DESIGNATED ORIGINAL

Certified By J. Gonik

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

EXAMINATION REPORT NOS. 50-317/84-10
50-318/84-10

FACILITY DOCKET NOS. 50-317 and 50-318

FACILITY LICENSE NOS. DPR-53 and DPR-69

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

FACILITIES: Calvert Cliffs Units 1 and 2

DATES: April 30, 1984 - May 4, 1984

CHIEF EXAMINER: DF Johnson
D. F. Johnson — May 24, 1984 Date

APPROVED BY:
Chief, Project Section 1D — June 5, 1984 Date

SUMMARY: Written examinations were administered to five SRO's and one RO candidate. Oral examinations were administered to the five SRO's and waivered for the RO candidate due to his successful passing of a previously administered oral examination on October 10, 1983. All candidates passed both the oral and written examinations.

REPORT DETAILS

TYPE OF EXAMS: Initial ___ Replacement _X_ Requalification ___

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail	Inst. Cert Pass/Fail	Fuel Handler Pass/Fail
Written Exam	1/0	5/0	/	/
Oral Exam	/	5/0	/	/
Simulator Exam	/	/	/	/
Overall	1/0	5/0	/	/

1. CHIEF EXAMINER AT SITE: D. F. Johnson
2. OTHER EXAMINERS: N. Dudley
3. PERSONS EXAMINED

RO

Wilson, John N.

SRO

Chandlee, Richard S.
Grooms, James V.
King, Dennis B.
Naley, Paul D.
Shick, Bruce B.

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

   Examiners noted that candidates were well prepared for examinations. There appears to be a general weakness in understanding the degree or magnitude of changes in plant parameters during severe transient conditions.

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

   No generic weaknesses were noted on the written exams. Overall grades were good.

3. Comments on availability and candidate familiarization with plant reference material:

   Several candidates were unable to properly classify events in accordance with the emergency plan procedures.

4. Comments on availability and candidate familiarization with plant design, procedure, T. S. changes and LERs:

   All candidates exhibited a thorough knowledge of Technical Specifications, recent LER's and procedures with exception of comments noted in item 1 and 3 above.

5. Comments on interface effectiveness with plant training staff and plant operations staff during exam period.

   The plant staff was extremely cooperat ve during all phases of the examination process.

6. Improvements noted in training programs as a result of prior operator licensing examinations/suggestions, etc:

   None noted.

7. Personnel Present at Exit Meeting:

NRC Personnel

D. F. Johnson, Chief Examiner

NRC Contractor Personnel

None

Facility Personnel

J. Hill, Supervisor, Operations Instructor
J. Yoe, Senior Operations Instructor (Nuclear)

8. Summary of NRC Comments made at exit interview:

At the conclusion of the administration of examinations, the Chief Examiner met with representatives of the plant staff to discuss the results of the oral examinations. They were informed that all candidates passed the operating portion of the examination.

Generic weaknesses noted in items 1 and 3 above were discussed in detail.

The Chief Examiner stated that the candidates were well prepared for the examinations.

9. Summary of facility comments and commitments made at exit interview:

None.

10. CHANGES MADE TO WRITTEN EXAM

At the conclusion of the written examination, the examiners met with J. Hill, J. Yoe, E. Chrzanowski, B. Hiestand and C. Andrews of the Operations Training Department to review the written examinations and answer keys. The license's comments and our resolution of these comments are enclosed.

Attachment:

Written Examination(s) and Answer Key(s) (SRO/RO)

The following is a list of the comments noted during the review of the SRO and RO examinations and the resolution of these comments.

RO EXAM

3.01 Answer key item a. added the words "process radiation monitor" to provide clarification and acceptance of this wording in addition to letdown line monitor, a plant specific terminology.

SRO EXAM

5.4 part b.2 deleted this question due to the terminology "saturated nucleate boiling" which was an unfamiliar term to the candidates and conflicted with question 5.4 part b.1., and is not a universally accepted term.

7.2 part b. deleted, the question as worded would not elicit the correct answer due to the fact that the LPSI pumps would be tripped on a RAS.

The following changes were made after discussions with the facility staff and verification of their comments.

5.7 the term 10^{-8} amps was changed to 10^{-4}% to reflect plant specific instrumentation.

5.9.b answer key was changed to accept an alternative plausible response that "fast fission is more effective than fast leakage" in answer to why beta-eff is less than beta.

6.2.k changed question from "air ejector S/G blowdown activity" to "condenser off-gas S/G blowdown activity" to reflect plant specific nomenclature.

6.3.b.4 added additional plausible response of "intermediate stop valves open" with respect to specific plant conditions.

6.5 changed answer A.3 to accept either "CEA function" or "planar radial peaking factor" plant specific terminology.

7.2 added additional plausible response A.6 "service water heat exchanger outlet returns to its pre-incident position" on a RAS.

7.4 placed brackets around 1578 psia and below 537°F to indicate setpoints not required for full credit.

7.10 added to part C answer an additional plausible response "isolate RCP bleedoff".

8.11 added to answer on part a. the shutdown margin for Unit 1 of 4.3 ΔK/K in addition to the 5.2 ΔK/K for Unit 2 since the question did not state which unit but asked for the minimum allowable shutdown margin for mode 1 operation.

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

FACILITY: _CALVERT_CLIFFS_________

REACTOR TYPE: _CE-PWR_______________

DATE ADMINISTERED: _84/05/01______________

EXAMINER: _SIBEIEB._G.____________

APPLICANT: ________________________

INSTRUCTIONS_TO_APPLICANT:

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	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
24.50	24.87	___________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.38	___________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
24.00	24.37	___________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.38	___________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
98.50	100.00	___________	________	TOTALS

FINAL GRADE _________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (3.00)

Compare the CALCULATED Estimated Critical Position (ECP) for a startup to be performed 4 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 HIGHER than, LOWER than, or SAME as the ECP.

a. One reactor coolant pump is stopped two minutes prior to criticality. (0.6)

b. The startup is delayed until 8 hours after the trip. (0.6)

c. The steam dump pressure setpoint is increased to a value just below the Steam Generator saftey valve setpoint. (0.6)

d. Condenser vacuum is reduced by 4 inches of Mercury. (0.6)

e. All Steam Generator levels are being raised by 5% as the ECP is reached. (0.6)

QUESTION 5.02 (2.00)

a. Explain the difference between available NPSH and minimum required NPSH. (1.0)

b. As the speed of a centrifugal pump increases, do the following increase, decrease, or remain the same?

1. Available NPSH

2. Minimum required NPSH (1.0)

QUESTION 5.03 (3.00)

Briefly EXPLAIN how the addition of 0.5% positive reactivity to a subcritical reactor would affect the following: (No calculations are required.)

a. THE CHANGE IN THE COUNT RATE: (if the reactor was slightly subcritical [shutdown margin = 1%] as compared to greatly subcritical [shutdown margin = 5%]). (1.5)

b. The TIME TO REACH A STABLE COUNT RATE: [for the different shutdown margin conditions in (a.) above.] (1.5)

QUESTION 5.04 (3.50)

a. What is the most significant type of heat transfer (conduction, convection, or radiation) taking place under each of the following conditions? Consider each condition separately.

1. Nucleate boiling.
2. Accident condition in which coolant is boiled and converted to steam in the reactor vessel.
3. Heat from fission thru the fuel rod.
4. Decay heat removal by natural circulation. (2.0)

b. Indicate on your answer sheet whether the following statements are TRUE or FALSE. No explanation is required.

1. For normal Pressurized Water Reactor (PWR) operation, NO bulk boiling (saturated nucleate boiling) occurs in the reactor vessel.

2. --- QUESTION DELETED---

3. The point at which the heat transfer COEFFICIENT is at its MAXIMUM value is called "departure from nucleate boiling".

4. As RCS pressure increases, a smaller heat flux (BTU/hr ft) occurs with a constant temperature difference (Twall - Tsat). (1.5)

QUESTION 5.05 (3.50)

a. Although the U238 resonance capture peaks broaden and flatten with increased fuel temperature, the area under the peak remains the same. Why then is there an increase in neutron capture as the fuel temperature is increased? (1.0)

b. Does the fuel temperature coefficient (PCM/ F) INCREASE or DECREASE as fuel temperature is increased? (0.5)

c. HOW AND WHY does the moderator temperature coefficient (MTC) change (more or less negative) as temperature is increased at a constant boron concentration, in an undermoderated core? (1.0)

d. HOW AND WHY does the MTC change as boron concentration is increased at a constant temperature, in an undermoderated core? (1.0)

QUESTION 5.06 (1.50)

Explain why reactor coolant system delta T can be used as a measure of reactor power, but secondary coolant delta T cannot. (1.5)

QUESTION 5.07 (3.50)

a. After criticality is achieved, the startup procedure requires the operator to level power at 10 -4 % to record critical data. The operator establishes a ZERO DPM SUR and verifies that the I/R indication is steady with no rod motion. After taking data, the operator notices that power is increasing. Explain why? (1.4)

b. Why was 10 -4 % selected as the point for taking critical data? (2.1)

QUESTION 5.08 (1.50)

HOW and WHY is fuel centerline temperature affected by the following: (consider each independently)

a. Fuel densification?

b. Fission product gas buildup IN the fuel pellet?

c. Clad creep? (1.5)

QUESTION 5.09 (2.00)

Beta is the fraction of all neutrons released by fission which are delayed:

a. When comparing the individual Beta's from thermal fission of U235, Pu239 and fast fission of U238, which Beta is largest? (0.5)

b. In a C-E PWR, why is Beta-eff less than Beta? (0.5)

c. For equivalent positive reactivity additions to a critical reactor, will the SUR be larger or smaller at EOL compared to BOL? Explain your answer. (1.0)

QUESTION 5.10 (1.00)

TRUE or FALSE?

a. Comparing a parallel-flow heat exchanger to a counter-flow heat exchanger, the temperature difference between the two fluids along the length of the heat exchanger tubes is more uniform for the parallel-flow heat exchanger. (0.5)

b. The heat transfer rate of a cross-flow heat exchanger is constant along the length of the heat exchanger tubes. (0.5)

QUESTION 6.01 (3.00)

The following concerns the control rod drive system:

a. The following concern failure of a CEA lift coil with the rod fully withdrawn;

1. Explain why the rod WILL or WILL NOT drop.

2. Explain why the rod WILL or WILL NOT move on a demand signal. (1.5)

b. What means exist to determine whether a control rod is withdrawing properly? (Five SEPERATE sytems indications required) (1.5)

QUESTION 6.02 (3.00)

Match the following indications/parameters you would use to differentiate between the two accidents within each group. (Assume steady state 70% reactor power).

1. Small LOCA outside containment AND S/G tube rupture (3 required)

2. Small LOCA inside containment AND feedwater break inside containment upstream check valve (7 required)

3. Steam break outside containment AND small LOCA outside containment (5 required)

a. Tave
b. RCS pressure
c. Przr. level
d. VCT level
e. Containment: pressure, temperature, humidity
f. Containment airborne
g. steam flow
h. steam pressure
i. feedwater flow
j. S/G level
k. Cond. off-gas/S/G blowdown activity (3.0)

QUESTION 6.03 (3.50)

The following concern the unit 1 main turbine:

a. When the master trip relay is energized due to a trip signal, what actuations are initiated? (Four required) (2.0)

b. What THREE events occur when the master reset pushbutton is actuated? (1.5)

QUESTION 6.04 (1.50)

What TWO conditions are required to generate a pipe rupture signal in the Aux. Feedwater system? (1.5)

QUESTION 6.05 (2.50)

Concerning the axial power distribution reactor trip;

a. What inputs are used to generate the APD signal and where is each input received from? (1.5)

b. What initiates an APD channel trip signal? (1.0)

QUESTION 6.06 (3.50)

Describe how the Steam Dump and Bypass system will function as a result of the following transients: (Be specific)

a. 10% load rejection from 35% power (1.5)

b. Turbine trip from 75% power (2.0)

QUESTION 6.07 (.25)

Which of the following IS NOT a direct input to the TM/LP trip circuit?

a. Reactor power

b. Primary flow

c. Primary temperature

d. Axial flux index (0.25)

QUESTION 6.08 (1.25)

a. What is the purpose of the TWO high range gamma monitors located in the containment? (0.25)

b. Where are these high range gamma monitors located in the containment? (1.0)

QUESTION 6.09 (2.25)

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 (0.75)

QUESTION 6.10 (1.25)

a. List the setpoint and range over which the high SUR trip is enabled. (0.60)

b. Why is the high SUR trip not required above and below this range? (0.65)

QUESTION 6.11 (3.00)

Describe the primary protection function or basis for the following reactor trips.

a. thermal margin/low pressure
b. axial power distribution
c. high reactor power
d. high rate of change of reactor power
e. high pressurizer pressure
f. containment high pressure
g. low steam generator water level
h. low steam generator pressure
i. low reator coolant flow
j. manual (3.0)

QUESTION 7.01 (2.50)

a. According to EOP-7 (Fuel Handling Incident) what are the FIVE required immediate action steps for a fuel handling incident on the CONTAINMENT side? (2.0)

b. What different action step is required if it happened in the auxiliary building? (0.5)

NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

QUESTION 7.02 (1.50)

An event that has happened at several PWR's, including Calvert Cliffs, was an inadvertently generated RAS while on shutdown cooling in mode 6.

a. List the 5 automatic actions that an RAS will cause. (1.5)

b. ---- DELETED-----

QUESTION 7.03 (1.50)

a. What are the Calvert Cliffs administrative limits concerning weekly, quarterly, and yearly whole body radiation dose? (0.9)

b. Whose approval is necessary prior to exceeding EACH of the above dose limits? (0.6)

QUESTION 7.04 (3.50)

The Plant is operating at full power when you receive the following radiation alarms: CONDENSER VACUUM PUMP DISCHARGE, STEAM GENERATOR BLOWDOWN, and STEAM GENERATOR BLOWDOWN RECOVERY.

a. What are your immediate actions? (0.5)

b. For the worst case accident in this situation, why would you expect a reactor trip and safety injection? Explain your answer assuming no operator action. (1.0)

c. If you determine that a reactor trip is imminent, what are your immediate actions? (4 required). (2.0)

QUESTION 7.05 (2.50)

The plant shutdown procedure, OP-4, contains the following NOTE (Reactor between 5% to 10% power): "With the Turbine Bypass Valve in automatic operation, the CEA's do not control primary system average temperature."

a. What is controlling Tave and reactor power at this power level? Explain how this control is performed. (1.5)

b. What controls Tave and reactor power at 50% power level? Explain. (1.0)

QUESTION 7.06 (3.00)

The following concern a loss of AC power (EOP 13):

a. The first automatic action states "all full length CEA's should be fully inserted", what are you required to do if this condition is not met? (1.0)

b. In addition to the automatic action in Part (a.) above, what automatic actions should occur? (1.5)

c. If the generator output breakers have not tripped automatically and if it cannot be confirmed that the turbine stop valves are shut, what are you required to do? (0.5)

QUESTION 7.07 (2.25)

According to AOP-5 (ECCS long term cooling core flush):

a. Core circulation MUST be in effect within 11 hours after a cold leg break. Why is a cold leg break more of problem than a hot leg break? (1.25)

b. If hot leg injection is used vice pressurize, injection how do you ensure that the minimum 40 gpm flow rate is achieved? (1.0)

QUESTION 7.08 (1.50)

In accordance with EOP-3 (Loss of main feedwater) each of the following situations requires reactor power to be reduced. What is the maximum allowed power level? (Assume 100% initial power)

a. One condensate pump available (0.5)

b. One condensate booster pump available (0.5)

c. One heater drain pump available (0.5)

QUESTION 7.09 (1.50)

The following questions concern the Emergency Diesel operations (OI-21):

a. If a diesel has been pre-lubed it should be run shortly thereafter. What is the reason for this precaution? (0.5)

b. If a SIAS signal starts the diesel automatically, the operator is cautioned to pay particularly close attention to two parameters, whose trip function have been bypassed. What are these two parameters? (0.5)

c. What special condition is required when running the diesel at speeds lower than normal? (0.5)

QUESTION 7.10 (2.75)

The following concern notes in the CVCS operating instruction;

a. Why are both letdown control valves NOT used simultaneously above 1500 psia? (1.0)

b. Prior to starting a charging pump, after maintenance has been performed on the fluid end, what is required? (0.5)

c. Explain the most probable cause AND the corrective action for an increasing VCT level following isolation of the charging and letdown system? (1.25)

QUESTION 7.11 (1.50)

A Reactor Coolant pump shall not be started when one or more loop Tc's are less than 275 F unless one of two possible conditons exist. What are these two conditions? (1.5)

QUESTION 8.01 (3.00)

Technical Specification 3.4.3 states 'two power operated relief valves (PORV's) and their associated block valves shall be operable'. For the following situations state what action is required by this tech. spec. and within what time limit. (Answer only for condtions that allow continued plant operation).

a. One or more PORV's inoperable

b. One or more block valves inoperable (3.0)

QUESTION 8.02 (1.50)

The Calvert Cliffs Technical Specifications indicate that the maximum linear heat rate shall not exceed 15.5 KW/ft. What TWO indications/conditions do the Tech. Specs. use to determine when this limit is exceeded? (1.5)

QUESTION 8.03 (2.00)

According to the GSO standing instructions there are special operating instructions when a loss of power to the #11 D.C. bus has occured. What are these operating instructions AND why are they necessary? (2.0)

QUESTION 8.04 (2.50)

a. What is a special work permit and how does it differ from a radiation work permit? (1.0)

b. Who must approve a special work permit? (0.5)

c. Who approves radiation work permits and how long are they valid? (1.0)

QUESTION 8.05 (1.50)

Fill in the blanks in the following statements concerning use of procedures. (Place your answer on your answer sheet)

a. A change to a fuel handling procedure that was initially reviewed by POSRC shall be approved by ____________ members of the plant management staff, at least _________ who holds an SRO license at that unit. (1.0)

b. All temporary procedure changes shall be reviewed by POSRC and the Plant Superintendent within ________ days of implementation. (0.5)

QUESTION 8.06 (2.50)

If while in mode 6 with the reactor vessel head removed and core alterations in progress, the reactor operator discovers that the boron concentration of the primary system is 2100 ppm.

a. As the shift supervisor what are your actions as outlined in the Tech. Specs.? (2.0)

b. How soon do you have to perform these actions? (0.5)

QUESTION 8.07 (1.50)

The minimum water level over the top of the reactor pressure vessel flange is 23 feet during movement of irradiated fuel assemblies and while irradiated CEA's are seated in the pressure vessel during refueling operations. What is the basis for this level? (1.5)

QUESTION 8.08 (2.00)

For the leakage conditions shown below, indicate whether you would CONTINUE TO OPERATE indefinitely or SHUTDOWN under specific time requirements. Assume no other leakage than that listed. Consider each item separately.

a. 0.5 gpm each, from five different valve packing glands

b. 0.2 gpm from a Th loop RTD weld

c. 1.3 gpm unknown leakage

d. 3 gpm seat leakage on a pressurizer safety valve (2.0)

QUESTION 8.09 (3.00)

For each of the following events explain briefly why the NRC SHOULD or SHOULD NOT be notified within 1 hr.

a. During instrument testing while at power, three pressurizer pressure safety channels are momentarily place in bypass.

/b. While at power, Tave momentarily dips to 510 F and then returns to normal.

c. Refueling water tank level falls below 400,000 gallons and cannot be restored.

d. During surveillance testing an expected actuation of LPIS train A occurs. (3.0)

QUESTION 8.10 (2.50)

a. If a Safety Limit has been violated, what THREE actions must be performed? (1.5)

b. What are the TWO Safety Limits for Calvert Cliffs unit 2? (1.0)

QUESTION 8.11 (3.00)

a. What is the minimum allowable shutdown margin for mode 1 operation? (0.5)

b. What are the required actions if the shutdown margin is determined to be less than the minimum allowable, while operating in mode 1? (1.0)

c. Give TWO reasons why adequate shutdown margin must be maintained during ALL modes of operation. (1.5)

ANSWER 5.01 (3.00)

a. SAME

b. HIGHER

c. HIGHER

d. SAME

e. LOWER [0.6 each] (3.0)

REFERENCE
C-E Reactor Theory Pgs. 166,204-206

ANSWER 5.02 (2.00)

a. Available NPSH is actual pump suction head minus Psat [0.5]
Minimum required NPSH is the NPSH required to prevent cavitation. [0.5] (1.0)

b. 1. Decreases
2. Increases (1.0)

REFERENCE
General Physics pgs. 319-320

ANSWER 5.03 (3.00)

a. The slightly (greatly) subcritical reactor will have a larger (smaller) increase in count rate. (1.5)

b. The slightly (greatly) subcritical reactor will take a longer (shorter) time to reach a stable count rate. (1.5)

REFERENCE
C-E Reactor Theory Pgs. 147-148

ANSWER 5.04 (3.50)

a. 1. Convection
2. Radiation/convection (large Delta T)
3. Conduction
4. Convection (natural) (2.0)

b. 1. True
2. --- DELETED---
3. False
4. True (1.5)

REFERENCE
General Physics pgs. 99-115

ANSWER 5.05 (3.50)

a. The neutron sees a significant absorption cross section over a wider range of energies (decrease in the fuel self shielding). (1.0)

b. Decrease. [less negative] (0.5)

c. MTC becomes more negative [0.25] because the density change per degrees-F is greater at higher temperatures [0.75]. (1.0)

d. MTC becomes less nega ve (decreases) [0.25] because the number of boron atoms (poison) in the core decreases more per F change at higher boron concentrations. [0.75] (1.0)

REFERENCE
C-E Reactor Theory Pgs. 159-168

ANSWER 5.06 (1.50)

Temperature rise in coolant is directly proportional to heat input as long as no phase change takes place. As vaporization takes place in the secondary, heat is added with no change in temperature. (1.5)

REFERENCE
General Physics

ANSWER 5.07 (3.50)

a. When rods were inserted only prompt neutrons were affected and reactor power was stablized.[0.7] During the time that the operator was taking critical data, the delayed neutrons contributed to the overall neutron population [0.7] thus power increases. (1.4)

b. 10 -4 % is above the point where source neutrons will have a significant effect [0.7] and below the POAH [0.7] so that power defect does not have any effect.[0.7] (2.1)

REFERENCE
C-E Reactor Theory pgs. 74-76

ANSWER 5.08 (1.50)

a. Fuel centerline temperature (FCT) INCREASES [0.2]. Densification results in fuel shrinkage and an increase in the gap between the fuel pellet and the gap [0.3].

b. FCT DECREASES [0.2] fission product gases cause a gradual swelling of the fuel pellets reducing the gap between the fuel and the clad [0.3].

c. FCT DECREASES [0.2] opposite reason of (a) above [0.3] (1.5)

REFERENCE
General Physics pgs. 235-241

ANSWER 5.09 (2.00)

a. U238 (0.5)

b. Beff is the % of the delayed neutron fraction (Beta) that reaches thermal energy. (Also accept that fast fission is more effective than fast leakage) (0.5)

c. Larger[0.5] due to Beta-eff at EOL decreasing due to the increased effect of the Pu-239 Beta. This causes T to decrease making SUR larger[0.5]. (1.0)

REFERENCE
C-E Reactor Theory Pgs. 151-153

ANSWER 5.10 (1.00)

a. False (0.5)

b. False (0.5)

REFERENCE
General Physics pgs. 163-169

ANSWER 6.01 (3.00)

a. The rod will not move up [0.4] because the lift coil is used to raise the upper gripper [0.35], the rod won't fall or insert [0.4] due to the action of the lower gripper [0.35]. (1.5)

b. 1. Primary CEA position indicating system
2. Secondary CEA position indicating system
3. Metrascope
4. Group Deviation
5. rod bottom lights
6. Power dependent insertion limit
7. Lower electric limit light [5 @ 0.3 each] (1.5)

REFERENCE
CEDS system description pgs. 8-11

ANSWER 6.02 (3.00)

1. i, k, j,

2. f, h, i, j, b, c, d

3. e, g, h, i, j [0.2 each] (3.0)

ANSWER 6.03 (3.50)

a. 1. Master trip solenoid valve is tripped

2. mechanical trip valve is tripped

3. FWRV's close and bypass valves go to 5% flow position

4. Loss of load trip

5. Trip signal sent to the Steam bypass and Atmosphere dump valves. [4 @ 0.5 each] (2.0)

b. 1. Mechanical trip valve is reset

2. Master trip valve resets

3. Emergency trip pressure solenoid is deenergized, removing the hydraulic lock in signal.

4. Intermediate stop valves open. [3 @ 0.5 each] (1.5)

REFERENCE
Unit 1 Turbine system description pgs. 36-38

ANSWER 6.04 (1.50)

1. Either the turbine AFW line flow or the motor driven AFW line flow to the steam generator is > 100gpm [0.75]

2. A 200 psid or higher differential pressure exists between the steam generator and the combined AFW line to the steam generator [0.75] (1.5)

REFERENCE
Aux. Feedwater system description pg. 8

ANSWER 6.05 (2.50)

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

2. ASI [0.25]-from the nuclear instruments [0.25]

3. CEA Function [0.25]-fixed input from the safety analysis [0.25] (1.5)

b. A channel trip occurs if the axial shape index (YI) exceeds a positive or negative limiting value [0.5] of the power-dependent insertion limits (YP or YN) [0.5] (1.0)

REFERENCE
Reactor protection system description pg. 21

ANSWER 6.06 (3.50)

a. The atmospheric dumps will not open because the turbine is not tripped [0.75]. The steam bypass valves will modulate to control steam pressure at 895 psia [0.75] (1.5)

b. When reactor power is > 63% [0.5] the RRS will supply a quick-opening signal to open the steam dump and bypass valves rapidly [0.5]. When Tave drops below 548 F the quick-open signal clears [0.5]. As temperature decreases the valves will modulate shut until they are completely shut at 535 F with exception of the bypass valves which will modulate to control 900 psia/535 F [0.5]. (2.0)

REFERENCE
Main Steam and MSIV system description pgs. 16-24

ANSWER 6.07 (.25)

b

REFERENCE
Reactor protection system description pg. 74

ANSWER 6.08 (1.25)

a. Provides the ability to check radiation levels in containment under accident conditions. (0.25)

b. Located on the 73 foot level [0.2], one near the steam generator #2 [0.4] and one near the pressurizer [0.4]. (1.0)

REFERENCE
Radiation monitoring system description pg. 13

ANSWER 6.09 (2.25)

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 service water heat exchanger saltwater outlet valve opens [0.25]

3. The turbine building SRW isolation valve shuts [0.25] (0.75)

REFERENCE
ESFAS system description pg. 9

ANSWER 6.10 (1.25)

a. 2.6 DPM 10- 4 % to 15% power. (0.60)

b. Below - high noise level. Above - other protection, variable over power, high press. (0.65)

REFERENCE
CCNPP sys.desc. 59

ANSWER 6.11 (3.00)

a. prevents reactor operations when DNBR < minimum design [.3]
b. prevents peak local power from damaging core(KW/FT fuel centerline melt[.3]
c. protect fuel cladding against reactivity excursions to rapid to be protected by high pressure or TM/LP (CEA ejection) [.3]
d. uncontrolled CEA withdrawal or boron dilution incident during startup or very low power levels [.3]
e. prevevt excessive blowdown of RCS by a PORV or safety valve opening to prevent the reactor from generating more heat than can be removed by the steam generators [.3]
f. ensures the RX is tripped on conditions which require safety injection [.3]
g. loss of feedwater accident,assures RCS pressure does not exceed design [.3]
h. protects against excessively high steam flow caused by a major steam leak [.3]
i. DNB core protection on a sudden flow decrease [.3]
j. permits the operator to trip when the reactor should be tripped prior to an automatic or any other condition conducive to safety [.3] (3.0)

REFERENCE
CCNPP sys. desc. #59

ANSWER 7.01 (2.50)

a. 1. Notify the control room and pass the word of the incident in the containment.

2. Insure the safety of any fuel being handled and evacuate the containment.

3. Initiate a containment radiation signal.

4. Shift the four containment cooling fans to fast speed with maximum cooling water flow.

5. Start all available containment iodine filters.
[0.4 each] (2.0)

b. Insure the charcoal filters in the fuel handling area ventilation exhaust system are in service. (0.5)

REFERENCE
EOP-7 pg. 2

ANSWER 7.02 (1.50)

a. 1. LPSI pump trips

2. Sump valves open

3. Mini-flow recirc valves shut

4. Salt water inlet valve to CCS heat exchanger opens

5. Salt water outlet valve from the CCS heat exchanger returns to its pre-incident position.

6. Service water HX saltwater outlet returns to its pre-incident postion. [5 @ 0.3 each] (1.5)

b. ---- DELETED -----

L>OC / N>XT / P>G / D>

REFERENCE
ESFAS System Description pgs. 105-108

ANSWER 7.03 (1.50)

a. Weekly 300 mrem
Quarterly 2.0 Rem
Yearly 4.0 Rem (0.9)

b. Immediate supervisor
General supervisor and General supervisor-radiation safety
General supervisor and General supervisor-radiation safety (0.6)

REFERENCE
CCI-800A pgs. 9-10

ANSWER 7.04 (3.50)

a. Ensure blowdown recovery system has diverted to the Misc. waste processing system. (0.5)

b. Double ended rupture would cause TM/LP trip
PRZ empties and SI starts when system press. reaches (1578 psia). (1.0)

c. If time permits -

1. Check all operable charging pumps are running.

2. Reduce primary system temperature (to below 537"F) and primary press. by inserting CEA's.

3. Manually trip the reactor and turbine upon receiving TM/LP pretrip.

4. Maintain pressurizer level > 101 inches. [0.5 each] (2.0)

REFERENCE
EOP 6 pgs. 3-4

ANSWER 7.05 (2.50)

a. Tave controlled by bypass valves set @ 900# which is saturation for 532 F. CEA's control reactor power with reactivity addition or removal. (1.5)

b. Reactor power controlled by steam demand. Tave controlled by rod movement. (1.0)

REFERENCE
OP-4 pg. 3

ANSWER 7.06 (3.00)

a. Increase RCS boron concentration by 200 ppm for each full length CEA that is not fully inserted. (1.0)

b. 1. Turbine has tripped and the generator output, and exciter field breakers have tripped [1.2]

2. Diesel generators have started [0.3] (1.5)

c. Shut the MSIV's (0.5)

REFERENCE
EOP-15 pg. 2

ANSWER 7.07 (2.25)

a. Because the cold leg break allows the safety injection water to bypass the core [0.75], allowing the boric acid to crystallize [0.25] and restrict core cooling flow [0.25]. (1.25)

b. The calculated difference between RCS pressure and containment pressure cannot exceed 160 psid. (1.0)

REFERENCE
AOP-5 pgs. 1-2

ANSWER 7.08 (1.50)

a. 50% (0.5)

b. 70% (0.5)

c. 80% (0.5)

REFERENCE
EOP-3 pg. 2

ANSWER 7.09 (1.50)

a. To preclude the oil trapped above the upper cylinders leaking past the rings into the combustion space. (0.5)

b. 1. Jacket coolant temperature
2. Jacket coolant pressure (0.5)

c. Must remove the excitation from the generator field (0.5)

REFERENCE
OI-21 pg. 1

ANSWER 7.10 (2.75)

a. Operations with both valves above 1500 psia will cause severe thermal transients in the letdown system. (1.0)

b. Fluid end must be vented prior to pump operations. (0.5)

c. RCP seal leakoff [0.75], open VCT drain valve (CVC-161) or isolate RCP bleedoff [0.5]. (1.25)

REFERENCE
OI-2A pgs, 2,4,18

ANSWER 7.11 (1.50)

1. The pressurizer water volume is less than 150 inches indicated level. [0.75]

2. The secondary water temperature of each steam generator is less than 46 F above RCS temperature in the reactor vessel [0.75] (1.5)

REFERENCE
OI-1A pgs. 3-4

ANSWER 8.01 (3.00)

a. Within one hour [0.5] either restore the PORV(s) to operable status [0.5] or close the associated block valve(s) and remove power from the block valve [0.5]. (1.5)

b. Within one hour [0.5] either restore the block valves to operable status [0.5] or close the block valves and remove power from the block valves [0.5]. (1.5)

REFERENCE
Calvert Cliffs Tech. Specs. pg. 3/4 4-4

ANSWER 8.02 (1.50)

a. Four or more coincident incore channel alarms

b. ASI outside of the power dependent control limits (1.5)

REFERENCE
Calvert Cliffs Tech Specs. pg. 3/4 2-1

ANSWER 8.03 (2.00)

An operator must be stationed at the unit 2 turbine front standard [0.5] in direct communication with the control room [0.5]. This is due to no automatic trip of the turbine [0.5] and loss of all remote and automatic electrical trip functions [0.5]. (2.0)

REFERENCE
GSO Standing instructions 83-12

ANSWER 8.04 (2.50)

a. Special work permits are short term authorizations to perform a specific job [0.5] vice a radiation work permit which covers routine work for extended time periods. [0.5] (1.0)

b. 1. Originating work supervisor
2. Rad-con (0.5)

c. Approved by Supervisor-Radiation control and General supervisor - radiation safety [0.5]. Valid for one year [0.5] (1.0)

REFERENCE
CCI-800A pgs. 25-27

ANSWER 8.05 (1.50)

a. two, one (1.0)

b. 14 (0.5)

REFERENCE
CCI-101I pg. 4

ANSWER 8.06 (2.50)

a. Suspend all operations involving core alterations or positive reactivity changes [1.0]. Initiate and continue boration at > 40 gpm of 2300 ppm boric acid until Keff is < 0.95 (or boron concentration is >2300 ppm) [1.0] (2.0)

b. Immediately (0.5)

REFERENCE
Calvert Cliffs Tech. Specs. pg. 3/4 9-1

ANSWER 8.07 (1.50)

The restrictions on minimum water level ensures that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. (1.5)

REFERENCE
Calvert CLiffs Tech. Specs. pg. B3/4 9-3

ANSWER 8.08 (2.00)

a. continue to operate

b. shutdown

c. shutdown

d. continue to operate [0.5 each] (2.0)

REFERENCE
Calvert Cliffs Tech. Specs. pg 3/4 4-14

ANSWER 8.09 (3.00)

a. Should report [0.35] since it prevented RPS from fulfilling its safety function [0.4]

b. No report [0.35] needed when an action statement for LCO is entered [0.4]

c. Should report [0.35] shutdown due to inability to meet LCO action statement requirement[0.4]

d. No report [0.35] for ESF actuation during surveillance testing [0.4] (3.0)

REFERENCE
Calvert Cliffs Tech. Specs. and CCI-118J pg. 1

ANSWER 8.10 (2.50)

a. 1. Facility placed in at least hot standby within one hour.

2. NRC operations center notified by phone as soon as possible and in all cases within one hour.

3. Safety limit violation report shall be prepared

[3 @ 0.5 each] (1.5)

b. 1. The combination of thermal power, pressurizer pressure and highest Tc shall not exceed the limits of the T. S. curve (2.1-1) [0.5]

2. RCS pressure shall not exceed 2750 psia [0.5] (1.0)

REFERENCE
Calvert Cliffs Tech. Specs. pgs. 6-13 and 2-1

ANSWER 8.11 (3.00)

a. 5.2% dK/K [Unit 2] 4.3% dK/K [Unit 1] (0.5)

b. Immediately initiate and continue boration at > 40 gpm of 2300 ppm boric acid solution until SDM is restored. (1.0)

c. 1. The reactor can be made subcritical from all operating conditions.
2. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits.
3. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. [any 2 @ 0.75 each] (1.5)

REFERENCE
Calvert Cliffs Tech. Specs pgs. 3/4 1-1 and B3/4 1-1