DETAILS

Examination Report Nos. 50-317/88-18
50-318/88-18

Facility Docket Nos. 50-317/318

Facility License Nos. DPR-53 and DPR-69

Licensee: Baltimore Gas & Electric Co.
P.O. Box 1475
Baltimore, MD 21203

Facility: Calvert Cliffs Nuclear Power Plant
Units 1 and 2

Examination Date: June 22, 1988

Chief Examiner: David M. Silk — 8/23/88
David M. Silk
Operations Engineer Examiner
Date

Approved By: for Larry E. Briggs — 8/24/88
Peter W. Eselgroth, Chief
PWR Section, Operations Branch
Division of Reactor Safety
Date

Summary: On June 22, 1988, a written examination was administered to one Senior Reactor Operator candidate from your staff who had previously passed the operating examination and had been granted a waiver. The candidate passed the written examination. No strengths or weaknesses in the training program could be identified because only one candidate was examined. No exit meeting was held because the written examination was administered in the Regional Office.

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TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

	SRO Pass/Fail
Written Exam	1/0
Operating Exam	Waived
Overall	1/0

Attachments:

1. Written Examination and Answer Key.
2. NRC RESOLUTION OF FACILITY COMMENTS FOR THE CALVERT CLIFFS SENIOR REACTOR OPERATOR EXAMINATION ADMINISTERED ON JUNE 22, 1988.
3. BALTIMORE GAS AND ELECTRIC COMPANY LETTER DATED JUNE 28, 1988.

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

FACILITY: CALVERT CLIFFS

REACTOR TYPE: PWR-CE

DATE ADMINSTERED: 88/06/22

EXAMINER: JAGGAR, F.

CANDIDATE

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00			5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00			6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00			7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00			8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.0		Final Grade	%	Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

6. Use only the paper provided for answers.

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

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

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

10. Skip at least three lines between each answer.

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

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

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 5.01 (1.50)

a. Why is the Shutdown Margin requirement greater in modes 1-4 than in mode 5? (1.0)

b. With the plant operating at 85% power and all systems in a normal configuration, the operator borates 100 PPM. Shutdown Margin will ...

1. Increase.
2. Decrease.
3. Remains unchanged. (0.5)

QUESTION 5.02 (2.00)

a. How will individual CEA rod worth change as moderator temperature is increased? (0.5)

b. If the worth of an individual CEA rod is measured at a flux level of 10(EE-5) % power, how is the worth of the CEA affected if flux level is raised to 10(EE-4)% power AND there is NO Change in relative flux distribution? Explain. (1.0)

c. Beside temperature and neutron flux, what are TWO design factors affecting the worth of an individual CEA? (0.5)

QUESTION 5.03 (1.50)

At BOC, power is reduced from 100% to 50% and stabilized, briefly explain HOW and WHY each of the following plant parameters will be affected over the next 5 hours. Assume all systems are in automatic, rod control is in manual sequential, and no operator action is taken.

a. RCS temperature

b. RCS pressure

c. Turbine Generator control valve position

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (1.50)

During end of cycle operation with the reactor at 100% power, all rods out, boron concentration at 10 ppm in the RCS, and Tavg 10 F less than Tref.

EXPLAIN HOW and WHY reducing the power to 95% would affect ASI for each of the following means of reducing power. Consider each separately.

1. Control rod insertion.

2. Boron addition.

3. Raising Tavg.

QUESTION 5.05 (2.00)

a. If a trip from 100% power occurs with Xenon at equilibrium, what is the approximate time interval after the trip that Xenon will again be at the 100% equilibrium value? (0.5)

b. How would this approximate time interval compare if the trip occurred from 50% equilibrium conditions? (0.5)

c. When a reactor is returned to 100% power from peak Xenon conditions, why does the Xenon reactivity value trend become less than the 100% equilibrium value for a short period of time? (1.0)

QUESTION 5.06 (2.00)

TRUE or FALSE?

a. Critical rod height does NOT depend on how fast control rods are withdrawn.

b. Critical rod height dictates the reactor power level when criticality is achieved.

c. The SLOWER the approach to criticality, the LOWER the reactor power level will be when reaching criticality.

d. When reactor power is in the SOURCE range, changes in power level do NOT cause a change in Tavg.

QUESTION 5.07 (2.00)

Unit 1 has just restarted following a refueling outage while Unit 2 is near EOC. Answer the following regarding the differences in plant response between the two units (explain your answers):

a. At a steady power level of 10EE(-8) amps during a startup, equal reactivity additions are made (approximately 100 pcm). Which Unit will have the higher steady state startup rate?

b. At 50% power, a control rod (100 pcm) drops. Assuming NO AUTOMATIC or OPERATOR ACTION, which Unit will have the lower steady state Tavg?

QUESTION 5.08 (1.50)

For the changes listed below (treat each one independently) indicate whether the Moderator Temperature Coefficient will become MORE NEGATIVE, LESS NEGATIVE or have NO EFFECT. (Assume all other parameters are constant)

a. Neutron flux peak shifts radially inward from the edge of the core.

b. Boron concentration decreases 100 ppm while core is at MOC.

c. Increased number of burnable poisons are inserted into the core.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.09 (2.00)

Indicate whether each of the following will INCREASE, DECREASE, or NOT AFFECT the Departure from Nucleate Boiling Ratio (DNBR). Assume all other parameters remain constant.

a. Primary coolant temperature decreases.

b. Primary coolant pressure decreases.

c. Primary coolant flow decreases.

d. Reactor power decreases.

QUESTION 5.10 (2.00)

a. Stable natural circulation conditions exist within the RCS with the following parameters:

Thot - Tcold = 25 F
SG pressure = 885 psig
Thot Subcooled Margin indicates 40 F subcooled

Determine RCS pressure. (Show steps used to arrive at your answer).

b. What will be the temperature of the PORV discharge line if a PORV opens when there is a steam bubble in the pressurizer, quench tank pressure is 20 psig, and pressurizer pressure is:

1. 1700 psig?

2. 700 psig?

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.11 (1.50)

What effect would each of the following failures have on a natural circulation cooldown which is underway at 490 F. Explain your answers and consider each failure independently.

a. The steam dump valve which is throttled to control cooldown rate fails open.

b. Level is lost in the pressurizer.

c. The auxiliary feedwater valve to one of the SG's fails shut.

QUESTION 5.12 (1.50)

HOW does the coast-down flow provided by the reactor coolant pump flywheels affect establishing natural circulation flow conditions, and WHY?

QUESTION 5.13 (1.50)

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

a. Power decreases with constant Tave.

b. Core age increases with constant power.

c. Pressurizer pressure increases with constant power.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.14 (2.50)

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

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

b. CETs are relatively stable (not increasing) and read 585 F, and a small constant feeding and steaming rate is occuring in the steam generators. In what regions of the RCS would the following heat removal mechanisms be used?

1.) Boiling.

2.) Superheating.

3.) Condensation. (1.0)

c. Describe how auxiliary spray flow could be used to determine or confirm the presence of RCS voiding. (0.5)

(***** END OF CATEGORY 5 *****)

QUESTION 6.01 (1.75)

a. State TWO conditions that will cause ALL charging pumps (except those in PULL-TO-LOCK) to start automatically. (1.0)

b. What are the THREE mechanisms employed in the CVCS system to reduce the Reactor Coolant System activity prior to cooldown? (0.75)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.02 (1.75)

For each of the following conditions (a., b., and c.), LIST those actuation signals (1. through 6.) that should have triggered. Consider each condition separately. Each condition may have for its answer none, one, or several actuation signal(s).

1. SIAS
2. CSAS
3. CIS
4. SGIS
5. RAS
6. AFAS

a. A steam-line break has occurred in containment and

containment pressure = 7 psig
containment radiation = background
S/G levels = -145 inches and -185 inches
PZR pressure = 1800 psia
RWT level = 30 ft
S/G pressure = 680 and 580 psia

b. A small-break LOCA has occurred and

containment pressure = 2 psig
containment radiation = 6 R/hr
S/G levels = -50 inches and -50 inches
PZR pressure = 1800 psia
RWT level = 20 ft
S/G pressure = 700 and 700 psia

c. A feedwater problem has occurred and

containment pressure = 1 psig
containment radiation = background
S/G levels = -50 inches and -150 inches
PZR pressure = 2200 psia
RWT level = 30 ft
S/G pressure = 720 and 650 psia

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (2.80)

The plant is at 100% power and all controls systems are in automatic. The controlling pressurizer level channel fails low. What system responses will occur and what reactor trip signal, if any, will be generated if no operator action is performed. Setpoints are not required.

QUESTION 6.04 (1.75)

The following pertain to the excore neutron detectors and associated instrumentation for Unit 1.

a. LIST THREE control/protective functions of the wide range channels. (0.75)

b. What are TWO functions (inputs to other systems) of the control channels? (1.0)

QUESTION 6.05 (2.50)

What automatic action(s) (other than alarms) occur (if any) when the following process radiation monitors exceed their High Levl setpoints? Consider each of the three monitors separately.

a. Liquid radwaste discharge monitor. (0.5)

b. Steam generator blowdown tank radiation monitor. (0.5)

c. Blowdown Recovery radiation monitor. (1.5)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.06 (2.00)

Will the plant trip as a result of the following instrument failures? Assume no operator action. Justify your answer.

a. SUR channels A and B fail HIGH during a reactor startup when the reactor is critical at 10E-6% power.

b. SG#11 level channel A fails LOW followed by SG#12 level channel B failing HIGH while at 80% power.

c. Loop #11 Tc channel A fails HIGH followed by loop #12 Th channel B failing HIGH while at 100% power.

d. The lower UIC detector for Safety channel B fails HIGH followed by Safety channel D upper detector failing HIGH at 80% power.

QUESTION 6.07 (1.70)

The following pertain to the Shutdown Cooling System (SDC).

a. STATE the pressure setpoint that interlocks both SDC suction header isolation valves preventing them from operating. (0.2)

b. What are the functions (bases) of the TWO relief valves (2485 psig and 315 psig) on the SDC suction header? (1.5)

QUESTION 6.08 (1.00)

Indicate whether each of the following is TRUE or FALSE concerning 480 volt Motor Control Center Operation.

a. In the event that 125 VDC control power is unavailable to a 480 VAC disconnect, it can be manually closed by charging the closing spring and depressing the close bar.

b. The "Breaker Improper Lineup" alarm is actuated when a third unit disconnect switch is closed with the associated 480 VAC circuit breaker racked out.

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (2.50)

a. State FOUR regulating control rod interlocks/limits that are in effect when the system is in MANUAL INDIVIDUAL control? (1.2)

b. WHAT are the TWO instrumentation signals/conditions that could provide a DROPPED ROD annunciator? (0.8)

c. If a loss of 125 VDC control power to the trip circuit breakers (TCBs) occurs, what component, if any, will ensure that the TCBs open if a trip signal is generated? (0.5)

QUESTION 6.10 (2.50)

What effect (INCREASE, DECREASE, NO EFFECT) will the following events have on the Thermal Margin / Low Pressure Trip SETPOINT. Consider each separately. Assume the plant is at 100% power.

a. Tcold Loop 1 fails LOW
b. A RCP trips
c. RCS pressure increases 25 psig
d. A Linear Power Range Channel (Safety) fails high
e. Delta-T PWR Calibrate pot is reduced

QUESTION 6.11 (1.75)

a. Assume that an inadvertant trip of the AFAS "A" channel is active for a period of only 15 seconds and then clears. Briefly describe how the AFAS signal is reset and the associated equipment returned to a normal configuration. (1.5)

b. How much time must have elapsed before the AFAS signal will seal in? (0.25)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

QUESTION 6.12 (3.00)

The following questions pertain to a loss of instrument air assuming normal, at power, initial conditions.

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

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

1. Main Feedwater Regulating Valves
2. Pressurizer spray valves
3. Letdown
4. Atmospheric Dump Valves
5. AFW regulating valves
6. EDG service water supply valves
7. Auxillary Spray valve
8. Turbine AFW pump speed (if operating) (2.0)

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

(***** END OF CATEGORY 6 *****)

QUESTION 7.01 (1.50)

For each operation listed, what is the manual mode of control NORMALLY used on the Control Element Drive System?

a. Withdrawing Shutdown Group CEA's during a reactor startup.

b. Inserting Regulating Group CEA's during reactor shutdown.

c. Recovering a droppped CEA.

QUESTION 7.02 (2.00)

The following questions pertain to information found in EOP-6, Steam Generator Tube Rupture.

a. State FOUR of the FIVE trends that can be used to identify the affected Steam Generator.

b. List the FOUR RMS monitors that can be used to verify that a Steam Generator Tube Rupture has occurred.

QUESTION 7.03 (2.50)

The following questions pertain to information found in OP-5, Plant Shutdown from HSB to Cold Shutdown.

a. Over a two hour period, the RCS steadily cooled down from 280 F to 230 F. Explain whether or not a cooldown limit has been exceeded. (1.5)

b. What is the minimum pressure for the RCS prior to blocking SIAS? (0.5)

c. What is the minimum allowable pressure in the Steam Generator prior to blocking SGIS? (0.5)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (2.50)

According to EOP-5, Loss of Coolant Accident:

a. State FOUR of the SIX Entry Conditions. (1.0)

b. State THREE of the FOUR positive indications of Core and/or RCS voiding. (0.75)

c. What are THREE of the FOUR conditions/parameters the operator must confirm/satisfy prior to stopping or throttling a safety injection train? (0.75)

QUESTION 7.05 (2.00)

a. Given a LOCA condition, how soon after SIAS actuation must "core flush" commence? (0.25)

b. State the TWO methods that are available for "core flush". Include a brief description of flow paths for each method. (1.5)

c. What is the reason that "core flush" is established? (0.25)

QUESTION 7.06 (2.00)

The following questions concern RCP Trip Strategy as stated in the EOPs:

State the number of RCP's that should be STOPPED given the following conditions. Consider each situation separately.

a. Small break LOCA with RCS pressure decreased to 1775 psia.

b. Small break LOCA with RCS pressure decreased to 1800 psia and a CIS actuated.

c. Total Loss of Feedwater event with RCS pressure decreased to 1800 psia.

d. Large break LOCA with RCS pressure decreased to 1400 psia.

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.07 (1.50)

The following pertain to information found in the Calvert Cliffs Technical Specifications.

a. State ALL dose rates that apply to designating an area as a High Radiation Area? (0.5)

b. How is entry controlled AND dose rates monitored for an area with a dose rate of 1500 mr/hr? (1.0)

QUESTION 7.08 (2.00)

The Functional Recovery Procedure, EOP 8, provides guidance for reactivity control. If CEAs cannot be inserted or driven in, reactor shutdown may be accomplished by boration.

a. List the TWO flow paths for charging water to the RCS.

b. List the TWO flow paths for supplying boric acid to the charging pumps.

QUESTION 7.09 (2.50)

The following pertain to information contained in EOP-4, Excess Steam Demand.

a. What THREE parameters are used to identify the affected S/G? (0.75)

b. State SIX of the EIGHT actions taken to isolate the S/G's. (1.5)

c. Why is it necessary to maximize RCS boration during the initial stages of an Excess Steam Demand event? (0.25)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (2.00)

Concerning information found in AOP-1A, Inadvertant Boron Dilution:

a. State FIVE of the SEVEN indications/alarms of a "Dilution While Critical". (1.0)

b. TRUE or FALSE?

Boration and CEA movement may be used CONCURRENTLY to maintain reactor power level constant. (0.5)

c. If CEA's have reached insertion limits as specified in Technical Specifications, what time restrictions are placed on boron addition? (0.5)

QUESTION 7.11 (1.50)

The following pertain to information found in AOP-1B, CEA Malfunctions.

a. How is programmed Tc maintained following a dropped CEA?

b. Based on Technical Specifications Fig. 3.1-3, Allowable Time to Realign CEA's...., what is the MAXIMUM allowable time to realign a dropped CEA?

c. During CEA realignment, what method is used to compensate for reactivity changes?

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.12 (2.00)

The following pertain to information found in OP-1A.

a. State the limit for each of the listed parameters where the Reactor Coolant Pump (RCP) must be stopped.

--Lower seal temperature.
--Motor thrust bearing temperature. (0.8)

b. What is the maximum time a RCP be run following a loss of CCW? (0.4)

c. Given the following RCP seal cavity pressures, state which seal (UPPER,MIDDLE, or LOWER) has probably failed. Consider each of the cases separately.

1.) Upper -- 50 psia
Middle -- 1400 psia

2.) Upper -- 700 psia
Middle -- 1975 psia (0.8)

QUESTION 7.13 (1.00)

In order to maintain the plant at 100% power, work must be performed inside the containment in a radiation field of 100 MREM/HR gamma and 100 MRAD/HR fast neutron. You will be accompanying the maintenance man into the work area. Assume you are 27 years old and have a lifetime exposure through last quarter of 29 REM on your NRC Form 4. So far this quarter, you have accumulated 1.0 REM.

How long may you stay in this area without exceeding the Calvert Cliffs administrative limits? Show all assumptions and calculations.

(***** END OF CATEGORY 7 *****)

QUESTION 8.01 (2.50)

a. Explain why the following leakages are IN or OUT of compliance with the Limiting Condition for Operation. Consider each INDIVIDUALLY and SEPARATELY.

1. Unidentified--0.65 GPM

2. Steam Generator Tube Leakage:

A--0.45 GPM
B--0.50 GPM

3. Variou[illegible]anual vent and drain valves seat and pack[illegible]gland leakage.-----------------------4.6 GPM

4. Back leakage through SI check valves detected by leak check on previous shift.-----------------3.9 GPM (1.0)

b. Evaluate Technical Specification compliance if the leakage in "a." is cumulative. (0.75)

c. Calvert Cliffs Technical Specifications require that systems be operable for detecting leakage from the RCS.

List the 3 systems or methods used to detect leakage to the RCS. (0.75)

QUESTION 8.02 (1.50)

State the approval requirements, by job title/position, for changes to the following types of procedures according to CCI-101J.

a. Emergency Operating Procedures.

b. Operating Instructions.

c. Nuclear Engineering Operating Guide.

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (3.00)

a. What combination of THREE parameters shall not exceed the limits of Fig 2.1-1 located in the Calvert Cliffs Technical Specifications to prevent exceeding the Reactor Core Safety Limit. (1.5)

b. What is the objective or basis for the Technical Specification Reactor Core Safety Limit? (0.5)

c. Assuming Mode 1 operation, state the action required by the Technical Specifications to be taken within one hour if the Reactor Core Safety Limit is exceeded. (1.0)

QUESTION 8.04 (2.00)

Unit 2 is in the process of a plant startup, reactor power is about 2%. You note that the loop temperatures are all 500 F. In accordance with the Unit 1 Technical Specifications:

a. What actions must be taken?

b. What are the FOUR bases for the Minimum Temperature for Criticality?

QUESTION 8.05 (2.50)

Answer the following in accordance with Calvert Cliffs Technical Specifications and procedure CCI-140D "Shift Staffing". Assume both units in Mode 1.

a. What actions must be taken if operating with a minimum crew composition and one of the Reactor Operators becomes incapacitated? (1.5)

b. Is the following action proper by the Shift Supervisor? Justify your answer.

Fifteen minutes before scheduled arrival of the on-coming shift, one of the four on-coming RO's calls in sick. The on-shift SS decides, due to the shift's present overtime condition, to call in a replacement. The SS also decides that since the replacement should arrive shortly after shift change (about 30 minutes) to send his people home and let the next shift start with three ROs. (1.0)

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

What are the TWO reasons identified by Technical Specifications for limiting the blow down of one steam generator in the event of a Main Steam Line Rupture?

QUESTION 8.07 (1.50)

Commensurate with Calvert Cliffs Technical Specifications, Primary Containment Integrity must be established in operational MODES 1,2,3, and 4.

What plant conditions constitute CONTAINMENT INTEGRITY ?

QUESTION 8.08 (2.00)

Unit 1 has been in Mode 6 for 30 days when the Refueling Pool is drained to facilitate installation of the reactor vessel head. After the pool has been drained, 11 LPSI pump develops a breaker problem and must be removed from service.

a. What actions must be taken AND in what time period according to Technical Specification 3.9.8.2?

b. What alternate action may be taken to comply with Technical Specifications?

QUESTION 8.09 (3.00)

a. Under what THREE conditions may the independent verification of safety tags be waived?

b. Wh[illegible] TWO people may authorize a waiver of the independent [illegible]fication of safety tags? Indicate people by title.

c. [illegible] multiple component tagout, when are the control board [illegible] to be tagged/untagged?

QUESTION 8.10 (2.00)

Given each of the events described below, along with "EAL Criteria", pages 14 and 15 of the EPIP, (included with the examination) classify each event separately to the correct EAL. Note that some events may not fall into any EAL. State any assumptions used in making a classification.

a. While operating at 100% power, steady state, a bomb threat is received stating that an explosive device has been placed in the control room. Upon conducting a search of the control room, what appears to be such a device is discovered. Examination of the device indicates that it may be triggered by radio control or movement. (0.5)

b. While moving fuel in the core, during refueling outage, all off-site power is lost due to line damage from high winds. After discussion with power dispatcher, offsite power will be restored in 20-24 hours. Onsite power systems operate according to design, with adequate fuel oil for at least 120 hours of operation. (0.5)

c. While operating at 100% power, steady state, a double ended rupture of one RCS cold leg occurs. (0.5)

d. While operating at 100% power, steady state, a main steam line break occurs downstream of containment but upstream of the main steam isolation valve. Subsequently, calculated steam generator tube leakage increases from 0.2 gpm at steady state to 40 gpm with no indication of secondary activity in the intact loop. (0.5)

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.11 (3.00)

For each of the following events explain briefly why the NRC SHOULD or SHOULD NOT be notified within 1 hr according to CCI-118-5.

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

b. While critical at 1% power on Unit 2, Tave drops below 515 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)

(***** END OF CATEGORY 8 *****)
(********** END OF EXAMINATION **********)

ANSWER 5.01 (1.50)

a. In mode 5 the plant is completely cooled down (below 200 deg. F) so that there is no need to have reserve shutdown margin to offset the positive reactivity addition resulting from a rapid cooldown accident while in modes 1-4. (1.0)

b. 1 (Increase) (0.5)

REFERENCE

RO-302-3-1; Calvert Cliffs: T. S. p 1-3
EO 8.5
3.5/3.9

004000K519 191004K107 191004K112 ..(KA's)

ANSWER 5.02 (2.00)

a. Worth will increase [0.5] with an increase in moderator temperature.

b. No change. [0.5] The absolute value of neutron flux will not change the worth. A shift in flux distribution is required. [0.5] (1.0)

c. CEA size, CEA absorber material, CEA location. [any 2, 0.25 ea)(0.5)

REFERENCE

CE Trng Ctr Rx Theory Notes, Pp. 181-183
RO-302-2-1
EO 4.7, 4.9
2.9/3.4 2.5/2.8

192005K107 001000K502 193003K125 ..(KA's)

ANSWER 5.03 (1.50)

a. Decreases (~15 F) [0.25] due to buildup of Xe [0.25]
b. Held constant [0.25] by PPCS spray and heaters [0.25]
c. Increases [0.25] due to lower S/G pressure [0.25]

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

Calvert Cliffs; SD #5, RCS, p 25
SD #23-1, Turbine Control and Protection System, p 54
RO-302-3-1 pp.75
EO 7.14
RO-62-1-1
LO 06201K602
3.5/3.6 3.2/3.5 3.6/4.1 3.6/3.6

039000K508 002000K510 001000K533 192008K124 193008K105 ..(KA's)

ANSWER 5.04 (1.50)

1. Adding negative reactivity to the top of the core causes the ASI to become more positive (power is driven to the bottom of the core).

2. A change in Th as power decreases is greater than the change in Tc. With a -MTC, less negative reactivity is inserted in the top of the core than the bottom due to positive reactivity feedback. Also, the MTC is more negative at the temperatures at the top of the core. ASI becomes more negative.

3. As SG steam flow is decreased, Tc increases. The hotter Tc entering the core reduces reactor power. At the lower reactor power there is a smaller delta T across the core. The net result is that Tc increases more than Th decreases. ~~But~~ MTC is more negative at the top of the core than at the bottom so the effects are ~~approximately off-setting~~ that power shifts to the top of the core and ASI ~~does not significantly change.~~ becomes more negative

[0.5 ea.]

REFERENCE

RO-302-4-0 pp. 5, 6, 7
EO 1.1, 2.2, 2.3
3.3/3.7 2.8/3.2

192005K111 015020K503 193008K122 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.05 (2.00)

a. ~ 24 Hrs. (accept 20 - 30 hrs.) (0.5)

b. Time would be shorter. (< 20 hrs.) (0.5)

c. Due to the time delay in Xenon production from the decay of Iodine. (1.0) (Will accept burnout as a correct response).

REFERENCE

NEOG 7 Fig. 1-II.D.4.a
RO-302-2-1 pp. 48,49
EO 12.6
3.1/3.1 3.4/3.4 3.4/3.4

192006K107 192006K106 192006K105 ..(KA's)

ANSWER 5.06 (2.00)

a. T
b. F
c. F
d. T [0 5 ea.]

REFERENCE

RO-302-3-1 p. 59
EO 7.9
3.8/3.9

192008K105 ..(KA's)

ANSWER 5.07 (2.00)

a. Unit 2 [0.5] due to a lower Beta effective coefficient at EOC [0.5]

b. Unit 1 [0.5] due to MTC being less negative, (so Tavg must decrease more to add + reactivity). [0.5]

REFERENCE

RO-302-3-1 pp. 65,66
EO 7.11
3.8/3.9 2.9/3.4 3.9/4.1

001000K510 001000K549 192008K120 ..(KA's)

ANSWER 5.08 (1.50)

a. Less
b. More
c. More (If assume same Boron conc., will accept no effect) [0.5 ea]

REFERENCE

RO -302-2-1 pp. 20-26
EO ᴗ.7
3.1/3.1 3.3/3.6

001000K526 192004K106 ..(KA's)

ANSWER 5.09 (2.00)

a. Increase

b. Decrease

c. Decrease

d. Increase [0.5 ea.]

REFERENCE

RO-301-13-0 pp. 26-28
EO 6.3
3.4/3.6

193008K105 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.10 (2.00)

a. Tcold = 503 F corresponding to saturation temperature for 700 psia [0.25]

Thot = Tcold + 25 = 528 F [0.25]

40 F subcooled = 528 + 40 = 568 F [0.25]

568 F corresponds to 1207.72 psia [0.25]

b. 1. ~~225~~ 260 F (~~220~~ 255 - ~~230~~ 265)
2. ~~410~~ 330 F (~~400~~ 320 - ~~420~~ 350) [0.5 ea.]

REFERENCE

Steam Tables
RO-301-8-0 p.4
TO 8.2
3.3/3.4 2.8/3.1 3.6/3.8

193003K115 193003K124 193003K125 ..(KA's)

ANSWER 5.11 (1.50)

a. Increase cooldown rate [0.2] since more energy is being removed from the primary. [0.3]

b. May interrupt natural circulation [0.2] since hot legs maybe voided. [0.3] OR No effect [0.2] if hot legs do not become voided [0.3]

c. Decrease cooldown rate [0.2] since SG tubes will become uncovered reducing heat removal. [0.3] (Will accept: No longer feeding with cold water).

REFERENCE

RO-301-14-0 (CAF)
EO 2.3
3.9/4.1

193008K123 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.12 (1.50)

Coast-down delays natural circulation [0.75] because it takes longer to establish a significant core differential temperature [0.75].

REFERENCE

RO-301-14-0 p. 5
EO 2.6
3.9/4.2

193008K121 ..(KA's)

ANSWER 5.13 (1.50)

a. Decrease (0.25), smaller delta T required to transfer more energy from RCS (0.25).

b. Decrease (0.25), fuel swelling and clad creep reduce clad gap which reduces delta T across gap and lowers center line temp (0.25).

c. No change (0.25), pressure has little effect on heat transfer in subcooled fluids (0.25). Accept increase if the assumption is stated that increasing pressure decreases nucleate boiling.

REFERENCE

RO-301-13-0 p. 6
EO 13.4, 13.3
RO-301-12-0 pp. 6,7
2.2*/2.4* 2.3*/2.4 2.5/2.5

193007K101 193008K130 193008K128 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

ANSWER 5.14 (2.50)

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

b. Boiling in the covered portion of the core [0.33]

Superheating in the uncovered portion of the core [0.33]

Condensation in the S/G U-tubes (reflux boiling) [0.33]

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

REFERENCE

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

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

c. EOP-5 Rev 0 pg. 17
EO 14.2.1; 14.2.7
4.0/4.6 4.0/4.4 4.5/4.9

000074K103 000074K311 000074A206 ..(KA's)

(***** END OF CATEGORY 5 *****)

ANSWER 6.01 (1.75)

a. The pumps will automatically start upon:

1. Pressurizer level deviation = -15 inches
2. Receipt of an SIAS [0.5 ea.]

b. 1. Degassification
2. Filtration
3. Demineralization (De-ionization) [0.25 ea.] (0.75)

REFERENCE

SD 6, Pp. 29, 20-22
Lesson Objectives 00603K505
00603K601
00604K501
2.9/3.8 3.0/3.9 3.6/4.0 3.8/4.0

004000K115 004000K101 004010A202 004000A101 ..(KA's)

ANSWER 6.02 (1.75)

a. SIAS, CSAS, CIS, SGIS [0.25 ea.]

b. CIS, SGIS [0.25 ea.]

c. None [0.25]

REFERENCE

SD-63 pp. 45-60 RO-34-1-1 pg. 7
Lesson Objectives - unavailable
3.8/3.9 4.2/4.4

013000K101 012000A301 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.03 (2.80)

PZR heaters deenergize
Letdown flow control valves close to minimum
Both backup pumps start
Auto makeup to VCT initiates (due to charging/letdown mismatch)
Sprays initiate (to reduce pressure from compressing the vapor space)
Reactor trips on high PZR pressure
PORVs open [0.4 pts each]

REFERENCE

RO-62-1-1 p. 23
SD-62 Fig. 62-8, Fig. 62-11
Lesson Objectives 06202K404
06202K405

3.6/3.9 3.1/3.8 3.7/4.0 3.8/3.9 3.2/3.4 3.3/3.7

011000K401 011000K301 011000K104 011000K103 011000K102
011000K101 ..(KA's)

ANSWER 6.04 (1.75)

a. SUR trip to the RPS
SUR enable
Zero mode bypass [0.25 ea]
b. Input to PRCs
Input to Internal Vibration Monitoring System. [0.5 ea]

REFERENCE

RO-57-1-2
SD-57 pp. 47-50, Fig. 57-14, pg. 52
Lesson Objectives - none available
4.1/4.2 3.7/3.9 3.1/3.5*

015000A202 015000A302 015000K101 ..(KA's)

ANSWER 6.05 (2.50)

a. The liquid discharge isolation valves are closed. [0.5]

b. Surface and bottom blowdown isolation valves are closed. [0.5]

c. Diverts blowdown recovery flow to the MWS.
Closes blowdown isolation valve to Circ. water.
Closes blowdown isolation valve to the condenser. [0.5 ea]

REFERENCE

SD-14B p. 6
SD-18 pp. 17, 27
RO-122-1-1
EO 1.3.1
4.0/4.3

073000K401 ..(KA's)

ANSWER 6.06 (2.00)

a. No trip [0.25]
The trip will not occur until power reaches 10E-4% power [0.25]

b. No trip [0.25]
Only the auctioneered LOW signal is selected, therefore only channel A will trip [0.25]

c. TRIP [0.25]
Due to Delta T decreasing and causing setpoint to change [0.25]

d. TRIP [0.25]
Both APD channels will trip [0.25]

REFERENCE

SD No 59, pgs 10-31
Lesson Objectives 05904K501
3.9/4.3 3.1/3.5

012000K603 012000K403 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.07 (1.70)

a. >/= 300 psia. [0.2]

b. 2485 psig - Protect piping from overpressure due to a sudden temperature increase in containment. [0.75]
315 psig - Protect piping from overpressure due to simultaneous operation of charging pumps and SDC with the pressurizer solid. [0.75]

REFERENCE

SD-7 pp. 42,43,47
Learning Objectives - not available
3.2/3.5* 3.2/3.5 3.8/4.1 2.9/3.1

005000K104 006000K409 005000K407 005000K402 ..(KA's)

ANSWER 6.08 (1.00)

a. true (0.5)

b. true (0.5)

REFERENCE

SD-53 pp. 19, 44
3.2/3.3

062000G009 ..(KA's)

ANSWER 6.09 (2.50)

a. Upper electrical limit
Lower electrical limit
CWP CEA withdrawal prohibit
CMI CEA motion inhibit [0.3 pts each]

b. 1. Rod drop from Reed switch
2. NI negative rate of power change from NI system. [0.4 pts each]

c. UV trip devices [0.5]

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

REFERENCE

SD-60 pp. 34, 39, FIg. 60-18
SD-57 p. 43 CO5 AA-24,AB-24
Learning Objectives RO-60-3-0 0600201
0600103
3.4/3.6 4.5/4.4 3.7/3.8 2.4/2.8 2.9/3.1

063000K201 001000K604 001000K407 001000K105 001000K103 ..(KA's)

ANSWER 6.10 (2.50)

a. No effect
b. Decrease (No Effect if only considering the change in flow)
c. No effect
d. Increase
e. No effect [0.5 pts each]

REFERENCE

SD-59 p. 12

3.9/4.3 3.3/3.8 2.9/2.9

012000K611 012000K501 012000K402 ..(KA's)

ANSWER 6.11 (1.75)

a. The AFAS signal will reset automatically.[0.5] MS-4071 will close when the signal clears, [0.5] the AFW pump must be manually stopped. [0.5]

b. The signal will seal-in after 18 seconds. [0.25]

REFERENCE

RO-34-1-1 pp. 7-9
Enabling Objective 1.02
4.1/4.4 3.7*/4.0 4.3*/4.5*

013000K404 013000A206 013000K107 ..(KA's)

(***** CATEGORY 6 CONTINUED ON NEXT PAGE *****)

ANSWER 6.12 (3.00)

a. 1. C
2. D
3. D [0.25 each]
4. D
5. D
6. A
7. D
8. A

b. 1. Auto valve to plant air system X-ties PA to IA at 85 psig IA pressure [0.5]

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

3. X-connect Units 1 and 2 plant air systems [0.5]

[any two for 0.5 each]

REFERENCE

S.D. 32 pg 15
AOP-7D pp. 1,3-6
S.D. 41 Fig A-7 to A-9
S.D. 39 pg 21
3.4/3.6 3.2/3.5

078000K402 078000K302 ..(KA's)

(***** END OF CATEGORY 6 *****)

ANSWER 7.01 (1.50)

a. Manual Group
b. Manual Sequential
c. Manual Individual [0.5 ea.]

REFERENCE

OP-2 p.11
OP-4 p. 6
AOP-1B
4.0/3.7 3.5/3.4

00000SG006 001000A403 ..(KA's)

ANSWER 7.02 (2.00)

a. 1. Mismatch in feed flow prior to the trip.
2. Unexplained increase in S/G level prior to the trip.
3. Main Steam Line RMS.
4. Post-trip S/G level changes.
5. S/G samples. [any 4 @ 0.25 ea.]

b. 1. Condenser off-gas RMS.
2. S/G Blowdown RMS.
3. MainSteam Line RMS.
4. Main Vent RMS. [4 @ 0.25 ea.]

REFERENCE

EOP-6 pp. 3, 10
4.4/4.6 4.5/4.8 3.9/4.2*

000038A204 000038A202 000038A203 ..(KA's)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

ANSWER 7.03 (2.50)

a. No [0.5]. Even though the RCS was cooling down at a rate of 25 F/hr, while below 250 F the RCS temperature did not exceed the 20 F/hr cool-down limit. [1.0]

b. 1760 psia. [0.5]

c. 720 psia. [0.5]

REFERENCE

OP-5 pp. 1, 4
3.4/3.9

002000SG10 ..(KA's)

ANSWER 7.04 (2.50)

a.
1. Unexplained decreasing presurizer level.
2. Unexplained decreasing pressurizer pressure.
3. Loss of RCS subcooled margin.
4. High Containment radiation alarm.
5. Increase in containment sump level.
6. Increase in containment sump alarm frequency.

[any 4 at 0.25 ea.]

b.
1. Letdown flow greater than charging flow.
2. Rapid increase in pressurizer level during an RCS pressure reduction.
3. Loss of subcooled margin as determined using CET temperatures.
4. "REACTOR VESSEL WATER LEVEL LOW" alarm.

[any 3 at 0.25 ea.]

c.
1. RCS subcooling > or = 30 F.
2. PZR level > 155 in., and stable.
3. At least 1 S/G available for heat removal
4. RVLMS indicates core covered. [any 3, 0.25 ea.]

REFERENCE

EOP-5 pp. 21, 11, 4
LO 201GA09
3.9/4.3 3.4/3.6 4.3*/4.5* 3.7*/3.5

000009A104 000011SG11 000009K310 000011A211 ..(KA's)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

ANSWER 7.05 (2.00)

a. 24 hours. (0.25)

b. Pressurizer injection [0.25]-- HPSI pump discharge to CVCS thru Auxiliary HPSI header to Aux. Spray to RCS thru surge line (loop 2 Th). [0.5]

Hot Leg Injection [0.25]-- LPSI pump from Containment Sump thru recirculation line and SDC return header into RCS loop 2 Th. [0.5]

c. Prevent restriction of core flow due to boron crystallization. (0.25)

REFERENCE

SD-7 pp.67, 68; EOP-5 p. 28
LO 201EK312
3.8/4.2

000011K313 ..(KA's)

ANSWER 7.06 (2.00)

a. None
b. All (4)
c. All (4)
d. Two [0.5 ea.]

REFERENCE

EOP-0 p. 5
LO201EK306
3.4/3.7 4.1/4.2

000011K314 000009K313 ..(KA's)

ANSWER 7.07 (1.50)

a. >100 mr/hr <1000 mr/hr >1000 mr/hr [0.25 ea]

b. SWP issued.
Radiation monitoring device indicating dose rate provided.
Locked barricades with keys maintained by Shift Supervisor or Supervisor-Radiation Control. [0.33 ea]

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

REFERENCE

CC Tech. Specs pp. 6-20, 6-21
2.8/3.4

194001K103 ..(KA's)

ANSWER 7.08 (2.00)

a. 1. Normal CVCS lineup.
   2. Charging thru the Aux. HPSI header. [0.5 ea]

b. 1. BAST-- via gravity feed and boric acid pumps.
   2. RWT-- to charging pump suction. [0.5 ea.]

REFERENCE

EOP-8 pp. 6-7
4.2/4.3 3.4/4.7 3.4/4.7

004000K123 004000K122 000029K311 ..(KA's)

ANSWER 7.09 (2.50)

a. S/G pressures
   T cold
   S/G Levels. [0.25 ea.]

b. 1. Shut MSIV's.
   2. Shut Feed Water isolation valves.
   3. Shut MSIV bypass valve's.
   4. Shut Blowdown valves.
   5. Shut AFW Steam supply valves.
   6. Shut AFW block valves.
   7. Shut Atmospheric dump valves.
   8. Shut upstream drains. [any 6 @ 0.25 each]

c. The cooldown adds positive reactivity and the boron adddition prevents a return to criticality. [0.25]

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

REFERENCE

EOP-4 pp.5, 6, 9
4.5/4.7 4.3/4.3 4.1/4.4

000040K105 000040A104 000040K304 ..(KA's)

ANSWER 7.10 (2.00)

a. 1. Increasing power level on Nuclear Instruments.
   2. Increasing Tc.
   3. Decreasing boron concentration indication.
   4. Demin. water flow alarm.
   5. BA flow alarm.
   6. Low BA pump discharge pressure alarm.
   7. Automatic withdrawal prohibit alarm.

[5 at 0.2 ea.]

b. TRUE [0.5]

c. Borate in 5 sec. increments. [0.5]

REFERENCE

AOP-1A p. 6
LO 00603003; 00603K401
3.8/3.9 3.8/4.3

004020A206 004000SG15 ..(KA's)

ANSWER 7.11 (1.50)

a. With turbine load.

b. 60 min.

c. Regulating CEA motion. [0.5ea.]

REFERENCE

AOP-1B pp. 5-6
T.S. Fig. 3.1-3
3.8/4.1

000003K304 ..(KA's)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

ANSWER 7.12 (2.00)

a. 200 F
   195 F [0.4 ea]

b. 10 min. [0.4]

c. 1.) UPPER
   2.) LOWER [0.4 ea.]

REFERENCE

RO-106-1-1 pp. 15-17
OP-1A p. 1
LO 00501K704
   00503K702
3.5/3.9 3.7/3.9

003000A202 003000A201 ..(KA's)

ANSWER 7.13 (1.00)

5(N-18) = 45 REM [0.25]
Total lifetime to date = 29 + 1 = 30 REM [0.25]
Total lifetime available = 45 - 30 = 15 REM
Total this quarter available = 2 - 1 = 1 REM
Quarterly limit is more restrictive than annual limit
gamma + neutron = total
0.1 REM/HR + (0.1 RAD/HR)(10 QF) = 1.1 REM/HR dose rate [0.25]
1.0 R/1.1 REM/HR = 0.91 HRS = 53 MIN [0.25]

REFERENCE

CCI-800B p. 9
3.1/3.4 ~~2.8/3.4~~ 3.3/3.5
194001K103 194001K10~~5~~4 ..(KA's)

(***** END OF CATEGORY 7 *****)

ANSWER 8.01 (2.50)

a. 1. In compliance, unidentified limit is 1 GPM. [0.25]
   2. In compliance, Total leakage less than 1.0 GPM [0.25]
   3. In compliance, identified leakage spec. is 10 GPM [0.25]
   4. In compliance, identified leakage spec. is 10 GPM [0.25]

b. Not in compliance [0.5] as total leakage is 10.1 GPM. [0.25] (0.75)

c. 1. Containment atmosphere pariculate.
   2. Containment sump level alarm.
   3. Containment atmosphere gaseous. [0.25 ea.]

REFERENCE

CC TS 3/4.4.6
2.6/3.8 3.6/3.8

002020K401 002000SG06 ..(KA's)

ANSWER 8.02 (1.50)

a. Two SRO license holders, one of whom must be the Shift Supervisor or Supervisor- Procedural Developement.

b. Same as a. above.

c. Two members of the plant management staff, one of which holds an SRO license on the affected unit.
[0.5 ea.]

REFERENCE

CCI-101J pg. 4
3.3/3.4

194001A101 ..(KA's)

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

ANSWER 8.03 (3.00)

a. Thermal Power, Pressurizer Pressure, highest loop Tc temperature. [0.5 ea.]

b. Maintain integrity of fuel cladding --OR--

To prevent the release of significant amounts of fission products to the primary coolant --OR--

To maintain the DNBR greater than 1.21 [0.5]

c. Be in HOT STANDBY; notify NRC Operations Center. [0.5 ea.]

REFERENCE

CC Technical Specifications pp. 2-1, B2-1, 6-13
3.6/4.1 2.6/3.8

002020G006 002020G005 ..(KA's)

ANSWER 8.04 (2.00)

a. ~~1.~~ Restore Tavg to > 515F [~~0.25~~ 0.5] within 15 minutes [~~0.25~~ 0.5] OR
~~2.~~ ~~or~~ be in Hot Standby [~~0.25~~ 0.5] within the next 15 minutes [~~0.25~~ 0.5]

b. 1. Ensure MTC is within analyzed range. [0.25]
2. Protective instrumentation is within normal operating range. [0.25]
3. Pressurizer is capable of being operable with steam bubble. [0.25]
4. Reactor vessel is above minimum RT-NDT temperatrue [0.25]

REFERENCE

CC TS U-2 pgs 3/4 1-7, B3/4 1-2
3.3/4.0 3.6/4.1
002000G005 002000G011 ..(KA's)

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

ANSWER 8.05 (2.50)

a. Obtain a qualified relief [0.75] within two hours [0.75]

b. Yes [0.50] the number of RO qualified personnel present would still meet TS requirements [0.50]

REFERENCE

CC TS, pg 6-5
CCI-140D pg 1
2.5/3.4

194001A103 ..(KA's)

ANSWER 8.06 (2.00)

a. Minimize the positive reactivity effects of the RCS cooldown associated with the blowdown. (1.0)

b. Limit the pressure rise w/in the containment in the event that the rupture occurs inside the containment. (1.0)

REFERENCE

CC Technical Specifications B 3/4 7-3
3.7/3.7

039000K405 ..(KA's)

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

ANSWER 8.07 (1.50)

Containment Integrity shall exist when:

(a) All penetrations required to be closed during accident conditions are either:
(1) Capable of being closed by an operable containment automatic isolation valve system, or (0.25)

(2) Closed by manual valves, blind flanges, or deactivated automatic control valves secured in their closed positions or as specified in the specifications. (0.25)

(b) All equipment hatches are closed and sealed. (0.25)

(c) Each air lock is operable (0.25)

(d) Containment leakage rated are w/in limits. (0.25)

(e) The sealing mechanism associated with each penetration is operable. (0.25)

REFERENCE

CC Technical Specifications, 1-2
3.1/3.9 3.4/4.1*

000069G008 103000G011 ..(KA's)

ANSWER 8.08 (2.00)

a. Initiate corrective action to return 11 LPSI pump to operable status within one hour.

One Spent Fuel Pool Cooling loop may be lined up to provide cooling flow.

[1.0 ea.]

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

REFERENCE

CC TS 3.9.8.2
LER 87-001EO 11
2.7/3.6

005000SG06 ..(KA's)

ANSWER 8.09 (3.00)

a. Emergency [0.4]
High radiation or hazards area [0.3]
Operationally tested and not a locked valve [0.3]
Not Safety Related or affects plant reliability

b. Shift Supervisor [0.5]
~~Safety Tagging Supervisor~~ [0.5]
Senior Control Room Op[illegible]

c. Tagged first [0.5]
Untagged last [0.5]

REFERENCE

Calvert Cliffs: CCI-112, L.O. 1.a, 1.b, 2
3.7/4.1

194001K102 ..(KA's)

ANSWER 8.10 (2.00)

a. Alert (CR evacuation may be anticipated) (0.5)

b. None (0.5)

c. S. A. E. (Failure of 2 FPB; G.A.E. if explain challenge to 3rd). (0.5)

d. S. A. E. (Failure of 2 FPB; MSLB w/ SGTR & uncontrolled release). (0.5)

REFERENCE

CC EPIP Sect. 3.0 pp. 14-15
3.4/4.4*

194001A116 ..(KA's)

ANSWER 8.11 (3.00)

a. Should report [0.35] plant is operated outside design basis [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 Instructions 113-5, p1
T.S. Unit 2, p 3/4 1-7
p 3/4 1-16
4.1*/3.9
194001A102 ..(KA's)

(***** END OF CATEGORY 8 *****)
(********** END OF EXAMINATION **********)

NRC RESOLUTION OF FACILITY COMMENTS FOR THE
CALVERT CLIFFS SENIOR REACTOR OPERATOR EXAMINATION
ADMINISTERED ON JUNE 22, 1988

Question Number: 5.04 3.

Facility Comment: 3) Raising Tavg.

The analysis of temperature change across the core is correct. However, the result would be in an increased negative ASI, power moving toward the top, as a result of the temperature change. Tc increasing adds negative reactivity to the lower half of the core. Th decreasing adds positive reactivity to the top of the core. Therefore power shifts to the top of the core. The more negative MTC in the upper core aids this process causing a larger power (ASI) swing.

Reference: Same as stated.

NRC Resolution: Agree with comment.

Reference stated supports facility comment and does not fully support answer key. Answer key is changed to reflect facility comment.

Question Number: 5.07

Facility Comment: The question used units of PCMS in lieu of % for reactivity. Although this did not confuse the candidate, it was inconsistent with material presented to build the exam. The unit of PCM is not used at Calvert Cliffs.

NRC Resolution: Comment Noted.

References to the unit PCM will be deleted from the question. In addition, units of AMPS are also deleted from the question.

Question Number: 5.10 b.

Facility Comment: b.1. 260 F (255-265) for 1700 psig
2. 330 F (320-380) for 700 psig

This is a constant enthalpy expansion to 35 psia. Additionally, units of psia & psig should remain consistent.

Reference: Steam Tables

NRC Resolution: Agree with comment.

Answer key changed to reflect proper answer.

Question Number: 5.14 a.

Facility Comment: a. Once through cooling is correct. However, the answer key states incorrectly that this method is inadequate. The references stated do not apply to this method of cooling and therefore cannot be used to verify the answer. A phone conversation with Dr. William Dove of CE verified this a viable cooling method. The answer should be "yes".

NRC Resolution: Agree with comment.

Answer key changed to YES.

Question Number: 6.04 b.

Facility Comment: b. "Control Channels" is not standard Calvert Cliffs terminology. Therefore, this question could be interpreted as reactor Regulating or the narrow range safety NI channels. If narrow range NIs are assumed, the answer would change to any two of the three lissted below.

1. Inhibit SUR trip
2. Enable loss of load trip
3. Enable APD trip

Reference: RO-57-1-3 pages 22 & 23

NRC Resolution: Disagree with comment.

Reference stated in facility comment (RO-57-1-3) was not supplied to the grader. Reference stated in the examination (RO-57-1-2 page 39) refers to the Power Range Control Channel. Question premise refers to "excore neutron detectors and associated instrumentation". During the examination the candidate asked "What is meant by Control Channels?", he was told they refer to Power Range NIs. No change to answer key.

Question Number: 7.13

Facility Comment: This knowlwdge is beyond that expected for an SRO at CCNPP. Radiation Protection personnel are on shift to provide the necessary support. We request this question be dropped.

NRC Resolution: Comment noted.

NUREG-1021, Operator Licensing Examiner Standards, lists items that are to be included in the scope of the written examination for an SRO. Included in this scope are
1.) ...describe methods for performing maintence so that he, his crew, and the general public are protected.
2.) Candidate should be familiar with the concept of ALARA and be able to demonstrate his knowledge regarding this concept.
NUREG-1122, K/A Catalog, lists, under Plant-wide Generics,
1.) K1.03 Knowlege of 10CFR20 and related facility radiation control requirements. --SRO value 3.4
2.) K1.04 Knowledge of facility ALARA program. --SRO value 3.5
The calculation of a stay time for performing maintence should be an evolution that a supervisor performs not only for himself but for the personnel who work for him. The Radiation Protection personnel on shift should support the supervisor in this function.

No change to answer key.

Comment on Note: Recommendation noted for future consideration.

ADDITIONAL COMMENTS MADE BY GRADER:

Question Number: 8.04 a.

Changed answer to require an OR response in order to obtain the requested information.

Question Number: 8.09 a.

Added an additional correct answer; to reflect current facility information.

Question Number: 8.09 b.

Changed the second answer; to reflect the common nomenclature of the facility.

ATTACHMENT 2

NRC RESOLUTION OF FACILITY COMMENTS FOR THE
CALVERT CLIFFS SENIOR REACTOR OPERATOR EXAMINATION
ADMINISTERED ON JUNE 22, 1988

Question Number: 5.04 3.

Facility Comment: 3) Raising Tavg.

The analysis of temperature change across the core is correct. However, the result would be in an increased negative ASI, power moving toward the top, as a result of the temperature change. Tc increasing adds negative reactivity to the lower half of the core. Th decreasing adds positive reactivity to the top of the core. Therefore power shifts to the top of the core. The more negative MTC in the upper core aids this process causing a larger power (ASI) swing.

Reference: Same as stated.

NRC Resolution: Agree with comment.

Reference stated supports facility comment and does not fully support answer key. Answer key is changed to reflect facility comment.

Question Number: 5.07

Facility Comment: The question used units of PCMS in lieu of % for reactivity. Although this did not confuse the candidate, it was inconsistent with material presented to build the exam. The unit of PCM is not used at Calvert Cliffs.

NRC Resolution: Comment Noted.

References to the unit PCM will be deleted from the question. In addition, units of AMPS are also deleted from the question.

Question Number: 5.10 b.

Facility Comment: b 1. 260 F (255-265) for 1700 psig
2. 330 F (320-380) for 700 psig

This is a constant enthalpy expansion to 35 psia. Additionally, units of psia & psig should remain consistent.

Reference: Steam Tables

NRC Resolution: Agree with comment.

Answer key changed to reflect proper answer.

Question Number: 5.14 a.

Facility Comment: a. Once through cooling is correct. However, the answer key states incorrectly that this method is inadequate. The references stated do not apply to this method of cooling and therefore cannot be used to verify the answer. A phone conversation with Dr. William Dove of CE verified this a viable cooling method. The answer should be "yes."

NRC Resolution: Agree with comment.

Answer key changed to YES.

Question Number: 6.04 b.

Facility Comment: b. "Control Channels" is not standard Calvert Cliffs terminology. Therefore, this question could be interpreted as reactor Regulating or the narrow range safety NI channels. If narrow range NIs are assumed, the answer would change to any two of the three listed below.

1. Inhibit SUR trip
2. Enable loss of load trip
3. Enable APD trip

Reference: RO-57-1-3 pages 22 & 23

NRC Resolution: Disagree with comment.

Reference stated in facility comment (RO-57-1-3) was not supplied to the NRC. Reference stated in the examination (RO-57-1-2 page 39) refers to the Power Range Control Channel. Question premise refers to "excore neutron detectors and associated instrumentation." During the examination the candidate asked "What is meant by Control Channels?," he was told they refer to Power Range NIs. No change to answer key.

Question Number: 7.13

Facility Comment: This knowledge is beyond that expected for an SRO at CCNPP. Radiation Protection personnel are on shift to provide the necessary support. We request this question be dropped.

NRC Resolution: Comment noted.

10 CFR 55.43 (b) (4), "Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions," is an area of knowledge which will be sampled on an SRO written examination. Examiner Standard ES-402 A.3 lists items that are to be included in the scope of the written examination for an SRO. Included in this scope are 1.) ...describe methods for performing maintenance so that he, his crew, and the general public are protected. 2.) Candidate should be familiar with the concept of ALARA and be able to demonstrate his knowledge regarding this concept.

NUREG-1122, K/A Catalog, lists, under Plant-wide Generics,

(1) K1.03 Knowledge of 10CFR20 and related facility radiation control requirements. --SRO value 3.4.

(2) K1.04 Knowledge of facility ALARA program. --SRO value 3.5.

The calculation of a stay time for performing maintenance should be an evolution that a supervisor performs, or reviews, not only for himself but for the personnel who work for him.

No change to answer key.

Comment on Note: Recommendation noted for future consideration.

ADDITIONAL COMMENTS MADE BY GRADER:

Question Number: 8.04 a.

Changed answer to require an OR response in order to obtain the requested information.

Question Number: 8.09 a.

Added an additional correct answer to reflect current facility information.

Question Number: 8.09 b.

Changed the second answer to reflect the common nomenclature of the facility.