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

EXAMINATION REPORT NOS. 50-317/86-12(OL) and 50-318/86-12(OL)

FACILITY DOCKET NOS. 50-317 and 50-318

FACILITY LICENSE NOS. DPR-53 and DPR-69

LICENSEE: Baltimore Gas and Electric Co.
P.O. Box 1475
Baltimore, Maryland 21203

FACILITY: Calvert Cliffs Nuclear Power Plant, Units 1 and 2

EXAMINATION DATES: August 11-15, 1986

CHIEF EXAMINER: D. H. Coe 10/16/86
D. H. Coe, Lead Reactor Engineer (Examiner) Date

REVIEWED BY: R. Keller 10/16/86
R. Keller, Chief, Projects Section 1C Date

APPROVED BY: H. Kister 10/20/86
H. Kister, Chief, Project Branch No. 1 Date

SUMMARY: Written and operating examinations were administered to six Reactor Operator (RO) candidates and four Senior Reactor Operator (SRO) candidates. One RO candidate failed the simulator portion of the operating examination and was denied a license. Two RO candidates failed both the written and operating examinations and were denied licenses.

REPORT DETAILS

TYPE OF EXAMS: Replacement

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	4 / 2	4 / 0
Oral Exam	4 / 2	4 / 0
Simulator Exam	3 / 3	4 / 0
Overall	3 / 3	4 / 0

1. CHIEF EXAMINER AT SITE: D. H. Coe, NRC

2. OTHER EXAMINERS: C. Y. Shiraki, NRC

   J. W. Upton, PNL

3. Summary of generic strengths or deficiencies noted on oral exams:

   Senior operator candidates did not always check their emergency actions against the Emergency Operating Procedures (EOPs) although, in most cases all required actions were taken.

   All candidates were very weak in their ability to determine plant radiation and contamination levels by reading the current Health Physics survey maps.

4. Summary of generic strengths or deficiencies noted from grading of written exams:

   RO examination

   The design purpose of the HPSI system during a Main Steam Line Rupture.

   The conditions which will energize the Emergency Diesel Generator Shutdown Relay in the presence of a SIAS.

Identification of the Vital Auxilliaries.

SRO examination

Reason why the 120 VAC manual transfer switch is normally locked in the INVERTER position.

5. Personnel Present at Exit Interview:

    NRC Personnel

    D. H. Coe, Chief Examiner
    C. Y. Shiraki

    Facility Personnel

    S. E. Jones, Jr., General Supervisor-Nuclear Training
    J. R. Hill, Supervisor Operations Training
    J. M. Yoe, Senior Instructor
    C. J. Andrews, Senior Simulator Instructor

6. Summary of NRC Comments made at exit interview:

The adequacy of the training and reference material sent to the NRC was very good. For future examinations an explicit list of differences between Unit 1 and Unit 2 was requested to be made part of the reference material.

This was the first operator examination at CCNPP which utilized a plant specific simulator. The performance of the simulator and of the training staff who operated it was completely acceptable for NRC examination purposes. Minor deficiencies of which the training staff is aware were an immediate blowdown radiation monitor alarm upon insertion of a Steam Generator Tube Leak and HPSI flow oscillations under certain degraded core conditions.

7. Summary of facility comments made at exit interview:

Facility comments regarding the operating exam and the written exam questions and answers were presented in writing. These are included as Attachment 3 to this report.

Attachments:

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on Written and Operating Exams
4. NRC Resolution of Facility Comments

ATTACHMENT 1

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: CALVERT CLIFFS

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 86/08/11

EXAMINER: COE, D.

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
25.00	25.00	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.00	25.00	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00	________	________	3. INSTRUMENTS AND CONTROLS
25.00	25.00	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.00		________ 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 1.01 (.50)

Which one of the following descriptions best supports the reason why Xenon reactivity increases sharply after a trip following 1000 hours of operation at 100% power ?

A) Xenon decays less rapidly due to a reduction in the neutron flux.
B) Xenon half-life is much shorter than Iodine half-life.
C) Iodine inventory is reduced by decay to Xenon which increases due to the reduction in neutron flux.
D) Due to reduced neutron absorption, Iodine concentration increases, and Xenon decays directly from Iodine, thus Xenon increases.

QUESTION 1.02 (1.00)

The following data is obtained during an initial approach to criticality following a refueling outage.

Boron concentration	wide range counts
2200 ppm	32 cps
2150 ppm	39 cps
2100 ppm	50 cps
2050 ppm	70 cps

Using Figure 1-1 as an answer page, construct a 1/M plot and predict the boron concentration at which criticality will occur. LABEL your AXES and next to EACH point on the plot, LIST the coordinates for that point.

QUESTION 1.03 (2.00)

Use Figure 1-2 as an answer page to :

a. Sketch the general shape of the SUR and the reactivity of a reactor during a subcritical to critical rod pull assuming the rod height and neutron flux levels are as shown by the SOLID lines. This is Case I. Assume the final rod height brings the reactor exactly critical. No calculations are necessary. [1.0]

b. On the graph of neutron flux, sketch the neutron flux change that would result if the rods were pulled as shown by the Case II dotted line. Assume final neutron flux levels in Case I and Case II are the same. No calculations are necessary. [1.0]

c. Clearly indicate which case results in a greater neutron flux level AT criticality. No calculations are necessary. [0.5]

QUESTION 1.04 (2.00)

Indicate the individual reactivity effect on the core from MTC and FTC during a Main Steam Line Break and a Continuous Rod Withdrawal Accident prior to reactor protective system action and for each specified time in life. ASSUME A POSITIVE MTC AT BOL. Answer "negative" for a negative reactivity effect, "positive" for a positive reactivity effect, or "no effect". All initial conditions are from Mode 1 operation. Format your answer as shown below.

		MTC	FTC
MSLB	BOL	____________	______________
	EOL	____________	______________
CRWA	BOL	____________	______________
	EOL	____________	______________

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (3.25)

During a reactor startup and after the reactor is critical the Wide Range Log Channel indication is observed to increase from 10E-5% to 10E-4% power in 100 seconds with no rod motion. If the effective delayed neutron fraction is 0.005 and assuming an average neutron precursor decay constant of 0.08 sec-1, how much reactivity was added after criticality? Give your answer in terms of delta k per k and show all calculations.

QUESTION 1.06 (2.50)

Compare the ACTUAL critical rod position for a startup to be performed 4 hours after a trip from extended 100% power near end-of-life, to the CALCULATED Estimated Critical Condition if the following events or conditions occurred. Consider each independently. Limit your answer to actual position is HIGHER than, LOWER than, or SAME as the ECC. Answer SAME if there is no NOTICEABLE difference to the operator.

a. An inadvertant RCS boration has been occuring for the last 4 hours. Boron concentration has increased by 100 ppm. [0.5]

b. The startup is delayed until 30 hours after the trip. [0.5]

c. The steam dump pressure setpoint is increased by 100 psi [0.5]

d. Pressurizer pressure is lowered by 100 psia. [0.5]

e. The ECC assumed a boron concentration 100 ppm higher than the actual boron concentration. [0.5]

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.07 (3.00)

The reactor is in the process of being started up. Power is at 10E-2% and CEA's are in manual group control. A malfunctioning steam generator safety valve lifts, causing an increase in steam flow to about 10% of total plant capacity. Assume no operator or reactor protective system action, and a negative MTC. Calculations are not necessary.

1. State HOW (increase, decrease, remain the same) and WHY each of the following parameters INITIALLY change.

2. State HOW (higher, lower, remain the same) and WHY the the final steady state value compares to the initial value → prior to the transient.

   a. Main steam pressure [1.0]
   b. Primary Tave [1.0]
   c. Reactor power [1.0]

QUESTION 1.08 (1.00)

a. What would pressurizer relief valve discharge temperature be if quench tank pressure is 15 PSIG, there is a steam bubble in the pressurizer and RCS pressure is 1650 PSIA? [0.5]

b. If quench tank pressure is 15 PSIG and RCS pressure is 2200 PSIA, will pressurizer relief valve discharge temperature be GREATER THAN, EQUAL TO, or LESS THAN that in part a? [0.5]

QUESTION 1.09 (2.50)

Answer the following questions assuming that a spurious reactor trip has just occured from extended 100% power operation (an inadvertant trip due to instrument malfunction) and that a pressurizer safety valve lifts and sticks open. Reactor Coolant Pumps are tripped according to procedure and RCS pressure drops to 1000 psia. Answer each part independently.

a. If natural circulation flow could NOT be attained, what other means of core cooling exists and will it be sufficient to cool the core? [1.0]

b. CETs are relatively stable (not increasing) and read 585F, and a constant small feeding and steaming rate is occuring in the steam generators. Briefly describe the THREE most probable heat transfer mechanisms taking place on the primary coolant as it traverses the core and RCS, and WHERE these regions are located. [1.0]

c. Describe how auxillary spray flow could be used to determine or confirm the presence of RCS voiding. [0.5]

QUESTION 1.10 (2.25)

There are THREE primary parameters that affect DNB and can be controlled by the reactor operator other than core flow or flux distribution. Briefly explain how an INCREASE in each of these parameters affects the DNBR. Assume other parameters remain constant.

QUESTION 1.11 (2.00)

Would fuel center line temperature INCREASE, DECREASE, or REMAIN THE SAME in each of the following situations? BRIEFLY EXPLAIN WHY.

a. Power decreases with constant Tave. [0.5]

b. Tave increases with constant power. [0.5]

c. Core age increases with constant power. [0.5]

d. Pressurizer pressure increases with constant power. [0.5]

QUESTION 1.12 (3.00)

The plant is in a Natural Circulation Mode of core cooling. As the fission product heat decays away, describe how and why you would expect the following RCS parameters to change. Assume that S/G pressure is being maintained at 900 psia.

a. Tcold [0.75]

b. Thot [0.75]

c. Core delta T [0.75]

d. Loop transit time [0.75]

(***** END OF CATEGORY 01 *****)

FIGURE 1-1

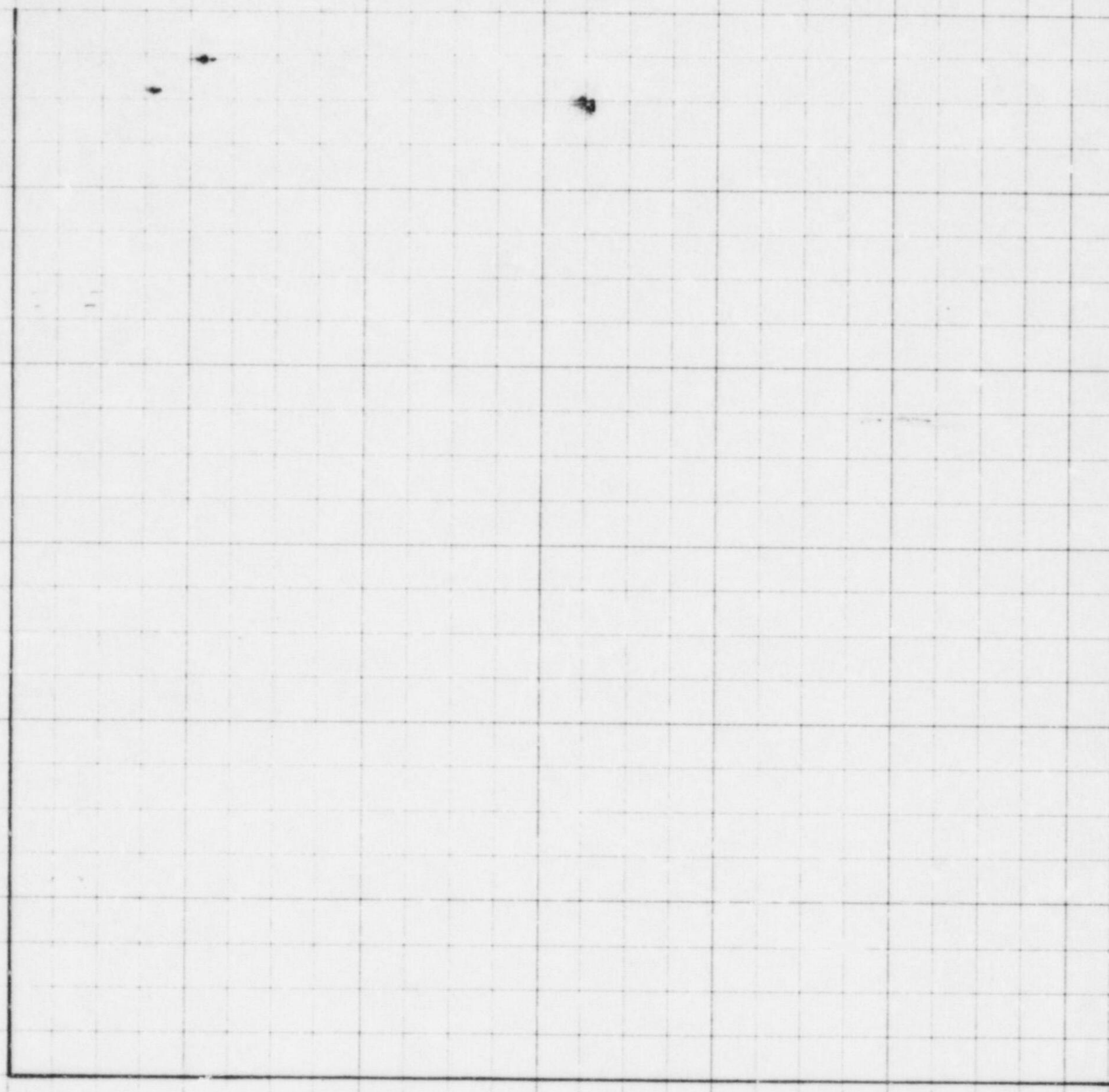
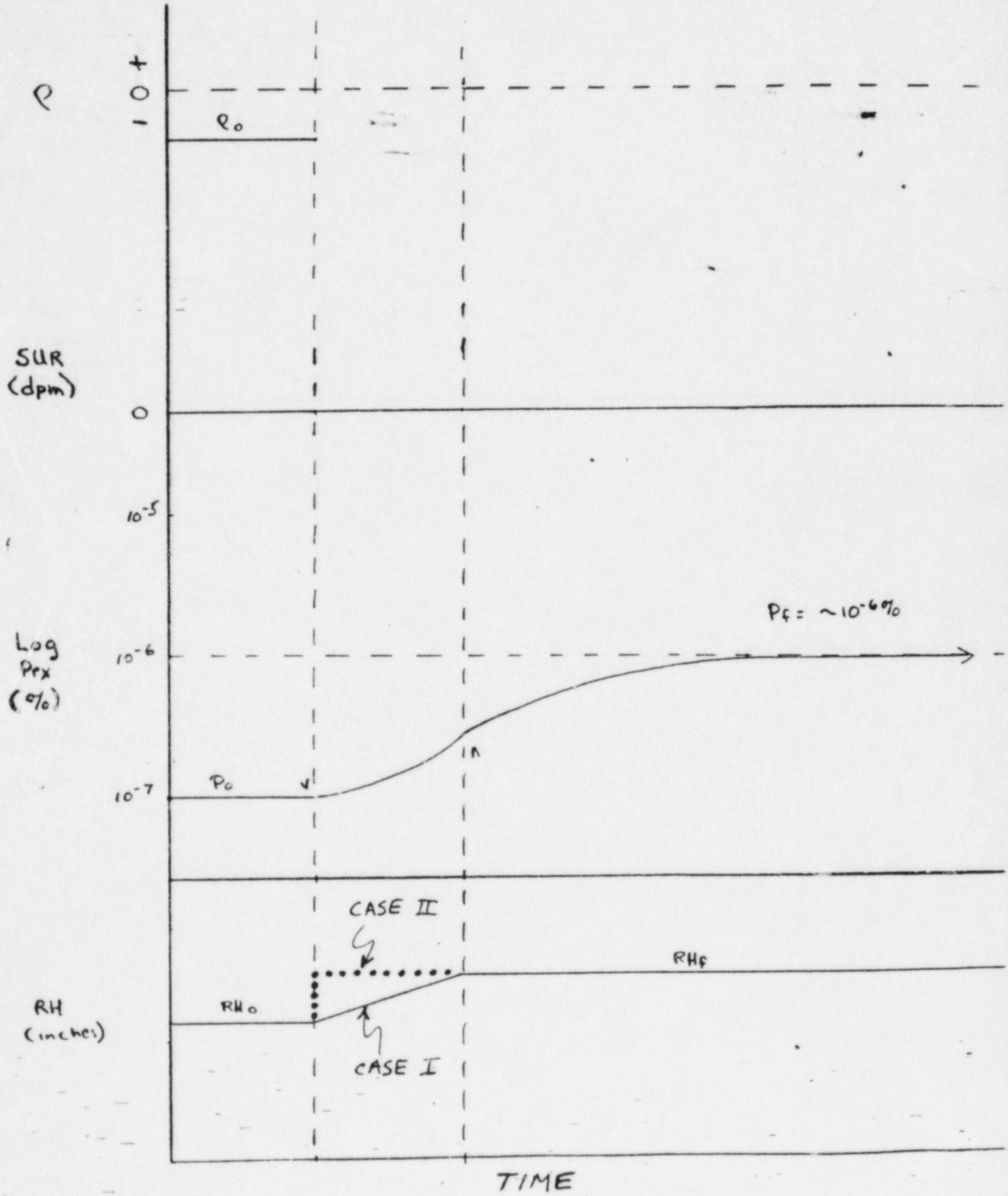


FIGURE 1-2

SOURCE RANGE BEHAVIOR

Subcritical to Critical Rod Pull



QUESTION 2.01 (1.50)

Describe how the Vessel Seal Leak Detection System is designed to monitor leakage and where the leakage goes upon leaving the seal.

QUESTION 2.02 (3.50)

Answer the following questions regarding the design of the Engineered Safety Features System:

a. The design purpose of the HPSI system during a Main Steam Line Rupture event is to (PREVENT/MITIGATE) damage to fuel. Choose one. [0.5]

b. In addition to core cooling and inventory maintenance, what other purpose does the HPSI system serve during a Main Steam Line Break? [0.5]

c. Describe two paths that reactor decay heat thermal energy would take ~~immediatel~~y following a Design Basis LOCI. Start with the core and continue to the ultimate heat sink. Be sure to list each component or system used to transfered this heat. [1.8]
One path - prior to CSAS, Second path - after RAS.

d. In order of preference list two means of removing Hydrogen from containment following a LOCI. [0.7]

QUESTION 2.03 (1.00)

What automatic actions should occur to the Containment Cooling Fans and their associated support systems following a CSAS?

QUESTION 2.04 (1.00)

Answer the following questions regarding the Emergency Diesel Generator (EDG) System:

a. What minimum number of EDG starts can be provided by the two starting air flasks associated with each diesel assuming they are initially full and are not recharged?[0.5]

b. How soon should the EDG be at rated speed and voltage following an auto start? [0.5]

QUESTION 2.05 (1.50)

a. How long should the Unit 1 main generator continue to supply the 500 KV Black Bus after a turbine trip if no other source of power to the Black Bus exists or is available? [1.0]

b. What component(s) is/are maintained energized during this time and why? [0.5]

QUESTION 2.06 (1.50)

Explain how power supply reliability is achieved for the 120 VAC instrument power buses. (Disregard computer power.)

QUESTION 2.07 (3.00)

Answer the following questions regarding a loss of instrument air assuming normal, at power, initial conditions.

a. How would a loss of instrument air header pressure due to a rupture just downstream of IA-144/146 (Air compressor isolation from air header) immediately affect the following components/systems. Choose ONE of the following for each component/system:

A fail open or flow maximum
B fail closed or flow stopped
C fail as is or flow cannot change
D no immediate effect or system functions normally

1) Main Feedwater Regulating Valves
2) Pressurizer spray valves
3) Letdown
4) Atmospheric Dump Valves
5) AFW regulating valves
6) EDG service water supply valves
7) Auxillary Spray valve [0.25 each]
8) Turbine AFW pump speed (if operating)

b. Describe two means of interconnecting the IA system with backup sources of air pressure. Indicate automatic setpoints, if any. Answer this question independently of part a above. [1.0]

QUESTION 2.08 (3.75)

For each system listed below, LIST the ESF related components which get cooling from the system immediately following a large break LOCI and STATE the effect on plant conditions of losing total system cooling flow to these components. Answer each part independently and assume no operator action.

a. Component Cooling Water
b. Service Water
c. Salt Water (do not include CCW and Service Water Heat Exchangers)

QUESTION 2.09 (4.25)

a. What will cause AND what action will result from a Recirculation Actuation Signal (RAS) ? [1.0]

b. It may be necessary to flush the core during long term cooling following a loss of coolant. Describe the two flow paths for performing this flush. [2.75]

c. Why is the flush in part b necessary? [0.5]

QUESTION 2.10 (2.00)

Describe what automatically happens in each of the following systems upon receiving a SIAS signal.

a. Chemical and Volume Control system [1.5]

b. Service water system [0.5]

QUESTION 2.11 (2.00)

Assume that a gaseous radioisotope is dissolved in the reactor coolant system. List the components in the flow path through which this gaseous radioisotope could be removed from the RCS, processed, and eventually released to the environment as part of a routine discharge.

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (1.50)

Will the plant trip as a result of the following simultaneous instrument failures? Explain your answers.

a. SUR channels A and B fail high during a startup, when reactor is critical at 10 -6%. [0.75]

b. SG-11 level channel A fails LOW and SG-12 level channel B fails HIGH while at 80% power. [0.75]

QUESTION 3.02 (2.00)

If one level indicator on each steam generator, 1-LT-1114A and 1-LT-1124B, which feed the logic matrix for the Auxiliary Feedwater Acuation System (AFAS) failed as is, would the AFAS be able to provide its protective function? Explain.

QUESTION 3.03 (2.00)

If Unit 1 is operating at 100% power and all root valves to the Steam Generator pressure safety channels are shut:

a. What two ESFAS/AFAS actuation signals would not function properly if a major steam leak developed in the containment? Include what equipment would not receive expected signals. [1.4]

b. What automatic Reactor Protection System trips are available to mitagate consequences of a major steam leak in the containment? [0.6]

QUESTION 3.04 (2.00)

The Unit 1 power plant is operating at 85% of full power with ASI = 0.0 and the CEA's at 90 inches on Group 5. Explain how (increase, decrease, remain the same) and WHY changes in the following parameters would affect the TM/LP pressure setpoint. Assume that the operators take any action necessary to maintain a constant electrical output. Consider each item separately.

a. ASI decreases (becomes negative) [1.0]

b. Bay-water temperature decreasing. [1.0]

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (2.50)

For each of the following sets of conditions, which ESFAS/AFAS signals, if any, should be actuated. Use the common abbreviations and answer each question independently.

a. A main steamline rupture (large break) occurs outside of containment upstream of S/G 11 MSIV while at 100% power. [1.0]

b. The reactor has been shutdown and is undergoing a cooldown. Pressurizer pressure is 1700 psia and the pressurizer pressure block has been activated. A large break LOCI occurs at the surge line.[1.0]

c. A containment area radiation monitor fails high. [0.5]

QUESTION 3.06 (2.00)

Briefly describe the automatic actions, if any, associated with the following radiation monitor channels. Do not include alarm and indication.

a. Condenser Air Removal Discharge Monitor [0.5]

b. Steam Generator Blowdown Recovery Radiation Monitor [0.5]

c. Main Steam Effluent Radiation Monitor (any of three channels) [0.5]

d. Control Room Ventilation Supply Radiation Monitor [0.5]

QUESTION 3.07 (1.50)

What actions should automatically occur in the Pressurizer Level and Pressure Control Systems if turbine generator load dropped from 100% to 70%? The decrease in load may be considered an instantaneous ~~trip~~ drop.

QUESTION 3.08 (2.00)

a. How many Hot and Cold Leg Temperature instruments are there in a single loop AND in what section of the primary loop are they located? [1.0]

b. What TWO systems are controlled by signals derived from loop temperatures? [1.0]

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.09 (2.00)

List the four conditions which will energize the Emergency Diesel Generator Shutdown Relay when a SIAS is present. Include coincidences if applicable.

QUESTION 3.10 (3.00)

Concerning the axial power distribution reactor trip;

a. What THREE inputs are used to generate the APD trip setpoint and where is each input received from? Do not include RCP combination input. [1.5]

b. What initiates an APD channel trip signal and how is the trip setpoint determined? [1.0]

c. In addition to being used to provide a reactor trip, what other function does the APD signal provide? Do not include indication. [0.5]

QUESTION 3.11 (2.25)

Briefly describe the instruments used to detect the following possible leakages:

a. Safety injection header check valve leakage. [0.75]

b. Letdown heat exchanger tube leakage. [0.75]

c. Pressurizer relief valve leakage. [0.75]

QUESTION 3.12 (2.25)

a. What automatically happens when reactor power decreases to a point where the EXTENDED RANGE MODE is activated for the wide range log channels? Explain why. [1.5]

b. At 10-4% increasing reactor power the level 2 bistable trips on. What THREE actions does this perform? [0.75]

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (1.50)

a. What are the Calvert Cliffs administrative limits concerning weekly, quarterly, and yearly whole body radiation dose for individuals older than 18 years? [0.9]

b. Whose approval is necessary prior to exceeding the weekly, the quarterly, or the yearly whole body radiation dose? [0.6]

QUESTION 4.02 (3.00)

Under what conditions should each of the following Emergency Operating Procedures be used?

a. EOP-1, Reactor Trip
b. EOP-2, Loss of Off-Site Power/Natural Circulation
c. EOP-4, Excess Steam Demand
d. EOP-8, Functional Recovery Procedure

QUESTION 4.03 (3.00)

For each of the below major steps of EOP-0 (Standard Post Trip Actions), list THREE parameters or conditions used to determine if the major step is satisfied. Include all parameter values where appropriate.

a. Verify RCS pressure and inventory control [1.5]

b. Verify core and RCS heat removal [1.5]

QUESTION 4.04 (3.00)

List SIX of the seven Vital Auxilaries.

A Vital Auxillary is one of the seven plant systems listed on EOP-0 which are necessary to carry out the EOP's.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.05 (3.00)

For each of the situations below, indicate whether the plant should be tripped immediately. For situations which do not require an immediate trip explain at what point a reactor trip, if any, is required assuming conditions continue to deteriorate and no operator action is taken. Assume the plant has been operating for 1 week at 90% power. Consider each situation separately.

a. The motor on the operating component cooling pump fails.

b. It is discovered that containment integrity has been breached when a blind flange is found improperly secured.

c. An unexplained dilution raises power by 5% and continues to rise.

d. Instrument air pressure drops to 45 psig.

e. The main journal bearing metal temperature is 230 F (5 F above the alarm set point) for the Unit 1 turbine.

f. The main journal bearing metal temperature is 225 F (5 F above the alarm set point) for the Unit 2 turbine.

QUESTION 4.06 (1.00)

Answer the following questions regarding AOP-9 ALTERNATE SAFE SHUTDOWN PROCEDURE/CONTROL ROOM EVACUATION.

a. Where do the Unit 1 Control Room Operator and Reactor Operator go INITIALLY if the control room must be evacuated? [0.5]

b. Who is designated to go to the Emergency Diesel Generator rooms? [0.5]

QUESTION 4.07 (2.00)

During a natural circulation cooldown, RCS voiding is indicated and the AOP-3F NATURAL CIRCULATION COOLDOWN actions of stopping letdown, stopping the cooldown, and pressurizing the RCS to maintain 200 degree subcooling are NOT effective in eliminating the RCS voids. What other TWO general methods could be used to reduce or eliminate the voided area?

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.08 (1.00)

A General Precaution in OI-2A CHEMICAL AND VOLUME CONTROL SYSTEM UNIT 2 states: "... letdown and charging flows should be started and stopped within 30 seconds of each other..." Why?

QUESTION 4.09 (3.00)

During the transition from shutdown cooling to RCP operation, the shutdown cooling pumps and the RCP's may both be de-energized for up to one hour. During this period of essentially no RCS flow, TWO adverse conditions may develop. What are these conditions and why do they adversely affect plant operations. One condition involves boron concentration and the other is related to RCS temperature.

QUESTION 4.10 (2.00)

If, following an inadvertant reactor trip, TWO CEA's do not fully insert, what actions should be taken. Include both the specific steps and the point at which the steps are considered complete.

QUESTION 4.11 (2.50)

a. In accordance with AOP-6A (High Reactor Coolant Activity) why can coolant activity increase during a heatup of the primary? [1.0]

b. What THREE indications do you have that coolant activity has increased? [1.5]

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

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

$W = v\ \Delta P$

$\Delta E = 931\ \Delta m$

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

$\dot{Q} = \dot{m} Cp \Delta t$

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

$Pwr = W_f \Delta h$

$I = I_o e^{-ux}$

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

$TVL = 1.3/u$

$HVL = -0.693/u$

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

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

$SUR = 26.06/T$

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

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

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

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

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\lambda\rho)$

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

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

$M = (1 - K_{effo})/(1 - K_{eff1})$

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

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

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

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

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

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^{\ 2} = I_2 d_2^{\ 2}$

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

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

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 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

ANSWER 1.01 (.50)

c (0.5)

REFERENCE
Reactor Theory Lesson RO-302-1-0 pp 37-39, 3-34 to 3-40
Learn. Obj. 3.1.2.3

ANSWER 1.02 (1.00)

1925 ppm (see Graph) [0.2 for correct axes, 0.1 for each correct point, 0.4 for correct answer]

REFERENCE
Lesson RO-302-2-0 Learn. Obj. 4.7.8 and pp 4-27 to 4-28

ANSWER 1.03 (2.00)

see graph

a. [0.5 for each trace]

b. [0.5]

c. [0.5]

REFERENCE
Reactor Theory module 4 Learn. Obj. 4.6.1 and pp 4-29 to 4-30

ANSWER 1.04 (2.00)

		MTC	FTC
MSLB	BOL	negative [0.25]	positive [0.25]
	EOL	positive [0.25]	negative [0.25]
CRWA	BOL	positive [0.25]	negative [0.25]
	EOL	negative [0.25]	negative [0.25]

FIGURE 1-1

ANSWER KEY

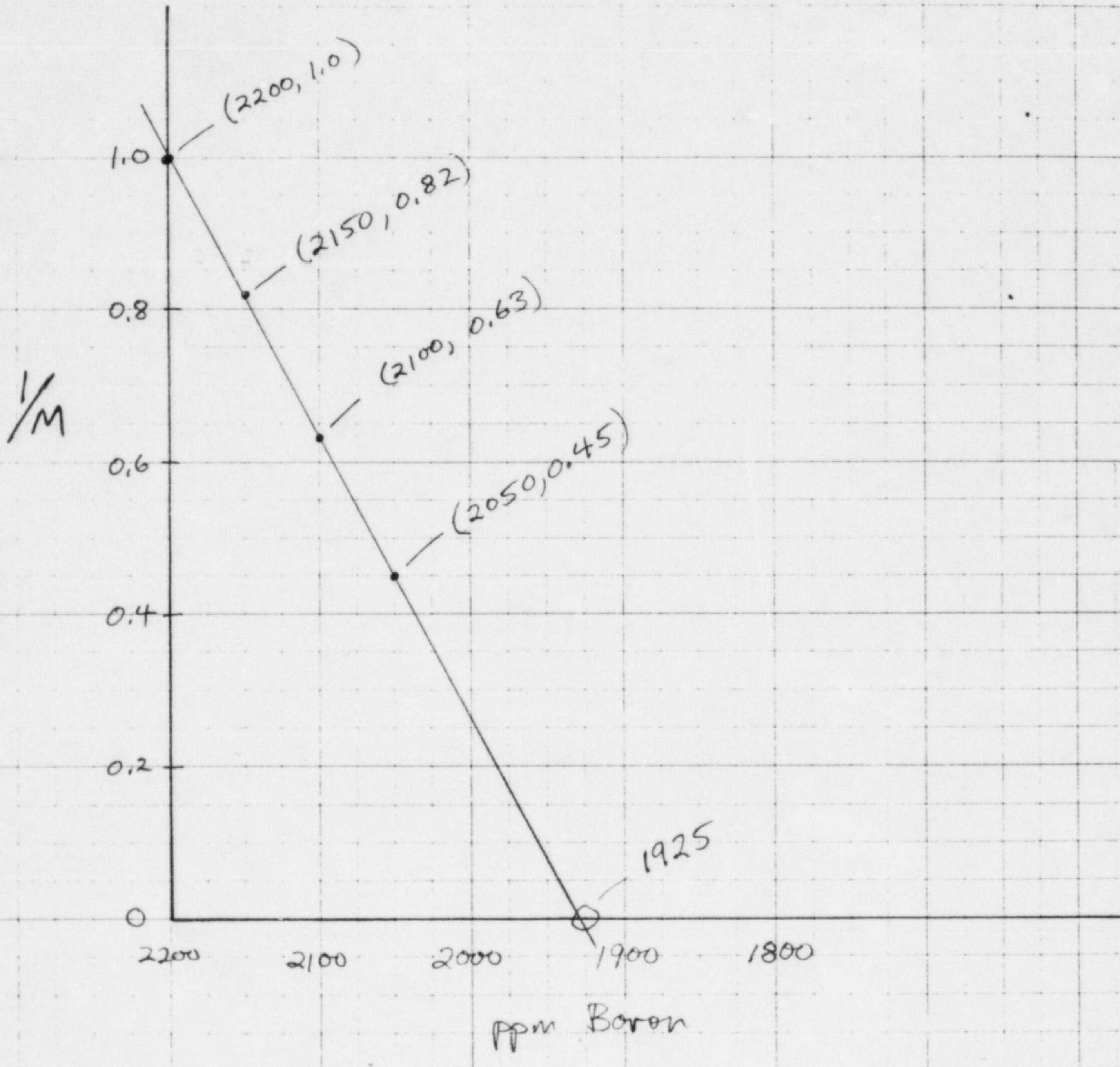
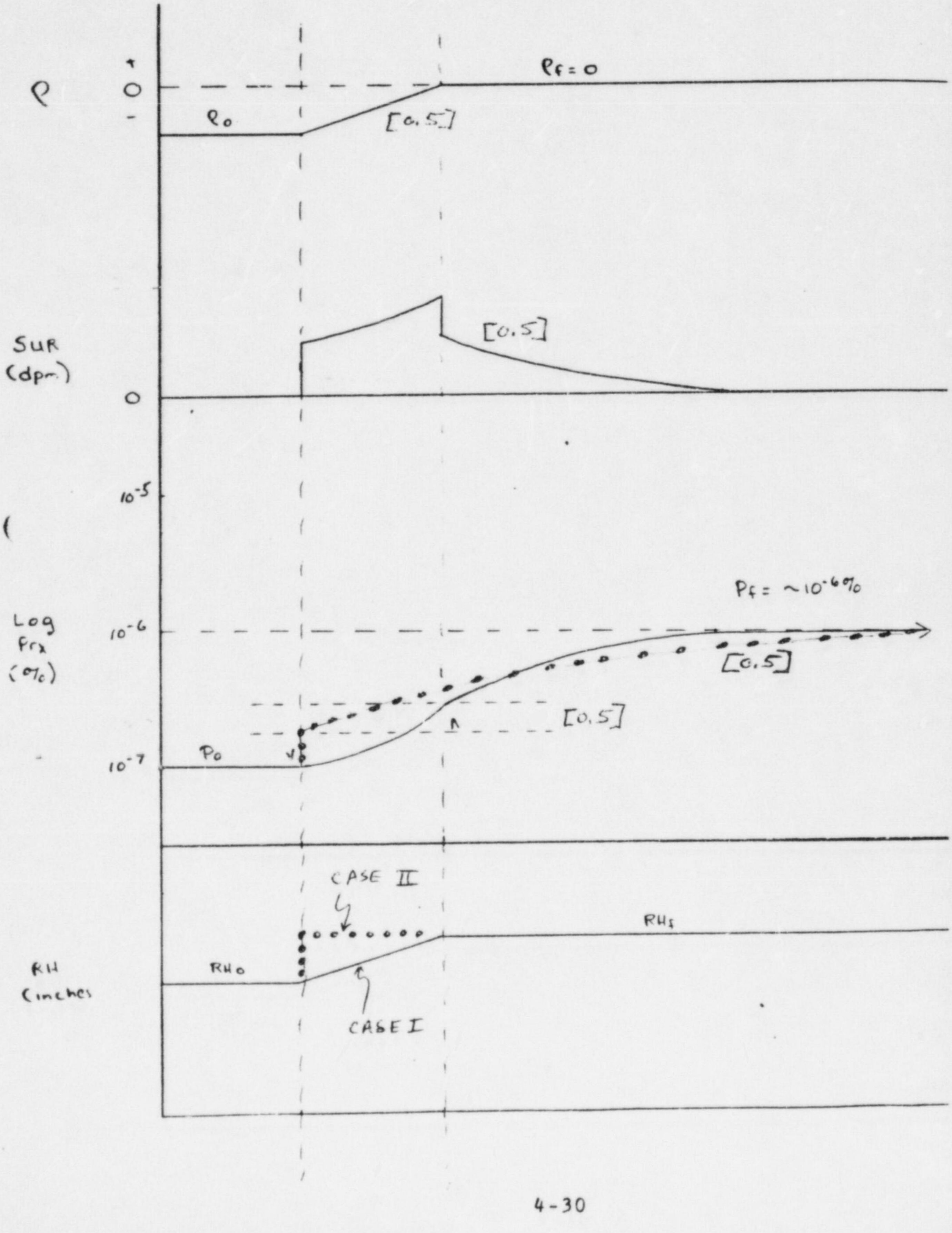


FIGURE 1-2

ANSWER KEY

SOURCE RANGE BEHAVIOR

Subcritical to Critical Rod Pull



REFERENCE
Reactor Theory module 3.0, Learn. Obj. 3.6.7c, 3.6.5, 3.7.3

ANSWER 1.05 (3.25)

$P = Po(10)^{sur(t)}$ [0.6]

$10^{-4} = 10^{-5} (10)^{1.666\ sur}$ [0.6]

solve for sur, sur = 0.6 DPM [0.3]

T = 26.06/sur = 26.06/0.6 = 43.43 sec [0.6]

rho = beta eff/(1 + lamba*bar X T) = 0.005/(1 + 0.08(43.43)) [0.8]

= 0.001117 = .112% delta k/k [0.35]

OR

$SUR = \frac{26 \lambda \rho}{\beta - \rho}$ [0,6]

$0.6 = \frac{26(.08)\rho}{.005 - \rho}$ [0.8]

$\rho = 0.112\% \frac{\Delta K}{K}$ (0.35]

REFERENCE
Reactor Theory module 4, Learn. Obj. 4.4.2 and pg 4-16

ANSWER 1.06 (2.50)

a. HIGHER

b. LOWER

c. HIGHER

d. SAME

e. LOWER [0.5 each]

REFERENCE
Reactor Theory module 3, Learn. Obj. 3.5, 3.12, 3.6, 3.10

ANSWERS -- CALVERT CLIFFS -86/08/11-COE,D.

ANSWER 1.07 (3.00)

a. Steam pressure decreases [0.3] due to thermal energy removal from the S/G [0.3]. Final value lower due to larger delta T (Tave - Tstm) required to drive more thermal energy across the S/G. [0.4]

b. Primary Tave decreases [0.3] due to Tstm now less than Tave which drives thermal energy from primary to secondary [0.3]. Final value lower due to positive reactivity needed from MTC to offset negative reactivity added by FTC [0.4]

c. Reactor power increases [0.3] due to positive reactivity from MTC [0.3]. Final value higher to supply new steam demand [0.4].

REFERENCE
Reactor Theory module 4, Learn. Obj. 4.7.12 and pg 4-35

ANSWER 1.08 (1.00)

a. 250 F [0.5]

b. equal to [0.5]

REFERENCE
Thermo module 9, Learn. Obj. 9.2
steam tables

ANSWER 1.09 (2.50)

a. Once through cooling [0.5] NO [0.5]

b. Boiling in the covered portion of the core [0.33]
Will accept "convection" for the covered region.
Superheating in the uncovered portion of the core [0.33]
Will accept "conduction" or "radiation" for the uncovered region.
Condensation in the S/G U-tubes (reflux boiling) [0.33]
will accept "convection" or "conduction" in this region.

c. Rapid increase in pressurizer level during Aux. sray [0.5]

REFERENCE
a. CEN-152 Rev 3 pg 5-31 and 5-34
EOP-5 Rev 0 pg 3

b. CEN-152 Rev 3 pg. 5-34 to 5-35 and pg. 5-91

c. EOP-5 Rev 0 pg. 17

ANSWER 1.10 (2.25)

1) Reactor power [0.25]. Increasing reactor power results in increased heat flux and DNBR decreases. [0.5]

2) Temperature [0.25]. If pressure is held constant and Tave is increased, subcooling will decrease. Therefore the heat flux required to reach DNB will decrease and the DNBR will decrease. [0.5]

3) Pressurizer pressure [0.25]. If Tave is held constant and pressure increased, subcooling increases and DNBR increases. [0.5]

REFERENCE
Thermo module 13.0, Learn. Obj. 13.6.3 and pp 25-28

ANSWER 1.11 (2.00)

a. Decreases, smaller delta T required to transfer energy to RCS. [0.5]

b. Increase, center line temperature responds to RCS temperature in order to maintain constant delta T across cladding. [0.5]

c. Decrease, fuel swelling and clad creep reduce clad gap which reduces delta T across gap and lowers center line temperature. [0.5]

d. No change, pressure has little effect on heat transfer in subcooled fluids. [0.5]

REFERENCE
Lesson Plan RO-301-13-0 module 13.0 Learn. Obj. 13.2, 13.3

ANSWER 1.12 (3.00)

a. Tcold will remain constant[0.25] Since it follows S/G saturation temperature. [0.5]

b. Thot will decrease[0.25] since less fission product heat is being produced than is being removed by the steam generators. [0.5]

c. Core delta T will decrease[0.25] since the amount of decay heat is decreasing. [0.5]

d. Loop transit time will increase [0.25] since the driving head for flow (core delta T) is decreasing. [0.5]

REFERENCE
Lesson Plan RO-301-14-0

ANSWERS -- CALVERT CLIFFS -86/08/11-COE,D.

ANSWER 2.01 (1.50)

Pressure is monitored in the line that taps off between the two concentric vessel "O" rings.[0.5] If the inner "O" ring leaks, the pressure will increase and an annunciator in the control room will be actuated [0.5]. Leakage goes to a manually isolated drain line inside containment [0.5] (will accept that leakage is held captive)

REFERENCE
S.D. 3 pg 7
plus CAF (no specific description found in reference material)

ANSWER 2.02 (3.50)

a. PREVENT [0.5]

b. To ensure adequate shutdown margin [0.5]

c. (1) Core -> Coolant recircing to sump [0.2] -> containment spray [0.2] -> SDC HX [0.2] -> CCW [0.2] -> Salt water system [0.1] -> Chesapeake Bay [0.1]
(2) Core -> boiloff to containment atmosphere [0.2] -> containment cooling fans [0.2] -> service water [0.2] -> salt water system [0.1] -> Chesapeake Bay [0.1] [1.8 total]

d. 1) H2 recombiners [0.25] 2) H2 purge system [0.25]
[0.2] for proper precedence

REFERENCE
a. S.D. 7 pg 58
b. S.D. 7 pg 1
c. S.D. 39 pg. 4, S.D. 38 pg. 1, S.D. 40 pg. 1
d. EOP 5 pg 14 and OI 41A and 41 B

ANSWER 2.03 (1.00)

All fans start or shift to low speed [0.5]
and 8 inch service water valves on cooler outlets receive an open signal. [0.5]

REFERENCE
S.D. 63 pg. 169
S.D. 39 pg. 20

ANSWER 2.04 (1.00)

a. four [0.5]

b. less than 10 sec[0.5]

REFERENCE
S.D. 48, pg. 46
Tech Spec 4.8.1.1.1.a.4

ANSWER 2.05 (1.50)

a. Until voltage drops to 80% of normal [0.5] or 20 seconds maximum [0.5]

b. To continue supplying RCP power to remove the initial decay heat [0.5]

REFERENCE
S.D. 50 pg. 44

ANSWER 2.06 (1.50)

Four (4) 120 VAC busses per unit are supplied directly by inverters.[0.25]
Each inverter is supplied by 125 VDC [0.5] or 120 VAC regulated power.[0.5]
Two inverters on each unit are supplied by ~~DC~~ AC power from the other unit. [0.25]

REFERENCE
Calvert Cliffs: SD No. 54, Figures 54-1 and 54-2

ANSWER 2.07 (3.00)

a. 1) C
2) B
3) B
4) D
5) D
6) A
7) D
8) A

b. 1) Autc 'alve to plant air system X-ties PA to IA at 85 psig IA pressure [0.5]

2) manual X-tie valve to Saltwater system air compressors. [0.5]

3) cross-connect Units 1&2 plant air systems [0.5]

[any 2 for 0.5 each]

REFERENCE
S.D. 32 pg 15
AOP-7D pp. 3-6
S.D. 41 Fig A-7 to A-9
S.D. 39 pg 21
AOP-7D pg 1

ANSWER 2.08 (3.75)

a. 1) SDC HX for contmt. spray [0.25]	ESF function of contmt. pressure/temp control met by 100% contmt. cooling fans capability [0.5]
2) HPSI/LPSI pump seals and bearings [0.25]	ESF function of injection flow met for up to 2 hours only without cooling [0.5]
b. 1) Contmt cooling fans [0.25]	ESF function of contmt. pressure/temp control met by 100% contmt. spray capability [0.5]
2) EDG cooling [0.25]	ESF function of emergency electrical power will be lost. [0.5]
c. 1) ECCS pump room air [0.25]	ESF function of HPSI/LPSI/CSP motor cooling lost [0.5]

REFERENCE
a. S.D. 40 pp 40-43
b. S.D. 7 pg 61 and S.D. 39 pg. 4
c. S.D. 38 pp. 3 and 29

ANSWER 2.09 (4.25)

a. 1. Caused by RWT level decreasing below approximately 30". [0.25]
(24" by T.S.)
2. Action that results: [0.25 ea.]

o Containment sump isolation valves open
o Both LPSI pumps stop
o Mini flow recirc. isolation valves receive a shut sig. [0.75]

b. 1. Containment sump > LPSI Pump > recirculation line > SDC return header > Hot leg [0.25 each]
AND
2. HPSI Pump > Aux. HPSI header > CVCS > Pzr. Aux. Spray > Surge line > Hot leg [0.25 each] [2.75]

c. To prevent boron precipitation and subsequent core flow channel blockage. [0.5]

REFERENCE
SD # 7&8, Pp. 65,67,68

ANSWER 2.10 (2.00)

a. 1. Boric acid pumps start [0.25]

2. Charging pumps start [0.25]

3. Boric acid storage tank is lined up to inject boric acid [0.25]

4. VCT makeup stop valve [0.25] and outlet valve shut [0.25]

5. Letdown line loop isolation valve shuts [0.25] (1.5)

b. 1. Two service water pumps start [0.25]

2. The turbine building SRW isolation valve shuts [0.25]

REFERENCE
ESFAS system description pg. 9

ANSWER 2.11 (2.00)

Letdown to degassifier through the CVCS. [0.3]
The degassifier removes the gas which is collected in the Waste Gas [0.3]
Surge Tank. [0.3]
Compressors move gas to [0.3] Waste Gas Decay Tanks. [0.3]
Vented through filters [0.2] and RMS before reaching main vent. [0.3]

REFERENCE
SD No. 14A: Waste Gas System, p 2, Fig. A-1

ANSWER 3.01 (1.50)

a. No [0.35] not until power reaches 10 -4%. [0.4]
b. No [0.35] channels auctioneer low signal therefore only channel A will trip. [0.4]

REFERENCE
SD No.: RPS, p 29, 31; Fig A-6, A-8

ANSWER 3.02 (2.00)

Yes protection would be provided. [0.8]
Three level detectors per SG are operable and would provide 2/3 redundancy to produce a trip signal. [1.2]

REFERENCE
SD No. 34: Auxiliary Feed System, p 45-47, 63-64

ANSWER 3.03 (2.00)

a. AFAS BLOCK: [0.2] AFW supply valves to faulted SG [0.2]
SGIS: [0.2] MSIV [0.2]
SGFP's [0.2]
Heater drain pumps [0.2]
Condensate booster pumps [0.2]

b. High power [0.2]
TM/LP [0.2] (ASGT)
Containment high pressure [0.2]
Low PZR pressure [0.2]
Low S/G ~~pressure~~ level [0.2]
[any 3 for 0.2 each]

REFERENCE
Preliminary Notification of Event or Unusual Occurrance PNO-I-85-55
DCS No. 50309/850808, Date August 1985.

ANSWER 3.04 (2.00)

a. Increase [0.5] because ASI decreasing will increase a penalty factor in the TM/LP calculation [0.5]

b. Decrease [0.5] because the reactor power had to decrease due to increasing turbine efficiency [0.5].

REFERENCE

S.D. 59 pp 10-17 and Fig A-6

ANSWER 3.05 (2.50)

a. SGIS [0.5] SIAS [0.5]

b. CIS [0.33] CSAS [0.33] SIAS [0.33]

c. none [0.5]

REFERENCE
S.D. Fig 63-6 to 63-14

ANSWER 3.06 (2.00)

a. none [0.5]

b. shuts B/D recovery discharge valves and diverts B/D flow to the Miscellaneous Waste Processing System [0.5]

c. none [0.5]

d. Outside dampers shut [0.2]
post LOCI fans start [0.1]
post LOCI filter dampers open [0.1]
~~normal control~~ room fans stop [0.1]
Kitchen & toilet

REFERENCE
S.D. 15 pp 46, 53, 63, 60

ANSWER 3.07 (1.50)

Letdown valve ramps open [0.5]
Backup heaters turn off (+9 inches) [0.25]
Backup heaters turn on (+12 inches) [0.25]
Spray valves open [0.5]

REFERENCE
S.D. 5 Fig. A-20, pg 65

ANSWER 3.08 (2.00)

a. Five T-hot in each loop located between Rx. Vessel and Steam Generator. [0.5]

Three T-cold per loop located between Coolant pump and Rx. Vessel. [0.5]

b. Pressurizer level, and Steam dump and by-pass system. [0.5 each] Will accept RRS, RPS, TM/LP, or ΔT ~~[2 req]~~ [2 required]

REFERENCE
SD 62, RCS Instrumentation; p 3

ANSWER 3.09 (2.00)

Engine overspeed [0.5]

~~2/3 [0.2]~~ lube oil low pressure switches trip ~~[0.3]~~ [0.5]

~~2/3 [0.2]~~ crankcase high pressure switches trip ~~[0.3]~~ [0.5]

cranking time control relay (start failure relay) time out [0.5]

REFERENCE
S.D. 48 pg 109

ANSWER 3.10 (3.00)

a. 1. Thermal power [0.25]-generated by the TM/LP calculator[0.25]

2. Nuclear power [0.25]-from the nuclear instruments [0.25]

3. CEA Function [0.25]-fixed input from the safety analysis [0.25]

b. A channel trip occurs if the axial shape index (YI) exceeds a positive or negative limiting value (Yp or Yn) [0.5] determined by Qmax, the largest of either NI power or thermal power [0.5]

c. Also generates the axial offset used in the TM/LP calculator [0.5]

REFERENCE
Reactor protection system description pg. 21,23 and fig. A-8

ANSWER 3.11 (2.25)

a. Press Transmitters downstream of the check valves leak off indicators. [0.75]

b. CCW head tank level + rad monitor. [0.75]

c. ~~Temp. detector downstream of relief valves. Hi temp alarm ; quench tank level and press; Acoustic monitors. [0.25 each]~~

Hi temp alarm [.15]
QT level [.15]
QT temp [.15]
QT press [.15]
Acoustic [.15]

REFERENCE
Safety Injection system description pg.26
CCW system description
Rx Core system description pg 7
RCS instrumentation system description pgs. 5&6

ANSWER 3.12 (2.25)

a. The dual section proportional counter is energized to provide additional neutron detecting sensitivity [1.0]. Also the indication changes from percent power to counts per second [0.5].(1.5)

b. 1. Enables the high SUR trip
2. Enables the SUR and TM/LP pretrip inputs to the CWP circuit
3. Energizes a level 2 status lamp [0.25 each] (1.5)
4. Enables PDIL function of the metrascope

any three

REFERENCE
Nuclear Inst. system description pgs. 20-21,23-24 and 25

ANSWERS -- CALVERT CLIFFS -86/08/11-COE,D.

ANSWER 4.01 (1.50)

a. Weekly 300 mrem
Quarterly 2.0 Rem — or 0.9 Rem (initial limit)
Yearly 4.0 Rem [0.9]

b. Individuals General supervisor and General supervisor-radiation safety [0.6]

REFERENCE
CCI-800B pgs. 9-10

ANSWER 4.02 (3.00)

a. Following Reactor Trip with no complications (no challenged safety functions)
b. Reactor shutdown, feed, condensate system, and all 4 RCPs unavailable because of loss of off-site power; or loss of all RCPs.
c. Unisolable leak upstream of either MSIV
d. One or more safety functions not met and/or diagnosis is not possible

[0.75 each]

REFERENCE
EOP 1, p 3
EOP 2, p 3
EOP 4, p 3
EOP 8, p 3

ANSWERS -- CALVERT CLIFFS -86/08/11-COE,D.

ANSWER 4.03 (3.00)

a. Verify pressurizer level [0.25] stabilizes between 80 to 180 inches [0.25].

Verify pressurizer pressure [0.25] stabilizes between 1850 and 2275 psia [0.25]

Verify RCS subcooling [0.25]greater than 30F [0.25]

b. Verify BOTH S/G levels [0.25] greater than -170 inches [0.25]

Verify feed rate is maintaining a constant or controlled increase in S/G level. [0.5]

(Verify proper operation of turbine bypass/ADVs)
S/G pressure stabilizes between 850 and 920 psia [0.25]

Tcold stabilizes between 525 and 535 degrees F [0.25]

REFERENCE
EOP-0

ANSWER 4.04 (3.00)

1. 4 KV vital buses (11, 21, 14, and 24)	[0.5]	
2. 125 vdc buses (11, 12, 21, and 22)	[0.5]	
3. 120 vac buses (21, 22, 23, and 24)	[0.5]	
4. Instrument Air pressure	[0.5]	
5. Component Cooling	[0.5]	
6. Service Water	[0.5]	
7. Salt Water	[0.5]	any 6

REFERENCE
EOP-0 pg 11

ANSWERS -- CALVERT CLIFFS -86/08/11-COE,D.

ANSWER 4.05 (3.00)

a. No trip [0.2] Trip if not restored in 10 min. [0.15] or alarm is recieved on RCP thrust bearing temperature. (>195 F) [0.15]
b. No trip. [0.2] Trip if Tech Spec requirements exceeded.[0.3]
c. No trip. [0.2] Trip if dilution raises power to RPS high power trip set point. [0.3]
d. Trip [0.5]
e. No trip. [0.2] Trip at 240 F. [0.3]
f. Trip reactor. [0.5]

REFERENCE
AOP 4, p 1
AOP 6, p 1-2
TS 3.6.1.1
~~AOP 7, p 4~~ AOP-1A
AOP 7D, p. 2
AOP 7E Unit 1, p 3
AOP 7E Unit 2, p 3

ANSWER 4.06 (1.00)

a. Unit 1 45 foot Switchgear room [0.5]
b. Outside operator (OSO) [0.5]

REFERENCE
Unit 1 AOP-9 pp 28-30

ANSWER 4.07 (2.00)

1) Cycle RCS pressure (within the limits of EOP Attachment 1) [1.0]

2) Operate Reactor Vessel head vent (per OI-1G) [1.0]

REFERENCE
Unit 1 AOP-3F pg 7

ANSWER 4.08 (1.00)

To minimize thermal transients in the system.

REFERENCE
Unit 2 OI-2A pg. 1

ANSWER 4.09 (3.00)

1) Dilution of RCS boron concentration [0.5] which would add positive reactivity [0.5] to the core as the stagnant slug of diluted water entered the core when RCP flow was restored. [0.5]

2) Core exit temperature [0.5] remains below saturation temperature[0.5] which otherwise would allow unintended boiling in the core region. [0.5]

Will accept other reasonable answers in absence of explicit reference.

REFERENCE
OP-1 pg 1 General Precaution F

ANSWER 4.10 (2.00)

Borate the RCS 400 ppm. [0.5]

by 1) opening charging pump suction direct feed valve (CVC-514-MOV) [0.5]

2) starting a boric acid pump [0.5]

3) starting all available charging pumps [0.5]

REFERENCE
EOP-0 pg 5

ANSWER 4.11 (2.50)

a. Coolant activity could increase due to activated corrosion products breaking loose from the metal surfaces and going into the coolant. (Will also accept fission product release) or Failed Fuel Monitor (1.0)

b. 1. Process radiation monitor countrate increase [0.5]
2. Process radiation monitor alarm [0.5]
3. Increase in coolant activity as determined by coolant sample [0.5] (1.5)

REFERENCE
AOP-~~10~~ 6A pg. 1

ATTACHMENT 2

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

FACILITY: CALVERT CLIFFS

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 86/08/12

EXAMINER: SHIRAKI

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	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	________	________	7. PROCEDURES - NORMAL. ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00		________ 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.

EQUATION SHEET

$\dot{\rho}_{RC} = (DRW)(\text{Rod speed})/\text{time}$

$\rho_{RL} = -\dot{\rho}_{RC}/\lambda_{eff}$

$\Delta t = (\rho_{RC} - \rho_0)/\dot{\rho}_{RC}$

$\rho_f = \dfrac{\rho_0(\beta_{eff} - \rho_0)}{\bar{\beta}_{eff} - \rho_f}$

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

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

$Pwr = W_f \dot{m}$

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

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

$SUR = 26.06/T$

$T = 1.44\ DT$

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

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

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

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

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

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

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

$\Sigma = N\sigma$

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

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}

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

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

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

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

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

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

QUESTION 5.01 (3.00)

Refer to FIGURE 5.1, a sketch of a typical auxiliary feedwater system utilizing two centrifugal pumps of identical characteristics and capacities. The plot of Volume Flow Rate versus Pressure shows the system with the "A" auxiliary feed pump in operation as the initial condition. (The "B" auxiliary feed pump discharge valve is shut.)

a) Show, on Figure 5.1, how the curve(s) will change as the PORV opens and reduces Steam Generator pressure by 50%.

b) Show, on Figure 5.1, how the curve(s) will change if the discharge valve is partially shut.

c) What effect will a decreased temperature of the water in the storage tank have on the Net Positive Suction Head of the pump?

QUESTION 5.02 (2.00)

What is the effect of the following operating transients on the cycle efficiency of the plant?

a) Decrease in Steam Demand

b) Loss of High Pressure Feedwater Heaters

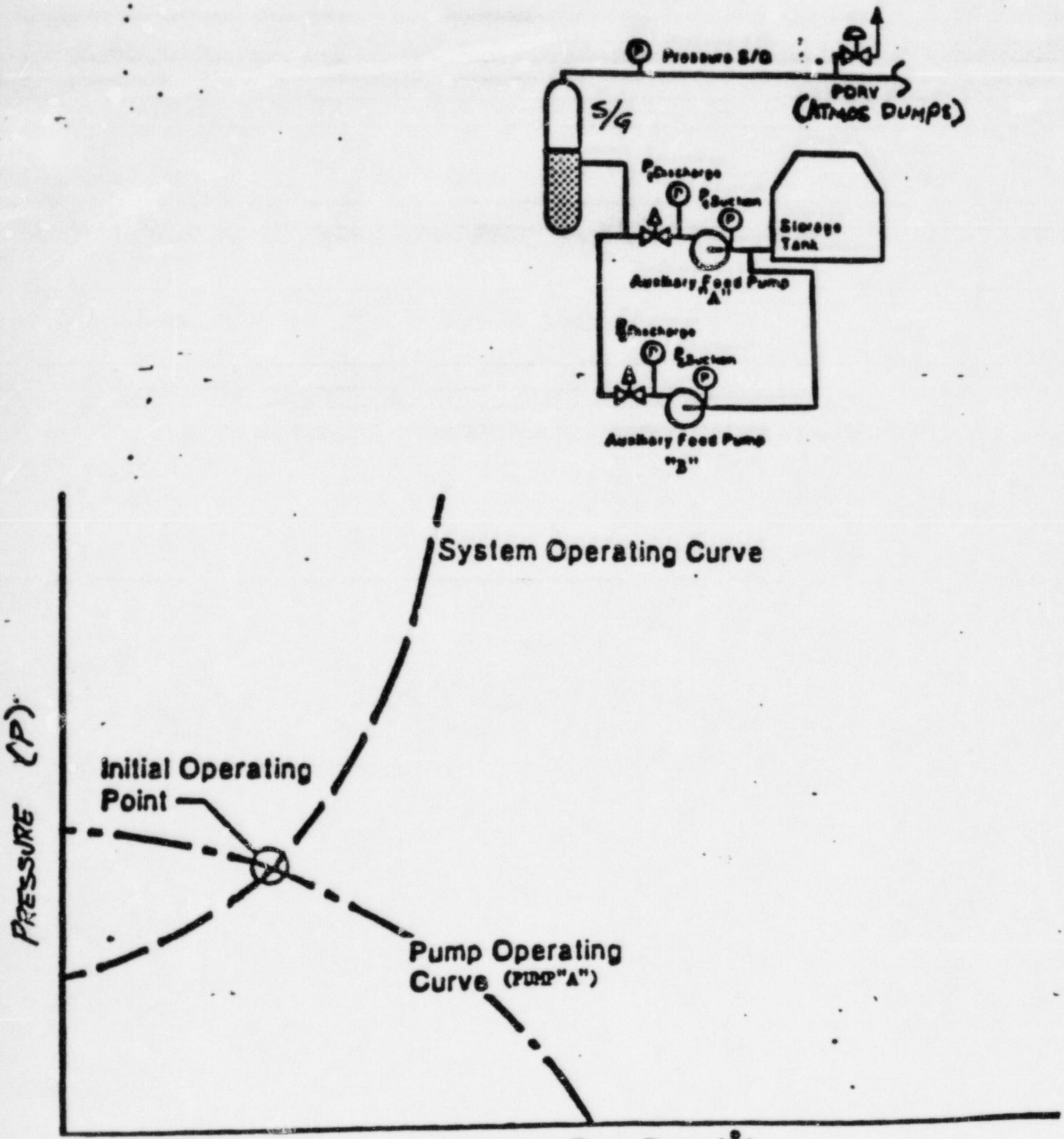
c) Increase in Sink Temperature (main condenser vacuum)

d) Loss of Reheat Steam

QUESTION 5.03 (2.00)

While critical at 10 E-4 %, a sudden reactivity insertion adds +0.4 % delta k/k to the reactor. If the operator starts inserting rods at 30 in/min with a differential rod worth (DRW) of 0.1 % delta k/k/in, how long would it take to turn power? Assume lambda eff = 0.1/sec and Beff = 0.007

FIGURE 5.1



Pressure S/G
PORV
(ATMOS DUMPS)
S/G
Storage Tank
Auxiliary Feed Pump "A"
Auxiliary Feed Pump "B"
System Operating Curve
Initial Operating Point
Pump Operating Curve (PUMP "A")
PRESSURE (P)
Volume Flow Rate ($\dot{V}$)

QUESTION 5.04 (2.50)

How does the variation of the following plant parameters affect the absolute value of differential boron worth (DBW)? Consider each question separately. Answer increase or decrease.

a) Higher Tavg

b) Increased boron concentration

c) Increased fission product poisons

d) Draw a graph showing the combined effect on the absolute value of differential boron worth of the decrease in boron concentration and increase in fission product poisons over core life. Assume constant Tavg and all control rods out.

QUESTION 5.05 (2.50)

Figure 5.2 is of Heat Flux versus Temperature Difference Between a Wall and the Bulk Fluid for an operating reactor. Note that there are two curves represented for two pressures (P1 < P2).

a) What is the principle type of heat transfer that is occurring at pressure P1 and 1.0 E4 BTU/Hr-ft between the wall and the bulk fluid?

b) What is the principle type of heat transfer that is occurring at pressure P2 and 1.0 E4 BTU/Hr-ft between the wall and the bulk fluid?

c) What pressure will yield a lower fuel center line temperature at 1.0 E4 BTU/Hr-ft?

d) What pressure will yield a higher fuel center line temperature at 3.0 E5 BTU/Hr-ft?

e) What type of heat transfer between the wall and the bulk fluid is occurring at Pressure P1 and 3.0 E5 BTU/Hr-ft?

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

QUESTION 5.06 (2.00)

During a normal increase in reactor power, the operator starts increasing reactor power and temperature with boron dilution or rod withdrawal and then increases steam demand by opening the turbine control valves.

a) As reactor power increases, what two coefficients oppose the reactivity added by the operator and cause power to level out?

b) What would be the final conditions of reactor power and coolant Tavg if steam demand were increased with no operator action to dilute or withdraw control rods?

QUESTION 5.07 (3.00)

Uranium fuel is the source of heat in the reactor.

a) Give three reasons for loading an excess of fuel at the beginning of core life.

b) What three means are used to control the resultant excess reactivity?

QUESTION 5.08 (1.50)

Compare the calculated Estimated Critical Condition (ECC) for a startup to be performed four hours after a trip from 100% power, to the actual control rod position, if the following events/conditions occurred. Consider each independently. Limit your answer to actual control rod position is higher than, lower than, or same as the ECC.

a) One reactor coolant pump is stopped two minutes prior to criticality.

b) The steam dump pressure setpoint is increased to a value just below the steam generator PORV setpoint.

c) All steam generator levels are being raised by 5% as the actual control rod position (criticality) is being reached.

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

QUESTION 5.09 (3.00)

After a secondary calorimetric and adjustment of the power range instruments, it is discovered that the Auxiliary Feedwater Pumps were operating.

a) How would this affect the indication of reactor power?

b) Give two reasons for this effect on indicated reactor power.

QUESTION 5.10 (1.50)

The severity of a rupture of a main steam line is affected by the moderator temperature coefficient (MTC).

a) How does the moderator temperature coefficient (MTC) differ from BOC to EOC?

b) What is the effect on RCS temperature of a main steam line break?

c) Why is a main steam line break a more severe accident at EOC than at BOC?

QUESTION 5.11 (2.00)

Following a reactor trip, the reactor settles into a start up rate (SUR) of about negative one third DPM.

a) What determines the value of this negative SUR?

b) Assuming Beff = 0.007 and reactivity of all rods = 8.5% p, calculate the power immediately after a trip from 100% power.

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

QUESTION 6.01 (3.50)

The atmospheric steam dump and turbine bypass controls provide automatic or operator control of the operation of the atmospheric steam dump and bypass valves during normal and emergency plant operation. Answer the following questions in reference to these valves.

a) When the reactor is operating at some power level between 8 percent and 63 percent, and the main turbine trips, how do the atmospheric steam dump and turbine bypass valves respond?

b) When the reactor is operating at some power level greater than 63 percent, and the main turbine trips, how do the atmospheric steam dump and turbine bypass valves respond?

c) What is the source of electrical operating power to the atmospheric steam dump and turbine bypass valves?

d) If the electrical operating power to the atmospheric steam dump and turbine bypass valves becomes unavailable for use, what feature(s) is(are) lost?

e) If the electrical operating power to the atmospheric steam dump and turbine bypass valves becomes unavailable for use, what capabilities for controlling the valves are still available in the Control Room?

QUESTION 6.02 (3.00)

If a no voltage or sustained undervoltage condition exists on a 4 kv ESF bus, the ESFAS generates the undervoltage, blocking, and sequencing signals necessary to automatically provide safe and reliable emergency power from the emergency diesel generators to the affected ESF bus.

a) If a loss of power occurs without an LOCI (SIAS not generated) how does the shutdown sequencer signal (SDS) respond?

b) If a loss of power occurs when an SIAS is present, what signals will have certain actuation subchannels blocked and then unblocked by the sequential actuation blocking signal (SASB) and the LOCI sequencer?

c) Assuming normal diesel generator switch lineup, how does the diesel generator 11 output breaker respond to an undervoltage condition on 4 kv bus 11?

d) What two conditions are necessary to produce an automatic closing of a diesel generator output breaker on to 4 kv bus 21?

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

QUESTION 6.03 (2.50)

During normal plant operation, the Containment Spray System is maintained in a standby mode with all of its components lined up for containment spray operation.

a) What two signals must be generated by the ESFAS in order to admit spray water into the containment?

b) How does the source of water differ during the injection mode and the recirculation mode of containment spray?

c) How is the trisodium phosphate carried into the containment spray system?

QUESTION 6.04 (3.00)

The Power Range Safety Instrumentation provides output signals to the Reactor Protection System (RPS) and the Internal Vibration Monitoring System.

a) If it is necessary to operate a channel with only one operable subchannel due to a failed detector, the (A + B)/2 switch is selected to the appropriate position. What are the three effects of this action?

b) At 15 percent increasing reactor power, the level 1 B/S trips off. What effect (enable or inhibit) does this have on the Loss of Load, APD, and HI SUR trips?

QUESTION 6.05 (3.00)

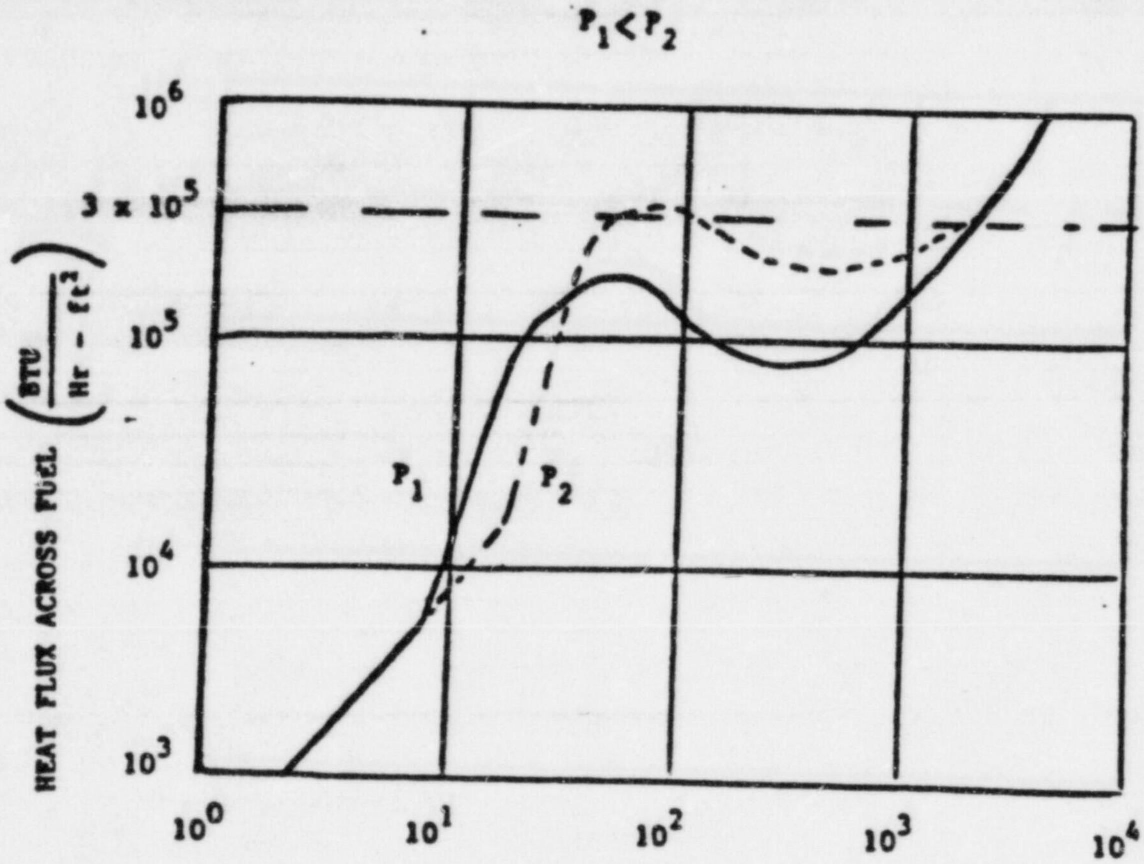
A portion of the water collected in the steam generator blowdown tank is circulated through the radiation monitor system.

a) What is the purpose of this radiation monitoring?

b) Assuming the ion exchangers are bypassed on high temperature, to what position will the following valves automatically move if the alarm setpoint is exceeded?

Surface blowdown isolation valve, bottom blowdown isolation valve, blowdown tank discharge valve to the condensers, blowdown tank discharge valve to the circulating water system, blowdown tank discharge valve to the miscellaneous waste processing system.

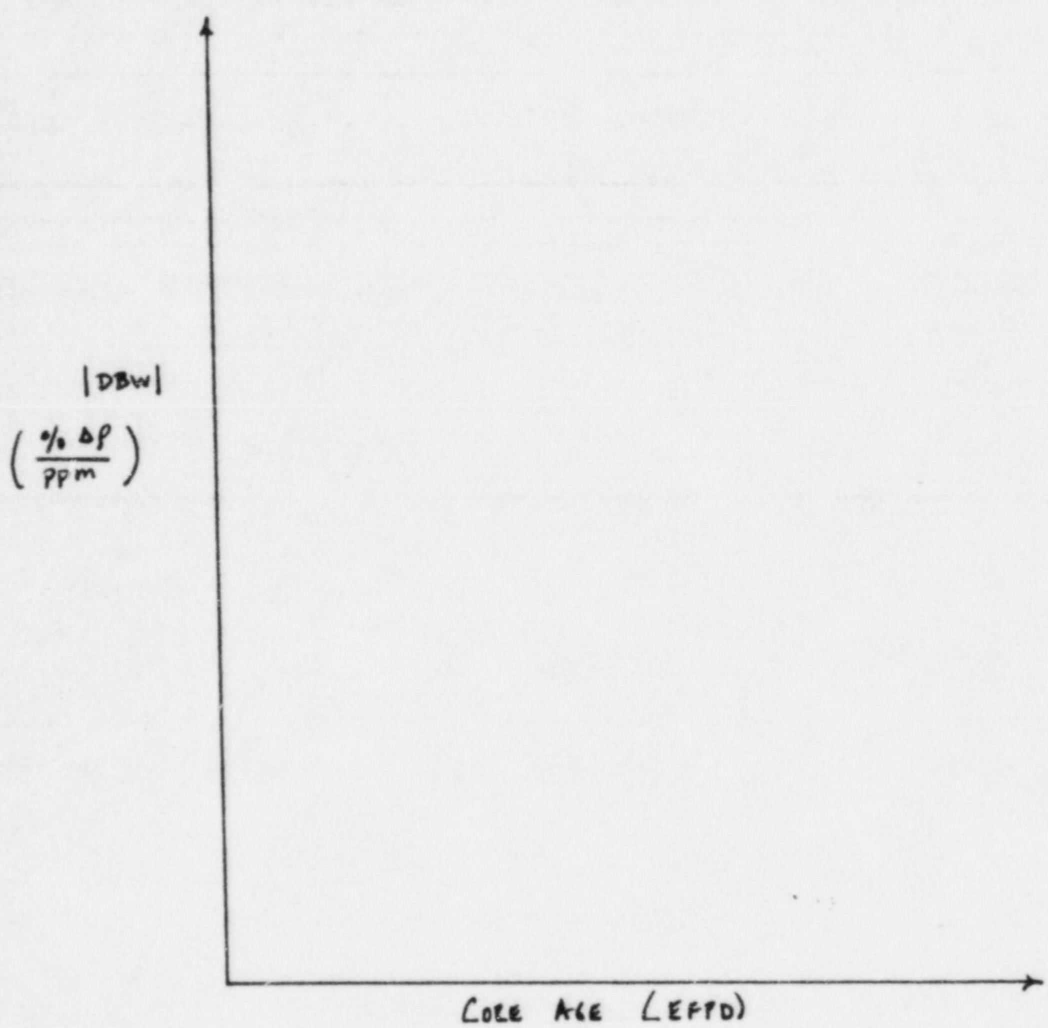
(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)



Temperature Difference Between Wall and Bulk Fluid (F°)

FIGURE 5.2

QUESTION 5.04



QUESTION 6.06 (2.50)

The feedwater regulating valves throttle the amount of feedwater going to each steam generator.

a) There are three modes (hard manual, soft manual, automatic) of operation of the feedwater regulating valve controllers. What is the function of each mode?

b) What is the function of the feedwater regulating valve override feature upon a main turbine trip?

QUESTION 6.07 (1.00)

The 120 VAC vital instrument buses are normally supplied by their associated single-phase inverters.

a) If an inverter is out of service, what is the source of power to the associated 120 VAC vital bus?

b) Changing the source of power to the associated 120 VAC vital instrument bus is accomplished with the manual transfer switch. Why is each manual transfer switch normally locked in the INVERTER position?

QUESTION 6.08 (2.00)

The Reactor Protective System (RPS) contains provisions for bypassing trip signals under certain conditions. One of the types of bypasses is the trip inhibit bypass.

a) Name one circumstance under which this bypass would be used.

b) How does the two-out-of-four channel coincidence logic change when an RPS bistable trip unit is place in BYPASS?

c) What two provisions ensure that not more than one channel of a trip unit is placed in BYPASS at a time?

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

QUESTION 6.09 (2.50)

Assume that the controlling pressure transmitter for the Pressurizer Pressure Control System has failed high while in service.

a) What alarm would occur in the Control Room?

b) How will the spray valve(s) and the heaters respond?

c) With no operator action, what will be the affect on the reactor plant?

QUESTION 6.10 (2.00)

The Instrument Air Service Header distributes instrument air to all of the IA System loads located throughout the plant.

a) To enter the containment, instrument air flows throughout containment isolation valve 1(2)-IA-2080. What ESFAS signal will automatically shut this valve?

b) In what position will containment isolation valve 1(2)-IA-2080 fail upon loss of operating power?

c) After passing into the containment, one of the IA branches passes through containment instrument air control valve 1(2)-IA-2085. In what position will this valve fail upon loss of air or power?

d) Valve 1(2)-IA-2085 will shut automatically when IA pressure drops to 75 psig. What is required to reopen the valve after IA pressure has been restored?

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

QUESTION 7.01 (3.00)

List the six Post Trip Immediate Actions.

QUESTION 7.02 (3.00)

The Functional Recovery Procedure (FRP) is designed to provide the Control Room with a systematic and structured response to plant casualties.

a) What is the objective of the FRP?

b) What are the four circumstances under which the Operator should enter the Functional Recovery Procedure?

QUESTION 7.03 (2.50)

During the performance of the Functional Recovery Procedure, the RCS is depressurized to maintain a maximum of 200 degrees F subcooling.

a) List the four methods of depressurization in order of preference.

b) During depressurization, what does a high, increasing pressurizer level indicate?

QUESTION 7.04 (3.00)

EOP-6, the steam generator tube rupture procedure contains the optimal recovery guidelines for this casualty.

a) List four indications of a steam generator tube rupture (SGTR).

b) Give two methods of identifying the SG with the ruptured tube.

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

QUESTION 7.05 (2.50)

During a loss of off site power, natural circulation must be established.

a) While establishing the SG as a heat sink, increased loop transport time causes a 5 to 10 minute delay in temperature responses to a plant change. What two plant parameters provide better indications of RCS response during this period?

b) Give three plant parameters that can be analyzed in order to verify natural circulation.

QUESTION 7.06 (3.00)

During a reactor startup, the regulating CEA's are withdrawn using the manual sequential mode.

a) What action is required for each CEA as the group reaches the upper group stop?

b) What is the maximum allowed sustained startup rate?

c) Within 15 minutes prior to achieving reactor criticality, what is the minimum RCS temperature (Tavg) allowed?

d) What are two Technical Specification bases for the temperature limitation of c) above?

QUESTION 7.07 (2.50)

During normal operation, the Auxiliary Feed System is maintained in a standby mode with its components lined up for automatic actuation.

a) What is the status of the two turbine driven AFW pumps during normal operation?

b) What three automatic actions are accomplished by an AFAS START A signal?

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

QUESTION 7.08 (3.50)

The letdown heat exchanger component cooling outlet control valve is controlled in manual due to poor throttling characteristics of the valve in the desired flow range.

a) Why might changing a major heat load on the component cooling system such as securing a liquid waste evaporator cause reactor power to increase?

b) What action can be taken during a major component cooling load change to preclude this power increase?

QUESTION 7.09 (2.00)

If, following an inadvertent reactor trip, TWO CEA's do not fully insert, what actions should be taken. Include both the specific steps and the point at which the steps are considered complete.

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

QUESTION 8.01 (2.50)

Changes to procedures require different levels of review depending on the initial approving official.

a) For a change to a Surveillance Test Procedure which was reviewed by POSRC, from whom must approval be obtained? Specify the number of individuals, qualification, and level of supervision.

b) Who must approve a change to an Emergency Operating Procedure? Specify the number of individuals, qualification and level of supervision.

c) If the change alters a step in which a QC hold was inserted, what further concurrence is required?

QUESTION 8.02 (2.00)

Radiation dose limits are closely monitored and controlled.

a) What is the maximum permissible occupational radiation dose for individuals 18 years of age and older per calendar quarter to the whole body, head and trunk, blood forming organs, lens of the eyes, or gonads.

b) What is the weekly administrative exposure limit for individuals 18 years of age or older to the whole body, head, trunk, blood forming organs, lens of the eyes or gonads.

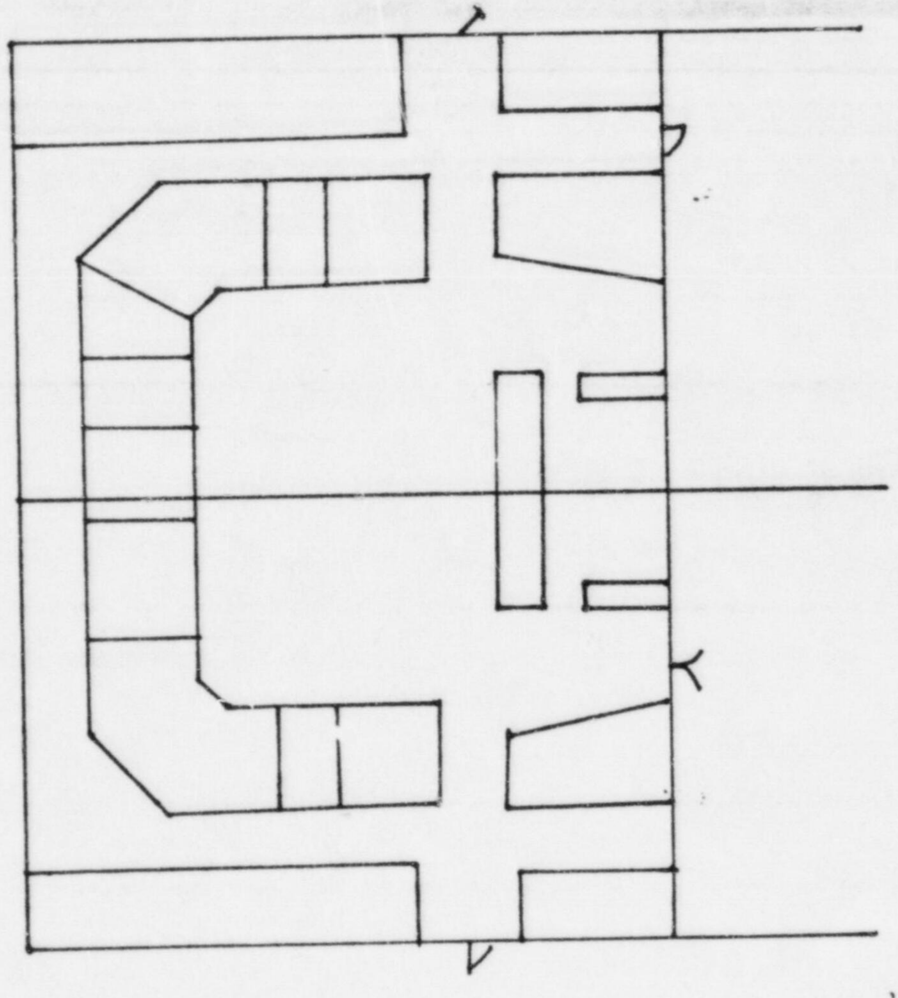
c) At what quarterly dose accumulation does the exposure limit of b) above change?

d) What is the new weekly exposure limit after the change that takes place in c) above?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

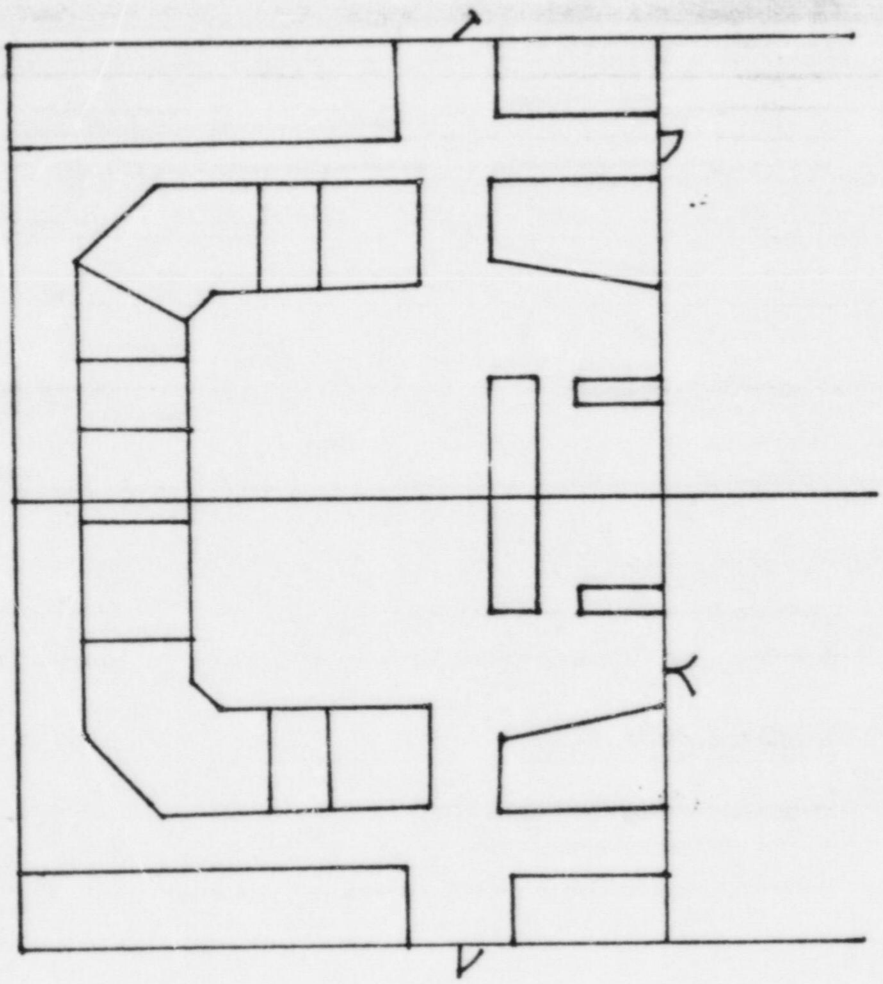
ROUTINE SURVEILLANCE AREA

UNIT 1 CONTROL ROOM OPERATOR

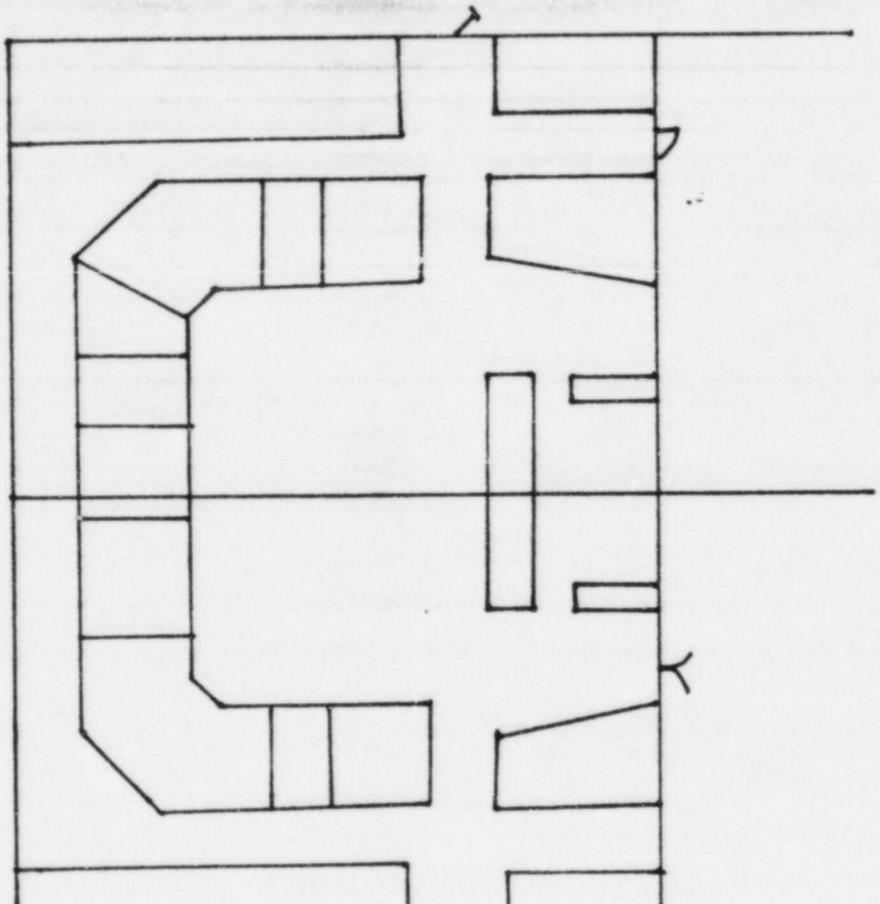


EMERGENCY SURVEILLANCE AREA

UNIT 1 CONTROL ROOM OPERATOR

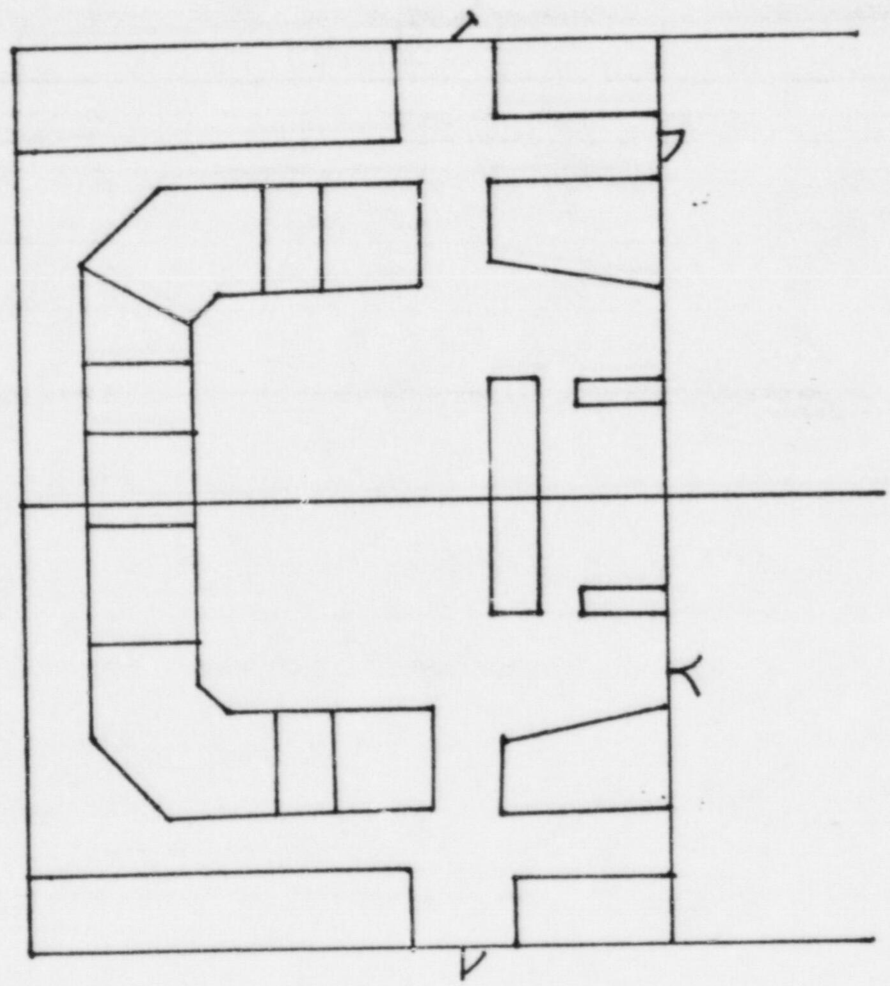


Routine Surveillance Area
Unit 2 Control Room Operator



EMERGENCY SURVEILLANCE AREA

UNIT 2 CONTROL ROOM OPERATOR



QUESTION 8.03 (1.50)

Fill in the blanks

Surveillance Requirements are required by Technical Specifications at a certain periodicity. Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed a) % of the surveillance interval, and

b. The combined time interval for any b) consecutive surveillance intervals not to exceed c) times the specified surveillance interval.

QUESTION 8.04 (3.00)

Technical Specification Safety Limits

a) For the Reactor Core, what three parameters are limited by a curve in the Technical Specifications?

b) What is the Technical Specification Safety Limit for Reactor Coolant System Pressure?

c) State two actions that are required in the event of a violation of a Safety Limit.

QUESTION 8.05 (3.50)

The Control Room Operator should not normally leave the area where continuous attention can be given to reactor operating conditions and where he has access to the reactor controls.

a) On the attached sheets, show the routine and emergency surveillance areas for Units 1 and 2.

b) If the Control Room Operator must leave the surveillance area for a short period of time, the individuals filling five other positions may relieve him. What are three of those positions?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

Shift Staffing Requirements

a) With both units operating in Modes 1 - 4, what is the minimum shift staffing (by license type) allowed?

b) For what period of time may shift crew composition be less than the minimum requirements in order to accommodate unexpected absence of on duty shift crew members?

c) If an absence such as that in b) is necessary, what action must be taken?

QUESTION 8.07 (1.50)

In accordance with 10 CFR 20, "Standards for Protection Against Radiation":

a) What is a Radiation Area?

b) What is a High Radiation Area?

QUESTION 8.08 (2.50)

In accordance with 10 CFR 55, "Operators' Licenses":

a) As defined in 10 CFR 55, when is an individual deemed to be operating the controls of a nuclear facility?

b) What are the "controls" as defined in 10 CFR 55?

c) According to the "Exemptions from License" provisions of 10 CFR 55, under what circumstances may an individual manipulate the reactor controls without a license?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.09 (1.00)

10 CFR 50.54(x) allows a licensee to take reasonable action that departs from a license condition or Technical Specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent.

a) What position or qualification must be held by the individual approving such an action?

QUESTION 8.10 (2.00)

Refer to 10 CFR 50.72 and/or 10 CFR 20.403 attached. Which of the following events require immediate notification (within a period of one hour or sooner) to the NRC Operations Center via the Emergency Notification System?

a) Personnel exposure to an individual's hands of 200 rem while performing steam generator tube repairs.

b) Personnel exposure to an individual's whole body of 30 rem while working in a steam generator.

c) Declaration of an "Unusual Event" at CCNPP.

d) Taking the plant from mode 1 to mode 3 to comply with Technical Specification requirements.

f) Discovery in mode 1 that two safety injection level instruments were improperly calibrated and that the levels have been above or below Technical Specification limits for two weeks.

QUESTION 8.11 (3.50)

The CCNPP Technical Specifications require that shutdown margin be greater than 3.5 % delta k/k while in modes 1 - 4.

a) What is shutdown margin?

b) What accident conditions are the basis for the shutdown margin restriction while in modes 1-4?

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

QUESTION B.10

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) *General requirements.*[1] (1) Each nuclear power reactor licensee licensed under § 50.21(b) or § 50.22 of this part shall notify the NRC Operations Center via the Emergency Notification System of:

(i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan;[2] or

(ii) Of those non-Emergency events specified in paragraph (b) of this section.

(2) If the Emergency Notification System is inoperative, the licensee shall make the required notifications via commercial telephone service, other dedicated telephone system, or any other method which will ensure that a report is made as soon as practical to the NRC Operations Center.[3]

(3) The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes.

(4) When making a report under paragraph (a)(3) of this section, the licensee shall identify:

(i) The Emergency Class declared; or

(ii) Either paragraph (b)(1), "One-Hour Report," or paragraph (b)(2), "Four-Hour Report," as the paragraph of this section requiring notification of the Non-Emergency Event.

(b) *Non-emergency events*—(1) *One-hour reports.* If not reported as a declaration of an Emergency Class under paragraph (a) of this section, the licensee shall notify the NRC as soon as practical and in all cases within one hour of the occurrence of any of the following:

(i)(A) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

(B) Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part.

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

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

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

(C) In a condition not covered by the plant's operating and emergency procedures.

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

(iv) Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal.

(v) Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system).

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

(2) *Four-hour reports.* If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following:

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

(ii) Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported.

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

(A) Shut down the reactor and maintain it in a safe shutdown condition,

(B) Remove residual heat,

(C) Control the release of radioactive material, or

(D) Mitigate the consequences of an accident.

(iv)(A) Any airborne radioactive release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour.

(B) Any liquid effluent release that exceeds 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour. (Immediate notifications made under this paragraph also satisfy the requirements of paragraphs (a)(2) and (b)(2) of § 20.403 of Part 20 of this chapter.)

(v) Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

(vi) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.

(c) *Followup notification.* With respect to the telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee, shall during the course of the event:

(1) *Immediately report* (i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.

(2) *Immediately report* (i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effective measures of response or protective measures taken, and (iii) information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

[48 FR 39046, Aug. 29, 1983; 48 FR 40882, Sept. 12, 1983]

[1] Other requirements for immediate notification of the NRC by licensed operating nuclear power reactors are contained elsewhere in this chapter, in particular, §§ 20.205, 20.403, 50.36, and 73.71.

[2] These Emergency Classes are addressed in Appendix E of this part.

[3] Commercial telephone number of the NRC Operations Center is (202) 951-0550.

QUESTION B.10

§ 20.403 Notifications of incidents.

(a) *Immediate notification.* Each licensee shall immediately report any events involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more or radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II of this part; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

(b) *Twenty-four hour notification.* Each licensee shall within 24 hours of discovery of the event, report any event involving licensed material possessed by the licensee that may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II of this part; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

(d) Reports made by licensees in response to the requirements of this section must be made as follows:

(1) Licensees that have an installed Emergency Notification System shall make the reports required by paragraphs (a) and (b) of this section to the NRC Operations Center in accordance with § 50.72 of this chapter.

(2) All other licensees shall make the reports required by paragraphs (a) and (b) of this section by telephone and by telegram, mailgram, or facsimile to the Administrator of the appropriate NRC Regional Office listed in Appendix D of this part.

[27 FR 5905, June 22, 1962, as amended at 28 FR 6823, July 3, 1963; 42 FR 43965, Sept. 1, 1977; 43 FR 2719, Jan. 19, 1978; 48 FR 33859, July 26, 1983]

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 5.01 (3.00)

a) Attached (1.0)

b) Attached (1.0)

c) Increases (1.0)

REFERENCE
CCNPP Lesson Plan # RO-301-9-0 - Steady Flow General Energy Equation

ANSWER 5.02 (2.00)

a) Increase

b) Decrease

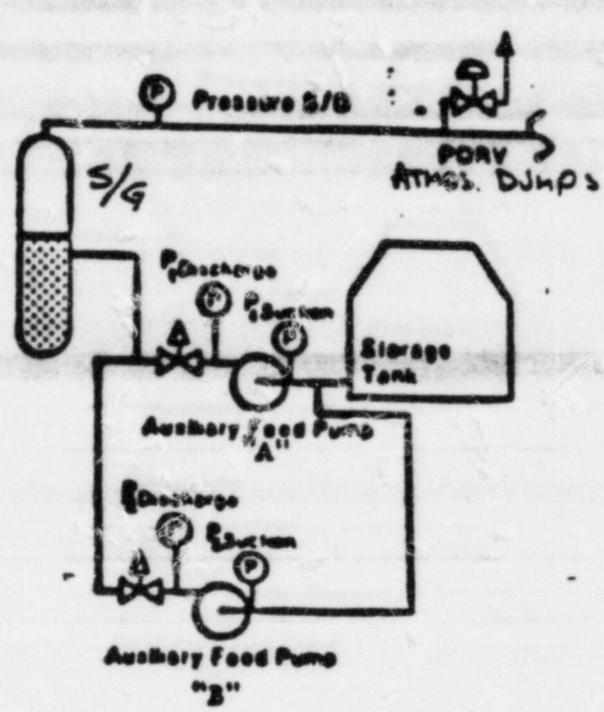
c) Decrease

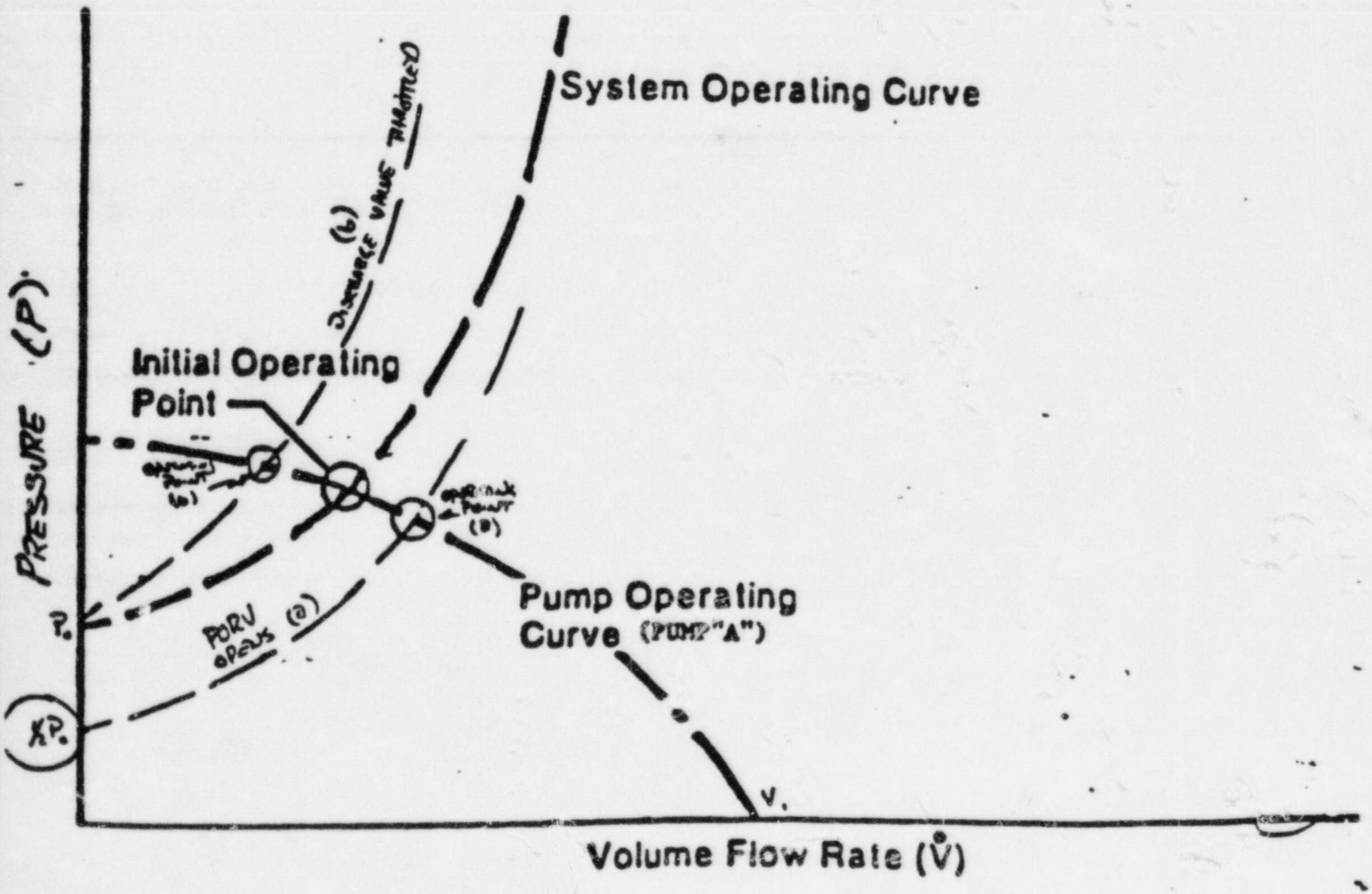
d) Decrease

REFERENCE
CCNPP Lesson Plan RO-301-10-0 - Plant Cycle Analysis

KEY

FIGURE 5.1





ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 5.03 (2.00)

The reactivity addition rate used to turn power is: (P = delta k/k)

$$\dot{\rho}pt = (DRW)(Rod\ Speed)/60\ sec/min$$

$$= (-0.1\ \%\ \rho/in)(30\ in/min)/60\ sec/min$$

$$\dot{\rho}pt = -0.05\ \%\ \rho/sec$$

The amount of reactivity to be overcome to turn power is:

$$\rho pt = -\dot{\rho}pt/\ \lambda\ eff$$

$$= -(-0.05\ \%\ \rho/sec)/0.1/sec$$

$$\rho pt = +0.5\ \%\ \rho$$

The time required to turn power is:

$$delta\ t = (\rho pt - \rho o)/\dot{\rho}pt$$

$$= (0.5\ \%\ \rho - 0.4\ \%\ \rho)/-.05\ \%\ \rho/sec$$

$$delta\ t = -2\ seconds$$

REFERENCE
CCNPP Lesson Plan RO-302-2-0 - Reactor Kinetics

ANSWER 5.04 (2.50)

a) The DBW decreases

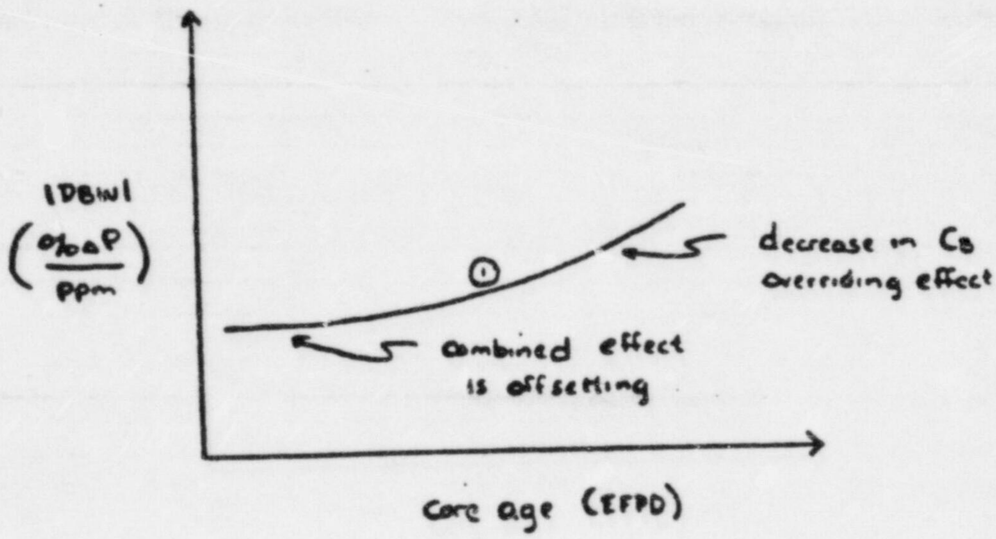
b) The DBW decreases

c) The DBW decreases

d) The graph should show that at BOL, the combined effect is offsetting and the absolute value of DBW is almost constant. At EOL, the decrease in boron concentration is the overriding effect, and the absolute value of DBW goes up.

REFERENCE
CCNPP Lesson Plan RO-302-1-0 - Reactivity Factors

QUESTION 5.04 KEY



ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 5.05 (2.50)

a) Nucleate boiling

b) Convection or single phase heat transfer

c) Pressure i

d) Pressure i

e) Radiant heat transter (film boiling or steam blanketing also acceotable)

REFERENCE
CCNPP Lesson Plan RO-301-13-0 - Reactor Heat Generation

ANSWER 5.06 (2.00)

a) Moderator temperature and fuel temperature coefficients

b) Increased reactor power and decreased coolant Tavg

REFERENCE
CCNPP Lesson Plan RO-302-2-0 - Reactor Kinetics

ANSWER 5.07 (3.00)

a) 1. Override temperature effects
2. Compensate for fuel depletion
3. Overcome effects of fission product poison buildup

b) 1. Control rods
2. Soluble boron
3. Burnable poisons (boron)

REFERENCE
CCNPP Lesson Plan RO-302-1-0 - Reactivity Factors

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 5.08 (1.50)

a) Same

b) Higher

c) Lower

REFERENCE
CCNPP Lesson Plan SRO-302-2-0 - Slowing Down Theory and Keff for SRO Upgrade

CCNPP Lesson Plan RO-302-1-0 - Reactivity Factors

ANSWER 5.09 (3.00)

a) Indicated reactor power would be lower than actual reactor power.

b) 1) Actual feedwater temperature would be lower than that used in the calorimetric calculation.

2) The feedwater mass flow used in the calculation would be lower than actual. (Since AFW flow bypasses feedflow indication.)

REFERENCE
CCNPP Lesson Plan RO-301-10-0 - Plant Cycle Analysis

ANSWER 5.10 (1.50)

a) The MTC is less negative at BOC than at EOC.

b) It causes a sudden cooling of the RCS.

c) Since at EOC the MTC is more negative, the cooling of the RCS adds more positive reactivity to the core at EOC than at BOC.

REFERENCE
CCNPP Lesson Plan RO-302-1-0 - Reactivity Factors

ANSWER 5.11 (2.00)

a) The longest lived delayed neutron precursors (DNP).

b) $Pf = Po(Beff - \rho o)/(Beff - \rho f)$

$= 100\%(0.007 - 0.0)/[0.007 - (-0.085)]$

$= 100\%(0.007)/0.092$

$Pf = 7.6\%$

REFERENCE
CCNPP Lesson Plan RO-302-2-0 - Reactor Kinetics

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 6.01 (3.50)

a) They modulate open .5

b) The go fully open .5

c) Engineered safety feature 125 VDC unit control panels .5

d) The quick opening feature and valve position indicator lamps are disabled. 1.0

e) The valves may still be automatically or operator controlled from the Control Room. 1.0

REFERENCE
CCNPP Main Steam System Description No. 19

ANSWER 6.02 (3.00)

a) The SDS automatically energizes selected essential equipment at 5-second intervals. .5

b) Safety injection actuation signal (SIAS), containment spray actuation signal (CSAS), and containment isolation signal (CIS). 1.5

c) Diesel generator 11 output breaker automatically shuts. 0.5

d) Undervoltage condition on bus 21 and SIAS for the unit. 0.5

REFERENCE
CCNPP Diesel Generator System Description No. 48

ANSWER 6.03 (2.50)

a) Containment Spray Actuation Signal and Safety Injection Actuation Signal

b) Injection mode - RWT
Recirculation mode - containment sump

c) Water on the containment floor dissolves it and carries it to the containment spray system.

REFERENCE
CCNPP Safety Injection and Containment Spray System Description No. 7

ANSWER 6.04 (3.00)

a) Actuate the power trip test interlock that results in tripping the high power, TM/LP, and APD trip units.

b) 1. Enable Loss of Load trip

2. Enable APD trip

3. Inhibit HI SUR trip

REFERENCE
CCNPP Nuclear Instrumentation System Description No. 57

ANSWER 6.05 (3.00)

a) Primary to secondary leak detection.

b) Surface and bottom blowdown isolation valves from the steam generator shut.

Blowdown tank discharge valves to the condensers and the Circulating Water System shut, blowdown tank discharge valve to the Miscellaneous Waste Processing System open.

REFERENCE
CCNPP Radiation Monitoring System Description No. 15

ANSWER 6.06 (2.50)

a) 1. (Hard Manual) - allows operation of the feedwater regulating valve by using the manual control knob.

2. (Soft Manual) - fine manual control of the feedwater regulating valve, or hold valve where it is to prevent transient.

3. (Automatic) - feedwater regulating valve automatically positioned to maintain steam generator setpoint level.

b) Main feedwater regulating valve shuts and the feedwater bypass valve opens to provide 5 percent of full power feedwater flow or allows the operator to regain control of the main feedwater bypass valve.

REFERENCE
CCNPP Main Feedwater System Description No. 32

ANSWER 6.07 (1.00)

a) One of the two 120 VAC backup buses

b) To prevent circuit overload

REFERENCE
CCNPP 120 VDC and 120 V Vital AC electrical power distribution System Description No. 54

ANSWER 6.08 (2.00)

a) One answer required

Individual trip unit is removed from the system, calibration or servicing.

b) Reverts to a two-out-of-three coincidence logic

c) Only one key for each type trip provided, key cannot be removed from the lock cylinder while the trip channel is in BYPASS.

REFERENCE
CCNPP Reactor Protective System Description No. 59

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 6.09 (2.50)

a) Pressurizer Pressure High/Low alarm

b) Heaters off, spray valve(s) open

c) Decreasing pressure until reactor trip and SIAS

REFERENCE
CCNPP Reactor Coolant System Description No. 62

ANSWER 6.10 (2.00)

a) Containment Isolation Signal

b) Fails as is

c) Fails shut

d) By using a key switch

REFERENCE
CCNPP Compressed Air System Description No. 41

ANSWER 7.01 (3.00)

a) Verify reactivity control

b) Verify RCS pressure and inventory control

c) Verify core and RCS heat removal

d) Verify 4 kv bus 11 or 14 energized

e) Verify normal containment environment

f) Verify normal radiation levels external to containment

REFERENCE
EOP-0

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 7.02 (3.00)

a) Satisfy the safety functions at risk.

Also acceptable: prevent core damage, stabilize the plant, provide additional safety function status information to determine the nature of the initiating event

b) 1. EOP-0 has been completed but an event diagnosis can not be made

2. An event diagnosis has been made and one of the EOP-1 through 7 procedures has been implemented but multiple safety functions are not meeting their acceptance criteria.

3. An event diagnosis has been made and one of the EOP-1 through 7 procedures has been implemented but all parameters for a single safety function are not meeting their acceptance criteria.

4. If a subset of parameters for a single safety function are violated, EOP-8 should be implemented unless all of the conditions below are met.

a. Reason for the violation has been established, and

b. Action has been identified that will return the parameters to within their acceptance criteria, and

c. The shift supervisor judges the recovery of the out of spec parameters to within acceptance criteria to be imminent.

REFERENCE
EOP-8

ANSWER 7.03 (2.50)

a) 1. Pressurizer spray

2. Auxiliary spray

3. Letdown depressurization

4. PORV depressurization

b) Voiding in the RCS (reactor vessel upper head region, SG tubes)

REFERENCE
EOP-8

ANSWER 7.04 (3.00)

a) Any four required

1. VCT level decreasing
2. Condenser OFF gas monitor alarm
3. SG blowdown high radiation monitor alarm
4. SG blowdown recovery radiation monitor alarm
5. Decreasing pessurizer pressure
6. Increased rate of level recovery in ruptured SG.
7. SG sampling results
8. Automatic isolation of SG blowdown
9. FW/Steam flow mismatch
10. RCS mass decrease with stable containment environment conditions.

b) Sample SG 11 and SG 12 for activity, identify SG with fastest increasing level, compare main steam line radiation monitor readings.

REFERENCE
EOP-6

ANSWER 7.05 (2.50)

a) Pressurizer level and pressure

b) Any three required

1. Thot - Tcold between 10 degrees and 50 degrees F

2. Tcold constant or decreasing

3. Thot constant or decreasing

4. CET temperature consistent with Thot

5. Steaming rate affects primary temperature

REFERENCE
EOP-2

ANSWER 7.06 (3.00)

a) The CEA's should be withdrawn individually to the upper electrical limit.

b) 1 DPM

c) 515 degrees F

d) Any two required

1. Moderator temperature coefficient is within its analyzed temperature range

2. Protective instrumentation is within its normal operating range

3. Pressurizer is capable of being in an operable status with a steam bubble

4. Reactor pressure vessel is above its minimum RT NDT temperature

REFERENCE
OP-2, Technical Specification Bases 3/4.1.1.5

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 7.07 (2.50)

a) One is aligned for automatic operation, and the other is in standby. 1.0

b) 1. Starts motor-driven AFW pump (Unit 1 only) .75

2. Opens 1(2)-MS-4071 .75

REFERENCE
CCNPP Auxiliary Feed Water System Description No. 34

ANSWER 7.08 (3.50)

a) Securing a heat load causes CCW temperature to decrease, causing letdown system temperature to decrease. Decreasing letdown system temperature increases the ability of the purification ion exchanger to absorb boron, causing boron concentration to decrease, adding positive reactivity to the reactor.

b) Shift the ion exchanger bypass valve to Bypass or control reactor power by using soluble boron or CEA's.

REFERENCE
CCNPP System Description No. 40j

ANSWER 7.09 (2.00)

Borate the RCS 100 ppm. [0.5]

by 1) opening charging pump suction direct feed valve (CVC-514-MOV) [0.5]

2) starting a boric acid pump [0.5]

3) starting all available charging pumps [0.5]

REFERENCE
EOP-0 pg 5

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 8.01 (2.50)

a) Two members of the Plant Management Staff, at least one of whom holds a SRO license on the Unit affected.

b) Two SRO's, one of whom must be the Shift Supervisor or GSO.

c) Concurrence of the SQCU must be obtained

REFERENCE
CCNPP CCI-101J

ANSWER 8.02 (2.00)

a) 2.00 rem

b) 300 mrem per week

c) 900 mrem

d) 150 mrem per week

REFERENCE
CCNPP CCI-800B

ANSWER 8.03 (1.50)

a) 25%

b) 3

c) 3.25

REFERENCE
CCNPP Technical Specifications

ANSWERS -- CALVERT CLIFFS -86/08/12-SHIRAKI

ANSWER 8.04 (3.00)

a) Thermal power, pressurizer pressure, highest operating loop cold leg coolant temperature

b) 2750 psia

c) Any two required

1. Facility placed in at least Hot Standby within one hour.

2. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. (The Vice President - Nuclear Energy and the OSSRC shall be notified within 24 hours.)

3. A Safety Limit Violation Report shall be prepared.

4. The Safety Limit Violation Report shall be submitted to the Commission, the OSSRC and the Vice President - Nuclear Energy within 14 days of the violation.

REFERENCE
CCNPP Technical Specifications

ANSWER 8.05 (3.50)

a) Attached

b) Any three required

Reactor Operator
Control Room Supervisor
Plant Watch Supervisor
Shift Supervisor's Assistant
Shift Supervisor

REFERENCE
CCNPP CCI 305B

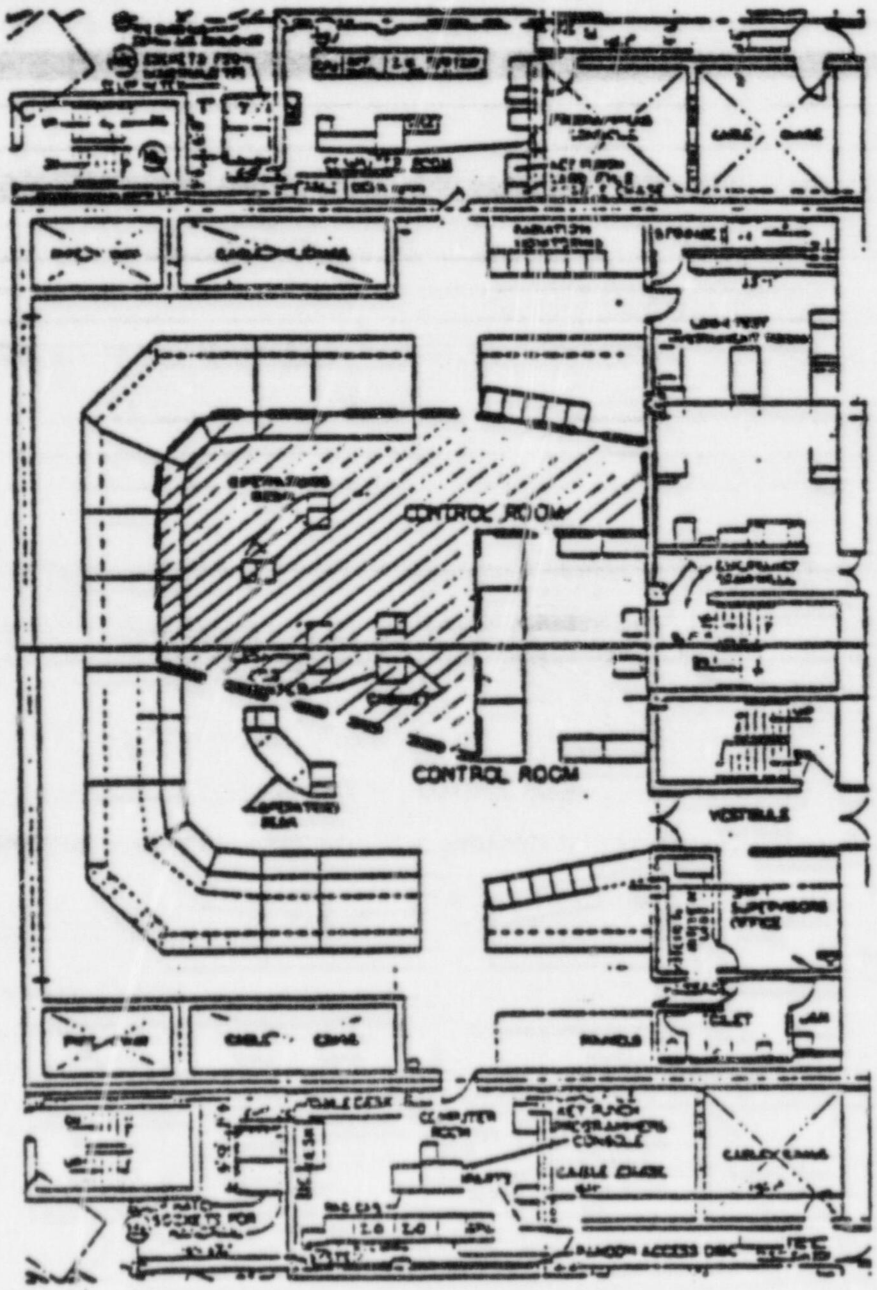
QUESTION 8.05 KEY

ATTACHMENT (1)

ROUTINE SURVEILLANCE AREA

FOR UNIT 1 CONTROL ROOM OPERATOR

(CROSS HATCHED AREA)

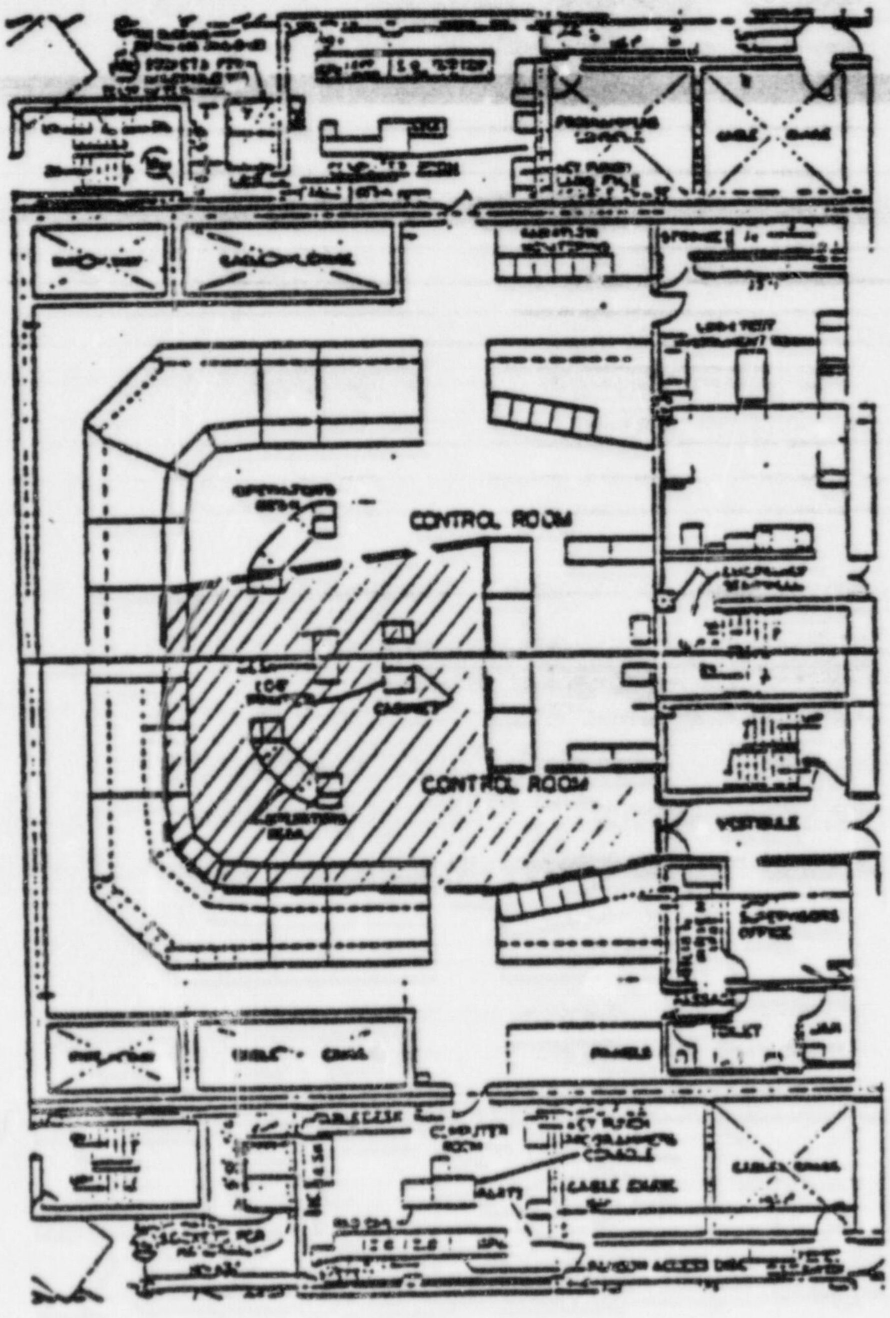


ATTACHMENT (2)

ROUTINE SURVEILLANCE AREA

FOR UNIT 2 CONTROL ROOM OPERATOR

(CROSS HATCHED AREA)



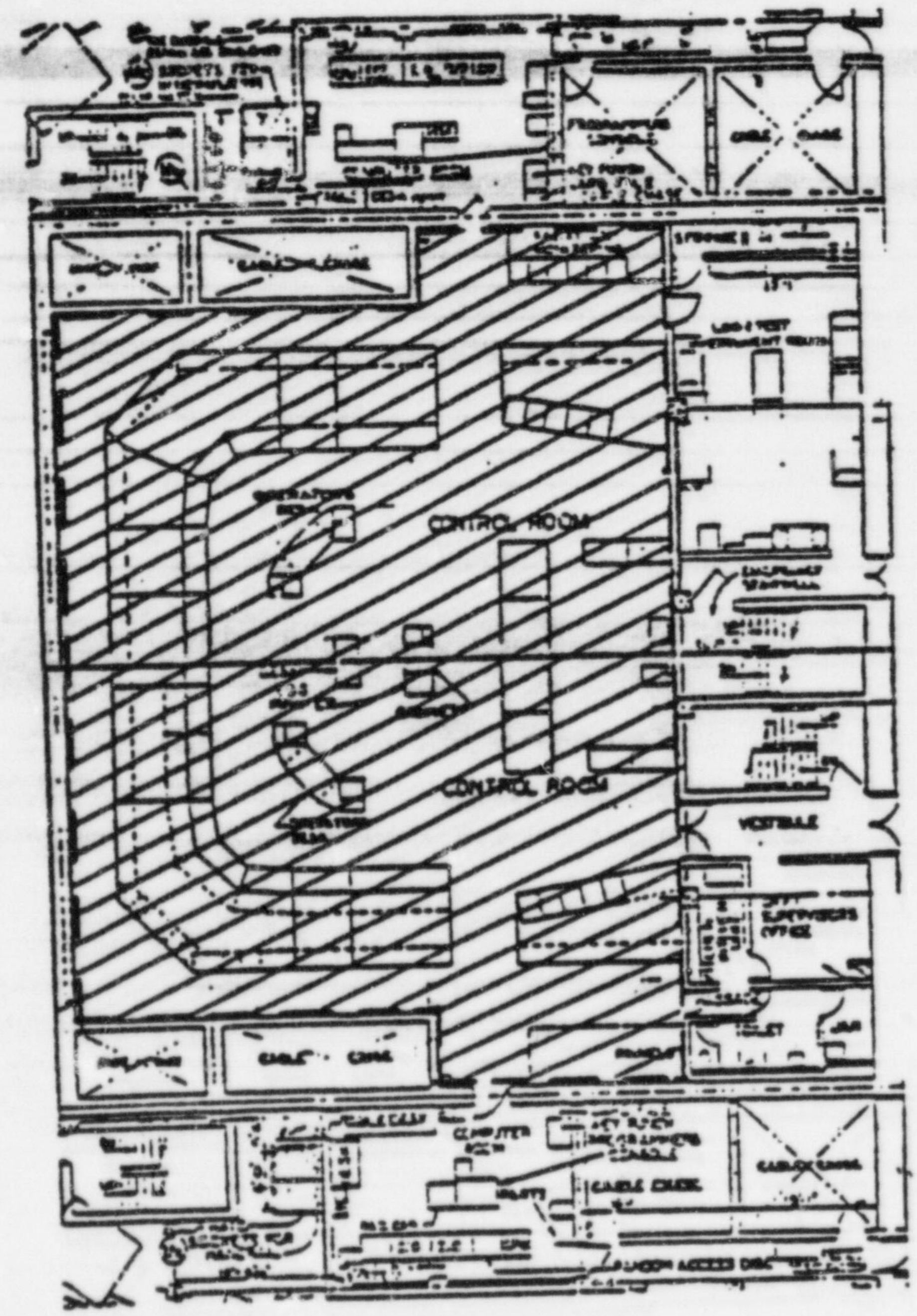
QUESTION 8.05 KEY CCI-305B

ATTACHMENT (3)

EMERGENCY SURVEILLANCE AREA

FOR UNIT 1 CONTROL ROOM OPERATOR

(CROSS HATCHED AREA)



QUESTION B.05 KEY

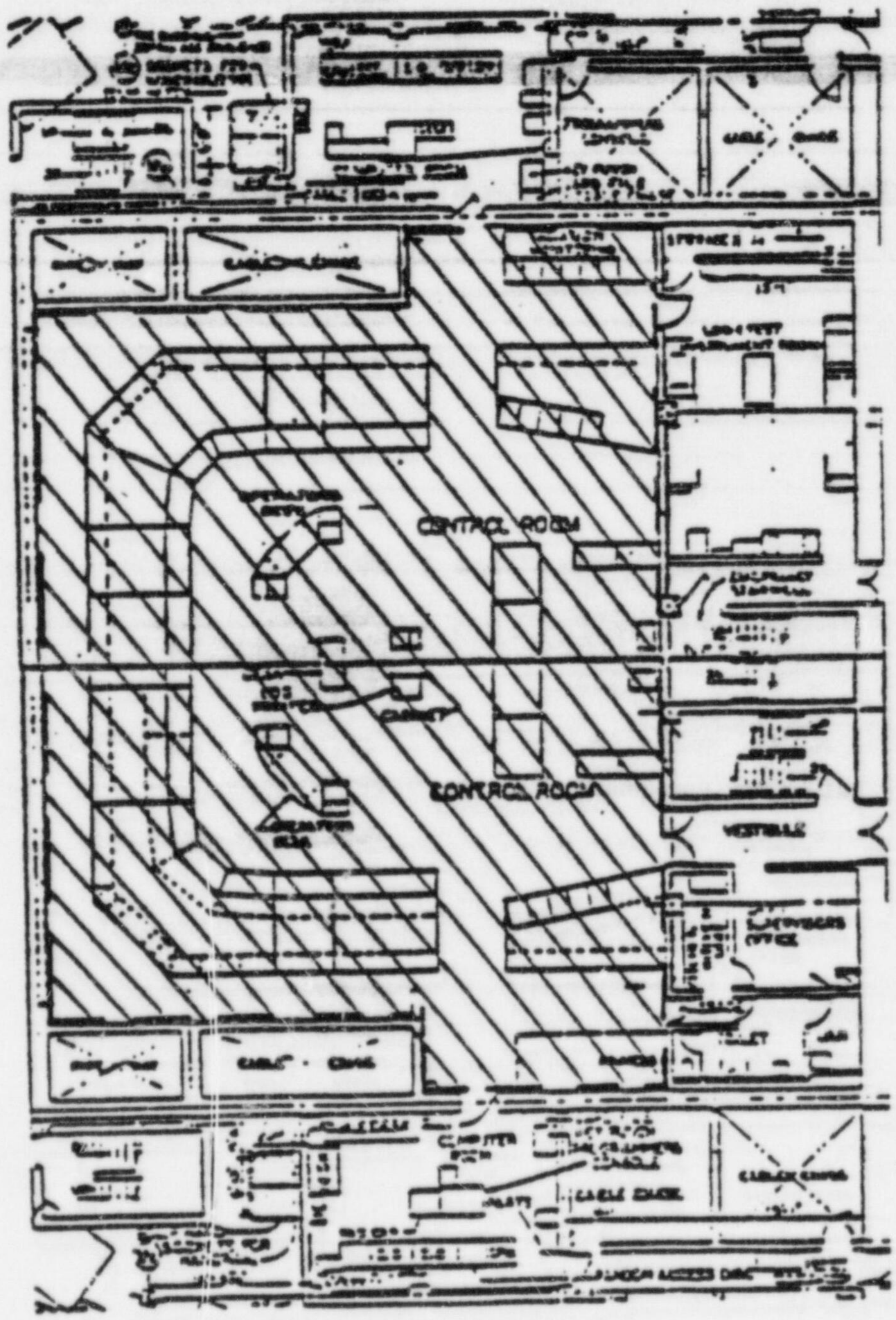
CCI-305B

ATTACHMENT (4)

EMERGENCY SURVEILLANCE AREA

FOR UNIT 2 CONTROL ROOM OPERATOR

(CROSS HATCHED AREA)



ANSWER 8.06 (2.00)

a) Three SRO's
Three RO's

b) 2 hours

c) Immediate action must be taken to restore the shift to the minimum number required.

REFERENCE
CCNPP CCI-140D

ANSWER 8.07 (1.50)

a) An area (accessible to personnel) where a major portion of the body could receive (greater than):

5 mrem in one hour
or
100 mrem in five (consecutive) days

b) An area (accessible to personnel) where a major portion of the body could receive (greater than):

100 mrem in one hour

REFERENCE
10 CFR 20.203(b) (2) and (b) (3)

ANSWER 8.08 (2.50)

a) An individual is deemed to operate the controls of a nuclear facility if he directly manipulates the controls or directs another to manipulate the controls.

b) "Controls" means apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor.

c) An individual may manipulate the controls as a part of his training to qualify for an operator license under the direction and in the presence of a licensed operator or senior operator.

REFERENCE
10 CFR 55.4(d) & (f) and 55.9(b)

ANSWER 8.09 (1.00)

a) A licensed senior reactor operator

REFERENCE
10 CFR 50.54(x) and (y)

ANSWER 8.10 (2.00)

B, C, D, and F require notification within one hour

REFERENCE
10 CFR 20.403(a) and 50.72(a)(1)i and 50.72(b)

ANSWER 8.11 (3.50)

a) Shutdown margin shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

b) Steam line rupture, no load conditions, (end of life)

REFERENCE
CCNPP Technical Specifications

QUESTION	VALUE	REFERENCE
05.01	3.00	CYS0000001
05.02	2.00	CYS0000002
05.03	2.00	CYS0000003
05.04	2.50	CYS0000004
05.05	2.50	CYS0000005
05.06	2.00	CYS0000006
05.07	3.00	CYS0000007
05.08	1.50	CYS0000009
05.09	3.00	CYS0000010
05.10	1.50	CYS0000011
05.11	2.00	CYS0000012
	25.00	
06.01	3.50	CYS0000022
06.02	3.00	CYS0000023
06.03	2.50	CYS0000024
06.04	3.00	CYS0000025
06.05	3.00	CYS0000026
06.06	2.50	CYS0000027
06.07	1.00	CYS0000028
06.08	2.00	CYS0000029
06.09	2.50	CYS0000030
06.10	2.00	CYS0000031
	25.00	
07.01	3.00	CYS0000032
07.02	3.00	CYS0000033
07.03	2.50	CYS0000036
07.04	3.00	CYS0000037
07.05	2.50	CYS0000038
07.06	3.00	CYS0000039
07.07	2.50	CYS0000040
07.08	3.50	CYS0000041
07.09	2.00	CYS0000055
	25.00	
08.01	2.50	CYS0000043
08.02	2.00	CYS0000044
08.03	1.50	CYS0000045
08.04	3.00	CYS0000046
08.05	3.50	CYS0000048
08.06	2.00	CYS0000049
08.07	1.50	CYS0000050
08.08	2.50	CYS0000051
08.09	1.00	CYS0000052
08.10	2.00	CYS0000053
08.11	3.50	CYS0000054
	25.00	
	100.00	

ATTACHMENT 3

BALTIMORE
GAS AND
ELECTRIC

CHARLES CENTER • P.O. BOX 1475 • BALTIMORE, MARYLAND 21203

QUALITY ASSURANCE & STAFF SERVICES DEPARTMENT
CALVERT CLIFFS NUCLEAR POWER PLANT
LUSBY, MARYLAND 20657

August 14, 1986

Mr. Douglas H. Coe
Reactor Engineer Examiner
U.S. NUCLEAR REGULATORY COMMISSION
Region 1
631 Park Avenue
King of Prussia, PA 19406

RE: NRC Exam Review

Dear Mr. Coe:

A detailed review of the Operator License examinations given at Calvert Cliffs on August 8, 1986 was conducted by the Operations Training Unit. Specific comments concerning individual examination questions are attached. Reference material has been provided where answers, provided by the staff, varied from those on the examination key. In addition to those responses, the following comments concern the examination in general.

I noted that the examination keys contained detailed references; specific lectures and objectives were used as source material for developing each question, system descriptions were identified by number and specific pages were referenced. This level of detail was extremely helpful in the review process.

I also welcomed your use of our learning objectives when formulating examination questions; the more extensive their use the closer the examination process will approach the actual requirements of the job. However, in three cases I feel that you misinterpreted the depth at which material was presented in support of an enabling objective. Specifically, the calculations related to reactivity, power turning, and power (Questions 1.05, 5.03 and 5.11 respectively) exceed our program's expectations. These questions are not performance based in that similar calculations will not be made by operators or senior operators in the control room. Although the training program included questions like these, they were included only to provide a foundation for the higher level terminal objectives. Furthermore, these tasks are not identified as RO or SRO knowledge requirements in NUREG-1122.

Question 5.03 was further complicated by the fact that the formula provided to solve the problem was incorrect.

When System Descriptions are used for examination questions, it must be noted that they contain information for everyone working at CCNPP. Specific details of circuit operation, for example, are intended for technicians who repair those systems. Cause and effect relationships of switch operation are operator oriented or performance based. This should be considered when extracting information from these references. Additionally, a disclaimer appears in each system description stating "This Document is for Instruction and Information Only. NOT TO BE USED FOR PLANT OPERATIONS." Despite this fact, operational questions were extracted from these references. Specifically, Questions 6.04 and 7.08 asked for operator actions which do not appear in approved plant procedures.

Questions involving Emergency Procedures were also more detailed than required by the Training Program. As requested, a policy statement was delivered at the April meeting in King of Prussia regarding proper use of these procedures. In spite of this, detailed sequence questions were asked which would have required candidates to have memorized portions of these lengthy procedures. Specifically, Questions 7.03a, 7.04a & 7.05a went beyond our policy statement. A better method of testing the understanding of these procedures is to provide a copy of the procedure to the candidate during the exam and ask questions concerning their use. This was done during a previous exam and proved very effective.

The final area of concern was the Operating Examination. This included the use of the plant simulator followed by a plant tour. Our first objection is the time involved in administering the exam. NUREG-1021 does not specify a minimum or maximum time to administer the exam but recommends a length of four to five hours. Instant SRO simulator exams lasted between five and six hours and were immediately followed by 2-1/2 and 3 hour plant tours. The average operating exam was therefore eight to nine hours in length. These lengthy sessions only tend to test the candidates' patience under stress. Knowledgeable candidates can be tested in a maximum of four hours and marginal candidates can be tested in five. Our second objection is the manner in which the simulator exams were administered. The simulator examinations combined both high and low probability scenarios with as many as five simultaneous casualties. This type of simulator session would challenge even the most seasoned operational crew, which is comprised of more than three individuals. Furthermore, this combination of events placed the plant in a condition which exceeds our safety analysis.

I request that a thorough evaluation be made of this examination practice. Performance-based training of licensed operators has been endorsed by both INPO and the NRC. This examination went beyond testing the performance ability of the candidates on both the written and operating examinations.

Sincerely,

S. E. Jones, Jr.
General Supervisor-
Nuclear Training

SEJ:JMY:rkb
Attachments

RO EXAM COMMENTS

1.05

An alternate method of calculating the reactivity inserted uses the formula:

$$SUR = \frac{26(\lambda)(p)}{B-p}$$

$$0.6 = \frac{26(0.08)p}{0.005-p}$$

$$p = .00112 \frac{\Delta k}{k}$$

The resultant answer agrees with the answer key.

1.09b.

The question asks for "THREE most probable heat transfer mechanisms". The attached pages of our approved reference states three mechanisms to transfer heat are:

conduction,
convection, and
radiation

The answer key lists its reference as CEN-152 REV. 3 and states the three to be:

boiling,
superheating, and
condensation

In reference to the three mechanisms, it is our belief that either answer should be accepted since both contain three mechanisms.

2.07b.

Another method used to interconnect the I.A. system to a backup source is by cross-connecting the Plant Air Systems between Units 1 and 2. This method is referenced in AOP-7D Page 1 discussion and should also be accepted.

3.08b.

The question asks for two systems controlled by signals provided by the loop temperatures asked for in part "a" of the question. The answer key identifies these systems as Presurizer Level and Steam Dump and Bypass. The system description referenced in the answer key lists these systems as Reactor Protection Systems and Reactor Regulating System.

It is our understanding that this was questioned by a few of the candidates during the written exam. They were told to be more specific as to where these temperatures are used. Since all candidates apparently were not given this information and because the answer key is limited to only the RRS, we feel that combinations of the following answers should be accepted.

1. Pressurizer level and Steam Dump and Bypass Systems
2. RRS & RPS
3. Pressurizer Level and RPS
4. Steam Dumps and RPS
5. TM/LP & RPS
6. ΔT Power and RRS

or any combination of the above

Reference System Description Nos. 59 and 62.

3.09

The answer key asks for the logic associated with the diesel trips. Although specified in the question as well, we do not feel the logic (2/3) is necessary operator knowledge. NUREG-1122, Page 3.7-11 k4.02 requires knowledge of the trips which would be demonstrated without requiring the logic of the trip.

3.11c.

Detection of pressurizer relief valve leakage could be identified by those items listed in the answer key but should also include quench tank temperature as indicated by AOP-2A II.F, Page 1.

3.12b.

The initial statement should be changed to say "...the level 2 bistable trips "<u>off</u>" instead of "on". Reference System Description 57, Page 25. By the same reference, the answer should include "enables the metrascope PDIL circuit".

4.01a.

The question asked for CCNPP administrative limits concerning weekly, quarterly and yearly whole body exposure. The answer key lists these limits as .3 REM, 2.0 REM and 4.0 REM respectively. These are the maximum amounts an individual could receive. Since the question did not state "maximum" limits, a candidate could answer .3 REM, .9 REM, and 4.0 REM, respectively and be correct. We, therefore, recommend either answer be accepted.

Reference CCI-800, Table-1, attached.

4.09

The question asks the basis for the precaution in OP-1, Page 1, Item F which states "no operations are permitted which cause dilution of the (RCS) boron concentration and core outlet temperature is maintained at least 10 °F below saturation temperature."

This condition, which is allowed by T.S. 3.4.1.3, is specifically identified by note ** at the bottom of page 3/4.4-2a of the Unit 1 Technical Specifications. In reviewing the T.S. basis, no mention is made of the conditions listed on the answer key. Therefore, in order for a candidate to receive credit for the question, the same conclusion as the answer key would have to be drawn.

Since no documentation as to the basis of this note can be found, we request any reasonable answer given by the candidates be accepted.

4.11b.

Students may refer to the Process Radiation Monitor as the Failed Fuel Monitor as a means of detecting high coolant activity. This is referenced in System Description 6, Page 3, Paragraph 3.

SRO EXAM COMMENTS

Formula Sheet

The equation for P_{pt}=(DRW) (Rod Speed)/time is incorrect. The formula should read P_{pt}=(DRW) (Rod Speed). This error on the formula sheet caused excessive delays by the candidates in solving the problem 5.03. The average delay noted by the individual candidates was approximately 20 minutes. Please correct this for future examinations.

6.04a.

The answer key outlines the effects of changing the A&B/2 switch position on the RPS. Since no procedure exists which allows the operator to change this switch position, for the initiating event as stated in the question, the proper action would be to bypass the power dependent trip units for that RPS channel. Namely to bypass the High Power, TM/LP, and APD trips. (Trip Units 1, 7, 10). If the switch were operated, it would result in actuation of the power trip test interlock (PTTI) and cause those trip units mentioned to fail to their trip conditions. Therefore, the expected response by the candidates would be actuation of the PTTI resulting in the trips indicated above. Candidates are not expected to know parts a.1 & 2 of your answer key and would answer as indicated.

6.06b.

The question asks the function of the feedwater override feature upon a turbine trip. This question is vague in that the candidate may answer the question two ways.

1. In accordance with the answer key naming those automatic actions which take place following a trip; or

2. By explaining the function of the "override" pushbutton located in 1/2C03 which will allow the operator to regain control of the main feedwater bypass valve.

Either answer is correct depending on how the candidate viewed the situation.
Reference System Description No. 32, Pages 32 & 33.

6.10d.

The question asks necessary actions to open 1(2)-CV-2085 after a loss of instrument air to the containment. This valve can be operated by a handswitch located in 27' switchgear room. This modification was made under Facility Change Request (FCR) 83-0060. Procedure AOP-7D, Page 7, Step 27 directs the operator to perform this action to restore instrument air to the containment.

7.04a.

The question asks for four indications of a SGTR and references those indications listed in EOP-6. There are more indications available to the operator than those listed. Examples:

1. Increased rate of level recovery in broken S/G.

2. S/G sampling results

3. Automatic isolation of S/G blowdown

4. Feed Flow/Steam Flow mismatch

5. RCS mass decrease with stable containment environment conditions.

The indications in the EOP are just a small sample of indications which might be used.

7.07b.

The reference material, System Description No. 34, incorrectly stated three actions accomplished by AFAS START 'A'. These actions were modified under FCR 84-1031 and eliminated the MFP turbine runback. In addition, the signal which shut both main feedwater isolations was never installed under the original FCR. A correct response would include only Items 1 and 2 of your answer key. (See attached FCR enclosure). The System Description will be udpated to include this information.

7.08b.

The answer key uses as a reference, a procedure or action discussed in the system description. This action, although appropriate, may or may not be taken by the operator because it is not in our approved procedure. In addition, the question asks what must be done to preclude the power increase not what can be done after the transient is over. We feel the appropriate response would be to bypass the ion exchanger or control reactor power by using soluble boron or CEAs. It would not be expected that the candidate would indicate returning the system(s) to normal since it wasn't asked in the question.

8.11b.

The question asks what accident provides the basis for the shutdown margin (SDM) requirements in modes 1-4. The technical specifications B3/4.1.1.1 and B3/4.1.1.2 page B3/4 1-1 state that the SDM requirements are based on the steam line rupture event initiated at no load conditions (hot zero power). The most restrictive case for this event occurs at EOC. The actual SDM requirement is based on any time in core life not just EOC. Therefore, we request that EOC be deleted from or made an optional part of the answer key.

8.02

The question asks the maximum permissible whole body exposure limit at CCNPP. The answer key states 1.25 REM/QTR. CCI-800 Table 1 (attached) allows a maximum whole body exposure of 2.0 REM/QTR. after an administrative review at .9 REM/QTR.
After exceeding .9 REM/QTR. the individual is limited by his/her supervisor depending on work needs as stated in Table 1.

ATTACHMENT 4

NRC Resolution of Facility Comments
on Written Examinations Administered
August 11, 1986

Question	Resolution
1.05	Alternate calculational methods, correctly performed, are acceptable.
1.09b	The responses "conduction", "convection", and "radiation" may be correct insofar as they correctly describe the processes taking place in the three regions of the RCS.
2.07b	Accepted per AOP-7D.
3.08b	Accepted per System Descriptions 59, 62.
3.09	Accepted based on NUREG-1122, K4.02, page 3.7-11.
3.11c	Accepted per AOP-2A
3.12b	Accepted per System Description 57.
4.01a	Accepted per CCI-800, Table 1
4.09	Will be considered during grading.
4.11b	Will be considered during grading.
Formula Sheet	The time variable is included to remind candidates that they need to convert from rod speed units of inches per minute to inches per second by dividing by 60 seconds per minute. Since this appears to have caused confusion and did not aid the candidates, the time variable will be deleted in future uses of the equation.
6.04.a	Answer key changed to: Actuate the power trip test interlock that results in tripping the high power, TM/LP, and APD trip units.
6.06.b	Either answer will be accepted.

Question	Resolution
6.10.d	Answer key will be changed to reflect new answer based on this reference. System Description No. 41, Compressed Air System, page 36 should be changed to show that a containment entry is no longer required to open the valve.
7.04.a	These indications will be added to the answer key. Any other reasonable indications given by the candidates will be considered.
7.07.b	The answer key will be changed to reflect this new information.
7.08.b	The answer key will be changed to reflect the use of soluble boron or CEA's and will not require a statement regarding returning the system(s) to normal.
8.11.b	EOC will not be required as part of the correct answer.
8.02	The question asks for the "maximum permissible occupational radiation dose," which, from paragraph A.1.a of Attachment (1) to CCI-800B, is 1.25 REM per calendar quarter. However, Paragraph C.1 and Table 1 of CCI-800B both give 2.0 REM/QTR as the administrative dose limit. Since the candidates are taught the administrative dose limits and since the reason for asking the question is to ensure that the candidates are aware of the limits imposed by the facility, 2.0 REM/QTR will be accepted as the correct answer.

NRC Response to General Facility Comments

Facility comment

Calculational questions (theory-type) related to reactivity, power turning, and power exceed our program's expectations and are not performance based.

NRC response

Calculational questions in which the formulae are provided to the candidate have always been considered an acceptable means of determining a candidates understanding of theoretical principals. It was never intended that questions of this nature be limited to calculation that operators would perform in the control room. These questions are often the best way of determining if higher level terminal learning objectives have been met. The knowledge requirements of NUREG-1122 require RO and SRO knowledge of "theoretical concepts" as they apply to various systems and components. It does not specify however, how this knowledge should be examined.

Facility comment

Questions involving Emergency Procedures were more detailed than required by our Training Program and required candidates to have memorized portions of these lengthy procedures.

NRC response

Questions of this nature are graded (for SRO candidates) on the basis of NUREG-1021 paragraph ES-402 A.3. This states "The candidate should be able to describe generally the objectives and methods used in the normal, offnormal, and emergency operating procedures and the methods used to perform the verifications." Although the examination answer key might be specific, a fully correct answer need not always be identical to the answer key.

Facility comment

The operating examination is too lengthy and only tends to test the candidates patience under stress. The simulator scenarios combined both high and low probability scenarios with as many as five simultaneous casualties which placed the plant in a condition which exceeds our safety analysis.

NRC response

The length of the operating examination is dependent upon several factors. First the minimum requirements of NUREG-1021 for the number of malfunctions to be given each type of candidate must be met. Second, the mix of candidates will affect the length of time required to meet the above requirements, i.e. three Instant SRO's will take longer than two RO's and one Upgrade SRO. Third, individual examiner judgement and candidate performance will affect the length of the plant walk-through portion of the examination.

Simulator scenarios are often written with industry experience in mind and the inclusion of five simultaneous casualties/malfunctions is not without basis in fact. With regard to exceeding the plant safety analysis, the NRC examines a candidate's ability to effectively use the plant procedures to adequately assure public health and safety. Emergency procedures are often written to account for casualties that exceed the safety analysis. Therefore, in order to determine a candidate's ability to use these procedures to their fullest extent, scenarios are written to include catastrophic events which consequently exceed the safety analysis.

Simulator examinations have been found to be excellent tools which the NRC can use to ensure newly licensed operators are able to handle themselves and their plant under challenging and adverse conditions. As efforts are made to improve these examinations, continuing feedback and dialogue with industry is sought and appreciated.