclear Training, Recirculation System, 4.1-38
(3.3/3.9/4.0/3.7)

205000K402 203000K411 203000K410 203000K402 ..(KA's)

ANSWER 2.02 (3.00)

a. Any 4 of the following @ 0.5 each;

1. Unit One Reactor building ventilation exhaust high radiation [0.25], 20 mR/hr [0.25]
2. Unit One Refueling Floor high radiation [0.25] 20 mR/hr [0.25]
3. Unit Two Refueling Floor high radiation [0.25] 20 mR/hr [0.25]
4. Unit One Drywell high pressure [0.25], 1.92 psig [0.25]
5. Reactor vessel low water level [0.25], -47 inches [0.25]

b. To remove heat generated by radioactive particles contained in the idle train. [1.0]

REFERENCE

EIH Nuclear Training, 3.3-6 and 3.3-9.
LP 13.2, LO 2, P. 4
LP 30.1, LO 3, P. 4
(3.7/2.6)

261000K402   261000K401   ..(KA's)

ANSWER   2.03   (2.50)

a. Any 4 of the following at 0.5 each;

1. Reactor Scram
2. Group I isolation
3. Swap Main Control Room Ventilation to pressurization mode.
4. Mechanical vacuum pump trip/isolates
5. Gland seal exhauster trip/isolates

b. 6 X Normal Background   [0.5]

REFERENCE

EIH, LP 14.1, LO 2, P. 6
SO-OPS-01-1987
(3.3/3.2/3.2)

272000K501   272000K402   288000K105   ..(KA's)

ANSWER   2.04   (2.00)

a. FALSE   [0.5]
b. FALSE   [0.5]
c. TRUE   [0.5]
d. FALSE   [0.5]

REFERENCE

EIH, LP LT-IH-01101-00, LO 5,8,9,& 11
Arnold System Description C-4, P. 15
(3.0/4.2/2.5/2.5)

211000K505 211000K409 211000K408 211000K402 ..(KA's)

ANSWER 2.05 (1.50)

a. Stabilizing valves (0.25) maintain constant flow through the pressure control valve (0.25) (thus maintaining RPV/drive water differential pressure constant) (0.5)

b. 1) Restricting orifice (0.25) (in the charging water header) limits flow (0.25) (to less than 200 gpm) (0.5)

2) Flow element for Flow Control Valves (0.25) is located between the pumps and the charging water header (0.25) (so a high charging flow closes the FCVs) (0.5)

REFERENCE

EIH Nuclear Training, 4.2-7
(2.5/2.6)

201001K402 201001K401 ..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.06 (2.00)

a. No response (operates normally) [0.25] because high water level trip is bypassed when operating from the Remote Shutdown Panel. [0.5]

b. Any 5 of the following @ 0.25 each:

1. Residual Heat Removal
2. RHR Service Water
3. Safety Relief
4. Plant Service Water
5. Control Rod Drive
6. Reactor Recirculation

REFERENCE

EIH, LT-IH-05201-00, PP. 8 & 14
(3.5/3.3)

217000K402 217000K102 ..(KA's)

ANSWER 2.07 (2.50)

a. GREEN Light - power available for the solenoid control valve [0.5]
RED Light - solenoid control valve has energized [0.5]
AMBER Light - high pressure in the relief valve tail pipe (tail pipe pressure swith actuated at greater than or equal to 85 PSIG) [0.5]

b. Manually reset by key lock [0.25] at P602 [0.25]

c. 1. Any SRV has opened (tail pipe pressure switch activated) [0.25]

2. Reactor pressure at high pressure setpoint (1044 or 1054 PSIG) [0.25]

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH Nuclear Training, Chapter 5.1, Pages 7, 21 and 22
(3.7/3.6/3.4)

239002K402 239001K610 239001K125 ..(KA's)

ANSWER 2.08 (2.00)

a. False
b. False
c. True
d. False

(4 @ 0.5 each)

REFERENCE

EIH Nuclear Training, Chapter 5.1, PP. 10-16
Lesson Plan 49.1, L.O. #7 and #4 P. 5
(3.1/3.2/3.0/2.4)

239003K407 239003K402 239003K401 239001K406 ..(KA's)

ANSWER 2.09 (1.00)

d (1.0)

REFERENCE

EIH Nuclear Training, Chapter 5.3, P. 8
(2.5/2.5/2.8/2.6/2.4/3.2)

256000K301 256000K110 256000K101 245000K502 245000K306
245000K102 ..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.10 (2.00)

a. Any 3 of the following @ 0.5 each:

1. Recirculation Pump Motor Windings
2. Recirculation MG Set Motor Coolers
3. Recirculation MG Set Generator Coolers
4. Reactor Water Sample Coolers

b. The 600V bus load shedding reset pushbutton must be reset. (0.5)

REFERENCE

EIH LP 9.1, LO #3 and #4, PP. 3, 4, & 8
(3.3/3.5)

295018K101 295018K201 ..(KA's)

ANSWER 2.11 (2.75)

a. Any 3 of the following @ 0.5 each:
1. High Drywell Pressure
2. Low Reactor Water Level
3. High Reactor Building Exhaust Radiation
4. High Refuel Floor Ventilation Exhaust Radiation

b. Any 5 of the following @ 0.25 each:
1. Particulate fission products in the drywell.
2. Gaseous fission products in the drywell.
3. Temperature in the drywell.
4. Pressure in the drywell.
5. Water temperature in the torus.
6. Water level in the torus.
7. Gamma Radiation levels in the Torus / Drywell

REFERENCE

EIH LP 51.1, LO #4 and #9, PP. 3 & 5
(3.2/3.7)

223001K403 223001K103 ..(KA's)

ANSWER 2.12 (2.00)

a. Service water to 1B D/G must be aligned manually from Div. I or Div. II PSW. [1.0]

b. 1. LOCA
2. LOSP
3. Condenser room flooding (3" above floor)
4. High division flow (20 psid increasing)
[4 @ 0.25 each]

REFERENCE

EIH LP 33.1, LO 3, P. 5 & 9
(3.3/2.9)

295018K301 295018K201 ..(KA's)

ANSWER 2.13 (1.50)

1. Close RHR Shutdown Cooling Isolation or Pump Shutdown Cooling Suction Valve (F006A).

2. Open Pump Torus Suction Valve (F004A).

3. Manually start RHR pump.
(3 @ 0.5 each)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH LP-IH-00701-00, LO 21, Table 00701-5
(3.6/3.9)
205000K108   203000K114   ..(KA's)

(***** END OF CATEGORY 2 *****)

ANSWER 3.01 (3.00)

a. The 20 non-calibrated jet pump differential pressure signals (0.5), from the jet pump throat pressure (0.25) and the below core plate tap (SLC injection line) (0.25), are converted to flow signals (by the square root extractors) (0.25). The jet pump flow signals are summed to obtain two recirc loop total flow signals and total core flow (0.25). [1.5 total]

b. When one pump trips, the circuit selects the summer which calculates the algebraic difference between the two loop flow signals. [1.0]

c. Flow being measured by the idle loop is backflow from the operating loop. [0.5]

REFERENCE

EIH, LP44.1, P. 11
(3.2/3.3)

216000K123 216000K110 ..(KA's)

ANSWER 3.02 (3.00)

a. 1. Less than 11 LPRM inputs to the channel.
2. APRM mode switch not in "Operate".
3. Any internal module (APRM or flow channel) unplugged.
(also accept the following as one of the required three:
Flow channel mode switch not in "Operate".)
(3 @ 0.5 each)

b. Any 3 of the following @ 0.5 each;

1. Reactor half-scram (full scram if 2 or more channels)
2. Rod Block
3. Annunciator on P603

REFERENCE

EIH LP 12.3, LO 4, P. 13
(4.0/3.6/3.3/3.7/4.1)

215005K402 215005K401 215005K116 215005K104 215005K101
..(KA's)

ANSWER 3.03 (2.00)

a. True [0.5]

b. False [0.5]

c. False [0.5]

d. True [0.5]

REFERENCE

EIH LP 12.1, LO 7, P.13
LP 12.2, LO 5, P. 8
LP 12.3, LO 3, PP. 11 & 12
(3.2/3.0/3.3/3.3/3.7/3.9)

212000K412 215005K407 212000K411 215005K116 215004A103
215004K406 ..(KA's)

ANSWER 3.04 1.0 (~~1.50~~)

a. Increase

b. Increase

~~c. Increase~~ Delete

[~~3~~ 2 @ 0.5 ea.] ~~(1.5)~~ (1.0)

REFERENCE

EIH, LP 44.1, LO 1, PP. 5-7
CPS: L. P. 74104 PP. 8 & 19. Enabling Objective 2.3.
(3.6/2.9)

216000K509 216000K507 ..(KA's)

ANSWER 3.05 (1.00)

c [1.0]

REFERENCE

EIH: GPNT, Vol VI, PP. 7.2-20
(3.0/3.6/3.8)

264000K101 262001K406 262001K401 ..(KA's)

ANSWER 3.06 (2.00)

a. 1. Pump suction valve (0.25) must be full open. (0.25)
   2. Suction pressure (0.25) must be greater than 34 psig. (0.25)
   3. Pump oil pressure (0.25) must be greater than 5 psig. (0.25)

b. Trip of a running pump. (0.5)

REFERENCE

EIH Nuclear Training, Chapter 5, P. 15
LP 2.1, LO #8, P. 6
(3.4/2.8)

256000K403 256000K401 ..(KA's)

ANSWER 3.07 (2.50)

Any 5 of the following at 0.5 each (0.25 for signal, 0.25 for setpoint):

1. Reactor vessel high water level - 58".
2. Low Condenser Vacuum - 22.3".
3. Thrust Bearing Wear - 40 PSIG Lube Oil Pressure.
4. Turbine Bearing Low Lube Oil Pressure - 4 PSIG.
5. Low Pump Suction Pressure - 160 PSIG (4 second T.D.)
6. RFP Bearing Low Lube Oil Pressure - 4 PSIG.
7. Mechanical Overspeed - 110% (6325 RPM).

REFERENCE

EIH LP 2.1, LO 4, P. 7
(2.5)

259001K405 ..(KA's)

ANSWER 3.08 (2.25)

Any 3 of the following;

1. Reactor Recirculation System (0.5)
   a. Total feed flow to #1 speed limiter. (0.125)
   b. Level and individual feedwater loop flow to #2 speed limiter. (0.125)

2. Process Computer (0.5) total steam flow and total feed flow for heat balance. (0.25)

3. Rod Worth Minimizer (0.5) total steam flow and total feed flow for power determination. (0.25)

4. Main Turbine (0.5) level for high water (58") trip. (0.25)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH Nuclear Training, Chapter 9.5, P. 15
LP 2.2
(3.1/2.6/3.6/2.9/3.2/3.1)

202002K109 259002K115 259002K114 259002K105 259002K107
201006K103 ..(KA's)

ANSWER 3.09 ~~(2.00)~~ 1.50

a. 50 PSIG Instrument Air Header Pressure.
b. 70 PSIG Instrument Air Header Pressure.
c. 90 PSIG Instrument Air Header Pressure.
~~d. 111 PSIG DW Pneumatic System Header Pressure.~~ delete
[~~4~~ 3 @ 0.5 EACH]

REFERENCE

EIH, LT-IH-03501-00, LO 11 & 12, Table 03501-9
(3.2/3.3/3.5/3.2)

295019K303 295019K302 295019K301 295019K214 ..(KA's)

ANSWER 3.10 (1.50)

1. Keylock switches (on P609 and P611) in bypass.

2. Turbine stop valves closed.

3. Mode switch not in Run.

(3 @ 0.5 ea = 1.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH Nuclear Training, 5.7-5
DAEC - System Description, A-6, Main Steam, page 36.
(3.8/3.1)

239001K402 239001K401 ..(KA's)

ANSWER 3.11 (1.00)

C [1.0]

REFERENCE

EIH, LP LT-IH-05402-01, LO 8, P. 33
(3.3)

201004K404 ..(KA's)

ANSWER 3.12 (1.00)

a. Operations supervisor [0.5]

b. power above the LPSP (30% power) [0.5]

REFERENCE

EIH, LP 54.2, LO 6 & 7, P. 10
(3.4/2.9)

201006K502 201006K404 ..(KA's)

ANSWER 3.13 (1.50)

The closed breaker will trip open [0.5] and the breaker being closed will trip open [0.5] resulting in a de-engergized bus. [0.5]

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH LP 27.1, LO 13, P. 13
(3.4/2.9/3.1)

262001K403 262001K402 262001K103 ..(KA's)

ANSWER 3.14 (2.50)

Any 5 of the following @ 0.5 each;

1. Emergency 4160V bus undervoltage
2. Proper output frequency
3. Proper output voltage
4. No electrical fault on bus.
5. Normal bus supply breaker open.
6. Alternate bus supply breaker open.

REFERENCE

EIH LP 28.1, LO 2, P. 7
(3.8)

264000K101 ..(KA's)

(***** END OF CATEGORY 3 *****)

ANSWER 4.01 (2.00)

1. Close the following valves; (valve numbers not required for full credit)

   - All MSIVs (B21-F022A(B,C,D) and B21-F028A(B,C,D)) [0.5]
   - Both Steam Drain Isolations (B21-F016 and B21-F019) [0.5]
   - Both Reactor Water Sample Isolations (B31-F019 and B31-F020) [0.5]

2. Open the RFP Bypass (N21-F113) [0.5]

REFERENCE

EIH, 34AB-OPS-055-2S
(3.8)

295016G010 ..(KA's)

ANSWER 4.02 (2.00)

a. Alert

b. (Notification of) Unusual Event

c. General Emergency

d. Site Area Emergency

(4 @ 0.5 each)

REFERENCE

EIH: GET Handbook, pp 57, 58, 60, 61; 10AC-MGR-006-0S
BFNP: BFN-IPD, IP-1, p 1; RQ 85/04/01
(4.3)

295030G011 ..(KA's)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

ANSWER 4.03 (2.00)

Any 4 of the following @ 0.5 each;

1. Prefilter differential pressure
2. Afterfilter differential pressure
3. System pressure
4. Post Accident radiation level
5. Offgas System flow rate
6. Hydrogen concentration

REFERENCE

EIH, 34AB-OPS-038-2
(3.8)

271000G015 ..(KA's)

ANSWER 4.04 (2.50)

a. 2
b. 4
c. 5
d. 3
e. 1

(5 @ 0.5 EACH)

REFERENCE

EIH, LT-IH-20101-00, LO 7, PP. 18 & 19
(3.9/3.9/3.9/3.8)

295006G012 295031G012 295025G012 295024G012 ..(KA's)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

ANSWER 4.05 (1.50)

1. At least one Core Spray pump is injecting, or
2. Reactor water level is above the top of active fuel, or
3. Steam cooling is in progress.

(3 @ 0.5 each)

REFERENCE

EIH, 31EO-EOP-001-2S 3.83, p. 3
(4.6/4.6)

295031A204 295031K101 ..(KA's)

ANSWER 4.06 (1.00)

Prevent overpressurization of the CRD suction piping. [1.0]

REFERENCE

EIH, 34SO-C11-005-2S
(3.2)

201001G010 ..(KA's)

ANSWER 4.07 (1.00)

To avoid excessive vibration on the piping returning to the main condenser. [1.0]

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH, 34SO-N21-007-2S, p. 15
(3.1/2.4)

256000K506 256000G010 ..(KA's)

ANSWER 4.08 (1.50)

a. RWCU system isolation (in which both demineralizers transfer to hold) [0.5]

b. Provide a pressure relief path for CRD water [1.0]

REFERENCE

EIH, 34SO-G31-003-2S, P. 15
(2.7)

204000K402 ..(KA's)

ANSWER 4.09 (1.00)

1. Significant amounts of H2 and O2 are present in the main condenser (which could result in a detonable mixture). [0.5]

2. The mechanical vacuum pump bypasses the holdup volume. [0.5]

REFERENCE

EIH, 34SO-N61-001-2S, p. 3
(3.1/2.7)

271000K507 271000G010 ..(KA's)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

ANSWER 4.10 (1.00)

Some machine gas pressure will be maintained (25-30 psig) by the bearing header oil supply to the seals [1.0] (so a load reduction is all that will be required).

REFERENCE

EIH, 34SO-N42-001-2S, p. 6
(2.8/2.8)

245000K610 245000K603 ..(KA's)

ANSWER 4.11 (1.50)

Any 3 of the following @ 0.5 each;

1. High area radiation levels (greater than 100 hr).
2. High Airborne radioactivity concentration (greater than 25% MPC)
3. Loose surface contamination (levels greater than 50,000 dmp/sq.cm)
4. Breach of a contaminated system.

REFERENCE

EIH, 62RP-RAD-006-OS, P. 3
(3.3)

294001K103 ..(KA's)

ANSWER 4.12 (1.00)

a. FALSE [0.5]
b. TRUE [0.5]

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH, 30AC-OPS-003-0S, PP. 16 & 17
(3.4)

294001A106 ..(KA's)

ANSWER 4.13 (2.00)

1. Tags included on a Temporary Release (for functional testing) [1.0]

2. Tags where all subclearances affecting them have been released and a Shift Supervisor (or Operations Supervisor) review has been performed. [1.0]

REFERENCE

EIH, 30AC-OPS-001-0S, PP. 15 & 18
(3.9)

294001K102 ..(KA's)

ANSWER 4.14 (3.50)

a. 1. Turn valve in closed direction (1/4 turn max to seat)

2. Turn valve in open direction (1/4 turn max to backseat)

3. Verify at remote (or local) position indication

4. Confirm locking device operability by attempting to move valve.

(4 @ 0.5 each)

b. 1. Reposition valve after getting SS concurrence

2. Install locking device and verify operability

3. Prepare a deviation report

(3 @ 0.5 each)

REFERENCE

EIH, 34GO-SUV-001-0S, P. 2
10AC-MGR-004-0
(3.7)

294001K101 ..(KA's)

ANSWER 4.15 (2.00)

a. 2 Minutes [0.5]

b. Shallow [1.0]

c. "Print Notch Error" function (of RWM) [0.5]

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

EIH: 42FH-ENG-010-1/2
(2.8/3.2)

201004K405 201006K507 ..(KA's)

ANSWER 4.16 (0.50)

FALSE [0.5]

REFERENCE

EIH: 34AB-FPX-053-2, pp 8
(3.5)
294001K116 ..(KA's)

(***** END OF CATEGORY 4 *****)
(********** END OF EXAMINATION **********)

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: HATCH 1&2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/02/08

EXAMINER: HOPPER, G.

CANDIDATE: ____________________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question 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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
27.00	25.65	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.25	23.99	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
28.00	26.60	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	23.75	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
105.25		________ Final Grade	________%	Totals

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only</u> <u>on</u> <u>one</u> <u>side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

   a. Assemble your examination as follows:

      (1) Exam questions on top.

      (2) Exam aids - figures, tables, etc.

      (3) Answer pages including figures which are part of the answer.

   b. Turn in your copy of the examination and all pages used to answer the examination questions.

   c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

   d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: HATCH 1&2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/02/08

EXAMINER: HOPPER, G.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question 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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
27.00	33.64	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.25	31.46	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
28.00	34.89	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
80.25		________ Final Grade	________%	Totals

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil only to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of each section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a new page, write only on one side of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least three lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility literature.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the examiner only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

   a. Assemble your examination as follows:

      (1) Exam questions on top.

      (2) Exam aids - figures, tables, etc.

      (3) Answer pages including figures which are part of the answer.

   b. Turn in your copy of the examination and all pages used to answer the examination questions.

   c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

   d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (2.00)

The reactor is operating at 80 % power.
ANSWER EACH ONE of the following questions as TRUE or FALSE.

a. INCREASING recirculation flow will cause reactor power to INCREASE.

b. DECREASING reactor pressure will cause a REDUCTION in reactor power.

c. LOSS of a heater string will cause power to DECREASE due to less subcooling of the feedwater.

d. INSERTING control rods will usually cause reactor pressure and generator output to DECREASE.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.02 (2.00)

Answer EACH ONE of the following statements TRUE or FALSE regarding reactivity coefficients.

a. An increase in flow through the reactor core will add negative reactivity by decreasing the void fraction and thus increasing reactor power.

b. As the burnable poison within a fuel bundle burns out, the VOID coefficient becomes more negative.

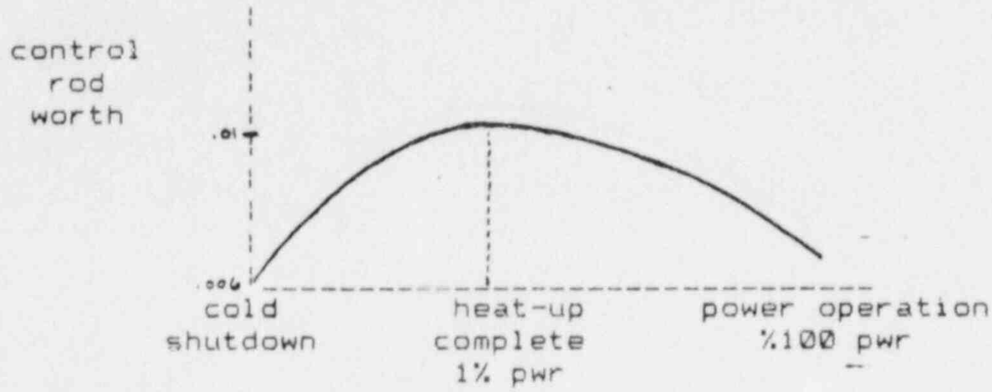
c. LATE in core life, the large reduction in fuel molecules and the decrease in moderator density during a plant HEAT-UP can lead to a positive reactivity addition.

d. As core age progresses, the DOPPLER coefficient becomes more negative due to plutonium-240 buildup.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.03 (2.00)

Answer EACH ONE of the following questions TRUE or FALSE concerning the graph of Contol Rod Worth During a Startup.



a. Control rod worth increases during heatup due to density decreases of the moderator which causes longer slowing down and thermal diffusion lengths, resulting in more thermal flux around a control blade.

b. Control rod worth decreases as power exceeds 1% due to the effects of rod shadowing. Withdrawal of rods increases the thermal diffusion length thereby increasing the flux around a control blade.

c. While heating-up, rod worth increase is due mainly to the effects of Bundle Coupling. Rod withdrawal couples fuel cells together making their effective size larger, resulting in increased leakage and a reduction in thermal flux.

d. Since control rods are worth more when the moderator is hot, fewer control rods must be withdrawn to go critical when the reactor is hot than when cold.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (2.00)

Answer EACH ONE of the following statements TRUE or FALSE.

a. Improper voiding and neutron thermalization caused by incorrect bypass flow could result in inaccurate indications of power.

b. A portion of the core bypass flow is provided by coolant holes in the lower core plate.

c. Increasing power by control rod withdrawal will increase core bypass flow.

d. A change in core pressure drop has a greater effect on core bypass flow than on channel flow.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.05 (2.00)

Match EACH ONE of the following terms in COLUMN A with the best corresponding definition in COLUMN B. Choose ONE (1) definition for each term.

COLUMN A

a. Keff

b. Shut Down Margin

c. K-excess

d. Reactivity

COLUMN B

1. An amount of fuel loaded into the core above that required for initial criticality to compensate for control rod motion.

2. A measure of the fractional change of the neutron multiplication factor.

3. 1-Keff/ Keff -1

4. A measure of how sub-critical the reactor is in terms of Keff.

5. The % of neutrons which are born delayed.

6. A measure of the availability of fuel loaded into the core above that is required for initial criticality

7. A ratio = $\frac{\text{\# neutrons produced from thermal fission}}{\text{\# neutrons produced from thermal fission preceeding generation}}$

QUESTION 5.06 (1.00)

Match EACH ONE of the following statements in COLUMN A with the corresponding acronym (abbreviation) in COLUMN B.

COLUMN A	COLUMN B
a. RPF*APF*LPF=	1) APLHGR
	2) MCPR
b. Thermal limit established to ensure PEAK CLADDING TEMPERATURES do not exceed 2200 F in event of a design basis LOCA.	3) GEXL
	4) TPF
	5) LHGR
	6) PCIOMR
	7) DNB
c. Safety limit that is not analyzed at <10% power or <800 psia.	8) MAPRAT
	9) PCI
d. A condition of maximum heat flux which occurs when steam bubbles combine to form a vapor film over a heated surface.	

QUESTION 5.07 (1.00)

FILL IN THE BLANKS.

Following a LOCA, ___(a)____ is the primary means of heat transfer from an uncovered fuel rod to a cooler fuel rod.

___(b)____ is the process of transferring heat between a fluid and a surface by the circulation or mixing of a fluid.

When heat is applied to a homogeneous material, the _____(c)_____ of the material's atoms is increased.

A _____(d)______ must exist in order for heat to be transferred through a material.

QUESTION 5.08 (2.00)

a. LIST TWO (2) conditions that are used to determine WHEN the reactor is critical during a startup. (1.0)

b. WHICH ONE (1) of the following statements DEFINES REACTIVITY ANOMALY ? (1.0)

1. An unexpected change in critical rod configuration due to normal Xenon transients.

2. A change in the critical rod configuration due to the buildup of Samarium in a new core.

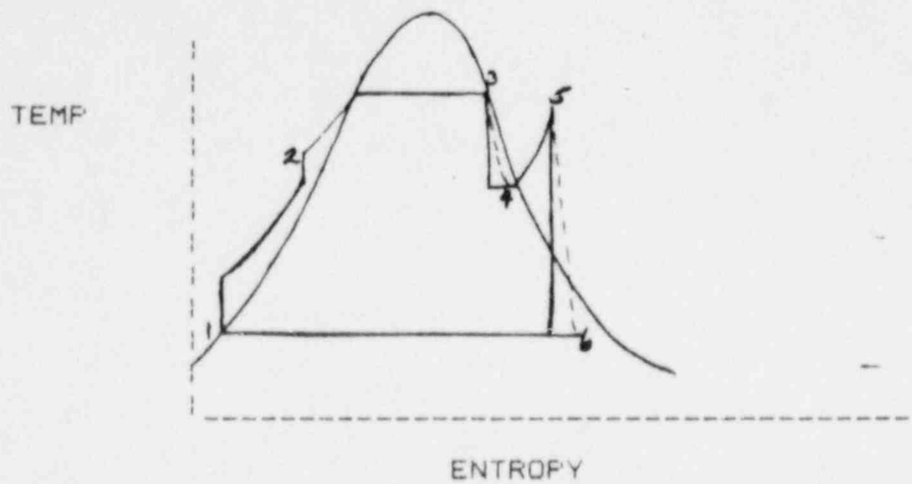
3. A reactivity equivalent of the difference between actual and expected rod configuration during steady state reactor power operation.

4. An unexpected change in reactor power induced via a reactivity change resulting from a plant transient.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.09 (1.50)

Utilizing the BWR Steam Cycle Diagram below, MATCH the line segments in COLUMN A with the associated plant component in COLUMN B.



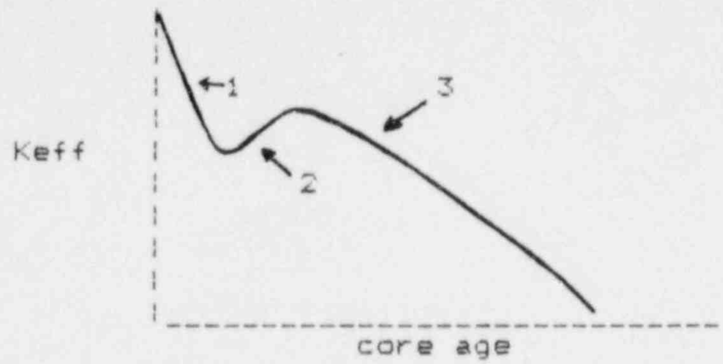
COLUMN A	COLUMN B
1-2	Low Pressure Turbine
2-3	Condenser
3-4	Pumps (condensate, booster, RFP's)
4-5	Heaters and Reactor
5-6	High Pressure Turbine
6-1	Moisture Separator (HP Turbine exhaust)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (1.50)

EXPLAIN WHY the curve of Keff vs. Core Age changes as it does at;

a. The DECREASE at point 1.
b. The TURNING UP at point 2.
c. The TURNING DOWN at point 3.



(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.11 (1.50)

A reactor scram occurs following several weeks of full power operation. You are the reactor operator performing the startup 9 hours later.

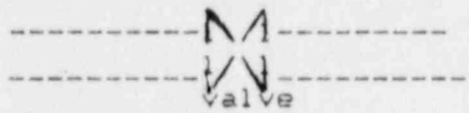
LIST TWO (2) concerns that you might have about reactor control and behavior when pulling rods, and EXPLAIN the reasons for your answers.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.12 (1.00)

Consider a high fluid velocity piping system with a pump running.

flow ->



a. EXPLAIN what would happen to the system piping if the valve were to suddenly (instantly) close. (.75)

b. Can a similar effect be achieved by suddenly OPENING the valve? (.25)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.13 (2.00)

The attached FIGURE (1) represents parameter changes for a plant transient on UNIT TWO. Use this figure and t'¬ following information to answer the questions below.

(1) Initial Power Level = 100 %
(2) The Main Turbine Generator Trips without the Bypass Valves opening
(3) No operator actions are taken

EXPLAIN the specific cause(s) of the following indications:

. The INCREASE of Reactor Power (point 1)
. Core Flow DECREASE (point 3)
c. Reactor Vessel Level DECREASE (point 6)
d. The INCREASE in Feedwater Flow (point 8)

QUESTION 5.14 (2.00)

The attached FIGURE (2) represents parameter changes for a plant transient on UNIT TWO. Use this figure and the following information to answer the questions below.

(1) Initial Reactor Power = 80 %
(2) The Master Recirc Controller Fails to MAXIMUM
(3) No operator actions are taken

EXPLAIN the specific cause(s) of the following indications:

a. The DECREASE in Reactor Power (point 2)
b. Reactor Water Level DECREASES (point 3)
c. Core Flow DECREASES (point 4)
d. Reactor Steam Flow DECREASES (point 6)

QUESTION 5.15 (1.00)

CALCULATE the equilibrium neutron count rate in a subcritical reactor after FOUR (4) generations given the following initial conditions:

Source = 100cps
Keff = .2
Assume generation 0 consists of only source neutrons and equilibrium is achieved after four generations

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.16 (1.50)

During a reactor shutdown tne reactor vessel pressure was decreased from 800 psig to 350 psig in 45 minutes.

a. CALCULATE the cooldown rate. (SHOW ALL WORK) (1.0)

b. STATE the technical specification maximum allowable cooldown rate. (.5)

QUESTION 5.17 (1.00)

a. DEFINE NPSH (Net Positive Suction Head).

b. EXPLAIN why operation of the recirculation pumps is not allowed with insufficient NPSH.

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (1.00)

Which ONE of the following conditions will NOT result in a shutdown of the SBGT System.

a. Manual shutdown

b. High temperature 225 deg F charcoal bed

c. High temperature 180 deg F heater inlet

d. Overloads in local control panel

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.02 (1.00)

The plant is operating normally at power when Pump A Controlled Leakage (FS "A") alarms LO (0.1 gpm) and you note an INCREASE in No.2 Recirc Pump seal pressure with NO CHANGE in No. 1 seal pressure. Which ONE of the following failures would cause these indications?

a. Failure of No. 1 seal

b. Failure of No. 2 seal

c. Plugging of the No. 1 internal restricting/breakdown orifice

d. Plugging of the No. 2 internal restricting/breakdown orifice

NOTE: NO OTHER ALARMS ARE PRESENT
FIGURE (3) IS ENCLOSED FOR REFERENCE

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (1.00)

The UNIT ONE Vital AC Power 120/240 V Distribution Cabinet 2A is normally supplied from 600 V Bus 2D through a Battery Charger and a Static Inverter. Which ONE of the following most accurately describes the response to the static inverter failing.

a. The power supply will automatically transfer to the alternate 600 V Bus 2C / Vital AC Transformer 2A.

b. The 125 VDC battery will maintain power to the Vital AC Cabinet for up to 5 hours.

c. The power supply can be manually transferred to the alternate 600 V Bus 2C / Alternate Static Inverter by depressing a a transfer pushbutton.

d. The power supply can be manually transferred to the alternate 600 V Bus 2C / Vital AC Transformer 2A by positioning the transfer switch to "Alternate".

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.04 (1.50)

Answer EACH ONE of the folowing questions concerning the ADS, TRUE OR FALSE.

a. On loss of power, the "B" logic circuit will automatically shift to the alternate 125 VDC power supply.

b. ADS is required to be operable when reactor pressure is greater than 150 psig.

c. The ADS Timer Reset Pushbutton will stop ADS permanently until the ADS Logic is reset.

QUESTION 6.05 (2.25)

ANSWER the following questions concerning PUMPS associated with the Condensate/Feedwater system.

a. MATCH the Pumps in column A with the Pump Trips in column B:
NOTE: EACH PUMP MAY HAVE MORE THAN ONE ANSWER (1.75)

COLUMN A	COLUMN B
Cond. Booster Pumps	1. Low suction Pressure of 34 psig
Reactor Feed Pumps	2. Low Condenser vacuum - 22.5"
Cond. Pumps	3. Low suction pressure 18" Hg
CRD Pumps	4. Low Oil Pressure of 5 psig
	5. Mechanical Overspeed - 110%
	6. Low Hotwell Level < 39"
	7. LOCA load shed

b. Why is there a 50 sec T.D. on the auto start of a Standby Condensate pump if any condensate pump is tripped coincident with a LOCA signal ? (.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.06 (.50)

Answer the following questions concerning the Core Spray sparger line break detection DP instrumentation:

a. DELETED

b. Is the normal (sparger intact) indication at rated power positive or negative with respect to zero ? (.25)

c. Is it normal for the A loop and B loop indications to have different values when operating at rated power ? (.25)

QUESTION 6.07 (1.00)

LIST the signals that will cause automatic isolation of the Off Gas System. (1.0)

QUESTION 6.08 (2.00)

Concerning the Reactor Manual Control System:

With the Refuel Platform over the core and the Reactor Mode Switch in Refuel, LIST FOUR (4) refueling interlocks which will result in a control rod withdrawal block. (BE SPECIFIC)

QUESTION 6.09 (2.00)

LIST FOUR (4) CONDITIONS that will AUTOMATICALLY TRIP a Diesel Generator at ANYTIME.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.10 (2.00)

Answer the following questions concerning the CRD System:

a. LIST the two (2) causes of a CRD Accumulator Trouble alarm (Setpoints NOT required) and EXPLAIN the action which must be taken to determine the cause. (1.0)

b. Shortly after resetting a reactor SCRAM, it is reported that Cooling Water flow is LOW; however, the CRD flow indicator is reading FULL SCALE. EXPLAIN this apparent discrepancy. (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.11 (1.ØØ)

Answer the following questions concerning Primary and Secondary Containment Systems for Unit 1:

a. STATE the Bases for inerting the Drywell and the Torus. ([illegible].Ø)

b. DELETED

QUESTION 6.12 (2.00)

With regards to the Main Steam Line Radiation Monitoring system:

LIST FOUR (4) automatic actions that occur when the trip setpoint of the MSL radiation monitor is reached.

QUESTION 6.13 (3.00)

a. LIST THREE (3) ways that the Rod Block Monitor (RBM) may be bypassed on UNIT 2. (1.50)

b. Discuss how the LPRM'S input into the RBM CHANNELS (number and level A,B,C,D) when a control rod near the center of the core is selected. (1.0)

c. Why are the level "A" detectors not utilized by the RBM SYSTEM ? (.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.14 (3.00)

With UNIT ONE operating at 100 % power, recirc in Master Manual, an operator inadvertently DECREASES the " PRESSURE SET " by 5 psi. DESCRIBE the INITIAL response and FINAL status of the following parameters due to this action. Briefly EXPLAIN the reason for EACH of your answers.
NOTE: ASSUME NO OPERATOR ACTIONS
FIGURE 9.4(12) IS ENCLOSED FOR REFERENCE

a. TCV position

b. BPV position

c. Power

d. Pressure

QUESTION 6.15 (1.00)

The RCIC Barometric Condenser maintains a vacuum via steam condensation by nozzle spray and a condenser vacuum pump. If the condenser vacuum pump were to fail, could steam condensation maintain a vacuum indefinitely ? EXPLAIN your answer.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.16 (1.00)

ANSWER the following questions concerning the MSIV Leakage Control System:

a. STATE the purpose of the MSIV-LCS.

B. EXPLAIN why Unit One does not have this system.

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (1.00)

The following parameter changes/annunciators are observed by the Reactor Operator:

RBCCW Temperature LOWER than normal
Low DP alarm (RBCCW to PSW)
RBCCW Surge Tank level INCREASING

WHICH ONE (1) of the following malfunctions would cau

a. RBCCW leak in the Drywell
b. Reactor Coolant leak into RBCCW via the NRHX
c. PSW leak in the RBCCW Heat Exchanger
d. RBCCW Fill Valve (F054) leakage into the RBCCW Surge Tank.

Ø 1 2 3 4 5 6 7 8 9 : ;

QUESTION 7.02 (1.50)

ANSWER the following questions concerning the Condenser Tube Leak abnormal operating procedure 34AB-OPS-028-2. (UNIT 2)

a. LIST TWO (2) ANNUNCIATOR ALARMS that would indicate a condenser tube leak. (.5)

b. If feedwater conductivity cannot be maintained less than .2 umhos/cm, your immediate actions should be to: (Select one answer) (1.0)

1. Reduce load as necessary until affected Hotwell Water Box is isolated.

2. Perform a normal shutdown within 8 hours.

3. Scram the Reactor and trip the RFPTs, Booster and Condensate Pumps.

4. Reduce power below 30 % ; be in a Hot Standby Condition within 24 hours if the leak cannot be isolated.

QUESTION 7.03 (2.00)

ANSWER EACH ONE of the following questions TRUE or FALSE concerning the Rod Sequence Control System:

a. The Group Notch Control Page self-test function is performed prior to reactor startup and before reducing reactor power below 30% .

b. RSCS restricts rod movement to four (4) notches from all other rods in selected group from 50 % rod density to a preset power level.

c. When an operator prepares to withdraw control rods for reactor startup the Sequence Mode Selector Switch must be set to NORMAL.

d. RSCS stops enforcing group notch constraints at 30% power; all rods in the selected group remain backlighted even when all rods are at the same notch position.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (1.00)

Determine if the following statements are TRUE or FALSE.

a. Control Room computer printouts may be used for scratch paper once the operator has determined the printout data is of no importance.

b. Chart recorders are required to be checked within one hour of shift change and marked with date, time, and operator initials.

QUESTION 7.05 (2.50)

ANSWER EACH ONE of the following with regards to the Primary Containment on UNIT ONE:

a. During a reactor plant startup, WHEN must the Oxygen concentration be less than 4% ? (1.0)

b. Upon increasing temperature of the suppression pool, STATE the temperature at which a Technical Specification Limiting Condition for Operation is first entered. (.5)

c. The Reactor shall be scrammed if Suppression Pool temperature reaches ___?____ . (.5)

d. During Reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the suppression pool temperature reaches ___?___ . (.5)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (2.50)

a. LIST FOUR (4) separate visual indications used to confirm correct orientation of a fuel bundle in the core. (2.0)

b. Briefly DESCRIBE how a bundle is checked for proper seating onto the fuel support piece. (.5)

QUESTION 7.07 (1.00)

The EOP Flowcharts direct an operator to the appropriate End Path Manual following any reactor scram. LIST TWO (2) cases (exceptions) where flowcharts direct the operator to a Normal Operating Procedure.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.08 (1.50)

Excluding Reactor Power decrease, LIST SIX (6) different indications that should be observed in the control room following SLC initiation.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (2.00)

In accordance with UNIT TWO procedure 34SO-B21-00102S,"Automatic Depressurization System and Low-Low Set System", the ADS may be initiated manually only if four conditions exist. LIST these FOUR (4) CONDITIONS.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (1.00)

STATE the TWO (2) alternative methods of scramming the Reactor if a Manual Scram was not possible PRIOR to evacuation of the control room.

QUESTION 7.11 (1.50)

LIST THREE (3) INDICATIONS which would be observable at the Rod Control Panel upon failure of the RPIS while at 20% power.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.12 (2.00)

LIST the IMMEDIATE OPERATOR ACTIONS for a loss of Plant Instrument and Service Air (34AB-OPS-020-2/1S). If differences exist between the procedures of UNIT 1 and UNIT 2, be sure to include actions for BOTH UNITS.

QUESTION 7.13 (1.00)

EXPLAIN what requirements must be met to close-out an LCO.

QUESTION 7.14 (1.50)

ANSWER the following questions concerning the Reactor Water Cleanup operating procedure (34SO-G31-003-2S).

a. STATE WHEN (under what conditions) the RWCU system can be operated without CRD Seal Purge flow. (.5)

b. EXPLAIN WHY the Demineralizer Bypass Valve (2G31-F044) must remain throttled open during operation of the RWCU system. (Assume a normal system lineup with the Reactor operating at power.) (1.0)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.15 (1.00)

With the Reactor operating at 80 % power a single (one) Recirculation pump TRIPS. The affected pumps discharge valve is required to oe closed by the subsequent operator actions of 34AB-OPS-032-2S. This same valve is then required to be throttled open within 5 MINUTES of the time it went closed. ANSWER the following questions:

a. EXPLAIN the reason for initially closing the pump Discharge Valve.

b. EXPLAIN WHY the Dischaige Valve is required to be Throttled open within 5 minutes.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.16 (2.50)

For EACH ONE of the following sets of plant conditions, IDENTIFY the EOP flowpath (BY NUMBER) specifically designed to address those conditions.

a. Reactor transients or failure of vital equipment while in hot standby of startup.

b. High radiation, loss of vital power, failure of vital equipment, or stuck open relief valve.

c. High radiation, loss of coolant, and loss of primary containment integrity.

d. Reactor transients or failure of vital equipment while in the RUN mode.

e. Failure of reactivity control systems.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

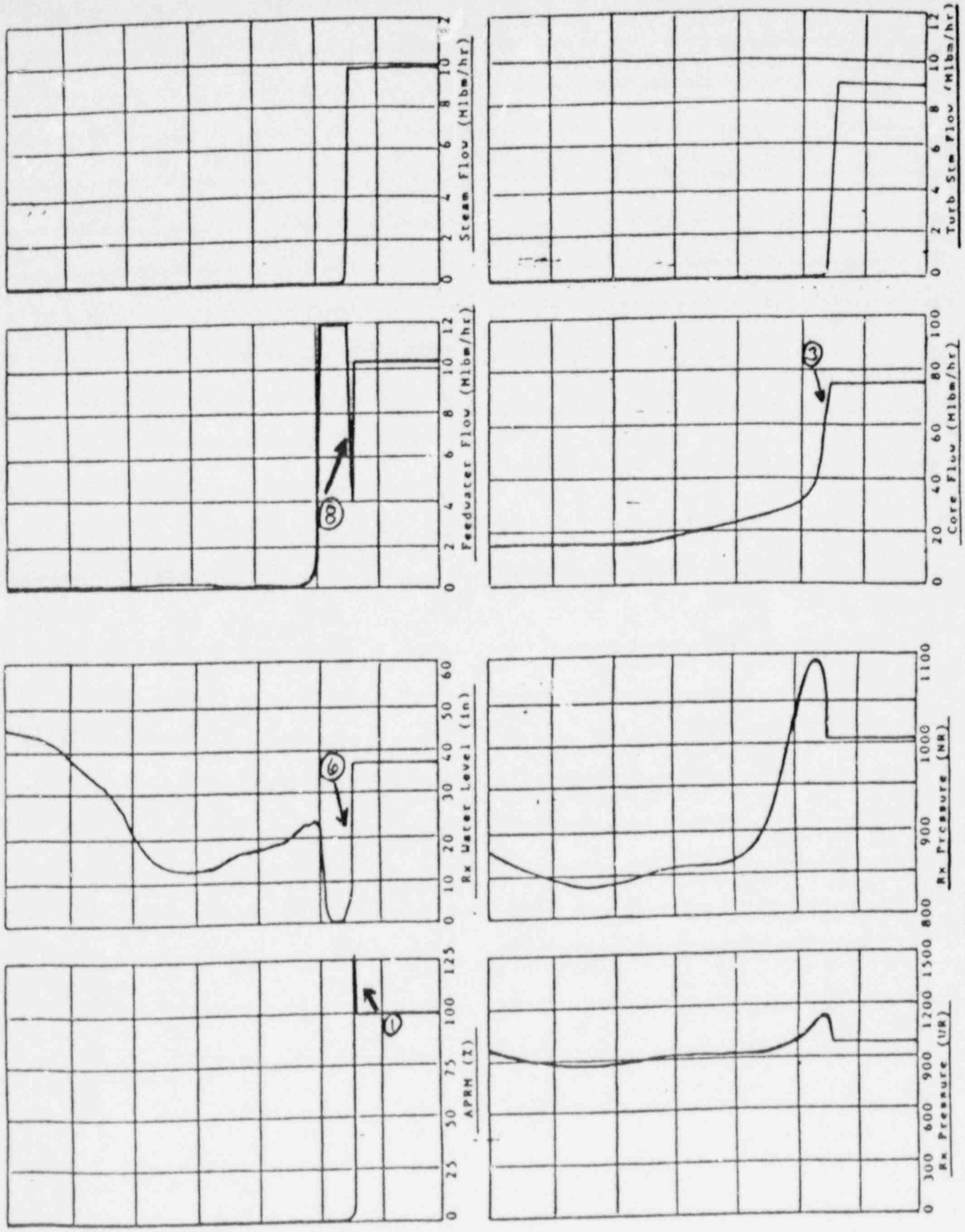
QUESTION 7.17 (2.50)

a. STATE the exposure rate limits (for a major portion of the body) which characterize EACH ONE of the following: (1.5)

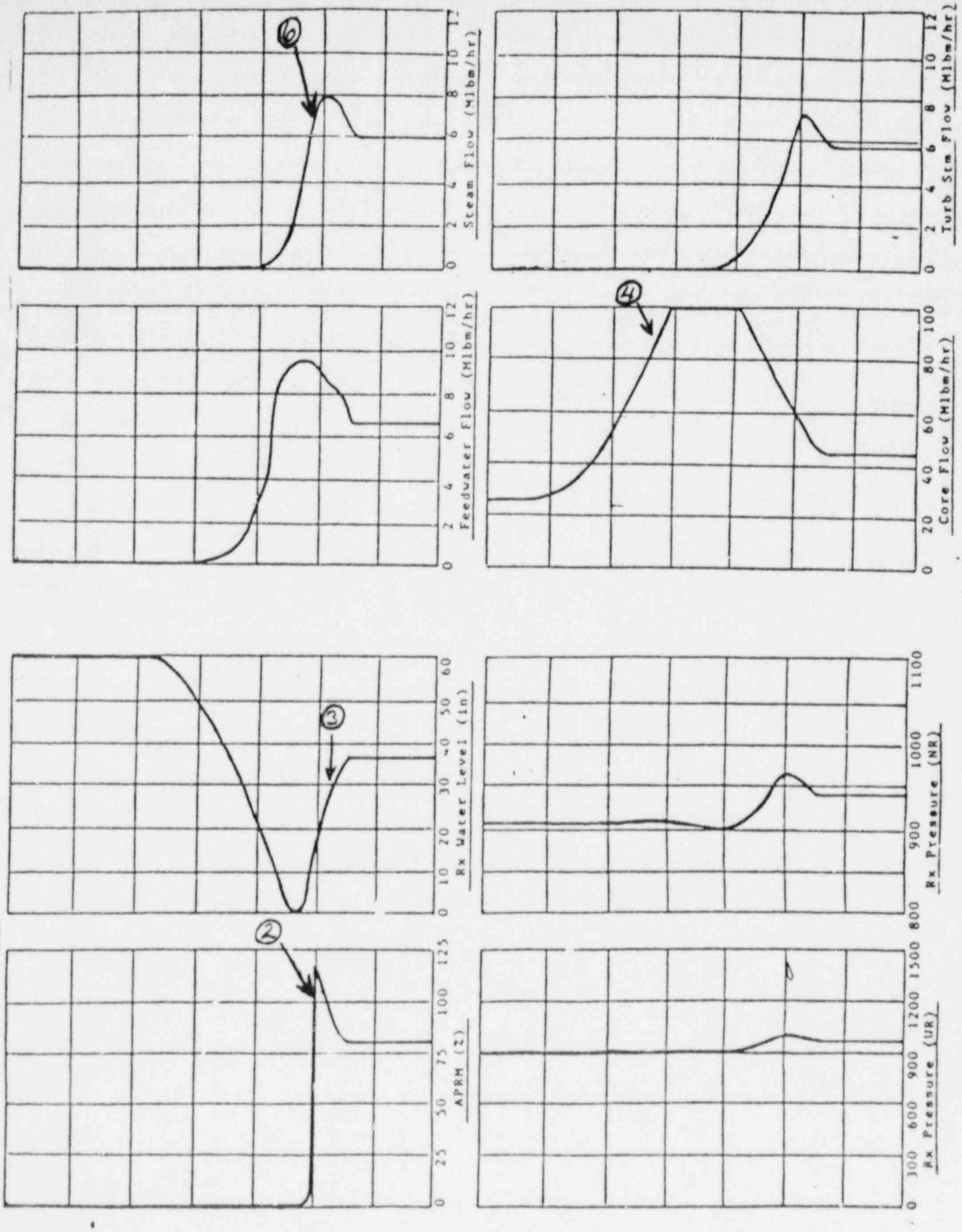
1. RADIATION AREA
2. HIGH RADIATION AREA
3. LOCKED HIGH RADIATION AREA

b. STATE the definition of extremities as it pertains to radiation exposure of personnel. (1.0)

(***** END OF CATEGORY 07 *****)
(************* END OF EXAMINATION ***************)



TURBINE TRIP WITHOUT BYPASS
FIGURE 1



MASTER RECIRC CONTROLLER FAILS TO MAXIMUM
FIGURE 2

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = UA\Delta T$

$Pwr = W_f\,\dot{m}$

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

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

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\lambda_{eff}\rho]$

$T = \ell^*/(\rho - \bar{\beta})$

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

$\rho = (K_{eff}-1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\rho = [\ell^*/TK_{eff}] + [\bar{\beta}/(1 + \lambda_{eff}T)]$

$P = \Sigma\phi V/(3 \times 10^{10})$

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

$\text{Cycle efficiency} = \frac{\text{Net Work (out)}}{\text{Energy (in)}}$

$A = \lambda N \qquad A = A_o e^{-\lambda t}$

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

$t_{1/2}(eff) = \frac{(t_{1/2})(t_b)}{(t_{1/2} + t_b)}$

$I = I_o e^{-\Sigma x}$

$I = I_o e^{-\mu x}$

$I = I_o\,10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1\ \text{seconds}^{-1}$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$R/hr = (0.5\ CE)/d^2\ (\text{meters})$

$R/hr = 6\ CE/d^2\ (\text{feet})$

WATER PARAMETERS

1 gal. = 8.345 lbm

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg

1 ft. H_2O = 0.4335 lbf/in^2

MISCELLANEOUS CONVERSIONS

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 BTU/hr

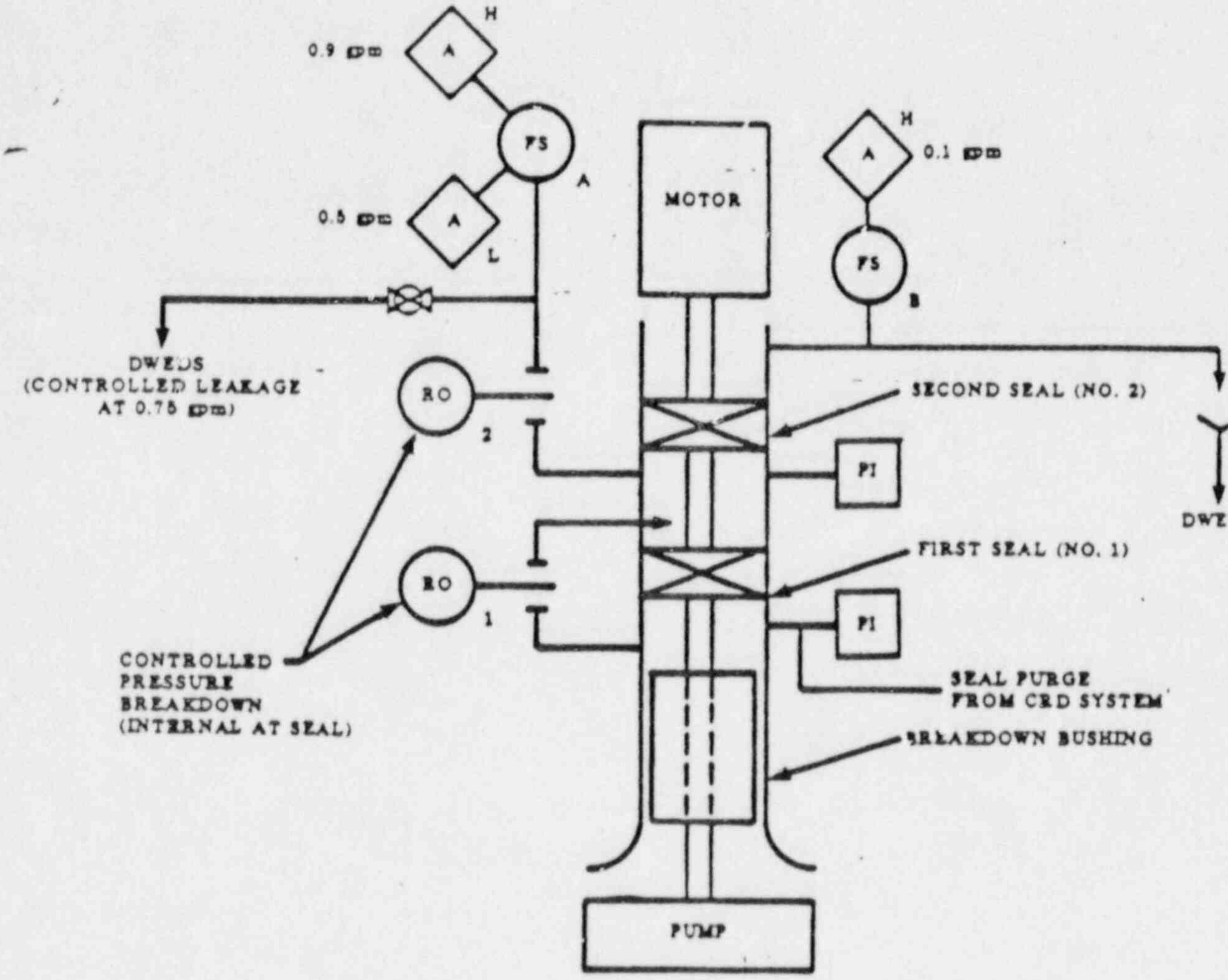
1 Mw = 3.41×10^6 Btu/hr

1 Btu = 778 ft-lbf

1 inch = 2.54 cm

$°F = 9/5\,°C + 32$

$°C = 5/9\,(°F - 32)$



FAILURE OF SEAL NO. 1 ONLY:	NO. 2 SEAL PRESSURE WOULD APPROACH NO. . SEAL PRESSURE. LEAKAGE THRU NO. 2 ORIFICE WILL GO TO ~ 1.1 gpm AND FS "A" WILL ALARM HI AT > 0.9 gpm.
FAILURE OF SEAL NO. 2 ONLY:	NO. 2 SEAL PRESSURE WOULD DROP DEPENDENT UPON MAGNITUDE OF FAILURE. LEAKAGE THRU FS "B" WOULD EXCEED 0.1 gpm AND ALARM HI.
FAILURE OF BOTH SEALS:	TOTAL LEAKAGE OUT OF THE SEAL ASSEMBLY WOULD APPROACH 60 gpm AS LIMITED BY THE BREAKDOWN BUSHING. BOTH FS "A" AND FS "B" WOULD ALARM HIGH. PRESSURE IN BOTH SEALS WOULD DROP DEPENDING UPON MAGNITUDE OF FAILURE. (NO. 1 PRESSURE MIGHT NOT DROP SIGNIFICANTLY UNLESS FAILURE WAS LARGE.)
PLUGGING OF NO. 1 INTERNAL "RO":	NO. 2 PRESSURE WOULD GO TOWARD ZERO AND FLOW THRU FS "A" WOULD APPROACH ZERO AND ALARM LOW AT 0.5 gpm.
PLUGGING OF NO. 2 INTERNAL "RO":	NO. 2 SEAL PRESSURE WOULD APPROACH NO. 1 SEAL PRESSURE. CONTROLLED LEAKAGE WOULD APPROACH ZERO AND ALARM LOW AT 0.1 gpm.

Figure (3)

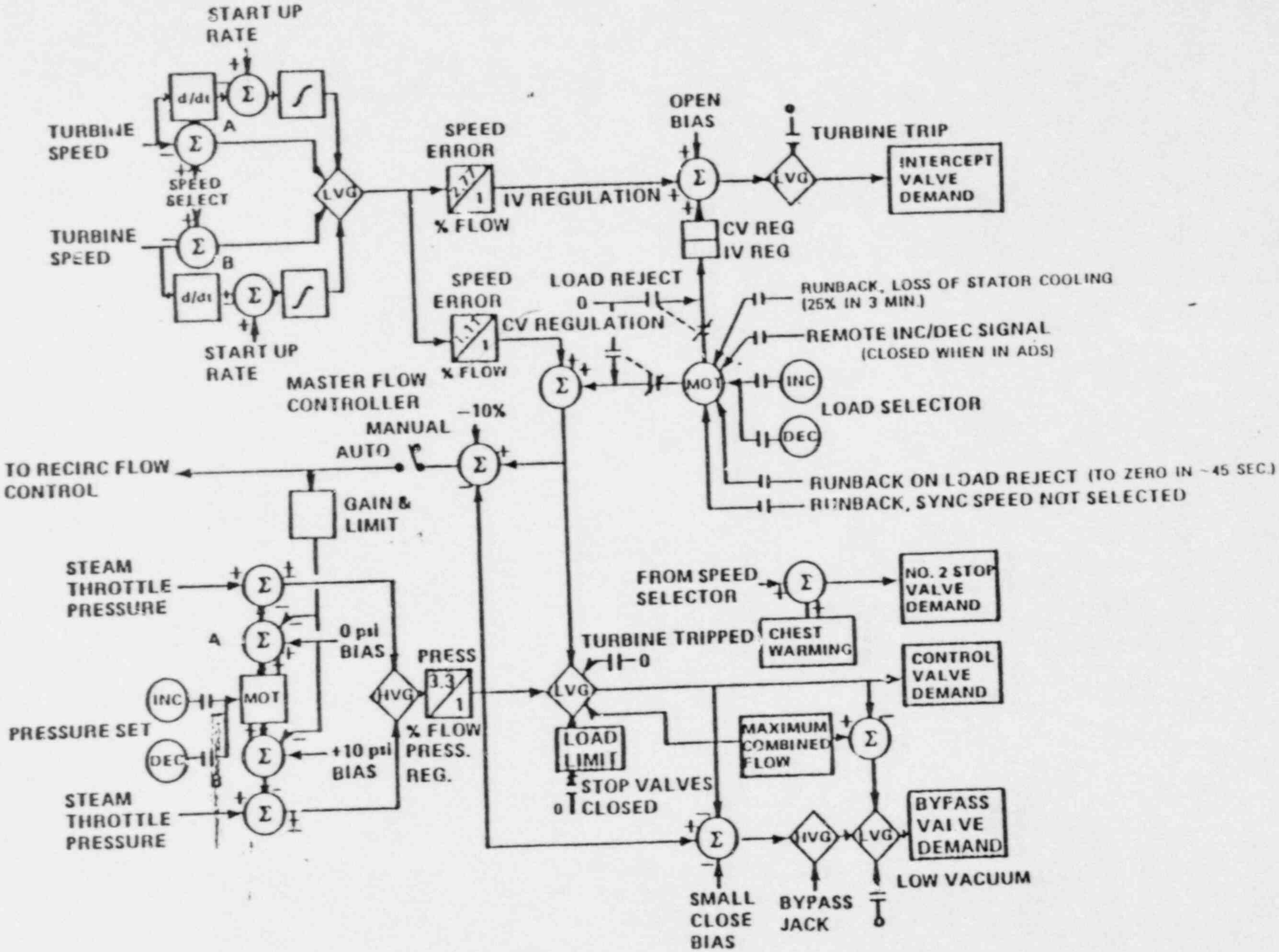


Figure 9.4(12) EHC Logic

ANSWER 5.01 (2.00)

a. True (.5 each)
b. True
c. False
d. True

REFERENCE
G.E. REACTOR THEORY. Chpt 4, LO 1.5,3.6, Chpt 7, LO 5.6,8.3
3.1/3.2 , 3.3/3.4 , 3.5/3.6
292008K119 292008K120 292008K122 ...(KA'S)

ANSWER 5.02 (2.00)

a. Falce
b. True
c. True
d. True (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 4, LO 1.5,3.6,4.3,6.3, Chpt 7, LO 5.6
2.5/2.6 , 2.1/2.2 , 2.5/2.6 , 1.9/2.12
292004K102 292004K109 292004K111 292004K113 ...(KA'S)

ANSWER 5.03 (2.00)

a. True
b. False
c. False
d. False (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt. 5, LO 2.5
2.5/2.6 , 2.6/2.9
292005K109 292005K112 ...(KA'S)

ANSWER 5.04 (2.00)

a. True
b. False
c. True
d. True (.5 each)

REFERENCE
G.E. HTFF, Chpt 8, LO 9.1,9.2 9.4,9.5
2.4/2.6
293008K133 ...(KA'S)

ANSWER 5.05 (2.00)

a. 7
b. 4
c. 6
d. 2 (.5 each)

REFERENCE
G.E. Reartor Theory, Chpt 1, LO 4.1,5.1,6.1
2.7/2.6 , 3.2/3.5 , 2.4/2.6 , 3.2/3.3
292002K108 292002K109 292002K110 292002K111 ...(KA'S)

ANSWER 5.06 (1.00)

a. TPF (4)
b. APLHGR (1)
c. MCPR (2)
d. DNB (7) (.25 each)

REFERENCE
G.E. HTFF,Chpt 8, LO 2.7, Chpt 9, LO 1.1,4.1,5.1
2.2/2.6 , 3.3/3.7 , 2.8/3.6 , 3.0/3.2
293008K109 293009K104 293009K110 293009K119 ...(KA'S)

ANSWERS -- HATCH 1&2 -88/02/08-HOPPER, G.

ANSWER 5.07 (1.00)

a. radiation
b. convection
c. kinetic energy
d. temperature gradient (^ T) (.25 each)

REFERENCE
G.E. HTFF, Chpt 7, LO 1.1,1.4,1.5
3.2/3.2
293007K101 ...(KA'S)

ANSWER 5.08 (2.00)

a.(Increasing power), stable positive period, no rod motion (.5 each)

b. 3 (1.0)

REFERENCE
G.E. Reactor Theory, Chpt. 7, LO 2.2,9.1
EIH: 34GO-OPS-001-2
4.3/4.3
292008K105 ...(KA'S)

ANSWER 5.09 (1.50)

1-2 Pumps (condensate,booster,feedwater)
2-3 Heaters and Reactor
3-4 High pressure turbine
4-5 Moisture Separator HP Turbine Exhaust
5-6 Low Pressure Turbine
6-1 Condenser (.25 each)

REFERENCE
G.E. HTFF, Chpt 5, LO 6.1
2.0/2.2
293002K105 ...(KA'S)

ANSWER 5.10 (1.50)

1. Fission product poison buildup.
2. Burnable poisons depletion is greater than fuel burnout. (FP buildup is approx. complete)
3. Fuel depletion overcomes the effect of poison burnout. (.5 each)

REFERENCE
G.E. Reactor Theory, Chpt 5, Chpt 6, LO 4.3
2.4/2.7
292007K103 ...(KA'S)

ANSWER 5.11 (1.50)

1) Reactivity effects of Peak Xenon during startup. (.25)

Increased neutron flux during startup will cause an increase in the burnup of Xenon. Depletion of Xenon's poison effects will lead to a positive reactivity insertion. (.50)

2) Abnormal flux distribution. (High worth peripheral rods) (.25)

Xenon concentration will be highest in regions of the core where neutron flux was the highest during the previous operational phase. Thermal neutron flux will be depressed in this region, and pushed toward prev'ously low power regions (.25). (Peak thermal neutron flux n ually occurs in the central regions of the core during startup.) High Xenon concentration causes the flux to be pushed to the periphery of the core, where previously low worth control rods now become high-worth control rods (.25).

REFERENCE
G.E.Reactor Theory, Chpt 6, LO 2.5,2.7
3.1/3.2
292006K114 ...(KA'S)

ANSWER 5.12 (1.00)

a. (The piping system would undergoe a fluid shock caused by the rapid change in flow) - FLUID HAMMER (water hammer)(.5) - which could result in extensive damage to the system.(.25)
b. YES. (.25)

REFERENCE
G.E. HTFF Chpt 6, LO 8.1
3.2/3.3
293006K105 ...(KA'S)

ANSWER 5.13 (2.00)

a. Increase due to pressure increase
b. Decrease due to RPT trip caused by Turbine Trip above 30% power. (.5 each)
c. Decrease due to void collapse on pressure increase.
d. Increase due to level drop (on scram).

REFERENCE
EIH: LP 200.1, VOL 8 CH 10.4
3.8/3.9 3.7/3.8 3.5/3.6
295005AA20 ...(KA'S)

ANSWER 5.14 (2.00)

a. Decrease due to Scram (.5)
b. Decrease due to rush of Downcomer water into Recirc Suction (.5)
c. Decrease due to Speed Limiter # 1 on feedflow decrease to less than 20 % (.5)
d. Decrease due to scram (.15) and EHC controlling reactor pressure (.35)

REFERENCE
EIH: LP 200.1, VOL 8 CH 10.4
3.8/4.0 3.3/3.6 4.1/4.1 3.5/3.6
202001A205 202001K102 202001K122 202001K416 ...(KA'S)

ANSWER 5.15 (1.00)

S/1-Keff 100/(1-.2) = 125

or

	fission	source	total
0	0	100	100
1	20	100	120
2	24	100	124
3	25	100	125
4	25	100	125

REFERENCE
G.E. Reactor Theory, Chpt 3, LO1.2,1.5
2.9/3.0 , 2.1/2.3
292003K101 292003K102 ...(KA'S)

ANSWER 5.16 (1.50)

a. 800 psig = 814.7 psia = 520.3 deg F (.25)
350 psig = 364.7 psia = 435.5 deg F (.25)
520.3 - 435.5 = 84.8 deg F (.25)
c/d rate = [84.8 deg/ 45 min]*60 min/hr = 113.. deg / hr (.25)

b. 100 deg F / HR (.5)

REFERENCE
Steam Tables
2.8/3.1
293003K123 ...(KA'S)

ANSWER 5.17 (1.00)

a. NPSH is defined as the difference between total pressure at the eye of a pump (or inlet of a valve) and saturation pressure.
or
NPSH= Pi - Psat (.5)

b. Cavitation would result in vibration (and noise) of the pump and pitting and corrosion of the pump parts,(especially the impeller). (.5)

REFERENCE
G.E. HTFF, Chpt 6, LO 10.8,10.9
2.7/2.8
293006K110 ...(KA'S)

ANSWER 6.01 (1.00)

c

REFERENCE
EIH: VOL 5 CH 3.3, LP 30.1 LO 4
3.0/3.1 3.8/3.8
261000A205 288000A301 ...(KA'S)

ANSWER 6.02 (1.00)

d

REFERENCE
BFNP: LP#7,P. 28
EIH: VOL 5 CH 4.1, LP 4.1 LO 24
GGNS SD B33-1, pp 5, 6; OP-B33-1-501, p 5; ARI B33-FAL-L603A
3.3/3.3
202001A109 ...(KA'S)

ANSWER 6.03 (1.00)

d

REFERENCE
EIH: VOL 6 CH 7.4, LP 27.1 LO 20
3.6/3.9 3.4/3.7
262001A401 262001K406 ...(KA'S)

ANSWER 6.04 (1.50)

a. False
b. True
c. False (.5 each)

REFERENCE
EIH: LT-IH 03801-01 EO 4,14,6
3.4/3.6 4.2/4.3 3.7/3.9
218000G001 218000K401 218000K606 ...(KA'S)

ANSWER 6.05 (2.25)

a. Cond. Booster Pumps - 1,4
Reactor Feed Pumps - 2,5
Cond. Pumps - 6 (.25 each)
CRD Pumps - 3,7

b. Prevents overloading of the startup transformer (2D) when the ECCS pumps are auto starting. (.5)

REFERENCE
EIH: VOL 6 CH 5.3, VOL 5 CH 4.2, LP 2.1 LO 4,7,8 LP 1.1 LO 7
2.8/2.8 3.3/3.3 3.4/3.4 3.2/3.3
256000A201 256000K302 256000K401 256000K403 ...(KA'S)

ANSWER 6.06 (.50)

a. DELETED

b. Negative (.25)

c. Yes (.25)

REFERENCE
BSEP: SSM, Core Spray, P.8 Lesson Objective 9
EIH: LT-IH-00801-00, 34AR-601-055-2, TO 200.64
3.0/3.2 2.4/2.6 2.8/3.0
209001K113 209001K404 209001K502 ...(KA'S)

ANSWER 6.07 (1.00)

Any combination of "Inop", "Downscale", or "Triple-HIgh" (.25 each) in both post treatment radiation monitor trip channels (.25).

REFERENCE
EIH: LP 31.1 LO 2, VOL 6 CH 6.8
3.1/3.3
271000K408 ...(KA'S)

ANSWER 6.08 (2.00)

a. Refuel Platform Grapple not full up
Fuel loaded on Refuel Platform Grapple
Fuel loaded on Frame Hoist
Fuel loaded on Trolley Hoist
Fuel Loaded on Service Platform Hoist
Selection of a second rod for movement with any other rod withdrawn

any 4 (.5 each)

REFERENCE
EIH: VOL 7 CH 9.2.1, VOL 6 CH 6.9, LP 1.2 LO 1,
3.5/3.5
201002K402 ...(KA'S)

ANSWER 6.09 (2.00)

Low lube oil pressure
Engine overspeed
Engine start failure
Differential Lockout (.5 each)

REFERENCE
EIH: VOL 6 CH 7.2, LP 2F.1 LO 1
4.0/4.2
264000K402 ...(KA'S)

ANSWER 6.10 (2.00)

a. Low Nitrogen Pressure (of 965 psig) (0.25)
High Water Level in the instrument block (of 60 ml) (0.25)

At the local control panel, the back lit button must be depressed. If the light goes out, the cause is water; if the light stays lit, the cause is gas pressure. (0.5)

b. The CRD FCV is downstream of the flow element. (0.25) All of the indicated flow is going through the Charging line to recharge the accumulators. (0.5) The sensed high flow is sending a signal to close the FCV, (and thus Cooling Water flow is low). (0.25)

REFERENCE
EIH: VOL 5 CH 4.2, LP 1.1 LO 10, 6
3.6/3.7

201001G007 ...(KA'S)

ANSWER 6.11 (1.00)

a. Reduce the concentration of oxygen to minimize the possibility of hydrogen combustion following a LOCA. (1.0)

b. DELETED

REFERENCE
BFNP: Lesson Plan 16, Objectives C, D, H, I, pp 1, 14, 26, & 32
EIH: LP 13.1 LO 4.20
3.7/3.8
223002G007 ...(KA'S)

ANSWER 6.12 (2.00)

Reactor Scram
Group I Isolation
Main control room ventilation swaps to pressurization mode
Mechanical vacuum pump trips and isolates
Gland seal exhauster trips (and isolates) (4 @ .5 each)

REFERENCE
EIH: VOL 7 CH 9.7, LP 14.1 LO 2
3.6/3.9 3.7/4.1 3.8/3.9
272000A301 272000K402 272000K403 ...(KA'S)

ANSWER 6.13 (3.00)

a. Power <27% UNIT 2
Selection of a peripheral rod
Manual operation of the RBM BYPASS joystick on P603 (.5 each)

b. One RBM channel will use two level "B" and "D" LPRM detectors and all four level "C" detectors. (.5) The other channel will use the remaining two "B" and "D" level detectors and the same four level "C" detectors. (.5)

c. The level "A" detectors are located at the bottom of the core. where transition boiling is not expected to occur.(.5) (Since the RBM protects against transition boiling, "A" level detectors are not necessary).

REFERENCE
EIH: VOL 7 CH 9.1.3, LP 12.3 LO 13
3.8/3.8 3.2/3.1 2.9/3.0 3.6/3.5
215002A304 215002G007 215002K102 215002K403 ...(KA'S)

ANSWER 6.14 (3.00)

INITIAL RESPONSE:
a. TCVs - Remain at 100% open
b. BPVs - Open (16.5%)
c. Power - Decreases
d. Pressure - Decreases (.25 each)

REASON: Above caused by PCU calling for approx. 115% steam flow (.5).
(950 - 91[illegible] x [illegible].3

FINAL STATUS:
a. TCVs - 100% position
b. BPVs - Shut
c. Power - Slightly lower
d. Pressure - Slightly lower (.25 each)

REASON: Above caused by the decrease in pressure and power causing BPVs to shut -- PCU cycling to new equilibrium state. (.5)
((945-915) x 3.3)

REFERENCE
EIH: VOL 7 CH 9.4, LP 19.1 LO 2,5,10
3.4/3.4 3.9/3.8 4.1/3.9 3.8/3.9
241000A101 241000A102 241000A114 241000K102 ...(KA'S)

ANSWER 6.15 (1.00)

NO. (.25)
The condenser would lose vacuum due to accummulation of noncondensible gases and could possibly overpressurize. (.75)

REFERENCE
EIH: VOL 5 CH 4.5, LP 39.1 LO 3
2.9/3.0 3.2/3.5
217000A209 217000K405 ...(KA'S)

ANSWER 6.16 (1.00)

a. The MSIV-LCS controls and minimizes the release of fission products (which could leak through the closed MSIV'S and bypass SBGT) following a LOCA. (.5)

b. Unit One does not have this system due to strict limits on MSIV leakage (11.5 SCFH per valve)(which meet 10 CFR off-site dose limits). (.5)

REFERENCE
EIH: LP 49.1 LO 1,9
3.2/3.3 2.6/3.7
239003G004 239003G006 ...(KA'S)

ANSWERS -- HATCH 1&2 -88/02/08-HOPPER, G.

ANSWER 7.01 (1.00)

c

REFERENCE
EIH: LP 9.1 LO 5

ANSWER 7.02 (1.50)

a. Powdex System Trouble
Contaminated Feedwater Alarm
Inlet High Conductivity
Polishing Demin High Conductivity
(any 2 @ .25 each)

b. 3 (1.0)

REFERENCE
EIH: LP 25.1 LO 4, 34AB-OPS-028-2N
3.7/3.8 3.1/3.9 2.8/3.1
256000A215 256000G011 256000G015 ...(KA'S)

ANSWER 7.03 (2.00)

a. True
b. False
c. False (.5 each)
d. True

REFERENCE
EIH: LT-IH-05402-01 LO 2.h,6a,7,10.c
3.5/3.2 3.1/3.1 3.5/3.7 3.9/3.9
201004A305 201004A402 201004G007 201004K103 ...(KA'S)

ANSWER 7.04 (1.00)

a. False
b. True (.5 each)

REFERENCE
EIH: 30AC-OPS-003-0S
3.2/3.4

ANSWERS -- HATCH 1&2 -88/02/08-HOPPER, G.

294001A106 ...(KA'S)

ANSWER 7.05 (2.50)

a. Within 24 hours subsequent to placing the reactor in the RUN MODE (following a shutdown). (1.0)
b. 95 deg F (.5)
c. 110 deg F (.5)
d. 120 deg F (.5)

REFERENCE
EIH: Technical Specifications 3.7.A.5 , 3.7
3.3/4.1
223001G005 ...(KA'S)

ANSWER 7.06 (2.50)

a. Channel Fastener in center of cell
Identification Lug points to center of cell
Spacer Buttons adjacent to control rod
Serial Numbers readable from center of cell
Cell to Cell symmetry
Location of Gad Rod end plugs
(any 4 @ .5 each)

b. A TV camera is mounted on the fuel grapple and lowered for visual inspection (.25)(to a position slightly above the fuel bail). The fuel grapple is lowered just above the bail handles and the core is scanned to ensure proper seating as indicated by the fuel grapple not striking any bail handles (.25)

REFERENCE
EIH: RO 1.5 LO 11, 42FH-ENG-012-2
3.0/3.7 3.1/3.3
234000G13 234000K505 ...(KA'S)

ANSWER 7.07 (1.00)

1. Scram signal was received while performing a scheduled test with the reactor shutdown. (.5)

2. Scram was initiated as part of a normal fast reactor shutdown procedure. (.5)

REFERENCE
EIH: LT-IH-20101-00 LO 11
4.1/4.0
295006G005 ...(KA'S)

ANSWER 7.08 (1.50)

1. Amber Squib-continuity lamp of both explosive valves extinguish.
2. Loss of continuity alarm annunciates.
3. RWCU outboard isolation valve closed
4. Selected SLC pump start light illuminates.
5. SLC discharge pressure increases.
6. SLC storage tank level decreases. (.25 each)

REFERENCE
EIH: MOD 1.3 EO 1.3.1.1
3.8/3.8 4.1/4.2 4.0/4.1 4.2/4.2
211000A303 211000A305 211000A306 211000A308 ...(KA'S)

ANSWER 7.09 (2.00)

1. ADS cannot initiate automatically (.5)
2. Reactor water level < -101 inches (.35) and is NOT INCREASING(.15).
3. Any low pressure injection system (Core Spray or RHR or Condensate system) is running (.5)
4. Reactor Pressure is > 350 psig (.5)

REFERENCE
EIH: 34SO-B21-001-2S, LT-IH-03801-01 EO 9.b
3.5/3.6 3.8/4.0 4.1/4.2
218000A204 218000G010 218000K402 ...(KA'S)

ANSWER 7.10 (1.00)

1. Open RPS Bus Breakers (CB3A,B) to Power Range Monitoring (APRM SCRAM)
2. Trip Mercoid Switches on Scram Discharge Instrument Volume Level Switches (HI-HI SCRAM) (Two switches on each side of Reactor Building cause full scram.)
3. Close scram air header isolation valve and vent the scram header.
4. Vent each CRD mechanism through its associated above piston vent valve.

(any 2 @ .5 each)

REFERENCE
EIH: LP 10.1 LO 9, 34AB-OPS-008-1(2)
3.8/3.9 4.1/4.2
295016AA10 295016AK30 ...(KA'S)

ANSWER 7.11 (1.50)

1. Loss of Four Rod Display
2. Rod Drift Indicator Light for all rods
3. Select rod Block
4. RWM Rod Block annunciator (if RWM in service)

(any 3 @ .5 each)

REFERENCE
EIH: 34AB-OPS-024-2
3.5/3.7 3.1/4.0 3.5/3.1 3.5/3.3
214000A303 214000G008 214000G011 214000G012 ...(KA'S)

ANSWER 7.12 (2.00)

1. If a Reactor Scram occurs, enter the EOP's. (.5)
2. Manually Scram the Reactor if any of the following occurs, then enter the EOP's. (.25)
   a. Indication of 4 or more control rods drifting into core. (.25)
   b. Scram Valve Pilot Air Header high/low pressure coincident with CRD HYD High Temp. (.25)
3. Maintain reactor water level between 15 and 45 inches (.25)
   a. Utilize LCV bypass as necessary on UNIT 2 (.25)
   b. Cycle the Startup Flow Control Isolation Valve and Feedwater Low Flow Control Bypass on UNIT 1 (.25)

REFERENCE
EIH: LT-IH-035W1-00 EO 16, 34AB-OPS-020-2
3.7/3.4 3.3/3.9 3.4/3.3
295019AK20 295019G005 295019G010 ...(KA'S)

ANSWER 7.13 (1.00)

(When a system or component is restored to an acceptable operable condition), the LCO can be cleared after all functional tests associated with each MWO (in the LCO) are confirmed to have been performed (.5) and the results are satisfactory. (.5)

REFERENCE
EIH: MOD 2.1 EO 2.1.3.2
3.9/4.5
2940001K10 ...(KA'S)

ANSWER 7.14 (1.50)

a. Operation is permitted if the CRD system is Out of Service (.25) and Reactor water temperature is less than 212 deg F (.25).
b. This valve provides a pressure relief path for CRD water (.5) in the event of a system isolation in which both demineralizers transfer to the HOLD condition (.5).

REFERENCE
EIH: 34SO-631-003-2S, LP 3.1 I.O 7.10
3.2/3.2 2.8/2.8 2.7/2.9 3.4/3.4

204000A213 204000G010 204000K116 204000K402 ...(KA'S)

ANSWER 7.15 (1.00)

a. Prevents reverse rotation of the affected pump. (.5)
b. Thermal Binding of the discharge valve may occur (.5)
(if the valve is not reopened within 5 minutes)

REFERENCE
EIH: 34AB-OPS-032-2S
3.2/3.2 3.6/3.7 3.5/3.7
202001A203 202001A223 202001G010 ...(KA'S)

ANSWER 7.16 (2.50)

a. 2
b. 4
c. 5
d. 3
e. 1
(.5 each)

REFERENCE
EIH: LT-IH-20101-00 LO 7
3.8/4.4 3.9/4.5 3.9/4.5 3.9/4.5
295006G012 295024G012 295025G012 295031G012 ...(KA'S)

ANSWER 7.17 (2.50)

a.
1. >5 mrem in any one hour or 100 mrem in 5 consecutive days
2. > 100 mrem in any one hour
3. > 1000 mrem/hr
(3 @ .5 each)
b. Hands and forearms, including the elbows, feet, ankles, and lower legs, including the knees. (equivalent answer accepted) (1.0)

REFERENCE
EIH: 60AC-HPX-004-0S, 10CFR20
3.3/3.8
294001K10[illegible] ...(KA'S)

QUESTION	VALUE	REFERENCE
05.01	2.00	GTH0000666
05.02	2.00	GTH0000658
05.03	2.00	GTH0000661
05.04	2.00	GTH0000669
05.05	2.00	GTH0000656
05.06	1.00	GTH0000659
05.07	1.00	GTH0000667
05.08	2.00	GTH0000662
05.09	1.50	GTH0000664
05.10	1.50	GTH0000660
05.11	1.50	GTH0000663
05.12	1.00	GTH0000668
05.13	2.00	GTH0000671
05.14	2.00	GTH0000672
05.15	1.00	GTH0000657
05.16	1.50	GTH0000670
05.17	1.00	GTH0000665
	27.00	
06.01	1.00	GTH0000679
06.02	1.00	GTH0000673
06.03	1.00	GTH0000685
06.04	1.50	GTH0000678
06.05	2.25	GTH0000681
06.06	.50	GTH0000676
06.07	1.00	GTH0000677
06.08	2.00	GTH0000684
06.09	2.00	GTH0000688
06.10	2.00	GTH0000674
06.11	1.00	GTH0000675
06.12	2.00	GTH0000680
06.13	3.00	GTH0000683
06.14	3.00	GTH0000686
06.15	1.00	GTH0000687
06.16	1.00	GTH0000682
	25.25	
07.01	1.00	GTH0000698
07.02	1.50	GTH0000703
07.03	2.00	GTH0000699
07.04	1.00	GTH0000705
07.05	2.50	GTH0000695
07.06	2.50	GTH0000690
07.07	1.06	GTH0000691
07.08	1.50	GTH0000693
07.09	2.00	GTH0000694
07.10	1.00	GTH0000696
07.11	1.50	GTH0000697

QUESTION	VALUE	REFERENCE
07.12	2.00	GTH0000702
07.13	1.00	GTH0000692
07.14	1.50	GTH0000700
07.15	1.00	GTH0000701
07.16	2.50	GTH0000704
07.17	2.50	GTH0000689
	28.00	
	80.25	

DOCKET NO 321

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: HATCH 1&2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/02/08

EXAMINER: PAYNE, C.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question 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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00 ~~26.00~~	100.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
25.00 ~~26.00~~		________ Final Grade	________%	Totals

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

____
Candidate's Signature

QUESTION 8.01 (1.00)

Which ONE (1) of the following describes the correct method for re-establishing the required SRM minimum count rate during Unit 2 core reload.

a. Spiral reload the core until 3 cps is established.

b. Load up to four new fuel assemblies next to each of the four SRMs to obtain the required count rate.

c. Prior to fuel reload, install neutron sources in the the source tubes to establish 3 cps.

d. Place offloaded fuel bundles (up to four) into their previous positions around each SRM until 3 cps is established

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (1.00)

Unit 1 is in Operational Condition 1 with NO outstanding deficiencies. The Surveillance Assignment Sheet identifies tomorrow as the "Latest Date" for the RCIC quarterly flow test. Prior to performing the scheduled surveillance, RCIC becomes INOPERABLE.

Which ONE (1) of the statements below accurately describes the surveillance requirements for this situation.

a. The surveillance must be performed immediately after returning the system to an OPERABLE status.

b. A MISSED SURVEILLANCE SHEET must be initiated since the surveillance will not be completed today as scheduled.

c. The surveillance shall NOT be documented as officially missed until the "Latest Date" has elapsed.

d. Since RCIC is already INOPERABLE, a MISSED SURVEILLANCE SHEET need not be issued to track the missed surveillance.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (1.50)

STATE the reporting requirements for EACH of the following situations:

a. Unit 1 reactor scrams from a Main Turbine Stop Valve Fast Closure, following a turbine trip. The HPCI system receives an auto initiation signal on low reactor water level but fails to inject to the reactor pressure vessel due to isolating on a Main Steam Line high dP isolation signal.

b. While making a tour of the Unit 1 Southeast Diagonal, a Plant Equipment Operator slips on the stairs at the 110 foot level resulting in a compound fracture of his left leg. The man is subsequently transported to Appling General Hospital for treatment.

c. With Unit 2 at full power at 2424 MWt, the functional test for MSIV closure procedure is being performed. While testing the "B MSIV Not Full Open" logic, it is found that a Division I circuit de-energized a Division II relay and the Division II logic de-energized a Division I relay, thus violating the divisional circuit separation criteria as addressed in the FSAR.

LIMIT YOUR RESPONSES TO LESS THAN 15 DAY REPORTING REQUIREMENTS.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.04 (1.50)

Concerning Temporary Procedure Changes:

a. You are the Unit 2 Shift Supervisor on duty and an instrument technician brings you the procedure for "Barksdale Pressure Switch Calibration". He points out a part of the procedure that he wants you to approve as a temporary change. He informs you that the work was just completed in accordance with the "changed", but not approved, procedure.

   (1) STATE whether you can approve this TCN and the work it covers (YES or NO). (0.25)

   (2) JUSTIFY your response. (0.50)

b. On August 19, the "PRIMARY CONTAINMENT ATMOSPHERE CONTROL SYSTEM" procedure was temporarily changed by the Shift Supervisor. On August 24, a procedure revision request to the procedure, consisting of the same items as the temporary changes, was reviewed by the PRB and approved and signed by the Plant Manager on September 4.

   (1) STATE whether any problem exists (YES or NO) with the administrative processing of the changes to the "PRIMARY CONTAINMENT ATMOSPHERE CONTROL SYSTEM" procedure. (0.25)

   (2) JUSTIFY your response. (0.50)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (2.00)

Following a TIP trace on Unit 2, the ball valve for the "C" drive mechanism fails to auto close. The TIP machine requires extra jogging to close this valve due to a sticking limit switch. The ball valve is subsequently closed.

a. STATE whether Primary Containment Integrity is satisfied. JUSTIFY your response and STATE any ACTION required. (1.5)

b. EXPLAIN any differences in your response if this situation had occurred on Unit 1. (0.5)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (1.00)

You are the Shift Supervisor while performing a normal reactor startup of Unit 2. At 10% power with the mode selector switch in STARTUP/HOT STANDBY, the Rod Worth Minimizer is declared INOPERABLE. A second licensed operator is stationed at the reactor control console to verify compliance with the required rod sequence check off list and the startup is continued to 100% power.

EXPLAIN WHY Technical Specifications HAVE or HAVE NOT been violated.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.07 (3.00)

For EACH of the following situations:

a. STATE whether the TS allows continued operation in Operational Condition 1.

b. JUSTIFY the basis, per TS, for your decision.

NOTE: ASSUME ALL OTHER COMPONENTS ARE FULLY OPERABLE

1. A fuel oil transfer is dismantled for maintenance on Diesel Generator 2A.

2. It is reported that 1B Standby Liquid Control Pump will not meet the minimum flow requirements per Technical Specifications.

3. TWO (2) Suppression Pool-Drywell Vacuum Breakers in Unit 2 cannot be CLOSED.

4. The leads are lifted from the motor controller for the "A" Fuel Oil Transfer Pump on Diesel Generator 1B.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.08 (3.00)

Unit 1 is operating at 48% power and 35% flow following a Recirculation Pump trip. LIST by section and paragraph ALL requirements per TS that must be considered to permit continued operation on a single loop. INCLUDE a summary of any ACTION you must initiate in order to comply with these requirements.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.09 (2.00)

Unit 1 is nearing the end of a major refueling and maintenance outage that has already overrun by two months. Significant pre-startup testing of plant systems is planned for the coming week. Management intends to cover them through use of extensive overtime by the staff. As Shift Supervisor you receive the following work schedule for one of your Plant Operators.

MONDAY	0700-1900
TUESDAY	0700-1700
WEDNESDAY	0700-2000
THURSDAY	0700-1700
FRIDAY	0000-0900
	1700-2300
SATURDAY	0600-2300
SUNDAY	-- OFF --

STATE the overtime restrictions, if any, that would be violated if the operator worked as scheduled and LIST the timeframe during which each violation would occur.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (2.00)

While operating at 100% power, a Group I isolation and reactor SCRAM occur on Unit 1. Data collected from the plant Process Computer and the plant operators indicate the following occurred:

(1) The Group I isolation was caused by technician error.

(2) The reactor SCRAM was caused by high reactor pressure.

(3) Both Reactor Feed Pumps tripped.

(4) Reactor water level decreased causing both HPCI and RCIC to auto start and inject to the reactor vessel.

(5) An operator secured HPCI, took manual control of RCIC and maintained reactor vessel level.

(6) A feed pump was restarted. After level control was transferred to the feed pump, RCIC was secured.

(7) Plant was placed in a normal HOT SHUTDOWN condition.

Which ONE (1) of the following, based on the information given above, is the correct statement concerning subsequent reactor operation. JUSTIFY your response

a. Power operation cannot resume because HPCI auto initiated and injected to the reactor vessel.

b. Power operation cannot resume because the feed pumps should not have tripped.

c. Power operation cannot resume because a safety limit may have been violated.

d. Power operation cannot resume until the MSIVs have been inspected due to the Group I isolation signal.

QUESTION 8.11 (2.00)

You have just assumed the 0000-0800 (2/8/88) shift as the Unit 1 Shift Supervisor. The plant is operating at 100% power with 98% core flow and the following equipment out of service (OOS):

	Date OOS
SLC Tank remote level indication	01/25/88
RHR Service Water Pump A	01/31/88
RBCCW Pump C	02/01/88
CRD Pump B	02/01/88
HPCI Equipment Area Coolers	02/06/88
Core Monitor	02/05/88
Turning Gear Motor (Main Turbine)	01/15/88
Condensate Pump A	02/05/88

Answer EACH of the following questions L sed on the above information:

a. LIST the out of service equipment which should have Limiting Conditions for Operation (LCOs) in effect and STATE the surveillance requirements that must be performed for each per Technical Specifications in order to allow continued operation in these conditions. (1.0)

b. STATE how long the reactor may remain in operation if NO repairs are completed. (0.5)

c. While performing the RHR quarterly full flow test, RHR Pump C is declared inoperable. STATE the TS requirements concerning plant operability. (BE SPECIFIC and REFERENCE THE TS BEING APPLIED!) (0.5)

QUESTION 8.12 (1.00)

You are the Unit 1 Shift Supervisor with the reactor operating at 80% power. CRD Pump A is out of service due to a faulty motor controller. Maintenance personnel inform you that they wish to commence work and will be disconnecting the electrical leads on the controller. You have the Maintenance Work Order (MWO) in front of you and notice that there is no Temporary Modification Sheet associated with the MWO.

LIST TWO (2) conditions that must be met in order for the work to be performed without a Temporary Modification Sheet.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.13 (1.00)

You are the Unit 2 Shift Supervisor with the plant in a Cold Shutdown Condition. Both loops of LPCI and "B" loop of CS have been sequentially aligned to take suction from the torus so that logic system functional tests can be performed. The following equipment is out of service:

A & B Main Condensate Pumps
HPCI
2A Standby Diesel Generator
All Suppression Pool level instrumentation
RWCU

DELETED

An I&C technician comes up to you and requests you align "A" CS loop to the torus so they can complete the logic system functional tests.

a. STATE whether you would align "A" loop of CS to the torus in this situation (YES or NO). (0.25)

b. JUSTIFY your response. (0.75)

QUESTION 8.14 (2.00)

a. Unit 2 is operating at 85% power and Plant Service Water (PSW) Pump "2A" has been out of service for two days. Maintenance continues, and the pump is expected to be returned to service not earlier than four days from today. A sizeable pipe break occurs in the "2B" and "2D" PSW pump room, removing both pumps from service. Repairs on these pumps are expected to take two weeks.

STATE the ACTION(S) you will take and REFERENCE all TS you use to develop your answer.

b. STATE what your ACTIUN(S) would be for the above scenario if instead of Operational Condition 1, the plant was in the REFUEL Condition. ASSUME heat losses are sufficient to maintain Operational Condition 5. REFERENCE all TS you use to develop your answer.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.15 (2.00)

For EACH of the situations below, STATE whether the plant is in an LCO and if so, STATE what ACTION must be taken:

a. You are the Unit 2 Shift Supervisor during normal full power operations when you receive the following information:

SLC Tank Temperature: 82 deg F
SLC Tank Concentration: 11.3 weight percent (1.0)

b. On a different occasion, you notice the SLC Tank level as indicated on Panel 2H11-P603 is 70%. (1.0)

SHOW ALL WORK AND REFERENCE ALL APPLICABLE SECTIONS OF THE TECHNICAL SPECIFICATIONS. CONSIDER EACH CASE SEPARATELY.

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

ANSWER 8.01 (1.00)

d

REFERENCE
EIH: U2 TS 4.9.2
LER #2-82-036

3.2/3.9 3.2/3.9
215004G005 215004G011 ...(KA'S)

ANSWER 8.02 (1.00)

a -or- c

REFERENCE
EIH: 40AC-REG-001-0S

3.1/3.7 3.4/3.6
217000G002 294001A106 ...(KA'S)

ANSWER 8.03 (1.50)

a. Within one hour (via ENS for valid signal for HPCI injection)
(4 hr report for RPS actuation)
(4 hr report for failure of HPCI)

b. None (if assume man NOT contaminated)
4 hr report (if assume man IS contaminated)

c. 1 hr report (for situation beyond design basis of FSAR or unanalyzed condition that significantly compromises plant safety)

(0.50 each)

REFERENCE
10 CFR 50.72
EIH: 40AC-REG-002-0S
LP 300.4, LO #B
Module 3.1, EO #3.1.3.1
LER #1-82-069,R1
LER #2-82-098

3.4/4.5 2.9/4.3
212000G003 261000G003 ...(KA'S)

ANSWER 8.04 (1.50)

a. (1) No (0.25)
(2) Procedure changes must be approved prior to performing the work. (0.50)

b. (1) Yes (0.25)
(2) Approval of the changes took too long (greater than 14 days from the original date of change). (0.50)

REFERENCE
EIH: 10AC-MGR-003-0S
U2 TS 6.8.3.c
LER #2-82-107
LER #1-82-077
LP 300.4, LO #A.7

2.9/3.4
294001A101 ...(KA'S)

ANSWER 8.05 (2.00)

a. (Primary Containment Integrity is) NOT satisfied. (0.5)
(Per TS 4.6.1.1) Primary Containment Integrity is violated if all penetrations to the containment are not capable of being closed by OPERABLE containment automatic isolation valves. (0.5) (To meet containment integrity and allow continued operation per TS 4.6.1.1 and 3.6.3), the ball valve must be deactivated in its isolated position. (0.5) (1.0)

b. (Per TS 1.0.T.3, Primary Containment Integrity is satisfied if all inoperable automatic isolation valves are deactivated in the isolated position.) No change from Unit 2 requirements! (0.5)

NOTE: Closing the shear valve accomplishes same effect and will be accepted in lieu of deactivating the ball valve.

ALSO ACCEPT:

If neither deactivating the ball valve nor closing the shear valve can be accomplished, an orderly shutdown shall be initiated and the reactor placed in the Cold Shutdown Condition within 24 hours.

REFERENCE
EIH: U2 TS 3.6.1.1
TS 4.6.1.1 (note 1)
U1 TS 3/4.7.D and Table 3.7-1
LP 300.1, LO #2

2.4/3.3 2.7/3.4
215001G005 215001G011 ...(KA'S)

ANSWER 8.06 (1.00)

TS have NOT been violated. TS 3.1.4.1 ACTION statement excepts TS 3.0.4 from being applicable. (0.5) Thus startup may continue and Condition 1 entered after a second operator is stationed. (0.5)

REFERENCE
EIH: U2 TS 3.1.4.1
TS 3.0.4
LP 300.1, LO #2

3.2/4.0 3.5/4.2
201006G005 201006G011 ...(KA'S)

ANSWER 8.07 (3.00)

1. Continued ops allowed (0.25). Only one of the two fuel oil transfer pumps is required per TS 3.8.1.1 (0.5).

2. Continued ops allowed (0.25). Per TS 3.4.B, plant is in a 7 day LCO provided the redundant component is operable (0.5).

3. Continued ops NOT allowed (0.25). This situation addresses circumstances in excess of those given in TS 3.6.4.1. Therefore, TS 3.0.3 applies and facility shall be placed in Hot Shutdown within 6 hours and in Cold Shutdown within the next 30 hours (0.5).

4. Continued ops allowed (0.25). (TS 4.9.A.2 requires both fuel oil transfer pumps to be operable for the D/G to be considered operable.) DG 1B is INOP and TS 3.9.B.2 applies limiting operation to 7 days if two 230 kV offsite transmission lines are available, both remaining DG's and associated buses are operable and increased SV requirements implemented per TS 4.9.B.2 (0.5).

REFERENCE
EIH: U2 TS 3.8.1.1
U1 TS 3.4.B & BASES
U2 TS 3.6.4.1
U1 TS 3/4.9.A.2
LP 300.1, LO #2

3.4/4.1 3.4/4.1
264000G005 264000G011 ...(KA'S)

ANSWER 8.08 (3.00)

1. 3.6.J.3 - requirements of Sections 1.1.A, 2.1.A, 3.1.A, 3.2.G, 3.11.A, and 3.11.C as applicable to single loop ops shall be met (or unit placed in HOT SHUTDOWN within 12 hours).

2. 3.6.J.4 - initiate action within 15 minutes to place power/flow below limits of Fig. 3.6-5.

3. (1.1.A, 3.11.C) - increase MCPR limits by 0.01 (to 1.08).

4. (2.1.A.1.C.(1), 3.1.A table 3.1-1) - adjust APRM setpoints.

5. (3.2.G table 3.2.7) - adjust APRM rod block setpoints.

6. 3.6 J.2 - verify operation below limits of Fig. 3.6-5 at least once per 24 hours and whenever thermal power has been changed by at least 5% of RTP and SS conditions have been reached.

(0.5 each)

7. 3.11.A - none (unless power is increased above 52% rated thermal power). (ANSWER NOT REQUIRED)

NOTE: Listing TS 3.6.J.1, 3.6.J.5, 3.6.D, & 3.6.E will not result in + or - credit

REFERENCE
EIH: U1 TS 3/4.6.J.2
LP 300.1, LO #2

3.4/3.4 3.3/4.0 3.4/4.2 3.6/3.7 3.4/4.2 3.4/4.2
202001A203 202002A201 202002G005 202002G011 ...(KA'S)

ANSWER 8.09 (2.00)

1. 0700 Thurs - 0700 Fri (0.25): work more than 16 hrs in 24 hr period (0.25).

2. 1700 Thurs - 0000 Fri (0.25): less than 8 hr break between work periods (0.25).

3. 0600 - 2300 Sat (0.25): work more than 16 hrs straight (0.25).

4. 0700 Mon - 2300 Sat (0.25): Work more than 72 hrs in seven day period (0.25). (Occurred at 2200 Friday)

NOTE: There are other violations that may have occurred in addition to those listed above. Any four = Full Credit.

REFERENCE
EIH: U1 TS 6.2.2.g

2.7/3.7
294001A103 ...(KA'S)

ANSWER 8.10 (2.00)

c (1.0)

On a Group I isolation, the SCRAM signal should come from MSIV closure and NOT high reactor pressure. (0.5)

TS (1.1.c) states that a Safety Limit shall be assumed to be exceeded when a SCRAM is accomplished by a means other than the expected scram signal. (0.5)

(Also, TS (1.0.DD) indicates that exceeding a Safety Limit requires Unit shutdown and review by the NRC before resumption of Unit operation.)

REFERENCE
EIH: U1 TS 1.0.DD, 1.1.C
42EN-ENG-011-0S

4.0/4.1 4.0/4.2 4.2/4.3 3.8/4.0 3.1/4.3 3.2/4.1 3.3/4.1
239001A203 239001A212 239001G001 239001K127 ...(KA'S)

ANSWER 8.11 (2.00)

a. RHR Service Water Pump A (TS 4.5.C.2) (0.1)
- remaining active components of both RHR subsystems shall be demonstrated to be operable immediately. (0.2)
- an operable service water pump shall be demonstrated to be operable daily. (0.2)

HPCI Equipment Area Cooler (HPCI INOP per TS 4.5.D.2 via 3.5.K.2) (0.1)
- the ADS actuation logic, the RCIC system, the RHR system LPCI mode, and the CS system shall be demonstrated to be operable immediately. (0.2)
- the RCIC system and ADS logic shall be demonstrated to be operable daily. (0.2)

b. 14 days past 2/6/88 (when HPCI pump was declared INOP) (or 2/20/88). (0.5)

c. Per TS 3.5.C.4, the reactor shall be placed in the Cold Shutdown Condition within 24 hours. (0.5)

TERMINAL TIMEOUT IN 30 SECONDS

REFERENCE
EIH: U1 TS 3.5
LP 300.1, LO #2

3.6/4.5 3.7/4.4 3.4/4.3
203000G011 206000G011 217000G011 ...(KA'S)

ANSWER 8.12 (1.00)

1. A functional test must be defined on the MWO that will test proper operation of the motor.

2. Work must be controlled by a clearance.

3. If functional test of a MWO requires that all wires and links be red-lined, or if it verifies the designed functions and/or setpoints of the subcomponent.

4. If the activities described (in 30AC-OPS-005-0S, Section 2.1) are adequately covered in another approved plant procedure. (Acceptability of other plant procedure is based on 4 specific guidelines given in 30AC-OPS-005-0S.)

(any 2 @ 0.5 each)

REFERENCE
EIH: 30AC-OPS-005-0S
LP300.4, LO #10

3.9/4.5
294001K102 ...(KA'S)

ANSWER 8.13 (1.00)

a. No (0.25)

b. TS 3/4.5.3 requires an OPERABLE flow path (in Condition 4 or 5) from either the SP or the CST. (0.25)

Since SP level instrumentation is INOP, then all systems lined up to the SP are considered INOP. (0.25)

Therefore, he must maintain the "A" loop of CS aligned to the CST (until "B" loop of CS is re-aligned to the CST. (0.25)

REFERENCE
EIH: U2 TS 3.5.3
LER #2-82-048

2.8/2.9 3.3/3.4 3.3/4.2 3.4/4.2
209001G005 209001G011 209001K410 209001K603 ...(KA'S)

DELETED

ANSWER 8.14 (2.00)

a. NONE of the ACTION statements listed in TS 3.7.1.2.a apply (0.5)

TS 3.0.3 must therefore be applied and the unit must be placed in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours. (0.5)

b. Per TS 3.7.1.2.b since they cannot restore both PSW loops with at least one pump in each loop to an OPERABLE status within 7 days, must declare the CS system, the LPCI system and the associated diesel generators inoperable and take the ACTION required by TS 3.5.3.1, 3.5.3.2, and 3.8.1.2. (1.0)

REFERENCE
EIH: U2 TS 3.7.1.2

4.2/4.2
294001A101 ...(KA'S)

ANSWER 8.15 (2.00)

a. From Fig. 3.1.5-1 in TS, the temperature vs. concentration is SATISFACTORY. NOT in an LCO. (1.0)

b. Converting percent level to gallons:
100% = 5150 gal => 70% = 3605 gal. (0.25)

Per TS 3/4.1.5, tank level must be greater than or equal to 3802 gal. for the system to be OPERABLE. (0.25)

Therefore, ARE in LCO 3.1.5.a & ACTION a.2 applies (restore system to OPERABLE status within 8 hrs. or be in at least HOT SHUTDOWN within the next 12 hrs.) (0.5)

REFERENCE
EIH: U2 TS 3.1.5
LT-IH-01101-00, EO #17

3.6/4.4 3.4/4.1
211000G005 211000G011 ...(KA'S)

QUESTION	VALUE	REFERENCE
08.01	1.00	DCP0001435
08.02	1.00	DCP0001436
08.03	1.50	DCP0001431
08.04	1.50	DCP0001433
08.05	2.00	DCP0001425
08.06	1.00	DCP0001426
08.07	3.00	DCP0001428
08.08	3.00	DCP0001430
08.09	2.00	DCP0001438
08.10	2.00	DCP0001440
08.11	2.00	DCP0001444
08.12	1.00	DCP0001445
08.13	1.00	DCP0001446
08.14	2.00	DCP0001447
08.15	2.00	DCP0001429
	26.00	
	26.00	

DOCKET NO 321

Georgia Power Company
Edwin I. Hatch Nuclear Plant
Post Office Box 439
Baxley, Georgia 31513
Telephone 912 367-7781
912 537-9444

ENCLOSURE 3

Georgia Power
the southern electric system

J. T. Beckham, Jr.
Vice President

February 15, 1988

PLANT E. I. HATCH
Plant Hatch Operator Licensing Examination
Rtype: A04.25
Log: LR-VPH-014-0288

U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Atlanta, Georgia 30323

ATTN: Caudle A. Julian,
Chief, Operations Branch
Division of Reactor Safety

Gentlemen:

In accordance with NUREG 1021, I have enclosed the formal comment submittal for the Plant Hatch Operator Licensing examination administered on February 8, 1988. Please call C. L. Coggin at 912-367-7851 if further information or clarification is desired.

J. T. Beckham, Jr.
J. T. Beckham, Jr.
Vice President Plant Hatch

JTB/RSG/gkb

xc: C. L. Coggin
E. M. Howard (w/o enclosure)

Plant E.I. Hatch
Utility Comments
Reactor Operator and Senior Operator
Written License Examination Comments
February 11, 1988

3.04c Utility Comment:
The question requires the effect of Recirc Loop operation on wide range instrumentation. The stem does not delineate any specific change in the Recirc operation. This causes the candidate to make assumptions that may not be consistent with the intent of the question. It is also noted that at Hatch the "Wide Range Instrumentation" is called Emergency Range Instrumentation. This may have caused some confusion.

Recommendation:
It is requested that 3.04c be deleted.

Reference:
E.I. Hatch,, LP 44.1

3.09d Utility Comment:
The question requires the pressure setpoint to isolate Nitrogen backup valves to Drywell Pneumatics System. The Drywell Pneumatic Air Compressors are not used at Plant Hatch. Therefore nitrogen is manually aligned to the Drywell Pneumatic System and any isolation is inconsequential to the operation of the system.

Recommendation:
It is requested that 3.09d be deleted

Reference:
E.I. Hatch LT-IH-03501-00

6.06a Utility Comment:
The question requires the high and low side pressure sensing points for Core Spray Line Break Instrumentation Detection System. The question does not state plant conditions, which will vary high and low side sensing. The answer key is given for cold conditions. At power, the answer is reversed due to Pitot Tube Effect.

Recommendation:
It is requested that 6.06a be deleted.

Reference:
E.I. Hatch LT-IH-00801-00

6.11b Utility Comment:
The question requires the reason for maintaining a Drywell to Torus Differential Pressure. Due to recent Hatch modifications, this is no longer required by the normal plant startup procedure.

Recommendation:
It is requested that 6.11b be deleted.

Reference:
34GO-OPS-001-1S

7.06b Utility Comment:
The question requires a description of the verification process for proper seating of fuel onto fuel support piece. Procedure allows for the use of the fuel grapple, in a scanning mode, to verify seating.

Recommendation:
It is requested that the key be revised to include the use of the Fuel Grapple for the verification.

Reference:
42FH-ERP-014-0S, 7.5.15 through 7.5.20

8.03c Utility Comment:
The question requires the necessary reporting requirements if RPS Divisional Circuit Separation criteria has been violated. Technical Specifications Amendment #86 deleted prompt and follow-up notifications from Technical Specifications. All necessary reports would be made per 40AC-REG-002-0S.

Recommendation:
Revise the answer key to accept One Hour Report per 40AC-REG-002-0S.

Reference:
Unit Two Technical Specifications sections 6.9.1.12 and 6.9.1.13 Amendment 86.
40AC-REG-002-0S

8.05 Utility Comment:

The question requires the Technical Specifications actions for the failure of a TIP ball valve. The failure of a TIP ball valve does not constitute a Primary Containment failure. It is the failure of a Primary Containment isolation valve. Answers should address requirements for inoperable Containment Isolations.

Recommendation:

It is requested that the answer key be revised to accept the following.

A. Yes, Primary Containment Integrity is maintained due to Ball Valve being closed. Only actions required would be those per Technical Specifications Section 3/4 6.3.

B. Unit One Technical Specifications would not require the deactivation of the Primary Containment Isolation Valve. It would only require the adherence to Section 3.7.D.2

References:

Unit 1 Technical Specifications; 3.7.D.2, Table 3.7-1
Unit 2 Technical Specifications; 3/4 6.3

8.07 Utility Comments:

The question requires an interpretation of Technical Specifications to determine if continued operation is allowed. The answer key states whether the equipment is operable/inoperable, and does not address if continued operation is allowed.

Recommendation:

It is requested that the key be revised to incorporate the Utility comments as follows:

Part 1; Yes - Operation may continue due to Technical Specifications requiring only one Fuel Oil Transfer Pump for Diesel Generator Operability.

-or-

Yes - Student may respond with a conservative interpretation of Technical Specifications Operability as defined in the definition section. This will result in the plant being in a 72 hr LCO due to an inoperable Diesel Generator.

Part 2; Yes - 1B SBLC Pump being inoperable puts the plant into a 7 day LCO per 3.4.B.

Part 3; Yes (this would place the plant in a shutdown LCO) - The plant will be in a 6 hours to Hot Shutdown and 30 hours to Cold Shutdown LCO per Technical Specifications section 3.0.3.

Part 4; Yes - One Diesel Generator being inoperable places the plant into a 7 day LCO per Unit 1 Technical Specifications section 3.5.G or should accept a 72 hour LCO if candidate addressed unit operation on Unit 2 with "B" Diesel Generator inoperable per Technical Specifications section 3.8.1.1.

References:

Unit 1 & 2 Technical Specifications.

8.13 Utility Comments:

The question requires an interpretation of Technical Specifications to allow alignment of "A" loop of Core Spray to the Torus. Since the plant is in condition 5, 3.0.5 does not apply. With the 2A Diesel inoperable, Core Spray "A" is inoperable, resulting in the initial conditions not being allowed by Technical Specifications. This situation may lead the candidate to make assumptions that may not be consistent with the intent of the question. The conditions specified in the referenced LER were without the "A" Diesel Generator being inoperable.

Recommendation:

It is requested that 8.13 be deleted.

References:

Unit 2 Technical Specifications

Section 8 Utility Comment:

Two primary factors are believed to have contributed to the greater than anticipated time required on section 8. First, the arrangement of the questions required frequent use of alternate unit Technical Specifications. This aspect could be improved by grouping questions dealing with each unit's Technical Specifications . Second, the importance of bounding questions becomes greater for open book examinations. Some questions may lead the candidate to go into greater depth than expected, by not specifying the scope or depth closely enough. Examples include: Question 8.03, regarding required reports, and question 8.09, regarding overtime restrictions.

TABLE 3.7-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
2	Suppression chamber exhaust valve bypass to standby gas treatment (T48-F339, T48-F338)		2	5	C	SC
2	Suppression chamber nitrogen make-up line (normal operation) (T48-F118B)		1	5	C	SC
2	Drywell and suppression chamber nitrogen supply line (inerting) (T48-F103)		1	5	C	SC
2	Drywell and suppression chamber nitrogen make-up line (normal operation) (T48-F104)		1	5	C	SC
2	Drywell equipment drain sump discharge (C11-F019, G11-F020)		2	15	O	GC
2	Drywell floor drain sump discharge (G11-F003, G11-F004)		2	15	O	GC
2	TIP Guide Tube (C51-J004)		1 each line	NA	C	SC
(c)	Drywell pneumatic system (P70-F002, P70-F003)		2	5	O	GC

NOTE: (C51-J004) IS MPL # FOR
TIP SKID MOUNT EQUIPMENT

TABLE 3.7-1 (Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES WHICH
RECEIVE A PRIMARY CONTAINMENT ISOLATION SIGNAL

Isolation Group (b)	Valve Identification (d)	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
6	RHR reactor shutdown cooling suction (supply) (E11-F008, E11-F009)	1	1	24	C	SC
6	RHR reactor head spray (E11-F022, E11-F023)	1	1	20/12	C	SC
3	HPCI - turbine steam (E41-F002, E41-F003)	1	1	50	O	GC
4	RCIC - turbine steam (E51-F007, E51-F008)	1	1	20	O	GC
5	Reactor water cleanup from recirculation loop (G31-F001, G31-F004)	1	1	30	O	GC
2	Post-accident sampling system supply (B21-F111, B21-F112)		2	5	C	SC
2	Post-accident sampling system return (E41-F122, E41-F121)		2	5	C	SC
2	Core spray test line to suppression pool (E21-F015A,B)		1 each line	50	C	SC

4.7.D.1. Surveillance of Operable Valves (Continued)

b. At least once per operating cycle the reactor coolant system instrument line excess flow check valves shall be tested for proper operation.

c. At least once per quarter:

(1) All normally open power-operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 75% of rated, the main steam line isolation valves shall be tripped (one at a time) and closure time verified.

d. At least once per week the main steam line power-operated isolation valves shall be exercised one at a time by partial closure and subsequent reopening.

3.7.D.2. Operation with Inoperable Valves

In the event any isolation valve specified in Table 3.7-1 becomes inoperable, reactor power operation may continue provided at least one isolation valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. Shutdown Requirements

If Specification 3.7.D.1. and 3.7.D.2. cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition w[illegible] in 24 hours.

2. Surveillance of Lines with an Inoperable Valve

Whenever an isolation valve listed in Table 3.7-1 is inoperable the position of at least one other isolation valve in each line having an inoperable isolation valve shall be verified to be in its isolated position daily.

4.7.C.1. <u>Surveillance While Integrity Maintained</u> (Cont'd)

b. Secondary containment capability to maintain a minimum 1/4-inch of water vacuum under calm wind (5 mph) conditions with each filter train flow rate not more than 4000 cfm shall be demonstrated at each refueling outage, prior to refueling.

3.7.C.2. <u>Violation of Secondary Containment Integrity</u>

a. Without Hatch-Unit 1 secondary containment integrity, restore Hatch - Unit 1 secondary containment integrity within 4 hours, or perform the following (as applicable):

(1) Suspend irradiated fuel and/or fuel cask handling in the Hatch-Unit 1 secondary containment.

(2) Be in at least Hot Shutdown within the next 12 hours and meet the Conditions of 3.7.C.1.a. within the next 24 hours.

b. Without Hatch-Unit 1 secondary containment, refer to the following Hatch-Unit 2 Technical Specification, for LCO's to be followed for Hatch-Unit 2:

(1) Section 3.6.5.1.

(2) Section 3.9.5.1.

2. <u>Surveillance After Integrity Violated</u>

After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment. The ability to maintain the remainder of the secondary containment at 1/4-inch of water vacuum pressure under calm (5 mph) wind conditions shall be confirmed.

D. <u>Primary Containment Isolation Valves</u>

1. <u>Valves Required to be Operable</u>

During reactor power operation, all primary containment isolation valves and all reactor coolant system instrument line excess flow check valves shall be operable except as stated in Specification 3.7.D.2.

D. <u>Primary Containment Isolation Valves</u>

1. <u>Surveillance of Operable Valves</u>

Surveillance of the primary containment isolation valves shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and the closure times.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
24. Traversing Incore Probe Isolation Valve Ball Valves	*	NA
25. Vacuum Relief Isolation Valves		
2T48-F309	6	5
2T48-F324	6	5
26. HPCI Pump Suction Isolation Valve		
2E41-F042	3	84

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group.
*Closes upon withdrawal of TIP. TIP automatic withdrawal is actuated by either low reactor vessel water level or high drywell pressure.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

B. MANUAL ISOLATION VALVES[*]

1. Main steam isolation valves
   2T32-F001B, F, K, P

2. RHR return to recirculation loop isolation valves
   2E11-F015A, B

3. LOCA H_2 recombiner isolation valves
   2T49-F002 A, B
   2T49-F004 A, B

4. Core spray isolation valves
   2E21-F005A, B

5. Service air isolation valves
   2P51-F651
   2P51-F513

6. RBCCW supply and return isolation valves
   2P42-F051
   2P42-F052

[*] Includes power operated valves which do not isolate automatically.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves specified in Table 3.6.3-1 shall be OPERABLE with isolation times as shown in Table 3.6.3-1.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment isolation valves specified in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable, provided that at least one isolation valve is maintained OPERABLE in each affected penetration that is open, and either:

1. The inoperable valve(s) is restored to OPERABLE status within 4 hours, or

2. Each affected penetration is isolated within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

3. Each affected penetration is isolated within 4 hours by use of at least one closed manual valve or blind flange.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves specified in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours;

1. The inoperable valve is returned to OPERABLE status, or

2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Note: Shear valve still operable

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verification of specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve specified in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve stops excess flow.

7.5.3 Set up the video equipment and lighting.

7.5.4 Attach the underwater television camera to the Refueling Platform Fuel Grapple.

7.5.5 POSITION the Refueling Platform over the fuel bundles to be observed.

7.5.6 OPERATE the Grapple LOWER Control to position the camera close enough to the fuel bundle to clearly discern the serial numbers.

7.5.7 Start the video recorder and indicate the coordinate of the first bundle in the row being observed and the direction of scan (i.e., North, South, East, or West) with an audible statement on the video tape.

7.5.8 Scan the core slowly, one row of fuel at a time, and record the serial number of each fuel bundle in the corresponding coordinate blank of the core map.

7.5.9 Record any discrepancies found on Attachment 9.

7.5.10 OPERATE the Grapple RAISE Control until a four bundle cell can be clearly discerned.

7.5.11 Indicate the coordinate of the fuel cell in the row being observed and the direction of scan with an audible statement on the video tape.

7.5.12 Scan the core slowly, one row of cells at a time, and confirm the following:

7.5.12.1 Each fuel bundle in the cell is oriented properly (the channel fastener clip is oriented toward the center of the control rod).

7.5.12.2 The twelve uncontrolled bundles on the core periphery have their channel fastener clips oriented radially outward from the core edge.

7.5.13 Record any discrepancies found on Attachment 9.

7.5.14 OPERATE the Grapple RAISE Control and raise the camera out of the water.

7.5.15 Remove the camera from the Fuel Grapple.

7.5.16 OPERATE the Grapple LOWER Control until the Fuel Grapple is located just above the fuel bale handles.

7.5.17 Scan the core slowly, one row of fuel at a time, and confirm that all fuel bundles are seated correctly as indicated by the Fuel Grapple not striking any fuel bale handles.

7.5.18 Record any discrepancies found on Attachment 9.

7.5.19 OPERATE the Grapple RAISE Control and position the Refueling Platform as necessary.

7.5.20 Complete Attachment 9 and attach the core map used.

a. During cold conditions:

- Pressure at point 2 is due to the weight of the water column within the core shroud (applied to the Hi side of the DP instrument).

- Pressure on the Lo side of the DP instrument is due to the weight of the water column in the piping from the Core Spray Header.

- DP is low or zero.

b. As rated temperature and pressure is reached the density of the water in the vessel decreases, therefore less applied pressure to the Hi side of the DP instrument resulting in a negative DP reading.

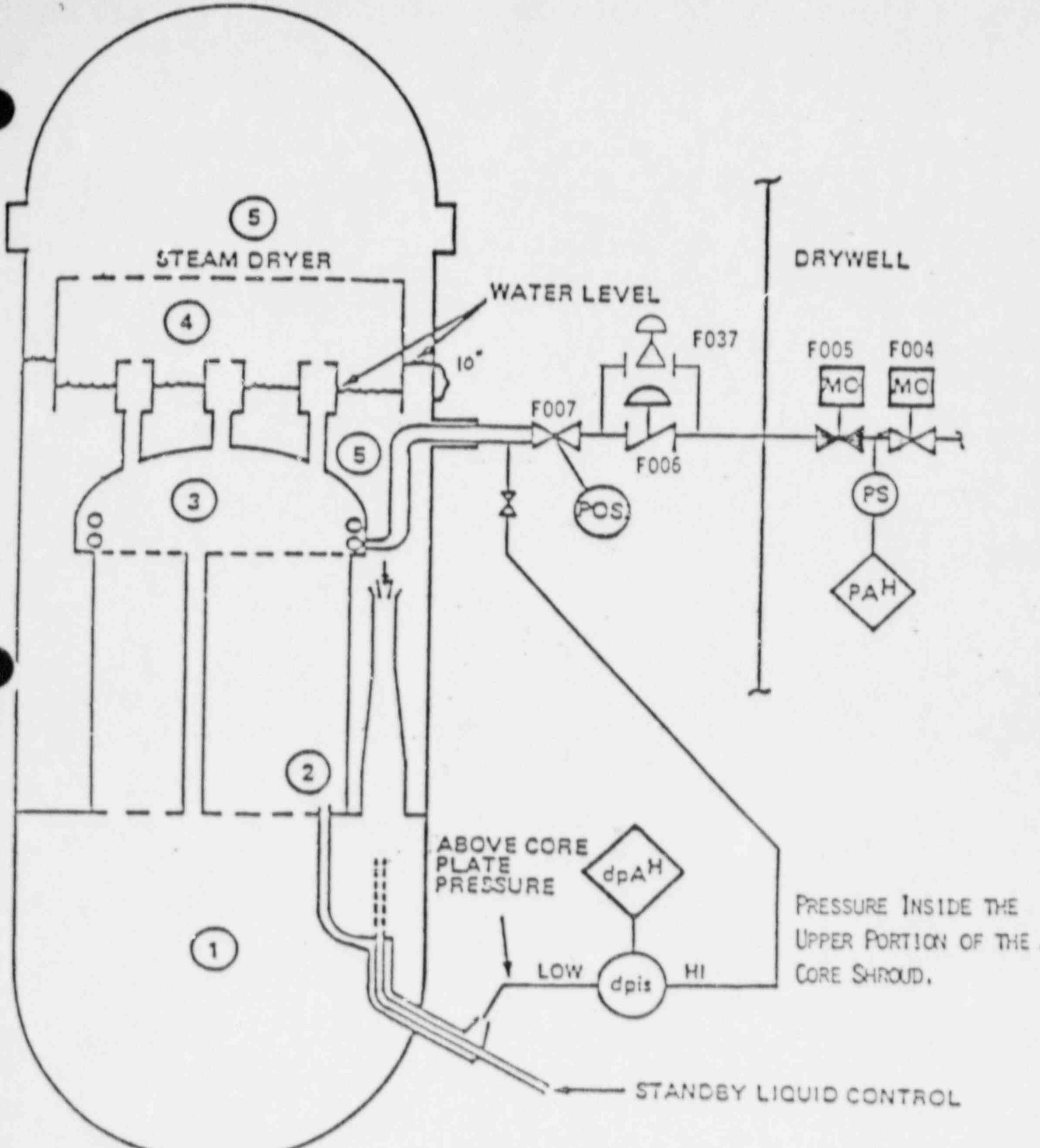
c. As core flow is increased, the pressure at point 3 increases, therefore more applied pressure to the Lo side of the DP instrument resulting in a greater negative DP reading. (nominal DP at rated is approximately -3.5 psid).

d. If a Core Spray line breaks internal to the vessel, pressure sensed will be between points 2 and 5; DP becomes less negative.

e. Alarm at 1 psid above normal DP

(1) Local indication

(2) Annunciator



TRANSPARENCY 00801-9

CORE SPRAY SYSTEM PIPE BREAK DETECTION INSTRUMENTATION

FOR TRAINING USE ONLY.

(This page is intentionally left blank.)

(This page is intentionally left blank.)

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee:	Georgia Power Company
Facility Licensee Docket No.:	50-321 and 50-366
Facility Licensee No.:	DPR-57 and NPF-5
Operating Tests administered at:	Edwin I. Hatch Nuclear Plant
Operating Tests Given On:	February 9-11, 1988

During the conduct of the simulator portion of the operating tests identified above, the following apparent performance and/or human factors discrepancies were observed:

1. "B" Recirc Controller has a dead spot that prevents shifting to AUTO without help from the Simulator Instructor.
2. The Annunciator Response Procedure for RBCCW Expansion Tank Low Level alarm was not available in the simulator control room.
3. The system lineup for the RCIC coolers was not in accordance with the procedure for all the IC's used on the simulator.
4. If RCIC trips (e.g. during SV testing) and is then reset, a subsequent restart of the system will result in a spurious injection into the reactor vessel.