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

EXAMINATION REPORT NO. 85-23

FACILITY DOCKET NOS. 50-317/50-318

FACILITY LICENSE NOS. DPR-53, DPR-69

LICENSEE: Baltimore Gas and Electric Company

Post Office Box 1475

Baltimore, Maryland 21203

FACILITY: Calvert Cliffs Unit 1 and 2

EXAMINATION DATES: August 20 to 23, 1985

CHIEF EXAMINER: Ward J Dudley — 9-14-95
Lead Reactor Engineer (Examiner) — Date

REVIEWED BY: RW Kelly — 9/18/85
Chief, Projects Section 1C — Date

APPROVED BY: WB Kester — 9/20/85
Chief, Project Branch No. 1 — Date

SUMMARY: Licensing examinations were administered to one RO and four SRO candidates. One RO and three SRO licenses will be issued. There was one SRO written examination failure. Generic weaknesses were identified in the use of Technical Specifications and the Emergency Response Plan. Generic strengths were noted in the use of Main Control Room Reference Material.

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REPORT DETAILS

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	1/0	3/1
Oral Exam	1/0	4/0
Overall	1/0	3/1

1. Chief Examiner at Site: N. Dudley

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

   - All candidates were familiar with Technical Specifications and action statements; however, some candidates had difficulty applying Technical Specifications to specific situations.
   - Some candidates incorrectly classified a steam generator tube rupture accompanied by a failed open atmospheric dump valve as an alert due to a lack of understanding of the definition of fission product boundaries.
   - Most candidates do not understand the difference between installed sources and source neutrons.
   - All candidates examined on the normal Shutdown Cooling lineup provided accurate responses without hesitation.

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

   - Most candidates were weak in basic electrical theory.
   - All candidates were confused as to the affect changing plant conditions have on the TM/LP trip set points.
   - Most candidates did not know the temperature on the Unit 2 turbine-generator main journal bearing at which the reactor should be tripped.

3. Comments on availability of, and candidate familiarization with, plant reference material in the control room:

   All candidates made good use of available reference material throughout the oral examinations.

4. Comments on availability of, and candidate familiarization with, design, procedure, and T. S. changes, and with LERs and recent significant events.

   All SRO candidates were knowledgable of the requirements for stationing an SRO in the containment during head left, the normal shutdown cooling lineup, the control requirements for the Steam Generator Isolation System block key during dooldown, and the appropriate instrument for reading Tc with the reactor vessel head removed. These items were noted as deficiencies in Inspection Reports Nos. 50-317/85-09 and 50-318/85-09. Inspection Reports Nos. 50-317/85-09, 50-318/85-09.

5. Personnel Present at Exit Interview:

   NRC Personnel

   N. Dudley, Lead Reactor Engineer (Examiner)

   Facility Personnel

   J. Hill, Supervisor of Operations Training
   R. Heibel, General Supervisor - Operation

6. Summary of NRC Comments made at exit interview:

   The number and type of examinations conducted were summarized. Weaknesses noted during the oral examinations in the application of the Technical Specifications and the Emergency Plan were described. The familiarity of candidates with recent plant problems was noted. Arrangements for the upcoming NRC requalification program evaluation were discussed. The facility was informed that the waiver of up to one month of the three month on-shift training prior to the examination was applicable on a case by case basis, however, would not be granted for entire licensing classes.

7. Summary of Facility questions and NRC responses at exit interview:

   The facility asked how may hours a candidate must spend in training in the control room to meet the three month on-shift eligibility requirement.

   The NRC responded that there was no specific number of hours; however, the facility was responsible for maintaining records to support three months of on-shift training for each candidate.

In January 1986, following the Unit 2 outage, new symptom based Emergency Operating Procedures (EOP) will be implemented. These procedures will replace the present event based EOP's. The facility requested that, as part of the NRC requalification program evaluation, the licensed operators be held responsible only for the new symptom based EOP's since they were the EOP's taught during requalification training.

The NRC stated that licensed operators would be examined only new symptom based EOP's.

8. CHANGES MADE TO WRITTEN EXAM DURING EXAMINATION REVIEW:

Question No.	Change	Reason
1.12b and 5.06	Add "or fluctuating power if voiding occurs at pump causing cavitation."	Question does not specify where in reactor vessel voiding occurs.
2.02	Change "DC" to "AC".	Provides correct source of backup power to inverters.
2.04b	Delete answer b. 2.	Saltwater valves are not considered part of component cooling water system.
2.08a	Change to "Ambient air which is cooled by service water in the CTMT air coolers".	Provides answer for RCP Motor instead of RCP Motor bearing.
2.10a	Change "leakoff return" to "recirc."	Provides correct nomenclature for facility.
4.07	Add "c. RAS cooling; verify proper RAS lineup and LPIS or HPIS flow".	Provides third means of cooling core during a LOCA.
4.08a	Add "or boron" after "CEA's".	Boron may also be used to control reactor power between 5% and 10%.
4.12c and 7.06	Change to "No trip; trip if not in hot standby in 6 hours."	Provides information included in T.S. action statement.

5.01	Add "or Q=m cpΔT" and "or using correct value for Cp".	Provides alternate method of calculating ΔT power.
5.03	Add "or 845/2700= 31.3%".	Uses rated thermal power to calculate efficiency.
5.04	Change "0.85" to "0.41". Change "68" to "33". Change "132" to "167".	Corrects math error.
6.04c	Change to "None".	Training material incorrectly includes an automatic action on Spent-Fuel Storage-Pool area radiation monitor.
7.03	Add "9. High or low steam flow. 10. SGIS initiation 11. AFAS block signal".	Includes additional indications in accordance with EOP-4.
7.04	Change to "a. Underload and cable slack; b. Dillon cell reading and high load interlock."	Corrects answers in accordance with defueling procedure.
7.04 reference	Change to "O.I. - 25c".	Provides procedure used for fuel handling.
7.09 b.3	Change "rapid" to "RCS".	Improves wording of answer.

Attchments:

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on Written Examinations

Attachment 1

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: CALVERT CLIFFS

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 85/08/20

EXAMINER: DUDLEY, N.

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
24.00	25.00	D. HOLM	_____	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
24.00	25.00	B. TILLET	_____	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
23.99	24.99	P. HURD	_____	3. INSTRUMENTS AND CONTROLS
24.00	25.00	C. ANDREWS	_____	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
95.99	100.00			TOTALS

FINAL GRADE _____%

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

_____
APPLICANT'S SIGNATURE

THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

QUESTION 1.01 (.50)

A centrifugal pump operating at 1800 rpm is pumping 400 gpm at a discharge head of 20 psi which requires power of 45 Kw. If the pump speed is increased to 2700 rpm, which one of the following will be true?

A. The discharge head will increase to 45 psi.
B. The power requirements will increase to 151 Kw.
C. The flow rate will increase to 800 gpm.
D. The pressure drop in the pumping system will increase to 30 psi.

QUESTION 1.02 (3.00)

What affect will the following events have on the Departure from Nucleate Boiling Ratio (DNBR)? Explain.

a. Dropping a rod at 100% power.
b. Reactor Coolant average temperature is reduced by 5 F while maintaining 100% power.

QUESTION 1.03 (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)

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

THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

QUESTION 1.04 (3.00)

Compairing a slightly subcritical reactor [shudown margin = 1%] to a greatly subcritical reactor [shutdown margin = 5%], explain how an addition of 0.5% of positive reactivity will affect the following.

a. THE CHANGE IN THE COUNT RATE

b. THE TIME TO REACH A STABLE COUNT RATE

QUESTION 1.05 (3.00)

The reactor has been operating at 50% for several days. CEAs are in manual control. An EH system malfunction causes the turbine control valves to open slightly, causing an increase in steam flow, which increases electrical output by 8 MW. With no operator action explain HOW and WHY each of the following parameters will change.

a. Steam generator pressure
b. Primary Tave

QUESTION 1.06 (.50)

If during accident conditions after a trip, the pressurizer pressure is 1600 psi, Th is 580 F, and Tc is 540 F, what approximate pressure reading should be seen on the subcooled meter?

A. 25 psi
B. 275 psi
C. 465 psi
D. 635 psi

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

QUESTION 1.07 (.50)

How long should it take to reach 1% power from an initial critical power level of 10 ^-4% if a constant 0.5 DPM startup rate is maintained? Choose the most correct answer.

A. 2 min
B. 4 min
C. 6 min
D. 8 min

QUESTION 1.08 (1.50)

Explain when AND why a plot of axial flux distribution for the Calvert Cliffs cores assumes a unique double-humped shape. (1.5)

QUESTION 1.09 (1.50)

a. What is the primary reason for having a start up neutron source for a new core ? (0.75)

b. Explain why a start up source is rapidly depleted after attaining high power conditions. (0.75)

QUESTION 1.10 (2.00)

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.

a. Nucleate boiling.
b. Accident condition in which coolant is boiled and converted to steam in the reactor vessel.
c. Heat from fission thru the fuel pellet.
d. Decay heat removal by natural circulation.

(xxxxx CATEGORY 01 CONTINUED ON NEXT PAGE xxxxx)

THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

QUESTION 1.11 (2.50)

a. If a trip from 100% power occurs with Xenon at equilibruim, what is the approximate time interval after trip that the operator should be concerned about Shutdown Margin decreasing below the SDM which existed when the trip occurred ? (0.75)

b. How would this approximate time interval compare if the trip occurred from 50% equilibruim conditions ? (0.75)

c. When a reactor is returned to 100% power from peak Xenon conditions,why does Xenon reactivity undershoot the equilibruim value? (1.0)

QUESTION 1.12 (3.00)

a. What will happen to the current drawn by a reactor coolant pump as the system is heated from 200 deg F to 564 deg F? Explain your answer.

b. What would happen to the current drawn by a reactor coolant pump if a void forms in the reactor vessel during accident conditions? Explain your answer.

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

QUESTION 2.01 (1.50)

a. What is the purpose of the TWO high range gamma monitors located in the containment? (0.5)

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

QUESTION 2.02 (1.50)

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

QUESTION 2.03 (2.00)

In addition to cooling RCS water during shutdown cooling operations, what THREE other systems use the shutdown cooling heat exchangers to provide a cooling function? Include when these would be used.

QUESTION 2.04 (2.50)

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 (1.0)

QUESTION 2.05 (2.00)

Why is running the diesel at less than normal operating speed not desired. Under what conditions may the diesel be run at less than normal operating speed for extended periods of time?

QUESTION 2.06 (1.50)

What will occur in the blowdown system if service water flow is lost to the blowdown heat exchanger during normal blowdown operations? Explain why?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.07 (3.00)

a. What will cause AND what action will result from a Recirculation Actuation Signal (RAS) ? (1.5)

b. It may be necessary to flush the core during long term cooling following a loss of coolant. Describe one flow path for performing this flush. (1.5)

QUESTION 2.08 (2.00)

Explain how each of the following components of a Reactor Coolant Pump (RCP) are cooled.

a. RCP motor
b. Thermal barrier area
c. Mechanical seals

QUESTION 2.09 (2.50)

a. Would containment design pressure and temperature limits be exceeded if the total Safety Injection System failed and all other Engineered Safety Features Systems fuctioned normally following a Loss of Coolant Incident (LOCI)? (0.5)

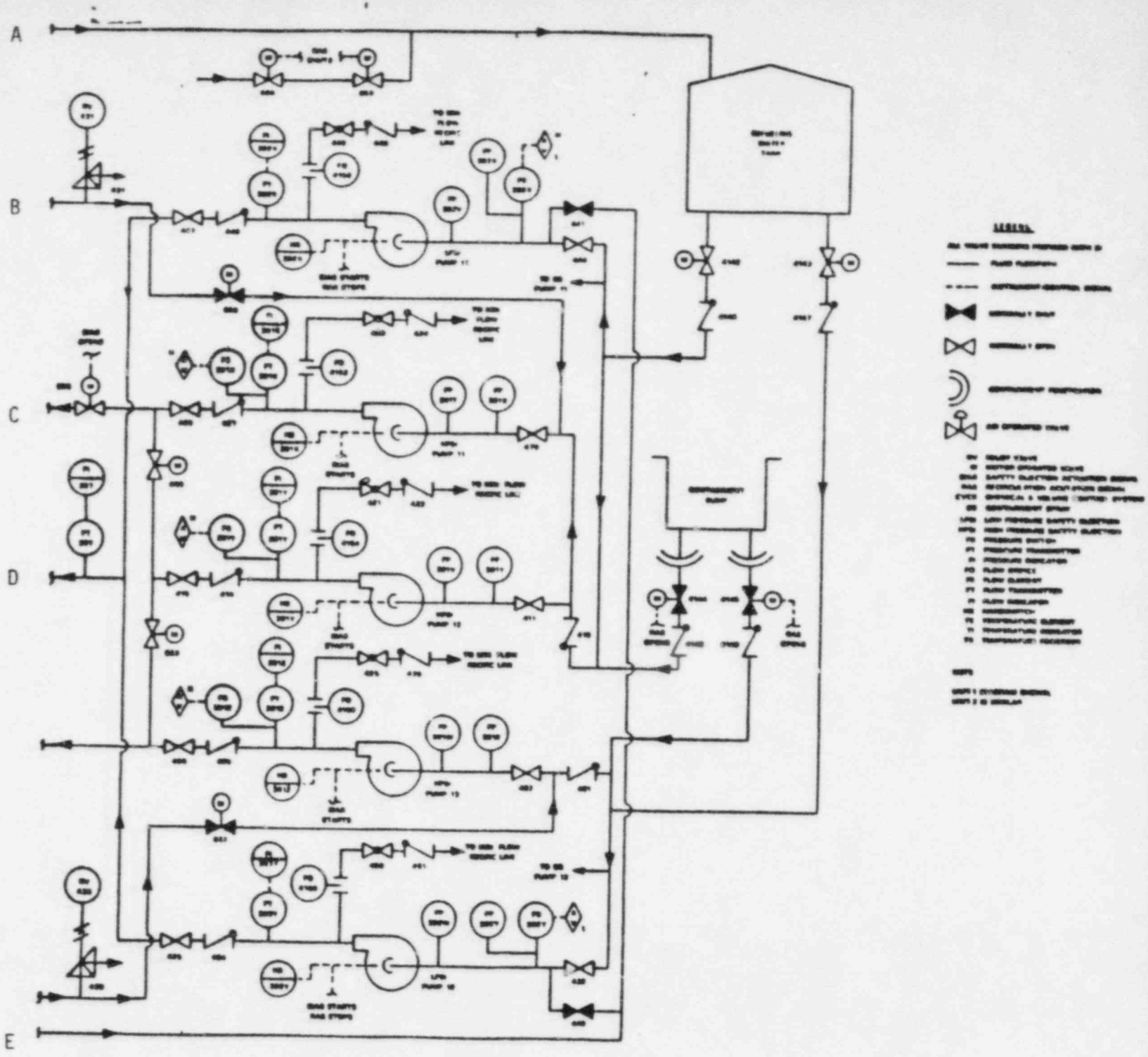
b. What two other systems would reduce containment pressure and temperature during a LOCI? Include how the systems perform their functions. (2.0)

QUESTION 2.10 (2.50)

a. For each of the indicated lines, A through E, on the attached figure, provide a description of where the line provides flow. (1.5)

b. Can the LPSI pump 11 be used to provide water to the Unit 2 Reactor Coolant System during shutdown cooling operations at Unit 2? Explain. (1.0)

(xxxxx CATEGORY 02 CONTINUED ON NEXT PAGE xxxxx)


A
B
C
D
E

QUESTION 2.11 (3.00)

a. What are the maximum allowable heatup and cooldown rates for the pressurizer? (1.2)

b. During a heatup, how often is the pressurizer temperature required to be verified? (0.6)

c. If during a cooldown the cooldown limit is exceeded and pressurizer temperature is 50 F below the limit, how can temperature be brought within limits? (1.2)

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

QUESTION 3.01 (.33)

As containment temperature increases, what will be the relationship between actual and indicated pressurizer level? Select the best answer.

A. Indicated level reads higher than actual level.
B. Indicated level increases reguardless of actual level.
C. Indicated level and actual level are the same.
D. Indicated level reads lower than actual level.

QUESTION 3.02 (.33)

Which of the following inputs will cause both a SIAS and CIAS initiation?

A. Low pressurizer pressures and high containment pressure.
B. High pressurizer pressure and high high containment pressure.
C. High pressurizer pressure and low steam generator level.
D. Low pressurizer pressure and variable overpower trip.

QUESTION 3.03 (.33)

The pressurizer level program is developed from: (Select the best answer)

A. The T ref signal
B. The T ave signal
C. The reactor power signal
D. The pressurizer pressure signal.

QUESTION 3.04 (2.00)

If after extended steady state operations at 100 % power the feed flow input signal to the feedwater control system [FIC-1111] fails low, how will the plant respond. Assume no operator action, follow transient to a stable plant condition and EXPLAIN the reason for the plant response.

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

QUESTION 3.05 (3.00)

a. Explain the logic that provides an Auxiliary Feed Actuating System (AFAS) signal and how it determines which pump(s) will start.

b. Explain how the AFAS can indicate that a rupture has occurred either in a steam line or an auxiliary feed line.

QUESTION 3.06 (2.25)

a. What type of detector(s) is/are used in the following monitors?

1. Main vent APDs
2. Main vent gaseous monitors
3. Containment APDs (0.75)

b. Which of the following monitor channels have automatic actions (other than indication and alarm) associated with them? Briefly describe the automatic actions.

1. Main vent APDs
2. Waste gas discharge monitor
3. Liquid waste discharge monitor
4. Component cooling radiation monitor (1.5)

QUESTION 3.07 (3.00)

a. How does an increase in the Thermal Margin/Low Pressure (TM/LP) trip setpoint affect the allowable DNBR ? (0.75)

b. List the three parameters that are measured and analyzed on the primary and secondary side of the Steam Generators to provide Reactor Protection signals. Also state what specific protection is provided. (2.25)

QUESTION 3.08 (3.25)

a. Describe the two methods of CEA position detection. (1.5)

b. What is the purpose of the CEA Group Deviation system(s) ? (0.75)

c. Describe how the system(s) function(s) to inhibit CEA motion. (1.0)

(xxxxx CATEGORY 03 CONTINUED ON NEXT PAGE xxxxx)

QUESTION 3.09 (3.50)

Concerning the axial power distribution reactor trip;

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

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

c. In addition to generating the AFD signal, what other functions does the AFD calculator provide? (1.0)

QUESTION 3.10 (2.50)

Explain how and why the pressurizer pressure control system would respond to a rapid insurge into the pressurizer which raised level 30 inches above the setpoint. Assume operating at 50% power and an initial pressure of 2250 psi.

QUESTION 3.11 (3.50)

a. What type of detectors makeup a wide range logarithmic nuclear instrument detector assembly? (0.75)

b. What type of detectors makeup the auxiliary wide range logarithmic instrumentation channel? (0.75)

c. Explain the function of the log count rate and campbell signal processing circuits for the wide range nuclear instrumentation. (2.0)

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

QUESTION 4.01 (.50)

Large overcooling transients can be characterized by: (Select the most correct answer.)

A. Decreasing primary system flow.
B. Increasing primary system boron concentration.
C. Decreasing primary system average temperature.
D. Increasing presurizer level.

QUESTION 4.02 (.50)

When performing shell warming of the main turbine, which of the following set of valves will be fully open: (Select the most correct answer.)

A. Stop valves
B. Control valves
C. Intercept valves
D. Intermediate stop valves.

QUESTION 4.03 (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 the weekly, the quarterly, or the yearly whole body radiation dose? (0.6)

QUESTION 4.04 (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)

(xxxxx CATEGORY 04 CONTINUED ON NEXT PAGE xxxxx)

QUESTION 4.05 (1.50)

Explain what will happen and what operator action is required for the unit 1 and unit 2 turbine upon experiencing a loss of power to the 11 DC bus. (1.5)

QUESTION 4.06 (1.50)

What affect would failure to equalize boron concentration between the Reactor Coolant System and the pressurizer have on normal plant power operations if Reactor Coolant System boron concentration is decreased 60 ppm?

QUESTION 4.07 (3.00)

The loss of reactor coolant emergency procedure discusses TWO modes of cooling the core depending on the size of the break. Briefly discuss how these two cooling modes would be verified operable by the operator.

QUESTION 4.08 (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 4.09 (2.50)

a. In accordance with AOP-8 (Excessive Reactor Coolant Leakage) what is the difference between a MAJOR leak and a MINOR leak? (1.0)

b. What is your required response for a major leak? (1.5)

(xxxxx CATEGORY 04 CONTINUED ON NEXT PAGE xxxxx)

QUESTION 4.10 (2.50)

The following concern a loss of AC power (EOP 15):

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 full insertion of the CEA's, what other automatic actions should occur? (1.5)

QUESTION 4.11 (3.00)

a. While operating at 80% reactor power you have indications of a steam line rupture from the #12 S/G. What immediate actions must you perform? (2.0)

b. Assume that in addition to the steam leak a safety injection actuation signal is received. What additional immediate actions must be performed? (1.0)

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

QUESTION 4.12 (3.50)

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. Assume plant has been operating for one week at 90% power and consider each situation seperately.

a. A rupture occurs in the Service Water subsystem.

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

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

d. An unexplained dilution raises power by 5%.

e. Instrument air pressure drops to 75 psig.

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

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

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

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 1.01 (.50)

1.12 a. Head increases to 45 psi.

ANSWER 1.02 (3.00)

a. The DNBR will decrease. [0.5]
The dropped rod will reduce the flux in the local area [0.5]
and will cause power to increase in other areas of the core. [0.5]

b. DNBR will increase. [0.5]
Reduced temperature will increase subcooling [0.5]
causing steam bubbles to quench more rapidly. [0.5]

REFERENCE
CE Thermal Hydraulics, p 14

ANSWER 1.03 (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 1.04 (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)

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

REFERENCE
C-E Reactor Theory Pgs. 147-148

ANSWER 1.05 (3.00)

a. SG pressure decreases [0.5]
SG pressure decreases as load is increased due to the removal of more energy. [1.0]

b. Tave will decrease. [0.5]
More energy is being removed from SG which will lower Tc [0.5]
Lower Tave will add positive reactivity to increase reactor power to match turbine load. [0.5]

ANSWER 1.06 (.50)

B. 275 psi

REFERENCE
Steam Tables

ANSWER 1.07 (.50)

D. 8 min

REFERENCE
CE Nuclear Physics, Reactor Theory, and Core Operating Characteristics, p 148

ANSWER 1.08 (1.50)

This will occur at the end of core life at high power conditions. [0.5] The peak in the upper portion of the core is due to the shift in power density to the top of the core caused by fuel depletion in the bottom.[0.5] The peak at the lower elevation is caused by the higher moderator density at the core inlet.[0.5] (1.5)

REFERENCE

CE Training Center Theory, P. 199

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 1.09 (1.50)

a. To provide an initial count rate for the nuclear instrumentation. (0.75)

c. A high neutron flux will result in the rapid burnout of the 94Pu-238. (0.75)

REFERENCE
CE Training Center Theory, Pp. 138,139

ANSWER 1.10 (2.00)

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

REFERENCE
General Physics pgs. 99-115

ANSWER 1.11 (2.50)

a. ~ 24 Hrs. (accept 20 - 30 hrs.) (0.75)

b. Time would be shorter. (< 20 hrs.) (0.75)

c. Due to the time delay in Xenon production from the decay of Iodine. (1.0)

REFERENCE
WJE 164
CE Training Center Theory,Pp. 204,206

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 1.12 (3.00)

a. Power decreases [0.5]
The pump is pumping the same volumetric flow rate but the density of the water decreases [0.5] which casuses a lower mass flow and less pump power. [0.5]

b. Power decreases. [0.5]
The pump is pumping against a lower discharge head due to the reduction of head losses in the area of the void. [1.0]

OR

FLUCTUATING POWER [0.5] IF VOIDS COLLAPSE AT PUMP, CAVITATION WOULD RESULT [1.0]

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 2.01 (1.50)

a. Provides the ability to check radiation levels in containment under accident conditions. (0.5)

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 2.02 (1.50)

There are 4 120 VAC busses per unit that are directly supplied by inverters. [0.25] Each of the inverters can be supplied by 125 VDC [0.5] or 120 VAC regulated power. [0.5] Two of the inverters on each unit are supplied by ~~DC~~ AC power from the other unit.[0.25]

REFERENCE
SD # 54, Fig's 54-1,54-2

ANSWER 2.03 (2.00)

1. Containment spray [0.4] during containment spray ops [0.26].

2. Spent fuel pool [0.4] when the complete core is removed and stored in the spent fuel pool [0.26].

3. Cooling water to HPSI suction [0.4] when HPSI is cavitating or during RAS [0.26]

REFERENCE
Safety Injection system decsription pgs. 10, 23 & 41

ANSWER 2.04 (2.50)

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 [0.5] ~~[0.33]~~

~~2. The service water heat exchanger saltwater outlet valve opens [0.33].~~

3. The turbine building SRW isolation valve shuts [0.5] ~~[0.33]~~ (1.0)

REFERENCE
ESFAS system description pg. 9

ANSWER 2.05 (2.00)

Operating at lower than normal operating speeds can damage the exciter-regulator [0.5] and the generator field [0.5]. Excitation must be removed from the generator field if operated at lower than normal speeds for maintenance [1.0]. (2.0)

REFERENCE
OI-21 pg. 1

ANSWER 2.06 (1.50)

Blowdown recovery flow is automatically diverted around the blowdown ion exchanger. [1.0]
Protects against overheating resin. [0.5]

REFERENCE
Blowdown system description pg. 4-6

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 2.07 (3.00)

a. 1. Caused by RWT level decreasing below approximately 30". (0.5)
(24" or T.S.)
2. Action that results: [0.33 ea.]

o Containment sump isolation valves open
o Both LPSI pumps stop
o Mini flow recirc. isolation valves receive a shut sig. (1.0)

b. 1. Containment sump > LPSI Pump > recirculation line > SDC return header > Hot leg [0.3 each]

or 2. HPSI Pump > Aux. HPSI header > CVCS > Pzr. Aux. Spray > Pzr. > Surge line > Hot leg [0.21 each] (1.5)

REFERENCE
SD # 788, Pp. 65,67,68

ANSWER 2.08 (2.00)

a. ~~Upper and lower oil reservoirs [0.3] are cooled by component cooling system. [0.33]~~ AMBIENT AIR [0.3] WHICH IS COOLED BY SERVICE WATER IN THE CTMT AIR COOLERS.
b. Cooled directly by component cooling. [0.66]
c. 1 gpm reactor coolant is passed through the pressure breakdown capillaries. [0.66]

REFERENCE
SD 5; RCS, p 16-19

ANSWER 2.09 (2.50)

a. No. [0.5] (temp and press would remain within limits)
b. CCS [0.4] condenses steam in CTMT removing heat and reducing and reduces pressure. [0.6]
CTMT Air Recirc System [0.5] removes heat by use of fans and cooling coils and thereby drop pressure.[0.5]

REFERENCE
SD 1; Containment System, p 38

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 2.10 (2.50)

a. A. SIT ~~leakoff~~ ~~return~~ RECIRC line
B. HX outlet
C. RCS loops or SIT
D. RCS loops or SIT
E. HX return header or RCS hot leg letdown [0.3 each]

b. No. [0.5] each unit has a seperate independent Safety Injection System. [0.5]

REFERENCE
SD 7 and 8; SI and CS Systems, p 2
Fig A-2

ANSWER 2.11 (3.00)

a. heatup at 100 F/hr [0.6]
cooldown at 200 F/hr [0.6]

b. every 30 minutes [0.6]

c. Stop cooldown. [0.6]
Turn on heaters [0.2]
Minimize spray [0.2]
Do not allow level to increase [0.2]

REFERENCE
SD 5; Reactor Coolant System, p 25

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 3.01 (.33)

~~3.9 mA~~ Indicated higher

ANSWER 3.02 (.33)

A. Low PZR press - high CTMT press.

ANSWER 3.03 (.33)

B. T ave signal

ANSWER 3.04 (2.00)

Reactor will trip on TG trip which trips on S/G Hi level [0.8]
Feedwater control valve will open [0.8]
and level dominant function will be too slow to prevent trip. [0.4]

REFERENCE
SD 17, SG System, p 18
SD 32, MF System, p 17-19

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 3.05 (3.00)

a. 1. Start signal is provided when 2/4 level indicating channels go below setpoint. (0.75)

2. The motor driven pump and the steam-driven pump that is valved in will start. (0.75)

b. Each of 4 S/G pressure channels is compared to it's corresponding channel on the other S/G. [0.25] When 2/4 channels reach the differential setpoint, a block signal is generated [0.25] and a steam line rupture alarm is actuated.[0.25] (0.75)

(The feed line break logic monitors 2/4 S/G pressure channels and 2 channels of AFW pressure to each S/G. Two channels of AFW flow to each S/G are also monitored. A pipe rupture signal is generated when):

AFW feed flow > 100GPM;[0.4] AND

S/G to feed line differential pressure > 175-200 psid [0.35] (0.75)

REFERENCE
SD # 34, Pp.5,6

ANSWER 3.06 (2.25)

a. 1. Scintillation detector
2. Geiger-Mueller tube
3. Scintillation detector [0.25 each] (0.75)

b. 1. none [0.25]
2. High alarm closes redundant waste gas discharge isolation valves [0.5]
3. High alarm closes two discharge isolation valves to stop discharge flow [0.5]
4. none [0.25] (1.5)

REFERENCE
Radiation Monitoring system description pgs. 20,24,39,43,45&48

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 3.07 (3.00)

a. A setpoint increase means that allowable DNBR has decreased (0.75)

b. 1. Coolant flow;[0.4] Prevent fuel damage should a loss of flow ccur. [0.35] (0.75)

2. S/G water level;[0.4] Ensure heat removal capability.[~~0.4~~ 0.35] (0.75)

3. S/G pressure;[0.4] Protect against excessive heat removal rate caused by steam rupture.[0.35] (0.75)

REFERENCE
WJE 166
General Physics HT&FF, Thermal Hydraulics sect. P.46
Sys. description # 59, Pp.14,29,30,31

ANSWER 3.08 (3.25)

a. 1. The primary indication utilizes up-down signals generated in the coil power programmers. The pulses are transmitted to an up-down counter in the plant computer. (0.75)

2. Secondary indication and CEA mimic display use two different sets of reed switches. (0.75)

b. To prevent power peaking in the core as a result of CEA deviation within a group. (0.75)

c. The system associated with the secondary indicating system auctioneers out the highest and lowest CEA'S within a group. [0.25]These signals are sent to a comparator and when a 4° deviation is sensed,[0.5] a bistable is tripped to actuate a motion inhibit interlock.[0.25] (1.0)

REFERENCE
WJE 169
System Description #60, Pp.25,26,29

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 3.09 (3.50)

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)

c. Also generates the allowable positive and negative ASI limits [0.5] and the axial offset used in the TM/LP calculator [0.5]. (1.0)

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

ANSWER 3.10 (2.50)

Power to proportional heaters go to zero. [0.6]
Spray valve ramps open until pressure is restored to 2300 psi. [0.6]
Power to proportional heaters decrease as pressure decreases below 2275 psi. [0.5]
Insurge will compress steam space raising pressure until spray condenses the steam to lower the pressure. [0.8]

REFERENCE
SD Reactor Coolant System Instrumentation, Figure 62-11

ANSWER 3.11 (3.50)

a. (2) proportional [0.4] and (1) fission chamber [0.35]
b. (2) fission chambers [0.75]
c. Log count rate produces a signal proportional to count rate upto 2 x 10 ^5 cps [0.8]
Cambelling circuit produces a signal proportional to power level above 6 x 10 ^2% power [0.8] and is combined with count rate signal to produce the power level signal. [0.4]

REFERENCE
SD 57; Nuclear Instrumentation System Description, p 4, 12, 19-21

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 4.01 (.50)

C. Decreasing primary system temperature.

ANSWER 4.02 (.50)

B. Control valves

ANSWER 4.03 (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 4.04 (1.50)

a. 50% (0.5)

b. 70% (0.5)

c. 80% (0.5)

REFERENCE
EOP-3 pg. 2

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

ANSWER 4.05 (1.50)

The unit 1 turbine will automatically trip [0.5]. The unit 2 turbine will not automatically trip but all remote and automatic electrical trips will be lost [0.5]. An operator must be stationed at the turbine front standard in direct communication with the control room to allow for manual tripping of the turbine [0.5] (1.5)

REFERENCE
GSO Standing Instructions

ANSWER 4.06 (1.50)

RCS temperature would decrease when an outsurge from the pressurizer occurred.

REFERENCE
OP-4, p 1

ANSWER 4.07 (3.00)

a. Natural circulation [0.5] verified by the observation of a temperature rise across the core, by the stabilization and decrease of Th and ability to change RCS temperature by feed/steam rates [1.5].

b. Core boiling [0.5] evidenced by the observation of saturation conditions in the RCS and an empty pressurizer [1.0].

c. RAS [illegible] [0.5] VERIFY PROPER RAS LINEUP AND [illegible] HPSI FLOW [1.0]

REFERENCE
EOP-5 pg. 2

ANSWER 4.08 (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 or boron concentration. (1.0)

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

REFERENCE
OP-4 pg. 3

ANSWER 4.09 (2.50)

a. If pressurizer level can be maintained with one charging pump you have minor leakage [0.5], if more than one charging pump is required, you have major leakage [0.5]. (1.0)

b. 1. Manually initiate or verify initiation of back-up charging pumps [0.75].
2. Trip the reactor if all charging pumps are running and pressurizer level is decreasing [0.75].
(Refer to Tech. Specs. for leakage limits)[0.25] (1.50)

REFERENCE
AOP-8 pgs. 1&4

ANSWER 4.10 (2.50)

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)

REFERENCE
EOP-15 pg. 2

ANSWER 4.11 (3.00)

a. 1. Trip reactor and turbine
2. Insure AFAS start, AFAS block and AFW pump starts
3. Isolate S/G
4. Emergency borate [0.5 each] (2.0)

b. 1. Stop all RCP's 5 seconds after the reactor is tripped
2. Implement EOP-2 [0.5 each] (1.0)

ANSWERS -- CALVERT CLIFFS -85/08/20-DUDLEY, N.

REFERENCE
EOP-4 pgs. 3-4

ANSWER 4.12 (3.50)

a. Trip reactor. [0.5]
b. Trip if not restored in 10 min. [0.3] or alarm received on RCP thrust bearaing temperature (>195 F). [0.2]
c. No trip. [0.2] ~~No condition will require trip.~~ [0.3]
d. No trip. [0.2] only if dilution raises power to RPS high power trip. [0.3]
e. No trip. [0.2] trip when pressure reaches 50 psig. [0.3]
f. No trip. [0.2] trip at 250 F. [0.3]
g. Trip reactor. [0.5]

REFERENCE
AOP 3, p 3
AOP 4, p 1
AOP 6, p 1, 2
AOP 7, p 4
AOP 7D, p 2
AOP 7E Unit 1, p 3
AOP 7E Unit 2, p 3

Attachment 2
MASTER

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: Calvert Cliffs

Reactor Type: PWR-CE

Date Administered: August 20, 1985

Examiner: J. D. Smith, J. W. Upton, Jr.

Candidate: Answer Key

INSTRUCTIONS TO APPLICANT:

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

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

Final Grade _____%

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

_____

Candidate's Signature

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

The following statements apply to Questions 5.01 through 5.05.

The Calvert Cliffs Unit 1 power plant has been operating continuously at a steady 100% of full power for 14 days. All control rods (CEAs) are fully withdrawn from the nuclear reactor core (ARO). All controlled parameters are equal to their respective programmed values. The fuel-burnup status is that the core has reached 10,000 MWD/MTU in cycle VII. The present boron concentration is 200 ppm and the inverse boron worth is 80 ppm/% $\Delta\rho$. With four (4) RCPs running, the total primary coolant flowrate is 134 M lbm/hr. The main generator is producing 845 MWe with a pf of 0.95 lagging. Use any of the provided figures and tables. Show your work and your procedures and indicate your assumptions.

Points Available

QUESTION 5.01

a. What is the "ΔT-power" as determined from the operating values in the primary loop? Give your answer in Btu/hr and in MW. (1.5)

b. If the flowrate of primary coolant through the core was reduced (without causing a trip), explain the effect that this would have on the ΔT across the core. (1.0)

- Section 5 continued on next page -

Points
Available

ANSWER 5.01

a. For 100% of full power,

T_H = 599.4°F [0.1]
T_{ave} = 572.5°F
T_C = 548°F [0.1]

$h(599.4°F) = 617$ Btu/lbm
$h(548°F) = 547$ Btu/lbm
$\Delta h = 70$ Btu/lbm
OR CORRECT VALUE FOR C_p [0.5].

~~(+0.5 for the procedure)~~

$Q = \dot{m}\Delta h$ OR $Q = \dot{m} C_p \Delta T$ [0.6]
$= (134 \times 10^6$ lbm/hr)(70 Btu/lbm) [0.1]
$= 9380 \times 10^6$ Btu/hr

~~(+0.75 for the procedure using the values above)~~

$$= \frac{9380 \times 10^6}{3600} \frac{\text{Btu}}{\text{sec}}$$

$$= \frac{9380 \times 10^6}{3600} (1055)\ \text{W} \quad [0.1]$$

$$= \frac{(9380)(1055)}{3600}\ \text{MW}$$

= 2748 MW

~~(+0.75 for the conversion and for the correct order-of-magnitude.)~~

~~(+1.5 max)~~

- Section 5 continued on next page -

Points
Available

ANSWER 5.01 (contd)

b. If the flowrate of the primary coolant is reduced, more heat will be added to the coolant as it passes through the core. This will raise the core ΔT **(+1.0)**.

Reference(s)

1. Calvert Cliffs: Systems Description No. 5, "Reactor Coolant System," Figure A-31 and p. 3.
2. Generic: Nuclear Power Plant Operator Training Program, "Thermodynamics, Fluid Flow and Heat Transfer for Nuclear Power Plants," Duke Power Company, pp. 56-57.
3. Generic: Steam Tables, C-E Power Systems.

- Section 5 continued on next page -

Points Available

QUESTION 5.02

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

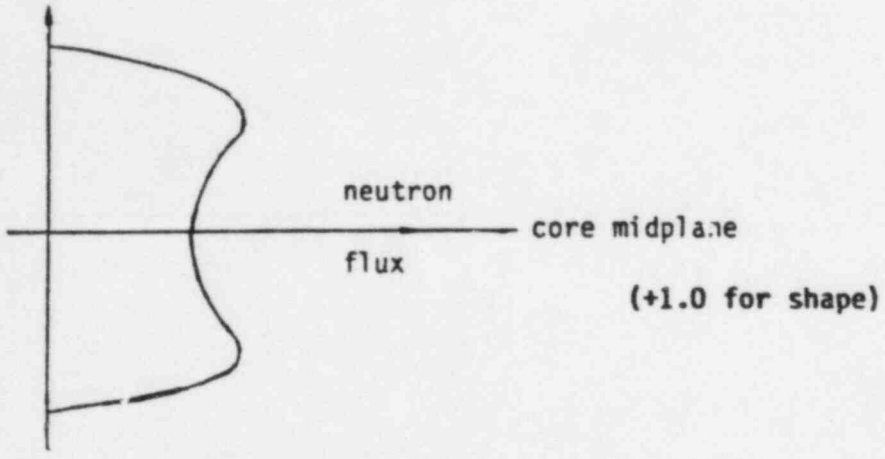
b. If the flowrate of the primary coolant through the core was reduced, explain the effect this would have on the axial neutron-flux shape. (1.0)

- Section 5 continued on next page -

Points
Available

ANSWER 5.02

a. core height



(+1.0 for shape)

With a fuel-burnup status of 10,000 MWD/MTU, the core is at the End-of-Life (EOL) **(+0.5)**. At EOL, the U-235 fuel has been disproportionally burned-up in the center of the core **(+0.5)**, which has depressed the BOL peak in the center of the core. Additionally, the fuel is disproportionally burned in the bottom half of the core and this shift of the relative fuel concentration to the top half of the core is responsible for the peak in the top half of the core **(+0.25)**. At full power, the moderator density decreases with core height. The higher moderator density in the bottom half of the core is responsible for the thermal-neutron peak in the bottom half of the core **(+0.25)**.

b. If the flowrate of the primary coolant was reduced, the temperatures of the reactor core would rise. The increase in temperature would be greater in the top half of the core than in the bottom half. This would cause the neutron flux distribution to shift to the bottom of the core. **(+1.0)**

Reference(s)

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

- Section 5 continued on next page -

Points Available

QUESTION 5.03

a. What fraction of the thermal power generated in the nuclear-reactor core is converted to electrical output power; i.e., what is the efficiency of the power plant? (1.0)

b. If the power plant is not 100% efficient, some of the thermal energy that is produced by the nuclear reaction is not converted into electrical output. What happens to this lost or unused energy? (1.0)

c. Determine the "apparent power" produced by the main generator. (1.0)

d. If the "MVAR loading" of the main generator was increased and the "real power" maintained constant, what would be the changes (qualitatively) in the pf (power factor) and in the "apparent power?" (1.0)

- Section 5 continued on next page -

Points Available

ANSWER 5.03

a. 845/2750 = .307
= 30.7% efficient OR 845/2700 = 31.3%

(+1.0 for the numerical procedure and for the correct order-of-magnitude)

b. The other 70% is "lost" or unused heat. Primarily, this heat is removed from the secondary fluid in the condenser by the water of the Circulating Water System **(+0.5)**. The heat absorbed by this cooling water is then transferred to the Bay **(+0.5)**. The heat that is not removed via the condenser cooling is dissipated by maintaining the temperature of the plant piping and components **(+0.5)**.

(+1.0 max)

c. Real power = (apparent power) (pf) **(+0.5)**

Apparent power = 845 M/0.95
= 889 M VA **(+0.5)**

d. The pf would decrease. **(+0.5)**
The apparent power would increase. **(+0.5)**

Reference(s)

1. Calvert Cliffs: System Description No. 35, "Circulating Water System, pp. 1-3 and Figures 35-1, 2 and 3.

2. C-E Training Center: "Pressurized Water Reactor Training Manual," Simulator Training Manual, Turbine-Generator Section, pp. 30-32.

- Section 5 continued on next page -

Points Available

QUESTION 5.04

a. If the control rods (CEAs) were inserted while in the manual sequential (MS) mode until Group 5 reached 50 in., and if the boron concentration were adjusted to maintain the present power level of 100%; what would be the new concentration of boron required for steady operation? Neglect any effect from changes in the xenon or samarium concentration. (1.5)

b. If the same maneuver was executed with the fuel burnup equal to 2000 MWD/MTU on cycle VII, the change in the boron concentration would be different. Explain what has changed and why. (1.5)

ANSWER 5.04

a. Using Figure 1-II.B.6,

$\Delta\rho$ rods = ~~0.85%~~ 0.47 (+0.5)

~~(0.85%)~~(80 ppm/%) = ~~68~~ 37 Δppm-boron (+0.5)

200-~~68~~ 37 = ~~132~~ 163 ppm (+0.5)

b. The boron worth (and inverse boron worth) would be different at BOL **(+0.25)**. In particular, the inverse boron worth is larger so that the change in the boron concentration for a 0.85% change in reactivity would be greater **(+0.25)**.

A core at BOL, as contrasted to EOL, has more U-235 fuel and correspondingly more boron. With more fuel and more boron in the core, a change in the boron concentration would have a smaller effect on the neutron population. Hence, the boron worth would be less and the inverse boron worth would be larger **(+1.0)**.

Reference(s)

1. Calvert Cliffs: NEOG-7, Rev. 10, Figure 1-II.B.6.
2. SONGS 2 and 3: "Reactor Theory Review," Student Handout, pp. 40-41.

- Section 5 continued on next page -

- Section 5 continued on next page -

Points Available

QUESTION 5.05

The Unit 1 power plant described prior to Question 5.01 (100% power, ARO, etc.) is to be taken from 100% to 80% of full power. Assume $T_{ave} = T_{ref}$ at both 100% and at 80% power.

a. What would be the magnitude of the change in reactivity (in % $\Delta\rho$) due to the change in power in the power plant. (0.5)

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

c. What five (5) factors must be considered in order to determine whether the reactor plant can be returned to 100% power immediately after a trip from 100%? (1.5)

- Section 5 continued on next page -

Points
Available

ANSWER 5.05

a. Using the power defect curve, Figure 1-II.C.1, reactivity change = 0.25% $\Delta\rho$. **(+0.5)**

b. See the attached figure, Figure 5.05 Answer. **(+1.5)**

c.
- CEA worth
- boron
- ~~z~~enon
- samarium
- power defect

(+0.3 each)

Reference(s)

1. Calvert Cliffs: NEOG-7, Rev. 10, Figure 1-II.C.1.
2. Generic: "Reactor Theory Review Student Handout," C-E Power Systems, Nuclear training, pp. 24-27.
3. Calvert Cliffs: NEOG 6, Attachment 6-3.
4. Calvert Cliffs: NEOG-7, Rev. 10, Figures 1-II.D.1 and 1-II.D.4.a.
5. Generic: "Reactor Theory Review Student Handout," C-E Power Systems, Nuclear Training, pp. 32-36.

- Section 5 continued on next page -

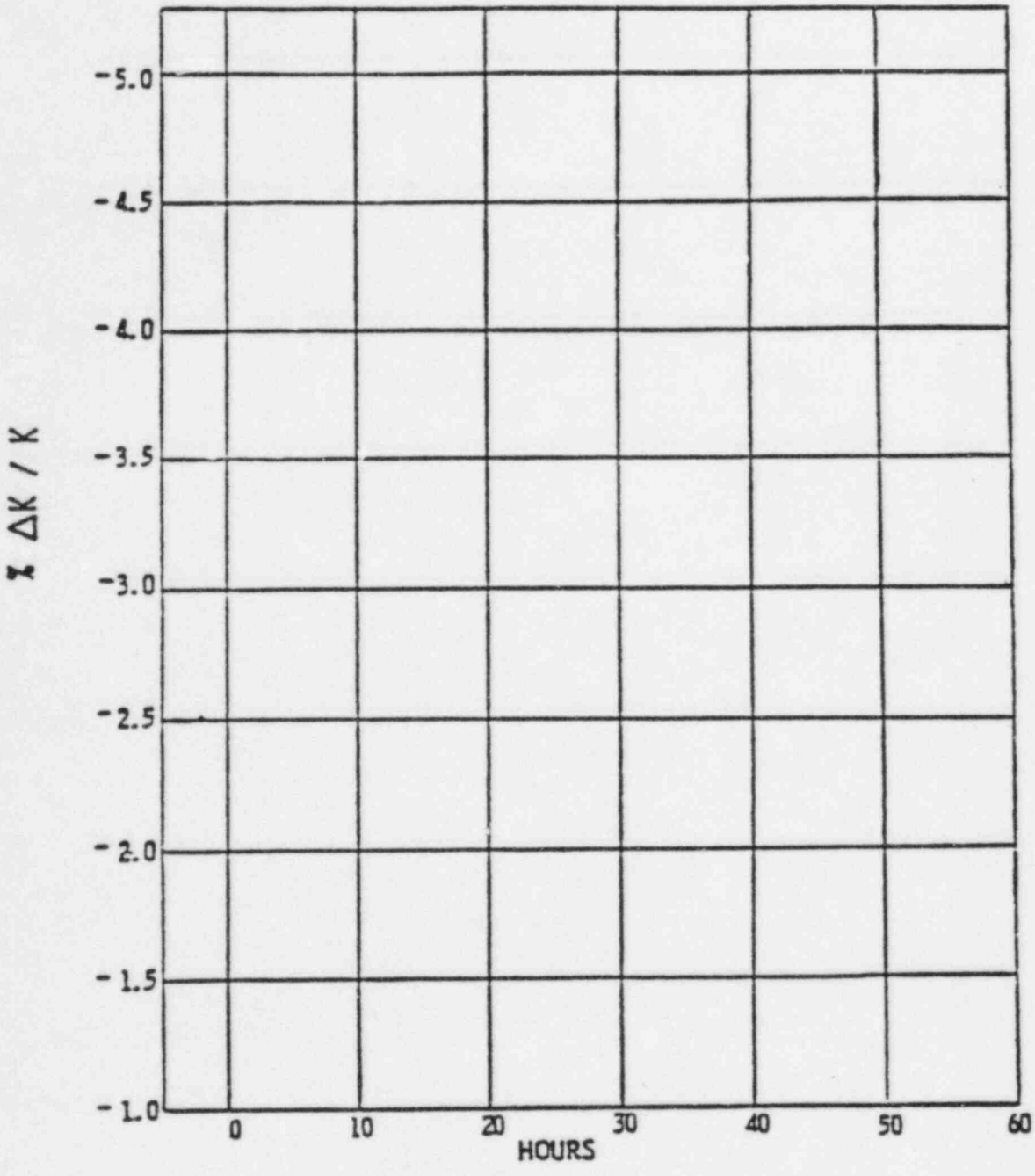


FIGURE 5.05 (QUESTION)

- Section 5 continued on next page -

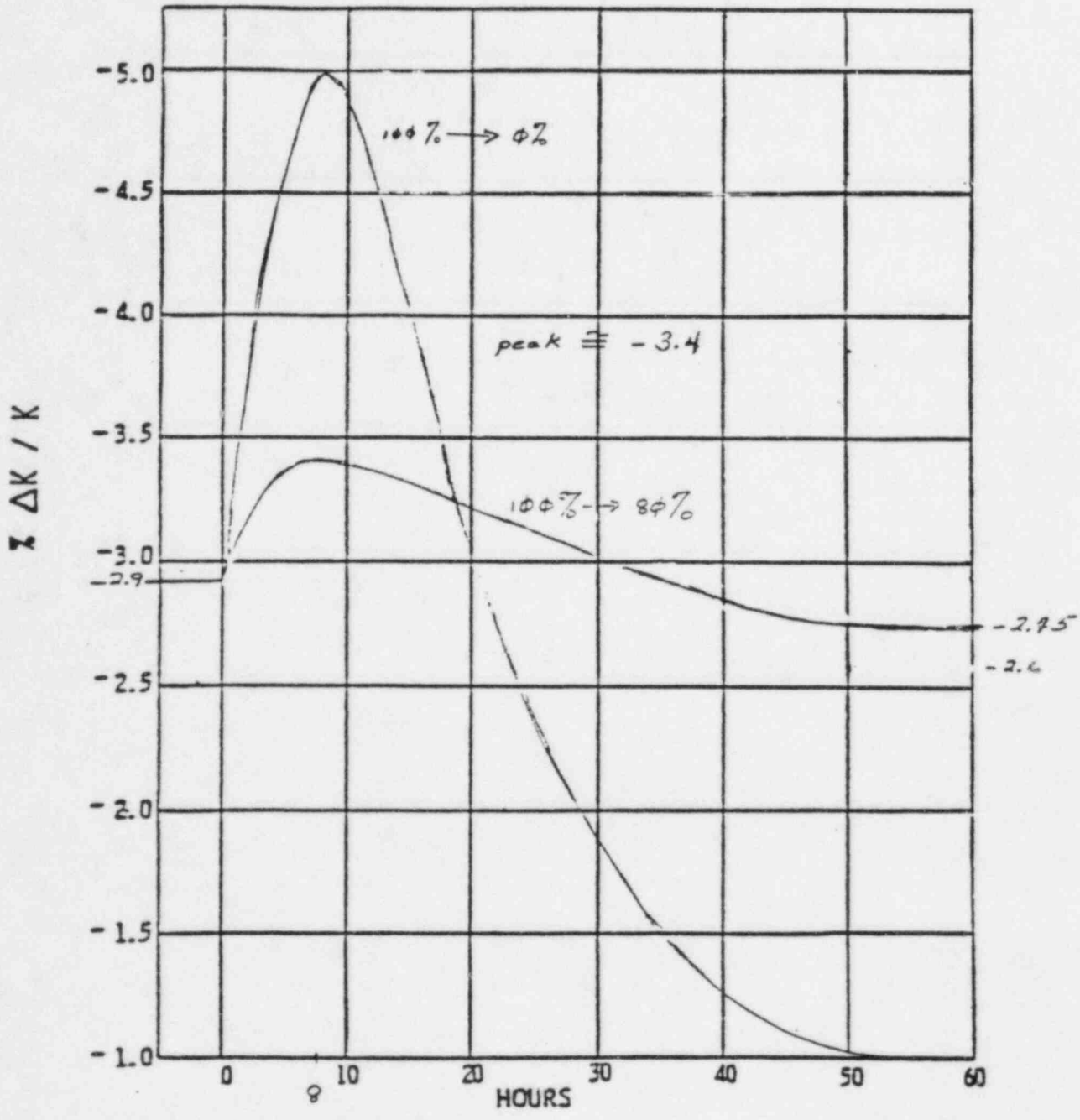


FIGURE 5.05 (ANSWER)

- Section 5 continued on next page -

Points Available

QUESTION 5.06

a. What will happen to the current drawn by the motor of a Reactor Coolant Pump as the system is heated from 200°F to 564°F? Explain your answer. (1.0)

b. What would happen to the current drawn by the motor of a Reactor Coolant Pump if a void forms in the reactor vessel during accident conditions? Explain your answer. (1.0)

ANSWER 5.06

a. Power decreases **(+0.33)**
The pump is pumping the same volumetric flow rate but the density of the water decreases **(+0.34)** which causes a lower mass flow and less pump power. **(+0.33)**

b. Power decreases **(+0.34)**
The pump is pumping against a lower discharge head due to the reduction of head losses in the area of the void. **(+0.66)**

Reference(s)

1. Calvert Cliffs: CAF.

CR
POWER FLUCTUATES [0.34] DUE TO VOID FORMING IN IMPELLER WHICH CAUSES CAVITATION [0.66]

- Section 5 continued on next page -

Points Available

QUESTION 5.07

Consider a centrifugal pump which is operating with the following values for the parameters power, flowrate and head:

power = 100 hp
flowrate = 1800 gpm
head = 150 ft.

a. If you doubled the speed of the pump, what would be the corresponding values for power, flowrate and head? (1.5)

b. In the situation where alternate core cooling is required (i.e., there is a SIAS), the reactor-coolant pressure may nevertheless exceed the LPSI pump discharge head. Describe the design feature of the Safety Injection System that exists to prevent the problem of running these centrifugal pumps with no flowrate. (1.0)

ANSWER 5.07

a. $flow_2 = flow_1 \times 2 = 3600$ gpm

$head_2 = head_1 \times 2^2 = 600$ ft

$power_2 = power_1 \times 2^3 = 800$ hp

(+0.5 each)

b. Each LPSI pump has a 40 gpm recirculation line that takes water upstream of the LPSI shutoff valve and returns the water to the RWT. **(+1.0)**

Reference(s)

1. Generic: Nuclear Power Plant Operator Training Program, "Thermodynamics, Fluid Flow and Heat Transfer for Nuclear Power Plants," Duke Power Company, p. 158.
2. Calvert Cliffs: System Description Nos. 7 and 8, "Safety Injection and Containment Spray Systems," July 1983, p. 18 and Figure A-2.

- Section 5 continued on next page -

QUESTION 5.08

What effect will the following events have on the Departure from Nucleate Boiling Ratio (DNBR)? Explain.

a. Dropping a rod at 100% power. (1.0)

b. Reactor coolant average temperature is reduced by 5°F while maintaining 100% power. (1.0)

ANSWER 5.08

a. The DNBR will decrease. **(+0.33)**
The dropped rod will reduce the flux in the local area **(+0.34)** and will cause power to increase in other areas of the core. **(+0.33)**

b. DNBR will increase. **(+0.33)**
Reduced temperature will increase subcooling **(+0.34)** causing steam bubbles to quench more rapidly. **(+0.33)**

Reference(s)

1. Calvert Cliffs: C-E Thermal Hydraulics, p. 14.

- Section 5 continued on next page -

Points Available

QUESTION 5.09

Comparing a slightly subcritical reactor (shutdown margin = 1%) to a greatly subcritical reactor (shutdown margin = 5%), explain how an addition of 0.5% of positive reactivity will affect the following:

a. The change in the count rate (1.0)
b. The time to reach a stable count rate. (1.0)

ANSWER 5.09

a. The slightly (greatly) subcritical reactor will have a larger (smaller) increase in count rate. **(+1.0)**

b. The slightly (greatly) subcritical reactor will take a longer (shorter) time to reach a stable count rate. **(+1.0)**

Reference(s)

1 Calvert Cliffs: C-E Reactor Theory, pp. 147-148.

- End of Section 5 -

6.0 PLANT SYSTEM DESIGN, CONTROL AND INSTRUMENTATION (25)

Points Available

QUESTION 6.01

This question refers to the operation of the Auxiliary Feedwater (AFW) System.

a. Explain the response of the AFW System to a continuously dropping level in S/G 11; include information on setpoints, the valves whose positions are affected and the pumps which are turned on or off. (1.5)

b. Explain how the Auxiliary Feedwater System actuation signal is developed; include sensors logic and output. (1.5)

- Section 6 continued on next page -

Points
Available

ANSWER 6.01

a. • starts the motor-driven pump **(+0.5)**

- opens the two steam-supply valves 4070,1 to the AFW System turbine pumps **(+0.5)**

- after a 60-second time delay, sends a signal to shut both main feedwater isolation valves and to run-back both Steam-Generator feed-pump turbines to a predetermined minimum speed setting **(+0.5).** (UNIT 1 RUN-BACK HAS BEEN REMOVED)

b. The AFS monitors four (4) channels of wide range level indication for each S/G **(+0.5).** Each low-level bistable module converts the analog level signal to a digital signal and compares it to a setpoint level of -170 inches **(0.5).** These bistable modules feed four logic matrices, two per S/G, which implement 2/4 logic. The output signals from the four low-level logic matrices enter two AFS START LOGIC MATRICES, each of which is an OR gate receiving one signal from a S/G 11 low-level matrix and one from S/G 12. In this case, both channels of AFS START would be actuated **(+0.5).**

Reference(s)

1. Calvert Cliffs: System Description No. 34, "Auxiliary Feedwater System," Figures 34-1, 34-3 and pp. 45-47, 63-64.

- Section 6 continued on next page -

Points Available

QUESTION 6.02

Calvert Cliffs Unit 1 has been operating at 40% of full power for the last 20 days when a reactor trip occurs.

a. Explain in detail the expected automatic response of the six (6) Steam-Dump/Turbine Bypass valves. (1.5)

b. Identify and explain two (2) indications (other than valve-position indications or limit lights) in the control room that could be used to verify that the Steam Dump/Turbine Bypass valves had opened excessively. (1.5)

ANSWER 6.02

a. Both steam-dumps ramp OPEN to lower T_{ave} to 535°F **(+0.6)**. Some bypass valves open sequentially to lower steam-header pressure to 900 psia or T_{ave} to 535°F **(+0.9)**.

b. Excessive cooldown would reduce T_{ave} and could reduce T_{ave} below the no-load value. The S/G pressure would be low due to the excessive cooldown. And the lowering of T_{ave} would produce a low pressure in the Pressurizer. **(+0.75 each, +1.5 max)**.

Reference(s)

1. Calvert Cliffs: System Description No. 19, "Main Steam and MSIV System," pp. 16-21.

- Section 6 continued on next page -

Points Available

QUESTION 6.04

If the high-level alarm setpoint is exceeded, what automatic action(s), if any, should occur for each of the following radiation instruments? Do not include the initiation of alarms.

a. CVCS-letdown radiation monitor (0.66)
b. Waste-Gas System discharge radiation monitor (0.67)
c. Spent-Fuel Storage-Pool area radiation monitor (0.67)

ANSWER 6.04

a. There is no automatic action. **(+0.66)**

b. The Waste-Gas System discharge isolation valves (discharge isolation valve and redundant isolation valve) should close. **(+0.67)**

c. ~~The Spent-Fuel Storage-Pool Ventilation System should auto-start. (+0.67)~~
NONE (0.67)

Reference(s)

1. Calvert Cliffs: System Description No. 14A, "Waste-Gas System," June 1981, pp. 11.
2. Calvert Cliffs: System Description No. 10, "Spent Fuel Pool and Spent Fuel Pool Cooling and Purification Systems," July 1983, p. 34.

- Section 6 continued on next page -

Points Available

QUESTION 6.05

Explain how power supply reliability is achieved for the 120 VAC instrument power buses. (Disregard computer power.) (1.5)

ANSWER 6.05

There are four (4) 120 VAC busses per unit that are directly supplied by inverters. **(+0.25)** Each of the inverters can be supplied by 125 VDC **(+0.5)** or 120 VAC regulated power. **(+0.5)** Two of the inverters on each unit are supplied by DC power from the other unit **(+0.25)**.

Reference(s)

1. Calvert Cliffs: System Description No. 54, Figures 54-1 and 54-2.

- Section 6 continued on next page -

Points Available

QUESTION 6.06

The following questions refer to the Component Cooling System.

a. The temperature of the water of the Component Cooling System must be maintained low enough to adequately carry the load. At what point in the System is the temperature monitored for control? To what value is this temperature controlled? How can this temperature be controlled? (2.0)

b. What is the expected response of the System to a SIAS at Unit 1 with the component cooling pump 13 selected to the 480 volt bus 11. Assume that there has not been an initiation of the CSAS and the CIS. (1.5)

ANSWER 6.06

a. It is the outlet water of the System heat exchanger that is controlled **(+0.5)**. It is controlled to 95°F **(+0.5)**. This control is accomplished by the automatic positioning of the temperature-controlled heat-exchanger bypass valve **(+0.5)** and/or by the manual control of the heat exchanger's salt-water outlet valve **(+0.5)**.

b. Upon the initiation of SIAS:

- component cooling pumps 11 and 12 start
- if the breaker for pump 11 does not close within 1 second, pump 13 starts
- the shutdown-cooling heat-exchanger outlet valves are opened
- the component-cooling heat-exchanger salt-water inlet and outlet valves are closed.

(+0.5 each, +1.5 max)

Reference(s)

1. Calvert Cliffs: System Description No. 40, "Component Cooling System," pp. 5, 17, 68.

- Section 6 continued on next page -

Points Available

QUESTION 6.07

Figure 6.07 (Question) on the next page shows the one-line electrical diagram for the Power Range Safety instrumentation.

a. Two (2) detectors are shown. What type of detectors are these and how are they constructed? (1.5)

b. What is the relationship between the output of amplifier-1 and its inputs? (What is the output of Amp 1?) (0.5)

c. What is the relationship between the output of amplifier-2 and its inputs? (What is the output of Amp 2?) (0.5)

d. What is the relationship between the output of amplifier-3 and its inputs? (What is the output of Amp 3?) (0.5)

e. The Internal Vibration Monitoring System (IVMS) receives information from the power range safety channels. What information is received and briefly describe what is done with the information to obtain an indication of vibration? (1.0)

- Section 6 continued on next page -

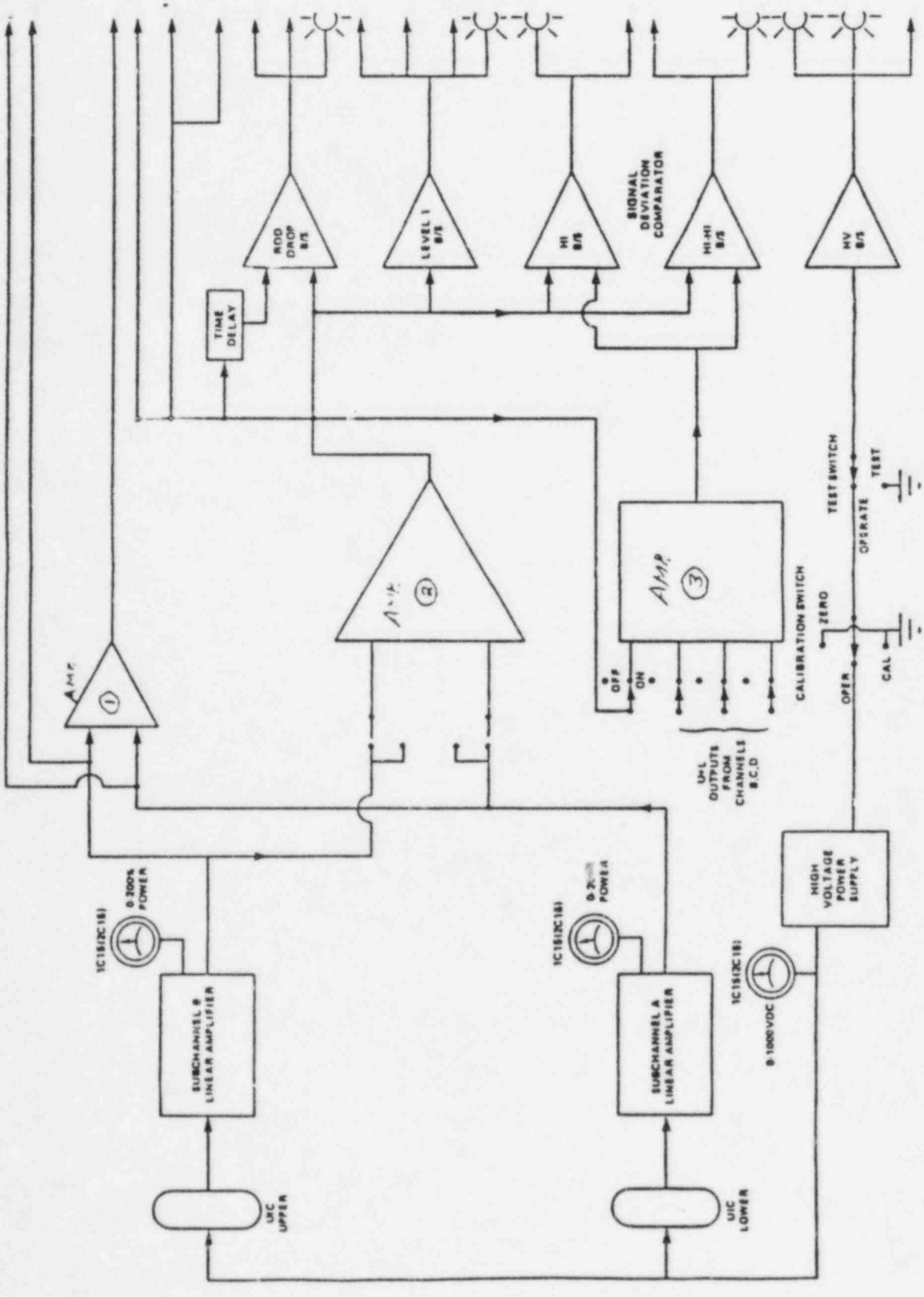


FIGURE 6.07 (QUESTION)

- Section 6 continued on next page -

Points
Available

ANSWER 6.07

a. The power range safety channels utilize UICs **(+0.5)**. Each UIC detector assembly consists of two independent UICs installed end-to-end in a common cylindrical housing. Each UIC contains a high voltage electrode and a signal electrode which connect to respective high voltage and signal cables. Nitrogen filler gas occupies the space between the two cylindrical electrodes. The UICs operate in the ionization region. The neutron-sensitive material used in the UIC is a coating on the electrode surfaces consisting of boron enriched with boron-10. **(+1.0)**

b. U-L **(+0.5)**

c. A+B/2 **(+0.5)**

d. sum U+L/8 **(+0.5)**

e. The A+B/2 output signal from each PR safety channel supplies an input to the IVMS **(+0.5)**. The system monitors reactor core and core-barrel motion by comparing the differential between input signal pairs and a frequency analysis of the signals is performed **(+0.5)**.

Reference(s)

1. Calvert Cliffs: System Description No. 57, "Nuclear Instrumentation, Revision 1, June 1984, pp. 35-36, Figure 57-14, pp. 39-42, 49-50.

- Section 6 continued on next page -

Points Available

QUESTION 6.08

a. What will cause AND what action will result from a Recirculation Actuation Signal (RAS)? (1.5)

b. It may be necessary to flush the core during long-term cooling following a loss of coolant. Describe one flow path for performing this flush. (1.5)

ANSWER 6.08

a. 1. Caused by RWT level decreasing below approximately 30" **(+0.5).**

2. Action that results:

- Containment sump isolation valves open
- Both LPSI pumps stop
- Mini flow recirculation isolation valves receive a shut signal.

(+0.33 each)

b. 1. Containment sump > LPSI Pump > recirculation line > SDC return header > hot leg **(+0.3 each)**, or

2. HPSI pump > Aux. HPSI header > CVCS > Pzr aux. spray > Pzr > surge line > hot leg **(+0.21 each).**

Reference(s)

1. Calvert Cliffs: SD #7 and 8, pp. 65,67,68.

- Section 6 continued on next page -

Points Available

QUESTION 6.09

a. Would containment design pressure and temperature limits be exceeded if the total Safety Injection System failed and all other Engineered Safety Features Systems functioned normally following a Loss of Coolant Incident (LOCI)? (0.5)

b. What two (2) other systems would reduce containment pressure and temperature during a LOCI? (0.5)

ANSWER 6.09

a. No (temperature and pressure would remain within limits). **(+0.5)**

b. CCS. **(+0.25)**
CTMT Air Recirculation System. **(+0.25)**

Reference(s)

1. Calvert Cliffs: SD #1; Containment System, p. 38.

- End of Section 6 -

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

Points Available

QUESTION 7.01

Unit 1 experiences a reactor trip from 100% power.

a. The first step of the Immediate Operator Actions is to depress the reactor trip buttons and to verify that two (2) results have taken place. List those two (2) responses. (1.0)

b. If the RPS fails to trip the reactor and de-energizing the CEDM MG sets still results in two (2) CEAs failing to fully insert, then what action should the operators take? (1.0)

c. What four (4) responses of the Turbine/Generator should be verified as part of the Immediate Operator Actions? (1.0)

d. If the Generator output breakers have not tripped and the Turbine stop valves cannot be confirmed to have shut, then what action(s) should the operators take? (0.5)

- Section 7 continued on next page -

Points
Available

ANSWER 7.01

a. • CEAs are fully inserted.
   • Reactor power is decreasing.

   (+0.5 each)

b. "Increase the boron concentration **(+0.5)** by 200 ppm for each CEA that fails to fully insert." Hence, increase the boron concentration by 400 ppm **(+0.5)**.

c. • Turbine is tripped.
   • Generator output breakers are open.
   • Generator field breaker is open.
   • Exciter field breaker is open.

   (+0.25 each)

d. Shut the MSIVs. **(+0.5)**

Reference(s)

1. Calvert Cliffs: EOP-1 (Reactor Trip), Rev. 7, pp. 2-4.

- Section 7 continued on next page -

Points Available

QUESTION 7.02

Explain what will happen and what operator action is required for the unit 1 and unit 2 turbine upon experiencing a loss of power to the 11 DC bus. (1.5)

ANSWER 7.02

The unit 1 turbine will automatically trip **(+0.5)**. The unit 2 turbine will not automatically trip but all remote and automatic electrical trips will be lost **(+0.5)**. An operator must be stationed at the turbine front standard in direct communication with the control room to allow for manual tripping of the turbine **(+0.5)**.

Reference(s)

1. Calvert Cliffs: GSO Standing Instructions.

- Section 7 continued on next page -

Points
Available

QUESTION 7.03

List six (6) control room indications of a steam line rupture on Unit 1. (3.0)

ANSWER 7.03

1. Rapid decrease in steam generator pressure
2. Reactor trip on low steam generator pressure
3. Rapid drop in reactor coolant temperature
4. Rapid drop in reactor coolant pressure
5. Loud noise and poor inplant visibility depending on the location of the rupture
6. Low pressurizer level
7. Safety Injection System Actuation Signal (SIAS)
8. Containment Isolation Signal (CIS) and Containment Spray Actuation Signal (CSAS) if location of rupture is inside containment

(+0.5 each, +3.0 max)

Reference(s)

1. Calvert Cliffs: EOP-4 (Steam Line Rupture), Rev. 9, I, p. 2.

- Section 7 continued on next page -

Points Available

QUESTION 7.04

a. What indication does a fuel handling operator have that a fuel bundle binds 1 in. from the bottom while loading fuel? (0.75)

b. What indication does a fuel handling operator have that a fuel bundle is mechanically bound during fuel removal? (0.75)

ANSWER 7.04

a. UNDERLOAD AND CABLE SLACK

b. ~~b.~~ The Dillon cell reading **(+0.35)** and high load interlock. **(+0.4)**

~~b. The tape measure attached to the gripper will not read properly. (+0.75) (TV surveillance after loading the core.)~~

Reference(s)

1. Calvert Cliffs: ~~CAF.~~ O.I. - 25 C

- Section 7 continued on next page -

Points Available

QUESTION 7.05

List the weekly administrative external dose limits for individuals 18 years of age or older. (2.5)

ANSWER 7.05

External Dose:

The weekly administrative exposure limits, for individuals 18 years of age or older, where a week is defined as 2400 h Friday through 2400 h Friday, are:

a. Dose to the whole body, head, trunk, blood forming organs, lens of eyes or gonads shall be limited to 300 mrem/week until the quarterly dose accumulation reaches 900 mrem (alert point and quarterly limit for 18-year olds) and 150 mrem/week for the balance of the quarter.

b. Dose to the skin of the whole body shall be limited to 2000 mrem/week.

c. Dose to the hands and forearms, feet and ankles shall be limited to 4500 mrem/week.

(+0.5 for each underlined limit)

Reference(s)

1. Calvert Cliffs: CCI-800A, Attachment (1), p. 9.

- Section 7 continued on next page -

Points Available

QUESTION 7.06

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. Assume plant has been operating for 1 week at 90% power and consider each situation separately.

a. A rupture occurs in the Service Water subsystem. (0.5)

b. The motor on the operating component cooling pump fails. (0.5)

c. It is discovered that containment integrity has been breached when a blind flange is found improperly secured. (0.5)

d. An unexplained dilution raises power by 5%. (0.5)

e. Instrument air pressure drops to 75 psig. (0.5)

f. The main journal bearing metal temperature is 230°F (5°F above the alarm set point) for the Unit 1 turbine. (0.5)

g. The main journal bearing metal temperature is 225°F (5°F above the alarm set point) for the Unit 2 turbine. (0.5)

- Section 7 continued on next page -

Points
Available

ANSWER 7.06

a. Trip reactor **(+0.5)**.

b. Trip if not restored in 10 min **(+0.3)** or alarm received on RCP thrust bearing temperature (>195°F) **(+0.2)**

c. No trip, **(+0.2)** ~~no condition will require trip. **(+0.3)**[4]~~ TRIP IF NOT IN HOT STANDBY IN 6 HOURS

d. No trip, **(+0.2)** only if dilution raises power to RPS high power trip **(+0.3)**.

e. No trip, **(+0.2)** trip when pressure reaches 50 psig **(+0.3)**.

f. No trip, **(+0.2)** trip at ~~250~~ 240°F **(+0.3)**.

g. Trip reactor **(+0.5)**.

Reference(s)

1. Calvert Cliffs: AOP 3, p. 3
2. Calvert Cliffs: AOP 4, p. 1.
3. Calvert Cliffs: AOP 6, pp. 1-2, T.S. 3.6.1.1
4. Calvert Cliffs: AOP 7, p. 4.
5. Calvert Cliffs: AOP 7D, p. 2.
6. Calvert Cliffs: AOP 7E Unit 1, p. 3.
7. Calvert Cliffs: AOP 7E Unit 2, p. 3.

- Section 7 continued on next page -

Points Available

QUESTION 7.07

Specify the requirements that an individual must satisfy to serve as an escort in a controlled area. (1.0)

ANSWER 7.07

The escort must be Controlled Area qualified and assigned to the plant for a minimum of 6 months or authorized by the SRC to act as a Controlled Area Escort. **(+1.0)**

Reference(s)

1. Calvert Cliffs: CCI-800A, Attachment (1), D, p. 32.

QUESTION 7.08

What requirements are imposed on the Pressurizer if RCS boron concentration is to be changed by 50 ppm or greater? (1.0)

ANSWER 7.08

Whenever boron concentration is changed by 50 ppm or greater, the Pressurizer spray valves shall be operated, consistent with pressure requirements, until the boron concentration of the Pressurizer is within 10 ppm of RCS concentration. **(+1.0)**

Reference(s)

1. Calvert Cliffs: OP-4 (Plant Shutdown from Power), Rev. 7, D, p. 1.

- Section 7 continued on next page -

Points Available

QUESTION 7.09

a. What four (4) indications are used to continually verify natural circulation? (1.6)

b. What operator actions can be used to affect each of the fcllowing conditions which are required to accomplish natural circulation?

1. Maintain RCS inventory (+0.4)
2. Maintain adequate Steam Generator water inventory (+0.8)
3. Maintain adequate RCS subcooling (+0.7)

ANSWER 7.09

a. ΔT is less than full power ΔT
T_c is constant or decreasing
T_h is stable
No abnormal difference between Th and core exit thermocouples

(+0.4 each)

b. 1. Proper operation of CVCS **(+0.4)**
2. Feeding the SG with main or auxiliary feedwater **(+0.4)**
Discharging steam using Turbine Bypass and/or Atm. dumps **(+0.4)**
3. Rapid cooldown rate **(+0.4)** Pzr press. maintained high **(+0.3)**
RCS

Reference(s)

1. Calvert Cliffs: EOP-12, Att (1), pp. 1-2.

- Section 7 continued on next page -

Points
Available

QUESTION 7.10

During operation at power a piping rupture occurs on the main header of the instrument air system. List the sequence of events and their set points that automatically take place for this event. (2.0)

ANSWER 7.10

a. The standby instrument air compressor starts at 90 psig.

b. The automatic plant air to instrument air cross connect valve opens at 85 psig (instrument air pressure).

c. The plant air header automatic isolation valve closes at 85 psig (plant air header pressure) causing the plant air compressor to discharge to the instrument air system only.

d. The other unit's plant air compressor automatically starts at 90 psig and supplies air to the affected units through normally open cross-connect valves.

(+0.5 each)

Reference(s)

1. Calvert Cliffs: AOP-7D (Loss of Instrument Air), Rev. 7, Discussion, p. 1.

- Section 7 continued on next page -

Points Available

QUESTION 7.11

According to the loss of load procedure, a sudden large reduction in power demand would likely be due to one (1) of four (4) malfunctions or system changes. List these four (4) occurrences. (2.0)

ANSWER 7.11

1. Spurious closure of a main stream isolation valve

2. Spurious closure of one (1) or more governor valves, without a reactor trip

3. Spurious closure of two (2) or more turbine stop valves, intercept valves or intermediate stop valves without a reactor trip

4. Separation from the interconnected system (loss of lines 5051 and 5052)

(+0.5 each)

Reference(s)

1. Calvert Cliffs: (Loss of Load), Rev. 7, Discussion, p. 1.

- End of Section 7 -

8.0 ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (25)

Points Available

QUESTION 8.01

While operating at power, it is determined that the shutdown margin is 4.1% Δk/k. What actions, if any, are required by Tech-Specs.? (2.0)

ANSWER 8.01

With the SHUTDOWN MARGIN <4.3% Δk/k, immediately **(+0.5)** initiate and continue boration at ≥40 gpm **(+0.5)** of 2300 ppm boric acid solution or equivalent **(+0.5)** until the required SHUTDOWN MARGIN is restored **(+0.5)**.

Reference(s)

1. Calvert Cliffs: Technical Specification, Section 3.1.1.1.

- Section 8 continued on next page -

Points Available

QUESTION 8.02

a. Complete the following table to indicate the minimum shift crew composition in the applicable modes for Unit 2 with Unit 1 in mode 5 or 6. (2.0)

MINIMUM SHIFT CREW COMPOSITION #

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

b. If the minimum shift-crew composition cannot be met, what action(s), if any, must be taken by the shift supervisor? Specify any time requirements associated with the action(s)? (1.0)

- Section 8 continued on next page -

Points
Available

ANSWER 8.02

a.

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

(+0.25 each)

b. The shift supervisor must immediately act to bring the composition to the required minimum **(+0.5)**. He has 2 hours **(+0.5)**.

Reference(s)

1. Calvert Cliffs: Technical Specification Administrative Controls, Table 6.2-1.

- Section 8 continued on next page -

Points Available

QUESTION 8.03

What are the four (4) conditions that must be met for the Refueling Water Tank to be considered operable in modes 1, 2, 3 and 4? (2.0)

ANSWER 8.03

The Refueling Water Tank shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons
b. A boron concentration of between 2300 and 2700 ppm
c. A minimum water temperature of 40°F
d. A maximum solution temperature of 100°F in mode 1.

(+0.5 each)

Reference(s)

1. Calvert Cliffs: Technical Specification, Section 3.5.4.

- Section 8 continued on next page -

Points
Available

QUESTION 8.04

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. (0.5)
b. 0.2 gpm from a T_h loop RTD weld (0.5)
c. 1.3 gpm unknown leakage (0.5)
d. 3 gpm seat leakage on a Pressurizer safety valve (0.5)

ANSWER 8.04

a. continue to operate
b. shutdown
c. shutdown
d. continue to operate

(+0.5 each)

Reference(s)

1. Calvert Cliffs: Technical Specifications, pp. 3/4, 4-14.

- Section 8 continued on next page -

Points Available

QUESTION 8.05

a. Under what two (2) conditions can a temporary electrical jumper be installed without meeting the logging requirements CCI-117D, the instruction that addresses temporary electrical jumpers? (1.25)

b. What are two (2) methods of verifying the proper removal of a temporary electrical jumper? (1.25)

ANSWER 8.05

a. 1. When approved procedures include instructions for installation and lifting of the jumper. **(+0.6)**

2. When installation and lifting is covered in a maintenance request. **(+0.6)**

b. 1. A second individual verifies. **(+0.6)**

2. Verify removal with a functional test. **(+0.6)**

Reference(s)

1. Calvert Cliffs: WJE 189.
2. Calvert Cliffs: CCI-117D, pp. 2,3,8.

- Section 8 continued on next page -

Points
Available

QUESTION 8.06

What provisions are required in order to make a temporary change to a procedure? (3.5)

ANSWER 8.06

Temporary changes to procedures may be made provided:

a. The intent of the original prccedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the POSRC and approved by the Plant Superintendent within 14 days of implementation.

(+0.5 for each underlined provision)

Reference(s)

1. Calvert Cliffs: Tecnnical Specification Administrative Control, Section 6.8.3.

- Section 8 continued on next page -

Points
Available

QUESTION 8.07

What are the four (4) administrative actions that shall be taken if RCS pressure reaches 2800 psia? (4.0)

ANSWER 8.07

The following actions shall be taken in the event a safety limit is violated.

a. The facility shall be placed in at least HOT STANDBY within 1 hour. **(+1.0)**

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Manager, Nuclear Power Department and the OSSRC, shall be notified within 24 hours. **(+1.0)**

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems and structures, and (3) corrective action taken to prevent recurrence. **(+1.0)**

d. The Safety Limit Violation Report shall be submitted to the Commission; the OSSRC; and the Manager, Nuclear Power Department within 14 days of the violation. **(+1.0)**

Reference(s)

1. Calvert Cliffs: Technical Specification Administrative Control, Section 6.8.1.

- Section 8 continued on next page -

Points Available

QUESTION 8.08

List the four (4) conditions which must be met in order that each Safety Injection Tank will be considered operable when in mode 1. (2.0)

ANSWER 8.08

Each reactor coolant system Safety Injection Tank shall be OPERABLE with:

a. The isolation valve open

b. A contained borated water volume of between 1113 and 1179 cubic feet of borated water (equivalent to tank levels of between 187 and 199 inches, respectively)

c. A boron concentration of between 2300 and 2700 ppm

d. A nitrogen cover-pressure of between 200 and 250 psig

(+0.5 each)

Reference(s)

1. Calvert Cliffs: Technical Specification, Section 3.5.1.

- Section 8 continued on next page -

Points Available

QUESTION 8.09

a. When should the Technical Support Center and the Operational Support Center be activated? (0.9)

b. What is the function of:

1. Technical Support Center? (0.8)
2. Operational Support Center? (0.8)

ANSWER 8.09

a. When an alert or higher emergency action level is declared. **(+0.9)**

b. 1. To analyze current and projected plant status and provide advice and assistance during an accident. **(+0.8)**

2. An assembly point for off-shift operators, HP personnel, and other emergency response personnel. **(+0.8)**

Reference(s)

1. Calvert Cliffs: WJE 195.
2. Calvert Cliffs: ERPIP Sections 4.1.2 and 4.1.3.

- Section 8 continued on next page -

Points Available

QUESTION 8.10

What are the Tech-Specs bases for the requirement that the reactor coolant system must have a minimum flow rate of 3000 gpm? (1.5)

ANSWER 8.10

A minimum flow rate of at least 3000 gpm provides adequate mixing, prevents stratification, and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 gpm will circulate an equivalent Reactor Coolant System volume of 9601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

(+0.5 for each underlined provision)

Reference(s)

1. Calvert Cliffs: Technical Specification Bases, Section 3/4.1.1.3 BORON DILUTION.

- End of Section 8 -

END OF EXAMINATION

Attachment 3

RO Test Section 1 D.A. Hocm

Question 1.12.6

CREDIT SHOULD BE GIVEN FOR:

↓ HEIGHT IN CORE LEVEL

-OR-

NO CHANGE FOR RV UH VOID.

Tilley

2.01 a agree

b. ANS MAY STATE 69 foot containment. This is the floor level you access to these monitors otherwise Agree

2.02 agree

2.03 agree

2.04 a. agree But VALVE NUMBERS MAY BE listed INSTEAD OF:

ANS 3 "Boric Acid storage TANK lined up TO inject Boric Acid"

MAY BE STATED "CVC-514-MOV OPEN which IS Boric Acid Direct MAKE UP VALVE And CVC-508-MOV OPEN And CVC-509-MOV OPEN which ARE THE Boric Acid gravity feed valves (see OI 2A ATTACHMENT I PAGE 25 of 35)

ANS 4 " VCT MAKE UP STOP VALVE And outlet VALVE"

MAY be STATED CVC-512-CV And CVC-501-MOV Shut see OI-2A ATTACH I PAGE 25 OF 35

ANS 5" Letdown Line LOOP ISO VALVE"

MAY STATE CVC-515-CV And CVC-516-CV see OI 2A ATT I PAGE 25 of 35

Tilley

2.04 B. disAgree ON Answer 2. You ASK what happens To The "service water system ON A SI, AS Then you LIST AN Answer From The SALT WATER system "SALTWATER outlet valve opens" agree with Ans #1 And #3

2.05 agree

2.06 agree (MAY STATE Hi TEMP BYPASS open.) 4088-CV see B10 sys Disc Fig A-3

2.07 a 1. agree MAY STATE Tech Spec VALUE ± 24 inches TABLE 3.3-4

i. agree

2.07 b agree in general MAY STATE VALVE Numbers See AOP-3A.

2.08 a disagree "you AS question FOR The "RCP MOTOR" which is cooled By CONTAINMENT AIR VIA CONTAINMENT COOLERS. YOUR Answer states the Reservoirs which cool The BEARINGS of The RCP's. YOUR Answer is not in Line with your Question

Tilley

2.08 B Agree
C Agree

2.09 a. agree
b agree

2.10 a. YOU STATE LEAKOFF LINE OR CAN BE A RECIRC LINE

b. agree (TO HPSI)

C. MAY BE AUX HPSI HEADER

D MAY BE MAIN HPSI HEADER

E. Agree OR SHUT DOWN COOLING RETURN

2.11 a agree

b. agree

C agree

RO Exam Section 3

3.01 a or b could be correct depending on assumptions if containment temp increases to normal from low then (c) would be incorrect and (d) would be most correct. Choice (c) assumes that the level and temperature you started at were those at which the instrument was calibrated.

3.02 OK

3.03 OK

3.04 Ok

3.05 a.) 1. OK
2. will mention that steam is isolated to 2nd AFW Turbine driven

b.) ok

3.06 a.) 1.) ok
2.) ok
3.) ok

b.) 1.) ok
2.) ok
3.) ok
4.) ok

3.07 a) ok

b.) For the specific protection portions I would be led to test - Low Flow fx Trip at setpoint
- Low S/G level Trip at -50"
- Low S/G pressure Trip at 685 psia

3.08 a.) 1.) ok
2.) ok

b) The purpose of the CEA Group Deviation System is to detect a CEA which is stuck or inoperable. the consequences of having a CEA Deviation is possible peaking in the core.

c.) ok

3.09 a) ok
b) ok
c) ok

3.10 ok

3.11 a.) ok
b.) ok
c.) ok

4.06 – outsurge caused initially by temperature decrease, need to add, power reduction and further temp. decrease

4.07 – question out of context, answer could talk of LAS cooling for large break

4.08 – a. add boron for power

4.11 – #13 AFW PP
EOP-2

4.12. C – could after one hour be forced to Shutdown to HSD w/in 6 HRS

f. 245°F AOP-7E

SRO Section 5 D.A. Helm

5.01 a) Candidates may use the equation

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

for this calculation and should receive full credit for this method.

5.03 a) 845 (2700) vice 2750 should be acceptable
Ref: TS 1.3 RTP is 2700 MWth

b) +1.5 vice +1.0? Last part, this means same as ambient heat losses?

5.04 a) 2.63%
−2.22%
.41% not .85% Ref: Fig 1-II.B.6

−OR−

0.5421
−0.120
0.422% Ref: Fig 1.II.B.5.e

(.41%)(80 ppm%) = 33Δppm
200 − 33 = (167 ppm)

THIS ROD INSERTION IS PROHIBITED BY ADIL CURVE. Ref: T.S. 3.1.3.6

5.06 b) Reason for decreased power

1. If the void is assumed by the candidate to be in the upper head, no change should be seen in RCP power since RCP does not pump significantly to the UH.

2. ~~A~~ Full credit should be given for decrease in power due to lower height of water in reactor vessel.

RO Section 1

~~1.12~~

1.12. b Same comments as SRO 5.06 b above.

(SRO) Section 6.0 Bruce Niestrand

6.1 a [redacted] Recent FCR Removal Runback (unit 1)

b. ok

6.2

a. ok } TC or True.
b. ok. } (how about SIAS or SGIS)

6.03

a. ok

b ok

c ok

d. ok.

6.04

a. ok

b. ok

c No Such Auto Start. (SD#15 Radmonitoring Pages 7-12)

6.05.

ok.

6.06

a ok.

b. ok

6.07

a [redacted] ok

b [redacted] ok.

c [redacted]. ok

d [redacted] Might also accept Average [redacted] A,B,C,D divided by 4.

e. ok

(ERG)

7.01 - [illegible]

7.02 - ok

7.03 - THERE ARE 15 INDICATIONS LISTED IN ESP-4
[illegible]

7.04 - TERMINATED [illegible]
c) [illegible]
[redacted]
b) [illegible] DIESEL CELL
[illegible]

7.05 - OK

7.06 - [illegible] AT <40°F [illegible] c) RCP-7B
b) RCP-1C
c) RCP-4A

c. [redacted] - TRIP IF NOT RESTORED
WITHIN ONE HOUR

7.07 - ok

7.08 - ok

7.09 - ok a & b MAINTAIN RCS SUBCOOLING [redacted] KEEP PRESSURE
OF COOLDOWN

7.10 ok 7.11 ok

8.01 – U-1 T.S. 3.1.1.1 is ≥ 3.5%

U-2 T.S 3.1.1.1 is ≥ 4.5%

8.06 include CCI 300F in answer

8.07 reference 6.7.1