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

EXAMINATION REPORT NO. 88-03(OL)

FACILITY DOCKET NO. 50-322

FACILITY LICENSE NO. NPF-36

LICENSEE: Long Island Lighting Company
Post Office Box 618
Wading River, New York 11792

FACILITY: Shoreham Nuclear Power Station

EXAMINATION DATES: May 16 to May 20, 1988

CHIEF EXAMINER: Allen G. Howe — 8-5-88
Allen G. Howe, Senior Operations Engineer — Date

APPROVED BY: Allen G. Howe for — 8-5-88
David J. Lange, Chief, BWR Section, Operations Branch, Division of Reactor Safety — Date

SUMMARY: Written examinations and operating tests were administered to three (3) senior reactor operator (SRO) candidates and nine (9) reactor operator (RO) candidates. Two (2) SRO candidates and seven (7) RO candidates passed these examinations. All other candidates failed the examinations.

DETAILS

TYPE OF EXAMINATIONS: Initial

EXAMINATION RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written	8/1	3/0
Operating	8/1	2/1
Overall	7/2	2/1

1. CHIEF EXAMINER AT SITE: Allen G. Howe, Senior Operations Engineer

2. OTHER EXAMINERS: D. Lange, Chief, BWR Section
T. Fish, Operations Engineer
D. Moon, PNL
C. Moore, PNL

3. The following is a summary of generic strengths or deficiencies noted on operating tests. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

STRENGTHS

A. General systems knowledge

B. Crew communications in the simulator

C. SRO knowledge of the Emergency Plan and Technical Specifications

DEFICIENCIES

A. Ability to explain the operation of various control systems

4. The following is a summary of generic strengths or deficiencies noted from the grading of written examinations. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

SRO STRENGTHS

A. Ability to calculate shutdown margins given changes in count rates. (Question 5.03)

B. Ability to predict the effects on the plant due to various failures of the APRM flow converters. (Question 6.07)

C. Understanding of the fuel preconditioning (PCIOMR) process and requirements. (Question 7.06)

D. Ability to classify events using the Emergency Plan Implementing Procedures. (Question 8.07)

SRO DEFICIENCIES

A. Ability to predict relative changes in radial power distribution in response to movements of shallow and deep control rods. Knowledge that CRD flow is used by the process computer for the heat balance calculation. (Questions 5.02 and 5.09)

B. Knowledge of the purpose of the CST connection to the RCIC pump suction when operating RHR in the steam condensing mode. (Question 6.04)

C. Ability to recognize entry conditions for the "Emergency Shutdown" procedure (SP29.010.01). Knowledge that below 110 deg. F the maximum allowable drywell pressure is not constant. (Questions 7.03 and 7.08)

D. Ability to determine actions required by technical specifications for inoperable control rods. (Question 8.06)

RO STRENGTHS

A. Ability to: predict the magnitude of power changes at different times in core life for the same reactivity addition, calculate the time to reach a specified given reactor period, predict changes in control rod worth due to changing other parameters, calculate cooldown rates, and predict critical power changes as plant parameters change. (Questions 1.01, 1.04, 1.05, 1.07, and 1.11)

B. Knowledge of how various air operated valves respond to a loss of air and the bases and controls for the Rod Block Monitor system. (Questions 2.01 and 2.11)

B. Knowledge of: what SCRAM signals are automatically bypassed and the operation of the Automatic Depressurization System. Ability to predict: the response of the reactor recirculation system to parameter changes and changes in reactor water level due to plant parameter changes. (Questions 3.01, 3.04, 3.07, and 3.08)

D. Knowledge of: technical specification requirements for recirculation pump speed mismatch and the administrative radiation dose limits. (Questions 4.09 and 4.12)

RO DEFICIENCIES

A. Ability to predict relative changes in radial power distribution in response to movements of shallow and deep control rods. Failure to recognize that power reductions result in less feedwater heating which in turn adds positive reactivity. Ability to determine if a pump will cavitate given the required NPSH and tank level at the suction. (Questions 1.02, 1.09, and 1.11)

B. Knowledge of: the operation of the emergency diesel generator shutdown system without starting air available and that the CST will provide NPSH to the RCIC pump when operating RHR in the steam condensing mode. Understanding of the interlocks for the RCIC pump suction. Knowledge that placing the Remote Shutdown System transfer switches in "EMERGENCY" bypasses most but not all trips and interlocks. (Questions 2.03, 2.04, 3.02, and 3.10)

C. Knowledge of: the reasons for preventing automatic transfer of the RCIC suction to the suppression pool per the "Loss of All AC Power" procedure SP 29.015.20 and the maximum allowable average core power for any transient authorized per Standing Order Number 30. (Questions 4.04 and 4.08)

5. Personnel Present at Exit Interview:

NRC Personnel

A. Howe, Chief Examiner
T. Fish, Operations Engineer

Facility Personnel

W. Steiger, Plant Manager
S. Skorupski, Assistant Vice President Nuclear Operations
M. Case, Operating Engineer
L. Calone, Manager Operations Training Division
H. McDaniel, Operations Training
H. Carter, Licensed Operator Training Supervisor
C. Thayer, Manager Operations and Simulator Training Department
K. Rottkamp, Manager, Facility Services Division

6. Summary of NRC comments made at exit interview:

The chief examiner thanked the training and operations staffs for their cooperation during the examination. The operators in the control room were also helpful in answering questions.

The examiners felt site access was smooth and that housekeeping was adequate.

Examination security for both the written and simulator portions were excellent.

The written examination review was discussed. The facility staff was advised where to send the formal comments. The facility staff also stated that the examination was of good quality.

The generic strengths and weaknesses noted on the operating examinations were discussed.

The simulator fidelity was reviewed. This is detailed in Attachment 5 of this report.

Attachments:

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on Written Examinations after Facility Review
4. NRC Response to Facility Comments
5. Simulation Facility Fidelity Report

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: SHOREHAM

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/05/16

EXAMINER: NRC REGION I

CANDIDATE: ANSWER KEY

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	_____	_____	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.00	25.00	_____	_____	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00	_____	_____	3. INSTRUMENTS AND CONTROLS
25.00	25.00	_____	_____	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.00		_____ Final Grade	_____%	Totals

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

_____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

6. Use only the paper provided for answers.

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

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

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

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

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

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

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 1.01 (3.00)

The reactor has been operating at 4% of rated core thermal power for several days with the main turbine on the turning gear. Reactor pressure is 921 psig. The reactor operator withdraws an in-sequence control rod and reactor power increases to 4.5%.

a. LIST the two (2) reactivity coefficients that had the greatest effect in initially turning the power increase. (0.5)

b. Once 4.5% power is initially obtained, with no further changes in control rod position or reactor recirculation flow, DESCRIBE HOW the magnitude of reactor power will behave (relative to 4.5%) initially during the next hour AND over the following 10 hours of operation. INCLUDE a brief explanation for this behavior. (Specific values for changes in magnitude are not required). (1.0)

c. If the same amount of reactivity were to be inserted at these same initial conditions but LATER in the fuel cycle, STATE whether the reactor period observed during the transient would be (GREATER THAN, LESS THAN, or EQUAL TO) the period observed earlier in the fuel cycle. SUPPORT your answer with an explanation. (1.5)

QUESTION 1.02 (2.50)

The reactor is operating at 95% of rated core thermal power. Control rod 18-35 is at notch position 06. Control rod 22-35 is at notch position 38.

a. STATE which of these two control rods is most likely to produce the greatest increase in core thermal power if withdrawn one additional notch. (0.5)

b. STATE which of these two control rods is most likely to produce the SMALLEST CHANGE in core RADIAL power distribution if withdrawn one notch. INCLUDE a brief explanation to support your answer. (1.0)

c. The reactor operator withdraws a shallow control rod one notch and notes that total core power decreased slightly in response to the rod movement. Briefly EXPLAIN HOW such a response is possible. (1.0)

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

QUESTION 1.03 (2.00)

Following a reactor scram from power, several control rods fail to insert to the full-in position. Within one hour, the reactor is determined to be subcritical with an actual shutdown margin (SDM) of 0.22% delta K/K.

a. If reactor coolant temperature and control rod positions remain constant during the next hour, WOULD actual SDM (INCREASE, DECREASE, or REMAIN THE SAME)? Briefly EXPLAIN your answer. (1.0)

b. During the next hour you notice reactor pressure is decreasing. WHAT effect would this have on actual SDM? Briefly EXPLAIN your answer. (1.0)

QUESTION 1.04 (2.00)

Reactor power is 60 on IRM range 2 with the MINIMUM permissible stable positive period allowed by procedure SP22.001.02. Heating power is determined to be 40 on IRM range 7.

a. HOW LONG will it take for reactor power to double if period remains constant? (1.0)

b. HOW LONG will it take for power to reach the point of adding heat if period remains constant? (1.0)

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

QUESTION 1.05 (2.50)

SELECT the appropriate response for each of the following statements concerning Control Rod Worth:

a. (MORE/LESS) control rods would need to be pulled to make the reactor critical at 545 deg F, as opposed to 140 deg F. (ASSUME initial rod pattern identical, and the same sequence is used in both cases.) (0.5)

b. An INCREASE in the Void Fraction will result in a/an (INCREASE/DECREASE) in individual control rod worth. (0.5)

c. Control rod worth will (INCREASE/DECREASE) with an INCREASE in moderator temperature. (0.5)

d. Control rod worth at End of Cycle would be (GREATER/LESS) than at the Beginning of Cycle. (0.5)

e. Control rod worth will (INCREASE/DECREASE) as the adjacent control rods are withdrawn. (0.5)

QUESTION 1.06 (3.00)

An EHC load reject occurs at 100% core thermal power with the EHC system aligned for normal 100% power generation. DESCRIBE HOW and WHY the following parameters respond initially AND then during the first five minutes subsequent to the opening of the generator output breaker.

a. Reactor Power (1.0)

b. Reactor Pressure (1.0)

c. Reactor Water Level (1.0)

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

QUESTION 1.07 (2.50)

As a reactor operator coming on shift, you are told that the previous shift performed a reactor shutdown and commenced a cooldown from 630 psig at 0630 hours. It is now 0730 hours and you note that wide range reactor pressure is 200 psig. Your shift is to place the reactor in shutdown cooling.

a. HAS the previous shift exceeded the maximum allowable cooldown rate allowed by procedure SP22.005.01, "Shutdown from 20% Power"? (INCLUDE in your answer the Cooldown Limit and the assumptions and calculations used.) (1.5)

b. HOW many more degrees of cooldown are necessary before RHR can be unisolated for shutdown cooling? (INCLUDE your assumptions and calculations.) (1.0)

QUESTION 1.08 (3.00)

WHAT are the three (3) "thermal limits" (specific values not required) observed during reactor operation and WHAT is the purpose for each one? (3.0)

QUESTION 1.09 (2.00)

The reactor has been operating at 95% power for several days. An operator RAPIDLY reduces reactor power to 60% by reducing the speed of the recirculation pumps. During the next 2 to 3 MINUTES the operator notices that the reactor power slowly increases to 63% with no operator action. EXPLAIN the cause of the power increase. (2.0)

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

QUESTION 1.10 (1.00)

An RHR pump is placed in ~~torus~~ suppression pool cooling. ASSUME the RHR pump is located 11 feet below the water level in the ~~torus~~ suppression pool and requires 5 feet NPSH to prevent cavitation.

a. WHAT is the definition of available NPSH? (0.5)

b. WILL the pump cavitate if the ~~torus~~ suppression pool water level decreased to only 4 feet above the RHR pump suction? Assume torus water temperature is 90 degrees F. EXPLAIN your answer. (0.5)

QUESTION 1.11 (1.50)

ANSWER the following questions concerning "CRITICAL POWER."

a. DEFINE "Critical Power." (0.75)

b. With the reactor at power, WHICH of the following conditions would tend to DECREASE the Critical Power level assuming all other variables remain unchanged?

(1) Reactor pressure is INCREASED.
(2) Total core flow is INCREASED.
(3) Inlet subcooling is INCREASED. (0.75)

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

QUESTION 2.01 (2.00)

STATE whether the following valves will (FAIL OPEN, FAIL CLOSED, or FAIL AS-IS) on a complete loss of instrument air:

a. RFP startup level control valves (0.4)

b. RFP minimum flow valves (0.4)

c. condensate pump minimum flow valves (0.4)

d. condenser normal makeup valve (0.4)

e. ~~SJA~~ SJAE steam supply valves (0.4)

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QUESTION 2.02 (2.50)

A loss of offsite power has occurred concurrently with a valid loss of coolant accident (LOCA) signal. Diesel generator G-101 properly autostarts.

a. SPECIFY the proper order in which the following components sequence onto emergency bus 101. INCLUDE the correct time delay:

(1) core spray pump
(2) RBSVS/CRAC water chiller
(3) RHR pump
(4) service water pump (1.5)

b. For the component start time delays, STATE the EVENT in the emergency diesel start/load sequence that initiates the timing sequence (i.e., serves as "time zero" for the time delay setpoint). (1.0)

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

QUESTION 2.03 (2.50)

Concerning the Emergency Diesel Generators (G-101, 102, 103):

a. LIST three (3) conditions that will automatically trip an emergency diesel generator if a valid LOCA signal is present with the diesel mode selector switch in "REMOTE." (DO NOT include a manual stop.) (1.5)

b. If an emergency diesel generator is running with the diesel mode selector switch in REMOTE and ALL starting air is completely lost, STATE whether that emergency diesel generator (WILL or WILL NOT) shutdown in response to the following conditions.

1. The operator depresses the control room manual stop pushbutton for that diesel. (0.5)

2. A valid automatic shutdown signal condition occurs for that diesel. (ASSUME an AUTO-START condition does not exist). (0.5)

QUESTION 2.04 (1.50)

The reactor has been operating for a month at rated core thermal power. A loss of all AC power occurs causing the reactor to scram. The decision to implement the steam condensing mode of RHR has been made.

a. STATE whether the steam condensing capability of two (2) RHR heat exchangers (IS/IS NOT) adequate to accommodate ALL of the reactor's decay heat immediately after the scram. (0.5)

b. The "A" loop of RHR and the RCIC pump are now operating in the steam condensing mode. The "A" RHR Heat Exchanger (HX) pressure control valve (PCV-003A) and level control valve (PCV-007A) controllers are in AUTOMATIC. INITIALLY ALL RCIC pump flow is condensate from the "A" RHR HX. If the reactor operator then doubles RCIC pump flow to rated flow, STATE whether the RCIC pump (WILL or WILL NOT) cavitate. EXPLAIN your answer. (1.0)

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

QUESTION 2.05 (3.00)

With regard to Low Pressure Coolant Injection (LPCI) system.

a. WHAT signals cause an automatic initiation of LPCI? (DO NOT include a manual initiation.) (1.0)

b. DESCRIBE the interlocks which must be satisfied in order to divert injection from the reactor to containment spray with a LPCI initiation signal present with the RHR inboard injection valves (MOV-037A/B) OPEN. (1.0)

c. LIST the two (2) modes of RHR system operation that require operator action to realign the system for LPCI operation if a valid LPCI initiation signal occurs. (1.0)

QUESTION 2.06 (3.00)

Concerning the Standby Liquid Control (SBLC) System:

a. LIST four (4) SBLC System indications available in the control room to confirm SBLC initiation/injection. (1.0)

b. EXPLAIN WHY a too rapid SBLC system injection rate is undesirable. (1.0)

c. DESCRIBE WHERE the SBLC system physically discharges in the reactor vessel relative to the core plate. (ABOVE/BELOW) (0.5)

d. WILL the SBLC pump, if running, automatically trip on any SBLC storage tank low level condition (YES/NO)? (0.5)

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

QUESTION 2.07 (2.50)

The Reactor Building Normal Ventilation System (RBNVS) is in service with a normal operating supply and exhaust fan lineup. The reactor operator must shift operating exhaust fans to allow maintenance.

a. DESCRIBE which RBNVS fan(s) and/or damper(s) will automatically respond to attempt to control reactor building to outside air differential pressure when the additional exhaust fan is STARTED. INCLUDE HOW they respond. (1.5)

b. If reactor building internal pressure inadvertently becomes greater than the outside air pressure during this shift of exhaust fans, DESCRIBE HOW the following components will respond (TRIP/REMAIN RUNNING):

(1) previously running reactor building supply fans (0.5)

(2) previously running reactor building exhaust fans (0.5)

QUESTION 2.08 (1.50)

Concerning the suppression pool to drywell vacuum breakers:

a. If two suppression pool to drywell vacuum breakers in series (e.g., RV93A and RV93B) were stuck open, DESCRIBE HOW this could cause primary containment to fail if a LOCA were to occur. (1.0)

b. CAN these valves be operated from the control room? (YES/NO) (0.5)

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

QUESTION 2.09 (1.50)

The reactor is in Operational Condition 2 with the reactor water cleanup (RWCU) system in service. The RWCU outboard containment isolation valve (1G33*M034) suddenly auto-isolates. The RWCU inboard containment isolation valve (1G33*M033) remains open.

a. STATE ALL possible conditions (i.e., signals) that could cause this particular isolation response, assuming the isolation logic for BOTH valves has properly functioned (setpoints NOT required). (0.5)

b. Assuming 1G33*M034 remains closed, STATE WHAT sampling requirements are now imposed to meet Technical Specification requirements. (1.0)

QUESTION 2.10 (3.00)

For each of the following statements regarding High Pressure Coolant Injection System (HPCI), INDICATE whether the statement is TRUE or FALSE, and EXPLAIN your answer.

a. In the event low pump suction pressure is sensed during HPCI system operation, the turbine will trip, and the signal must be manually reset before the turbine will restart, if an initiation signal is still present. (1.0)

b. Upon a HPCI system isolation, due to low steam pressure, the system cannot restart until the pressure rises above the isolation setpoint and the isolation signal is reset. (1.0)

c. If the HPCI turbine trips due to an overspeed condition, it will restart when the speed coasts down to between 3000 and 4000 RPM, if an initiation signal is still present. (1.0)

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

QUESTION 2.11 (2.00)

For the Rod Block Monitor (RBM), PROVIDE answers to the following questions:

a. WHAT adverse condition is the system designed to prevent? (1.0)

b. When the Meter Function Switch on the Back Panel 937 Meter Section is in the "Count" position, WHAT are the "units" of the indication on the meter and WHAT can be calculated by utilizing the indicated value? (1.0)

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

QUESTION 3.01 (3.00)

Following a reactor SCRAM, some scram signals are bypassed by operator or automatic actions. For each of the following scram signals, STATE ALL the condition(s) that must be in effect for a bypass to occur:

a. main steam line isolation scram (0.75)

b. reactor mode switch in SHUTDOWN scram (0.75)

c. turbine control valve fast-closure scram (0.75)

d. scram discharge volume high-level scram (0.75)

QUESTION 3.02 (3.00)

An automatic RCIC initiation has occurred. Subsequently, RCIC injection was automatically terminated due to high reactor water level.

a. WHAT components in the RCIC system functioned to terminate the injection? (1.0)

b. Assuming no operator action, HOW will RCIC respond if reactor water level subsequently decreased to -45 inches? (0.5)

c. If an RCIC flow functional test had been in progress when the initial automatic initiation signal had been received, HOW would the SYSTEM have responded? (0.5)

d. If, following the initial automatic initiation, the RCIC turbine had tripped on mechanical overspeed, COULD it be reset from the Control Room? (YES/NO) (0.5)

e. HOW can the operator override the interlock associated with the RCIC pump suction valve from the suppression pool valve (1E51*MOV-32)? (0.5)

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

QUESTION 3.03 (2.00)

The reactor is shutdown. Both loops of RHR inadvertently initiate in the LPCI mode and inject at rated flow. ANSWER the following questions concerning fuel zone level instrumentation and indication at these conditions.

a. Fuel zone level indication would indicate (UPSCALE/DOWNSCALE). (0.5)

b. Fuel zone instrument sensed differential pressure would (INCREASE/DECREASE/NOT BE AFFECTED). (0.75)

c. Fuel zone instrument variable leg pressure subsequently (INCREASES/DECREASES/REMAINS THE SAME) as a result of LPCI flow. (0.75)

QUESTION 3.04 (2.50)

For the following situations, STATE whether the Automatic Depressurization System (ADS) relief valves will (OPEN, CLOSE or REMAIN AS-IS). CONSIDER each set of conditions separately.

a. ADS initiating signal (low level) occurs, the ADS timer times out, and ADS valves open. Then reactor water level rises to +10 inches. (0.5)

b. ADS initiating parameters are present and ADS valves open. Then the ADS timer reset buttons are depressed. (0.5)

c. ADS initiating parameters are present and ADS valves open. Then a DC power failure occurs that affects all busses supplying solenoid power to the ADS valves. (0.5)

d. ADS initiating parameters are present and a loss of instrument air supply to the drywell has occurred. Then, the ~~120~~ 105-second timer times out. (0.5)

e. ADS initiating parameters are present and all ECCS pumps are secured except for CS pump B which is running with a discharge pressure of 135 psig. Then, the ~~120~~ 105-second timer times out. (0.5)

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

QUESTION 3.05 (2.75)

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ng operation of the APRM flow units: (Assume 100% power)

the OUTPUT of one APRM flow unit fails UPSCALE, SELECT WHICH one of the following automatic responses that should occur. (1.0)

(1) rod block but no half scram
(2) half scram but no rod block
(3) rod block and half scram
(4) no rod block and no half scram.

b. If the OUTPUT of one APRM flow unit fails DOWNSCALE, SELECT WHICH one of the following automatic responses that should occur. (1.0)

(1) rod block but no half scram
(2) half scram but no rod block
(3) rod block and half scram
(4) no rod block and no half scram.

c. From the following list, SELECT the process flow that is actually measured by the flow transmitters that provide flow signals to the Recirculation Flow Units.

(1) total core flow
(2) recirculation loop "Driving" (pump) flow
(3) recirculation loop jet pump flow (0.75)

QUESTION 3.06 (3.00)

Consider the Rod Block Monitor system (RBM):

a. WHAT is the purpose of the null sequence control circuit? (1.5)

b. WHAT are three (3) ways the RBM trips are BYPASSED? (1.5)

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

QUESTION 3.07 (2.25)

The reactor is operating at 100% power with the recirculation system in Master Manual Control. EXPLAIN HOW and WHY EACH recirculation pump responds to the following conditions. Where applicable, PROVIDE specific values.

a. Reactor water level decreases to 25 inches following a feedwater pump trip. (1.0)

b. Full-open signal from recirculation pump 'A' discharge valve is lost. (1.25)

QUESTION 3.08 (1.50)

The reactor is operating at 100% power under steady state conditions with the feedwater control system in 3-element control. For the following loss of signal failures to the feedwater control system, INDICATE in which direction reactor water level will initially respond (INCREASE or DECREASE). ASSUME no operator action.

a. loss of one feedwater flow input (0.5)

b. complete loss of steam flow input (0.5)

c. loss of reactor water level input (0.5)

QUESTION 3.09 (3.00)

The following concern the Source Range Monitors.

a. STATE the difference between the retract permissive indicating light on Panel 1H11*PNL-603 and the retract permissive light on the SRM drawer. (INCLUDE setpoints.) (0.5)

b. Except for inop, LIST ALL the conditions that will generate a rod block in the SRM circuit. (INCLUDE setpoints AND when the rod block is not bypassed.) (2.25)

c. DESCRIBE HOW the SRMs can be used to produce a reactor scram signal. (0.25)

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

QUESTION 3.10 (2.00)

Consider the remote shutdown panel. ANSWER TRUE or FALSE for each of the following statements concerning the remote shutdown panel (RSP):

a. RSP instrumentation includes indication for drywell temperature, suppression pool level, AND SRV air header pressure. (0.5)

b. If the remote shutdown transfer switch (RSTS) for the RHR system is in the EMERGENCY position, the RHR pump will NOT automatically start when a valid low reactor water level initiation signal occurs. (0.5)

c. When the RSTS for the RHR system is in the EMERGENCY position, the RHR pump CANNOT be stopped or started from the main control room. (0.5)

d. When the RSTS for the RCIC system is in the EMERGENCY position, all the RCIC system interlocks are bypassed. (0.5)

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

QUESTION 4.01 (2.50)

Procedure SP 22.001.01, "Startup-Cold Shutdown to 20 Percent," places administrative restrictions upon reactor operation. Concerning these restrictions:

a. STATE the purpose for limiting the maximum permissible control rod drive (CRD) hydraulic system charging water header pressure to 1510 psig. (1.0)

b. STATE the maximum allowable interval (time) at which heatup rate must be verified to be within limits. (0.5)

c. STATE the purpose for ensuring PRESSURE SET is set above reactor pressure before condenser vacuum increases above 7" Hg. (1.0)

QUESTION 4.02 (2.50)

LIST ALL the entry conditions for SP29.023.02, "Secondary Containment Control Emergency Procedure." INCLUDE setpoints where applicable. (2.5)

QUESTION 4.03 (2.50)

The reactor was in hot standby with a bottom head drain temperature of 520 deg F. The high pressure coolant injection (HPCI) system auto initiated on a valid initiation signal while the reactor core isolation cooling (RCIC) system remained in standby. Based upon the responses of these two systems alone, SPECIFY WHICH entry condition(s) was/were met AND also WHICH procedure(s) should have been entered. ASSUME the HPCI and RCIC systems are properly aligned in standby for automatic initiation and are fully operable. (2.5)

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

QUESTION 4.04 (3.00)

A loss of ALL AC power has occurred. The "immediate actions" of emergency procedure SP 29.015.02, "Loss of All AC Power" are complete. The "subsequent actions" are now being performed. Reactor water level has been stabilized at +30" using the RCIC system alone (HPCI has been secured in accordance with the procedure).

a. STATE the reason WHY the procedure instructs the operator to depressurize the reactor as quickly as possible. (1.0)

b. STATE the reason WHY the procedure instructs the operator to secure the RCIC vacuum pump though RCIC is feeding the reactor pressure vessel to maintain water level. (1.0)

c. STATE one (1) reason WHY the procedure instructs the operator to PREVENT automatic suction transfer of the RCIC pump to the suppression pool. (1.0)

QUESTION 4.05 (1.50)

In accordance with SP 12.012.01, "Radiation Work Permits" (RWPs),

a. STATE the color of a correctly posted RWP copy at the job site. (0.5)

b. To perform his rounds, an equipment operator requires access to an unlocked room that has been designated as a "Radiation Area" by the health physics department. STATE whether an RWP (WILL or WILL NOT) have to be INITIATED. (ASSUME the health physics department has determined there is no loose surface or airborne contamination.) (0.5)

c. STATE the TYPE of RWP used by operators in performing surveillances during routine operating conditions. (0.5)

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

QUESTION 4.06 (3.00)

a. The reactor is operating at rated core thermal power. The main condenser low vacuum alarm ("COND VACUUM LO") is received.

1. STATE the immediate action(s) required by emergency procedure SP 29.012.01, "Loss of Condenser Vacuum." (0.6)

2. LIST the four (4) automatic actions that will be initiated by a low vacuum condition. INCLUDE setpoints. (Assume a complete loss of condensor vacuum occurs.) (1.6)

b. LIST two (2) reasons WHY procedures do not allow condenser air removal pumps to maintain condenser vacuum when reactor power is above 4%. (0.8)

QUESTION 4.07 (2.00)

The reactor is operating at rated core thermal power. The reactor building closed loop cooling water (RBCLCW) head tank "low-low" level alarm ("RBCLCW HD TK A(B) LEV LO-LO") is received. RBCLCW has isolated from all nonsafety loads.

a. LIST ALL immediate actions that are required by SP 29.017.01, "Loss of RBCLCW." (1.5)

b. STATE HOW LONG continued operation of the Reactor Recirc MG sets is allowed in this condition. (0.5)

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

QUESTION 4.08 (2.00)

Standing Order Number 30 provides specific guidance concerning reactor operation while the 5% Low Power License is in effect. Concerning the requirements of this standing order:

a. STATE the MAXIMUM allowable "average core power" for ANY transient. (0.5)

b. DESCRIBE HOW "average core power" is to be determined from plant instrumentation. (1.0)

c. SPECIFY the locations of the indications to be used for reactor power determination. (0.5)

QUESTION 4.09 (1.00)

Per Technical Specifications, recirculation pump speeds shall be maintained within:

a. _____ % of each other with core flow > 70%. (0.5)

b. _____ % of each other with core flow < 70%. (0.5)

QUESTION 4.10 (1.50)

Procedure SP 21.001.01, "Shift Operations," requires that only "ACTIVE" licensed operators may assume the watch.

a. SPECIFY ALL quarterly watch standing requirements that must be met for a licensed operator to maintain his "active" status. (1.0)

b. STATE which document must be signed by an individual if he is to receive credit towards maintaining his "active" status when he completes a watch in the capacity of his license. (0.5)

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

QUESTION 4.11 (1.00)

Concerning Procedure SP 12.011.01, "Station Equipment Clearance Permits (SECPs)":

a. STATE whether or not the second individual performing the independent verification of the placement of hold-off tags for an SECP is required to accompany the individual performing the SECP. (0.5)

b. STATE the criterion for determining whether or not a lifted lead requires an SECP hold-off tag in addition to a lifted lead/jumper tag. (0.5)

QUESTION 4.12 (2.50)

Concerning radiation exposure control limits:

a. An individual has a current NRC Form 4 on file, he is 45 years old, his lifetime whole body exposure is 131 REM, and it is Jan. 1.

   1. WHAT is his allowable FEDERAL whole body exposure for the first quarter? (0.5)

   2. WHAT is his allowable FEDERAL whole body exposure for the year? (0.5)

b. WHAT is the allowable FEDERAL whole body exposure per quarter for an individual (45 years old) who does NOT have a current NRC form 4 on file? (0.5)

c. WHAT are the "Administrative Radiation Dose Guides" per SP61.012.01, "Personnel Dose Limits" for

   1. whole body per week during outages? (0.5)

   2. whole body per quarter? (0.5)

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

ANSWER 1.01 (3.00)

a. 1. void coefficient [+0.25]
   2. Doppler coefficient [+0.25]

b. Reactor power will first increase greater than 4.5% [+0.25] as xenon is initially depleted faster than it is produced [+0.25]. Reactor power will subsequently stabilize at less than 4.5% [+0.25] as the production of xenon from the decay of iodine restores xenon concentration to a new higher equilibrium value [+0.25].

c. (Reactor period would) LESS THAN [+0.5]. Increased core age results in a higher fraction of core power being produced by Pu-239 [+0.25] which reduces the value of BETA-EFFECTIVE [+0.5]. From the period equation, as BETA-EFFECTIVE decreases, for a given reactivity insertion, the resultant reactor period decreases [+0.25].

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 11, LO CB and Lesson 15, LO CD, CH.

2. GE: Reactor Theory, Chapters 3, 4, and 6.

292003K106 292004K114 292006K105 292006K106 ...(KA'S)

ANSWER 1.02 (2.50)

a. (Control rod) 18-35 [+0.5]

b. (Control rod) 22-35 [+0.33]. The rod shadowing effects of adjacent rods [+0.33] will be more significant in dampening radial power because of the higher local control rod density relative to the control rod tip (of control rod 22-35) [+0.34].

c. When a shallow rod is withdrawn a notch, the void fraction in the adjacent fuel channels increases throughout the entire boiling length [+0.33]. Though local power will increase where the rod tip was withdrawn [+0.33], it is possible for the negative reactivity of the increased voiding to be the dominate effect (causing net total power to decrease) [+0.34].

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 13, LO CB, CC.

2. GE: Reactor Theory, Chapter 5.
292005K104 292005K112 292008K119 ...(KA'S)

ANSWER 1.03 (2.00)

a. increase [+0.5] due to Xe build in [+0.5]

b. The decrease in reactor pressure is directly related to a decrease in reactor coolant temperature. Due to the negative value of the moderator temperature coefficient, reactivity increases with decreasing moderator temperature [+0.5]. Hence, SDM decreases [+0.5].

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 12, LO CB.
292002K110 292002K114 ...(KA'S)

ANSWER 1.04 (2.00)

a. (From SP22.001.02) minimum period equals 60 seconds [+0.5].
Thus doubling time is 60/1.443 = 41.6 seconds [+0.5].

b. 60 on range 2 is equal to 0.06 on range 7

P(t) = P(o)e**-t/T [+0.25]

P(o) = 0.06, P(t) = 40, period = 60 seconds [+0.25]

t = 60 ln 40/0.06
= 390 seconds or 6.5 minutes [+0.5]

REFERENCE

1. Shoreham: HL-602-SH1, Lesson 15, LO CA.

2. Shoreham: SP22.001.02.

3. GE: Reactor Theory, Chapter 3.
292003K108 ...(KA'S)

ANSWER 1.05 (2.50)

a. more
b. decrease
c. increase
d. less
e. increase

[+0.5] each

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 12, LO CG.
292005K109 ...(KA'S)

ANSWER 1.06 (3.00)

a. Reactor power will rapidly increase due to the pressure increase [+0.5]. Power will then decrease due to the (TCV fast closure) scram [+0.5]. DRM 5/26/88

b. Reactor pressure will rapidly increase due to the rapid closure of the TCVs [+0.5]. Pressure will then decrease due to the scram and the opening of the bypass valves which will then attempt to maintain reactor pressure at 920 psig [+0.5].

c. Reactor water level will initially drop due to collapsing of voids [+0.5]. The feed control system will respond to increase level and level should then rise to the level controller setpoint (level may overshoot causing feed pumps to trip) [+0.5].

REFERENCE

1. Shoreham: HL-900-SH1; HL-657-SH1, LO C3.
241000K101 241000K102 241000K103 ...(KA'S)

ANSWER 1.07 (2.50)

a. The previous shift DID EXCEED the cooldown limit [+0.5] of 90 degrees F/hr [+0.5]. ALSO ACCEPT "100 degrees F/hr" DRM 5/26/88

(Tsat for 630 psig = 494 degrees F;
Tsat for 200 psig = 388 degrees F;
cooldown rate = (494-388) degrees F/1 hour
= 106 degrees F/hr) [+0.5]

b. 35 to 64 degrees F (of cooldown required depending on assumptions)

Tsat for 200 psig = 388 degrees F;
Tsat for 125 psig = 353 degrees F;
Tsat for 80 psig = 324 degrees F;
(388-353) = 35 degrees F;
(388-324) = 64 degrees F [+1.0]

REFERENCE

1. Shoreham: HL-901-SH1, LO CB; HL-121-SH1, LO CC.
205000K402 ...(KA'S)

ANSWER 1.08 (3.00)

1. LHGR - Linear Heat Generation Rate [+0.5] designed to (limit the pin power at any node in the reactor to) limit the fuel clad strain to less than 1% plastic strain [+0.5].

2. APLHGR - Average Planer Linear Heat Generation Rate [+0.5] designed to assure the maximum fuel clad temperature (following a design basis accident) will not exceed 2200 degrees F [+0.5].

3. MCPR - Minimum Critical Power Ratio [+0.5] designed (to limit the power of any fuel element) to prevent any point in the bundle from experiencing the onset of transition boiling [+0.5].

REFERENCE

1. Shoreham: HL-904-SH1, LO CA and CD.

2. GE: HTFF, Chapter 9.

293009K107 293009K111 293009K119 ...(KA'S)

ANSWER 1.09 (2.00)

The reactor is now producing less steam to go to the turbine. There will be less extraction steam going to the feedwater heater [+1.0]. Therefore, less feedwater heating will occur resulting in colder feedwater entering the vessel [+0.5] which will cause reactor power to increase (about 3%) from the positive reactivity addition (alpha m) [+0.5].

REFERENCE

1. Shoreham: HL-901-SH1, Lesson 4, LO CA.

2. GE: HTFF, Chapter 5.

3. GE: Rx Theory, Chapter 7.

292008K120 292008K121 2930005K10 ...(KA'S)

ANSWER 1.10 (1.00)

a. available NPSH is inlet pressure minus saturation pressure [+0.5]

b. No [+0.25], due to the contribution from atmospheric pressure [+0.25].

REFERENCE

1. Shoreham: HL-902-SH1, LO.
293003K123 293006K109 293006K110 ...(KA'S)

ANSWER 1.11 (1.50)

a. The assembly (bundle) power that would cause the onset of transition boiling [+0.75].

b. (1) [+0.75]

REFERENCE

1. Shoreham: HL-904-SH1, Lesson 1, LO CA.
293009K117 293009K122 293009K123 293009K124 ...(KA'S)

ANSWER 2.01 (2.00)

a. fail closed
b. fail open
c. fail closed
d. fail open
e. fail closed

[+0.4] each

REFERENCE

1. Shoreham: HL-109-SH1, LO M.1.

2. Shoreham: HL-117-SH1, LO CC.
295019AK20 295019AK21 ...(KA'S)

ANSWER 2.02 (2.50)

a.	(1)	RHR pump (3)	2 second T.D.	[+0.2]
	(2)	core spray pump (1)	7 second T.D.	[+0.2]
	(3)	service water pump (4) and	12 second T.D.	[+0.2]
		RBSVS/CRAC water chillers (2)	12 second T.D.	[+0.2]

(Point awards above are for T.D. values only. [+0.7] for correct order, no partial credit)

b. The timing sequence is initiated by the closing of the diesel generator output breaker. [+1.0]

REFERENCE

1. Shoreham: HL-307-SH1, LO B.1.b.
262001A304 262001K301 264000K506 ...(KA'S)

ANSWER 2.03 (2.50)

a. 1. overspeed [+0.5]
   2. generator phase differential [+0.5]
   3. generator overcurrent [+0.5]

b. 1. will not [+0.5]
   2. will not [+0.5]

REFERENCE

1. Shoreham: HL-307-SH1, LO A, C.4, D.
264000K106 264000K601 ...(KA'S)

ANSWER 2.04 (1.50)

a. IS NOT [+0.5]

b. The RCIC pump WILL NOT cavitate [+0.25]. CST inventory will maintain RCIC pump suction pressure [+0.75].

REFERENCE

1. Shoreham: HL-121-SH1, LO CC.

2. Shoreham: 23.121.01.
217000A101 217000K101 217000K105 217000SG1 ...(KA'S)

ANSWER 2.05 (3.00)

a. Hi drywell pressure 1.69 psig OR [+0.5]

Lo Lo Rx water level -132.5" [+0.5]

(NOTE: Do not penalize if RPV low pressure permissive of 338 psig is also listed.)

b. 1. containment spray valve manual override keyswitch in "MANUAL" [+0.5]

2. containment spray valve accident control switch in "MANUAL" [+0.5] (until seal-in status light is lit)

c. 1. shutdown cooling mode [+0.5]

2. fuel pool cooling mode [+0.5]

REFERENCE

1. Shoreham: HL-204-SH1, LO CC.
226001K403 ...(KA'S)

ANSWER 2.06 (3.00)

a. 1. squib valve continuity circuit indicator lamp extinguishes

2. squib valve loss of continuity annunciator

3. SBLC pump discharge pressure greater than reactor pressure

4. SBLC pump running indication (ON)

5. SBLC storage tank level decreasing

6. Continuity amp meters in back of panel 603 should read zero.

Any four (4) [+0.25] each, +1.0 maximum.

b. A too rapid injection rate could cause insufficient mixing and uneven concentrations of boron circulating in the core [+0.5] leading to power oscillations ("chugging") [+0.5].

c. below (the core plate) [+0.5]

d. No [+0.5]

REFERENCE

1. Shoreham: HL-123-SH1, LO B.1, F.
211000K106 211000K403 211000K405 211000K506 ...(KA'S)

ANSWER 2.07 (2.50)

a. The exhaust damper [+0.5] of the operating supply fan [+0.5] will modulate further open [+0.5].

b. (1) trip [+0.5]
(2) remain running [+0.5]

REFERENCE

1. Shoreham: HL-405/418-SH1, LO I.D.5.
261000K101 261000K401 ...(KA'S)

ANSWER 2.08 (1.50)

a. Primary containment could be overpressurized [+0.5] because of steam bypassing the suppression pool, pressurizing containment [+0.5].

b. Yes [+0.5]

REFERENCE

1. Shoreham: HL-654-SH1, LO CB.
223001K405 223001K501 223001K503 ...(KA'S)

ANSWER 2.09 (1.50)

a. 1. RWCU nonregenerative heat exchanger outlet (filter demineralizer inlet) temperature high [+0.25]

2. standby liquid control (SLC) system initiation [+0.25]

b. (In-line) conductivity must be sampled periodically [+1.0]. (continuous conductivity indication has been lost)

REFERENCE

1. Shoreham: Technical Specifications 3.4.4.

2. Shoreham: HL-709-SH1, LO CE, CF, CG.
204000K404 204000K507 204000SG11 204000SG5 ...(KA'S)

ANSWER 2.10 (3.00)

a. False [+0.5] - Once the low suction pressure signal is clear, the turbine will auto restart if the initiation signals are still present [+0.5].

b. True [+0.5] - The low steam pressure auto isolation signal seals in, and must be manually reset (using the AUTO ISOLATION SIGNAL RESET ~~pushbuttons~~ Keylock switches on the *PNL-601 after the reason for the isolation has been determined and corrected) [+0.5]. BM 5/24/88

c. True [+0.5] - The oil pressure will be restored when the turbine coasts down, thereby causing the stop valve to open [+0.5].

REFERENCE

1. Shoreham: High Pressure Coolant Injection System Procedure 23.202.01, Rev. 18.

2. Shoreham: HL-202-SH1, LO CI.

206000K401 ...(KA'S)

ANSWER 2.11 (2.00)

a. local fuel damage (by generating a rod withdrawal block) [+1.0] ALT ANS: OVERPOWERING LOCAL REGIONS OF THE CORE AT >30% PWR. [illegible] 5/24/88

b. units = volts [+0.5], number of operable LPRM inputs can be calculated (by using ~~5~~ 1 volts per operable input) [+0.5] BM 5/24/88

REFERENCE

1. Shoreham: HL-603,604,652-SH1, LO C.

215002A402 215002K102 215002SG04 ...(KA'S)

ANSWER 3.01 (3.00)

a. bypassed when the mode switch is NOT in run

b. auto bypassed after (10 sec.) time delay

c. auto bypassed if reactor power <30 percent (as indicated by turbine first stage pressure of 109 psig)

d. manual bypass switch~~es~~ in BYPASS with mode switch in SHUTDOWN or REFUEL

[+0.75] each

REFERENCE

1. Shoreham: HL-611-SH1, LO D.
212000K412 212004K408 ...(KA'S)

ANSWER 3.02 (3.00)

a. closure of the RCIC steam supply stop valve (MOV-43) [+0.5] and closure of ~~trip and throttle~~ injection (pump discharge) valve (MOV-~~44~~ 35) [+0.5]

b. RCIC will automatically initiate (and inject to the RPV) [+0.5]

c. align in RCIC starting mode and inject [+0.5]

d. no (locally) [+0.5]

e. by placing the suppression pool suction valve control switch in the CLOSE position [+0.5]

REFERENCE

1. Shoreham: HL-119-SH1, LO I.
217000A201 217000K202 217000K402 ...(KA'S)

ANSWER 3.03 (2.00)

a. upscale [+0.5]
b. decrease [+0.75]
c. increases [+0.75]

REFERENCE

1. Shoreham: HL-621-SH1, LO I.D.5.
216000K105 ...(KA'S)

ANSWER 3.04 (2.50)

a. remain as is
b. close
c. close
d. open
e. remain as is

[+0.5] each

REFERENCE

1. Shoreham: HL-201-SH1, LO F.
218000A205 218000K404 218000K501 218000K602 ...(KA'S)

ANSWER 3.05 (2.75)

a. (1) (rod block) [+1.0]

b. (3) (rod block and half scram) [+1.0]

c. (2) (recirculation loop "Driving" flow) [+0.75]

REFERENCE

1. Shoreham: HL-603-SH1, LO I.C.
215005K110 215005K116 215005K607 ...(KA'S)

ANSWER 3.06 (3.00)

a. adjusts the gain of the RBM channel [+0.75] to compensate for either variations in local power --OR-- bypassed LPRMs [+0.75] for either answer

b. 1. manual operation of RBM bypass switch [+0.5]
   2. edge rod selected [+0.5]
   3. reference APRM downscale (less than 30%) [+0.5]

REFERENCE

1. Shoreham: HL-606-SH1, LO CA.
215005A403 215005K403 215005K502 ...(KA'S)

ANSWER 3.07 (2.25)

a. Both recirculation pumps run back to 45% speed [+0.5] due to the automatic runback interlock with speed limiter (#2) [+0.5].

b. Recirculation pump 'A' trips [+0.5] due to the discharge valve not-full-open interlock with the MG set drive motor breaker [+0.25]. Recirculation pump 'B' speed will be unaffected [+0.5].

REFERENCE

1. Shoreham: HL-658-SH1, LO CG.
202002K305 202002K604 ...(KA'S)

ANSWER 3.08 (1.50)

a. increases [+0.5]
b. decreases [+0.5]
c. increases [+0.5]

REFERENCE

1. Shoreham: HL-656-SH1, LO C.
259002K301 259002K604 259002K605 295002K603 ...(KA'S)

ANSWER 3.09 (3.00)

a. Panel-603 = light on for greater than or equal to 100 cps
back panel = light off for greater than or equal to 100 cps
[+0.5]

b. SRM high, greater than 1 x 10**5 [+0.25], rod block below range 8 on IRM and not in run [+0.5]

SRM downscale, less than 3 cps [+0.25], rod block below range 3 on IRM and not in run [+0.5]

SRM retract not permitted less than 100 cps [+0.25], rod block if detector not full in and below range 3 on IRM and not in run [+0.5]

c. Removal of the shorting links (will cause any single SRM to produce a full scram signal at 2 x 10**5 cps). [+0.25]

REFERENCE

1. Shoreham: HL-G01-SH1, LO C and D.
215004K101 215004K401 215004K402 215004K405 ...(KA'S)

ANSWER 3.10 (2.00)

a. true
b. true
c. true
d. false (some trips and valve interlocks are not bypassed)

[+0.5] each

REFERENCE

1. Shoreham: HL-133-SH1, LO C and E.
295016AK20 ...(KA'S)

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

ANSWERS -- SHOREHAM -88/05/16-NRC REGION I

ANSWER 4.01 (2.50)

a. To prevent control rod drive mechanism damage [+0.5] during a scram [+0.5].

b. 15 minutes [+0.5]

c. To prevent inadvertent bypass valve operation [+1.0].

REFERENCE

1. Shoreham: SP 22.001.01; HL-106-SH1 LO J.
216000SG1 241000SG1 ...(KA'S)

ANSWER 4.02 (2.50)

1. Rx building differential pressure at or above 0 inches of water

2. Rx building exhaust radiation level above maximum normal operating radiation level

3. Rx building floor drain sump water level above maximum normal operating (table) value

4. any secondary containment area temperature above maximum normal operating (table) value

5. any secondary containment area radiation level above maximum normal operating (table) value

[+0.5] each

REFERENCE

1. Shoreham: SP29.023.02.

2. Shoreham: HL-944-SH3, LO I.B.
295032SG11 295033SG11 295035SG11 ...(KA'S)

ANSWER 4.03 (2.50)

1. the entry condition was high drywell pressure >1.69 psig [+1.0]

2. the emergency procedures which should have been entered were:

   a. "Emergency Shutdown" (SP 29.010.01) [+0.5]
   b. "Reactor Pressure Vessel (RPV) Control" (SP 29.023.01) [+0.5]

   c. "Primary Containment Control" (SP 29.023.03) [+0.5]

REFERENCE

1. Shoreham: HL-944-SH1, LO I.B.

2. Shoreham: HL-944-SH2, LO I.B.

295024SG11 ...(KA'S)

ANSWER 4.04 (3.00)

a. Early RPV depressurization will result in suppression pool and drywell (containment) temperatures [+0.25] and pressures [+0.25] remaining below designed limitations [+0.5].

(ALTERNATE ANSWER: to limit the total heat load [+0.5] placed upon the primary containment [+0.5])

b. (The RCIC vacuum pump is secured) to prolong [+0.5] the use of the division I battery [+0.5].

(ALTERNATE ANSWER: to reduce the load [+0.5] upon the Division I battery [+0.5])

c. 1. to slow the rate of containment temperature and pressure rise

2. to avoid failure of the RCIC turbine due to high lube oil temperatures

(Either 1. or 2. for [+1.0])

REFERENCE

1. Shoreham: SP 29.015.02.
295003AK20 295003AK30 ...(KA'S)

ANSWER 4.05 (1.50)

a. white [+0.5]

b. will not [+0.5]

c. extended (RWP) [+0.5]

REFERENCE

1. Shoreham: SP 12.012.01
294001K103 ...(KA'S)

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

ANSWERS -- SHOREHAM -88/05/16-NRC REGION I

ANSWER 4.06 (3.00)

a. 1. commence power reduction (in accordance with SP22.004.01, "Operation Between 20% and 100% Power") [+0.6]

2. a. main turbine trip [+0.2] at 22.5" Hg vac [+0.2]

b. reactor feed pump (turbine) trip [+0.2] at 20" Hg vac [+0.2]

c. MSIV (/main steam drain) isolation [+0.2] at 8.5" Hg vac [+0.2]

d. turbine bypass valve (TBV) isolation [+0.2] at 7" Hg vac [+0.2]

b. 1. to avoid hydrogen explosion (above 4% power) [+0.4]

2. the level of radioactivity in the noncondensable condenser gases is significant (above 4% power) [+0.4] (ALTERNATE ANSWER: condenser air removal pump exhaust is not treated prior to release)

REFERENCE

1. Shoreham: SP 29.012.01.

2. Shoreham: HL-701/714-SH1, LO E.4.

295002AK20 ...(KA'S)

ANSWER 4.07 (2.00)

a. 1. trip the operating RWCU pump [+0.25]

2. isolate the RWCU system from containment (close MOVs 033 and 034) [+0.25]

3. reduce reactor recirc pump speed to minimum [+0.25]

4. trip both reactor recirc pumps [+0.25]

5. initiate emergency shutdown procedure (SP 29.010.01) [+0.25]

6. trip CRD pumps after all control rods are verified inserted [+0.25]

b. 10 minutes [+0.5]

REFERENCE

1. Shoreham: SP 29.017.01.
295018SG10 295018SG11 ...(KA'S)

ANSWER 4.08 (2.00)

a. 5% [+0.5]

b. (Average core power is) the average of all APRM readings. [+1.0]

c. APRM readings are to be taken at panel 608 [+0.5] (also allow "backpanel").

REFERENCE

1. Shoreham: Standing Order No. 30.
294001A103 ...(KA'S)

ANSWER 4.09 (1.00)

a. 5% [+0.5]
b. 10% [+0.5]

REFERENCE

1. Shoreham: Technical Specifications, Section 3.4.1.3.

2. Shoreham: HL-120-SH1, LO C.5, L.
202002SG6 ...(KA'S)

ANSWER 4.10 (1.50)

a. A licensed operator must perform his duties (of his license) for seven 8-hour shifts [+0.5] or five 12-hour shifts [+0.5].

b. the shift turnover sheet [+0.5]

REFERENCE

1. Shoreham: SP 21.001.01.

2. Shoreham: SP 21.002.01.
294001A103 ...(KA'S)

ANSWER 4.11 (1.00)

a. is not required [+0.5]

b. Any lead that in normal service could be exposed to voltages in excess of 130 volts (requires a hold-off tag). [+0.5]

REFERENCE

1. Shoreham: SP 12.011.01.

2. Shoreham: SP 12.035.01.
294000K102 ...(KA'S)

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

ANSWERS -- SHOREHAM -88/05/16-NRC REGION I

ANSWER 4.12 (2.50)

a. 1. 3000 mrem [+0.5]
   2. ~~4000~~ mrem [+0.5]
   6750 RM 5/26/88

b. 1250 mrem [+0.5]

c. 1. 500 mrem/wk [+0.5]
   2. 1000 mrem/quarter [+0.5]

REFERENCE

1. Shoreham: SP61.012.01.

2. 10CFR20.101.
29400K103 ...(KA'S)

$f = ma$

$w = mg$

$E = mc^2$

$KE = 1/2\ mv^2$

$PE = mgh$

$V_f = V_o + at$

$W = V\ \Delta P$

$\Delta E = 931\ \Delta m$

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

$Q = UA\Delta h$

$Pwr = W_f \Delta h$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

$\Sigma = \sigma N$

$v = s/t$

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

$a = (V_f - V_o)/t$

$w = \theta/t$

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

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

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

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

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

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

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

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 $lbf/in.^2$

Miscellaneous Conversions

1 curie = 3.7×10^{10}dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

Master

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

FACILITY: SHOREHAM

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/05/16

EXAMINER: NRC REGION I

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
100.00		_____ Final Grade	_____ %	Totals

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

_____
Candidate's Signature

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

FACILITY: SHOREHAM

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/05/16

EXAMINER: NRC REGION I

CANDIDATE: ANSWER KEY

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.

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25.00	25.00	_____	_____	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00		_____ Final Grade	_____ %	Totals

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

_____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

6. Use only the paper provided for answers.

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

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

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

10. Skip at least three lines between each answer.

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

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

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 5.01 (3.00)

The reactor has been operating at 4% of rated core thermal power for several days with the main turbine on the turning gear. Reactor pressure is 921 psig. The reactor operator withdraws an in-sequence control rod and reactor power increases to 4.5%.

a. LIST the two (2) reactivity coefficients that had the greatest effect in initially turning the power increase. (0.5)

b. Once 4.5% power is initially obtained, with no further changes in control rod position or reactor recirculation flow, DESCRIBE HOW the magnitude of reactor power will behave (relative to 4.5%) initially during the next hour AND over the following 10 hours of operation. INCLUDE a brief explanation for this behavior. (Specific values for changes in magnitude are not required). (1.0)

c. If the same amount of reactivity were to be inserted at these same initial conditions but LATER in the fuel cycle, STATE whether the reactor period observed during the transient would be (GREATER THAN, LESS THAN, or EQUAL TO) the period observed earlier in the fuel cycle. SUPPORT your answer with an explanation. (1.5)

QUESTION 5.02 (2.50)

The reactor is operating at 95% of rated core thermal power. Control rod 18-35 is at notch position 06. Control rod 22-35 is at notch position 38.

a. STATE which of these two control rods is most likely to produce the greatest increase in core thermal power if withdrawn one additional notch. (0.5)

b. STATE which of these two control rods is most likely to produce the SMALLEST CHANGE in core RADIAL power distribution if withdrawn one notch. INCLUDE a brief explanation to support your answer. (1.0)

c. The reactor operator withdraws a shallow control rod one notch and notes that total core power decreased slightly in response to the rod movement. Briefly EXPLAIN HOW such a response is possible. (1.0)

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

QUESTION 5.03 (2.00)

You have been informed that the measured Shutdown Margin (SDM) for your reactor is four percent delta K/K. The average indication is 200 cps on the SRM instrumentation. Control rods are withdrawn and the average count rate on the SRM's increases to 1000 cps. WHAT is the new Shutdown Margin? (SHOW all work) (2.0)

QUESTION 5.04 (2.00)

The reactor is critical on a positive stable period of 100 seconds.

a. WHAT is the doubling time if period remains constant? (0.5)

b. If IRM Channel "B" indicates "10" on range 3, HOW long can IRM Channel "B" be left on range 3 before an IRM upscale rod block is initiated by IRM Channel "B" if period remains constant? (SHOW all work) (1.5)

QUESTION 5.05 (2.50)

SELECT the appropriate response for each of the following statements concerning Control Rod Worth:

a. (MORE/LESS) control rods would need to be pulled to make the reactor critical at 545 deg F, as opposed to 140 deg F. (ASSUME initial rod pattern identical, and the same sequence is used in both cases.) (0.5)

b. An INCREASE in the Void Fraction will result in a/an (INCREASE/DECREASE) in individual control rod worth. (0.5)

c. Control rod worth will (INCREASE/DECREASE) with an INCREASE in moderator temperature. (0.5)

d. Control rod worth at End of Cycle would be (GREATER/LESS) than at the Beginning of Cycle. (0.5)

e. Control rod worth will (INCREASE/DECREASE) as the adjacent control rods are withdrawn. (0.5)

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

QUESTION 5.06 (3.00)

An EHC load reject occurs at 100% core thermal power with the EHC system aligned for normal 100% power generation. DESCRIBE HOW and WHY the following parameters respond initially AND then during the first five minutes subsequent to the opening of the generator output breaker.

a. Reactor Power (1.0)

b. Reactor Pressure (1.0)

c. Reactor Water Level (1.0)

QUESTION 5.07 (2.00)

The reactor is critical and a heatup is in progress. Reactor pressure is 40 psig. If the heatup progresses at the maximum rate allowed by procedure SP 22.001.01, "Startup-Cold Shutdown to 20 Percent," HOW long from now will it be until reactor pressure will be adequate to allow warming of the HPCI turbine steam supply piping? (STATE all assumptions. ASSUME heatup rate is uniform. PROVIDE a calculation to support your answer.) (2.0)

QUESTION 5.08 (2.25)

Concerning Technical Specification safety limits:

a. STATE the basis for the reactor coolant system Technical Specification pressure safety limit of 1325 psig. INCLUDE a discussion concerning the considerations made in selecting this value. (1.5)

b. STATE which core thermal limit is specifically limited as a "safety limit" during FULL POWER operation. (0.75)

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

QUESTION 5.09 (3.00)

Reactor recirculation pump speed is 30%. Each reactor recirculation pump motor is drawing 0.18 MW of power.

a. If the reactor operator increases reactor recirc pump speed (both pumps) to 45%, HOW much power will be consumed by both pumps at 45% speed? (Using basic pump laws, PROVIDE a simple calculation and express your answer in MW.) (2.0)

b. The process computer does not measure actual steam flow rate in the heat balance calculation of core thermal power because of inherent inaccuracies in the measurement of steam flow. STATE HOW steam flow rate is determined by the process computer for the determination of core thermal power. (1.0)

QUESTION 5.10 (1.25)

The reactor is in shutdown cooling with a bottom head drain temperature of 180 deg F. The only operating residual heat removal pump trips. (Reactor recirculation pumps are not running.)

a. SELECT from the following the MINIMUM elevation at which reactor water level must be maintained to ensure a FLOWPATH for natural circulation exists (ASSUME no reactor recirculation pumps or RHR pumps are running). (0.75)

(1) The elevation where the MAIN STEAM LINES are submerged.

(2) The elevation where the STEAM SEPARATOR is submerged.

(3) The elevation where the STEAM DRYER is submerged.

(4) The elevation where the CORE TOP GUIDE is submerged.

b. In addition to thermal stresses that stratification could cause on the reactor vessel and components, STATE the major concern associated with stratification in this condition. (0.5)

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

QUESTION 5.11 (1.50)

ANSWER the following questions concerning "CRITICAL POWER."

a. DEFINE "Critical Power." (0.75)

b. With the reactor at power, WHICH of the following conditions would tend to DECREASE the Critical Power level assuming all other variables remain unchanged?

(1) Reactor pressure is INCREASED.
(2) Total core flow is INCREASED.
(3) Inlet subcooling is INCREASED. (0.75)

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

QUESTION 6.01 (3.00)

Concerning the Emergency Diesel Generators (G-101, 102, 103):

a. If a diesel generator is supplying its associated bus and control power for the generator output breaker is lost, WILL that breaker trip open if a fault then occurs causing generator breaker current to exceed its instantaneous overcurrent setpoint (YES/NO)? (0.5)

b. LIST three (3) conditions that will automatically trip an emergency diesel generator if a valid LOCA signal is present with the diesel mode selector switch in "REMOTE." (DO NOT include a manual stop.) (1.5)

c. If an emergency diesel generator is running with the diesel mode selector switch in REMOTE and ALL starting air is completely lost, STATE whether that emergency diesel generator WILL or WILL NOT shutdown in response to the following conditions.

   1. The operator depresses the control room manual stop pushbutton for that diesel. (0.5)

   2. A valid automatic shutdown signal condition occurs for that diesel. (ASSUME an AUTO-START condition does not exist). (0.5)

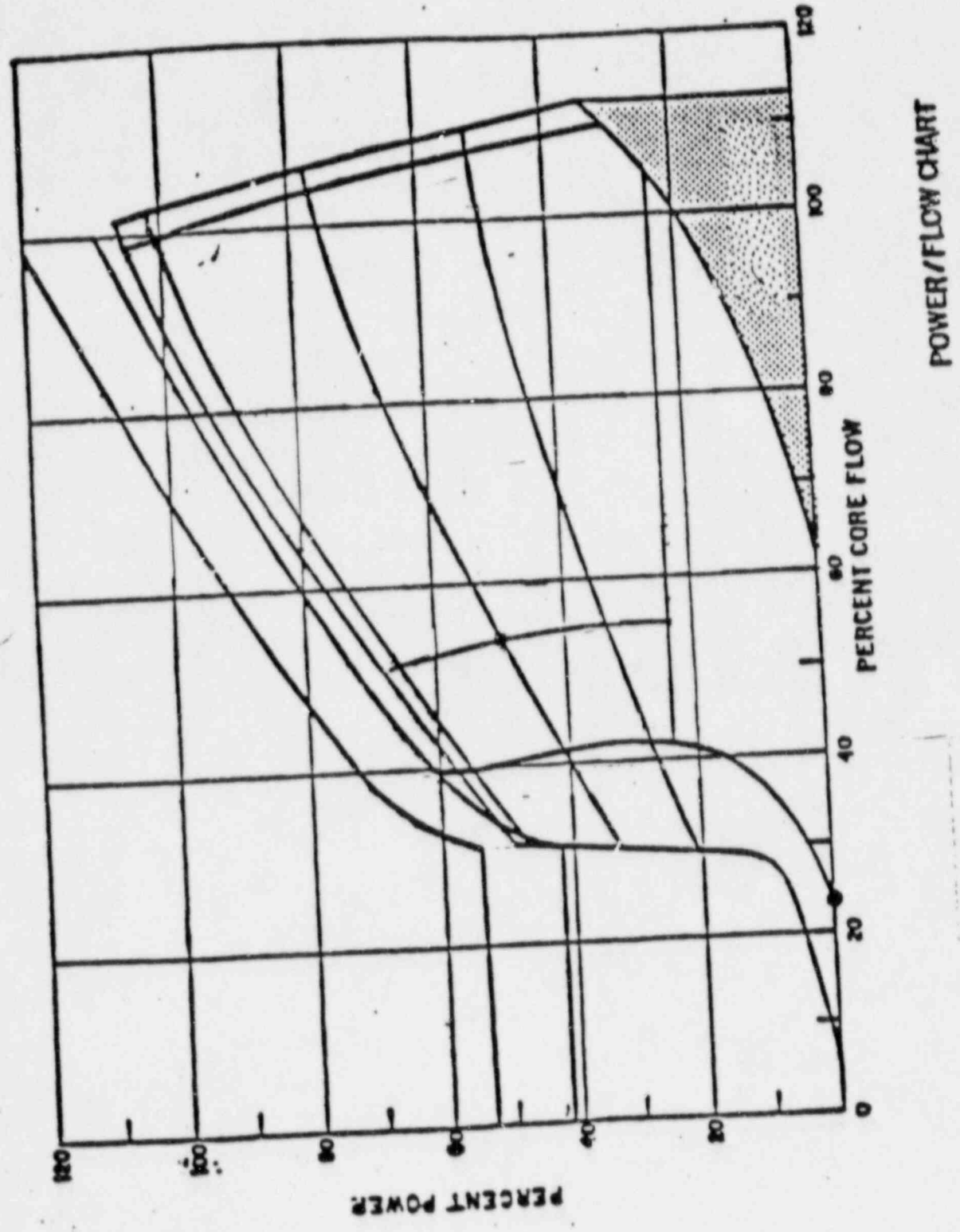
QUESTION 6.02 (1.50)

The reactor is operating at rated core thermal power and rated total core flow. Both reactor recirculation pumps trip. The reactor does not scram.

a. STATE the approximate power level at which the reactor will stabilize. (Refer to Figure 6.02 as necessary.) (0.5)

b. Recirculation loop temperatures must be within 50 deg F of each other prior to startup of either reactor recirculation pump. STATE the basis for this limit placed on the temperature differential. INCLUDE in your statement the vessel components/regions that are most limiting. (1.0)

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



POWER/FLOW CHART

QUESTION 6.03 (2.00)

The reactor is shutdown. Both loops of RHR inadvertently initiate in the LPCI mode and inject at rated flow. ANSWER the following questions concerning fuel zone level instrumentation and indication at these conditions.

a. Fuel zone level indication would indicate (UPSCALE/DOWNSCALE). (0.5)

b. Fuel zone instrument sensed differential pressure would (INCREASE/DECREASE/NOT BE AFFECTED). (0.75)

c. Fuel zone instrument variable leg pressure subsequently (INCREASES/DECREASES/REMAINS THE SAME) as a result of LPCI flow. (0.75)

QUESTION 6.04 (1.75)

The reactor has been operating for a month at rated core thermal power. A loss of all AC power occurs causing the reactor to scram. The decision to implement the steam condensing mode of RHR has been made.

a. STATE whether the steam condensing capability of two (2) RHR heat exchangers (IS/IS NOT) adequate to accommodate ALL of the reactor's decay heat immediately after the scram. (0.5)

b. The "A" loop of RHR and the RCIC pump are now operating in the steam condensing mode. The "A" RHR Heat Exchanger (HX) pressure control valve (PCV-003A) and level control valve (PCV-007A) controllers are in AUTOMATIC. INITIALLY ALL RCIC pump flow is condensate from the "A" RHR HX. If the reactor operator then doubles RCIC pump flow to rated flow, STATE whether the RCIC pump WILL or WILL NOT cavitate. EXPLAIN your answer. (1.25)

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

QUESTION 6.05 (3.00)

With regard to Low Pressure Coolant Injection (LPCI) system.

a. WHAT signals cause an automatic initiation of LPCI? (DO NOT include a manual initiation.) (1.0)

b. DESCRIBE the interlocks which must be satisfied in order to divert injection from the reactor to containment spray with a LPCI initiation signal present with the RHR inboard injection valves (MOV-037A/B) OPEN. (1.0)

c. LIST the two (2) modes of RHR system operation that require operator action to realign the system for LPCI operation if a valid LPCI initiation signal occurs. (1.0)

QUESTION 6.06 (3.00)

Concerning the Standby Liquid Control (SBLC) System:

a. LIST four (4) SBLC System indications available in the control room to confirm SBLC initiation/injection. (1.0)

b. EXPLAIN WHY a too rapid SBLC system injection rate is undesirable. (1.0)

c. DESCRIBE WHERE the SBLC system physically discharges in the reactor vessel relative to the core plate. (ABOVE/BELOW) (0.5)

d. WILL the SBLC pump, if running, automatically trip on any SBLC storage tank low level condition (YES/NO)? (0.5)

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

QUESTION 6.07 (2.75)

Concerning operation of the APRM flow units:

a. If the OUTPUT of one APRM flow unit fails UPSCALE, SELECT WHICH one of the following automatic response should occur. (1.0)

(1) rod block but no half scram
(2) half scram but no rod block
(3) rod block and half scram
(4) no rod block and no half scram.

b. If the OUTPUT of one APRM flow unit fails DOWNSCALE, SELECT WHICH one of the following automatic response should occur. (1.0)

(1) rod block but no half scram
(2) half scram but no rod block
(3) rod block and half scram
(4) no rod block and no half scram.

c. From the following list, SELECT the process flow that is actually measured by the flow transmitters that provide flow signals to the Recirculation Flow Units.

(1) total core flow
(2) recirculation loop "Driving" (pump) flow
(3) recirculation loop jet pump flow (0.75)

QUESTION 6.08 (2.50)

A loss of offsite power has occurred concurrently with a valid loss of coolant accident (LOCA) signal. Diesel generator G-101 properly autostarts.

a. SPECIFY the proper order in which the following components sequence onto emergency bus 101. INCLUDE the correct time delay:

(1) core spray pump
(2) RBSVS/CRAC water chiller
(3) RHR pump
(4) service water pump (1.5)

b. For the component start time delays, STATE the EVENT in the emergency diesel start/load sequence that initiates the timing sequence (i.e., serves as "time zero" for the time delay setpoint). (1.0)

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

QUESTION 6.09 (2.50)

The Reactor Building Normal Ventilation System (RBNVS) is in service with a normal operating supply and exhaust fan lineup. The reactor operator must shift operating exhaust fans to allow maintenance.

a. DESCRIBE which RBNVS fan(s) and/or damper(s) will automatically respond to attempt to control reactor building to outside air differential pressure when the additional exhaust fan is STARTED. INCLUDE HOW they respond. (1.5)

b. If reactor building internal pressure inadvertently becomes greater than outside air pressure during this shift of exhaust fans, DESCRIBE HOW the following components will respond (TRIP/REMAIN RUNNING):

(1) previously running reactor building supply fans (0.5)

(2) previously running reactor building exhaust fans (0.5)

QUESTION 6.10 (1.50)

Concerning the suppression pool to drywell vacuum breakers:

a. If two suppression pool to drywell vacuum breakers in series (e.g., RV93A and RV93B) were stuck open, DESCRIBE HOW this could cause primary containment to fail if a LOCA were to occur. (1.0)

b. CAN these valves be operated from the control room? (YES/NO) (0.5)

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

QUESTION 6.11 (1.50)

The reactor is in Operational Condition 2 with the reactor water cleanup (RWCU) system in service. The RWCU outboard containment isolation valve (1G33*M034) suddenly auto-isolates. The RWCU inboard containment isolation valve (1G33*M033) remains open.

a. STATE ALL possible conditions (i.e., signals) that could cause this particular isolation response, assuming the isolation logic for BOTH valves has properly functioned (setpoints NOT required). (0.5)

b. Assuming 1G33*M034 remains closed, STATE WHAT sampling requirements are now imposed to meet Technical Specification Requirements. (1.0)

(***** END [illegible] *****)

QUESTION 7.01 (2.50)

Procedure SP 22.001.01, "Startup-Cold Shutdown to 20 Percent," places administrative restrictions upon reactor operation. Concerning these restrictions:

a. STATE the minimum permissible stable period. (0.5)

b. STATE the purpose for limiting the maximum permissible control rod drive (CRD) hydraulic system charging water header pressure to 1510 psig. (1.0)

c. STATE the maximum allowable interval (time) at which heatup rate must be verified to be within limits. (0.5)

d. STATE the purpose for ensuring PRESSURE SET is set above reactor pressure before condenser vacuum increases above 7" Hg. (0.5)

QUESTION 7.02 (3.00)

The reactor has been operating for several days at 4.5% of rated power. The shift chemist notifies you that primary coolant chemistry analysis indicates that the specific activity is 0.3 microcuries per gram dose equivalent I-131.

a. STATE ALL immediate actions, if any, that you are required to take. (1.0)

b. If this level of activity in the primary coolant persists for greater than 48 hours, procedure requires the reactor be shutdown with MSIVs closed. STATE the reason for requiring MSIVs to be closed. (1.0)

c. STATE the two (2) symptoms of fuel cladding failure that require the immediate actions of SP 29.008.01, "Fuel Cladding Failure." (1.0)

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

QUESTION 7.03 (2.50)

The reactor was in hot standby with a bottom head drain temperature of 520 deg F. The high pressure coolant injection (HPCI) system auto initiated on a valid initiation signal while the reactor core isolation cooling (RCIC) system remained in standby. Based upon the responses of these two systems alone, SPECIFY WHICH entry condition(s) was/were met AND also WHICH procedure(s) should have been entered. ASSUME the HPCI and RCIC systems are properly aligned in standby for automatic initiation and are fully operable. (2.5)

QUESTION 7.04 (2.50)

In accordance with SP 12.012.01, "Radiation Work Permits" (RWPs),

a. LIST the TWO (2) individuals by title that can sign for final approval of an RWP. (1.0)

b. STATE the color of a correctly posted RWP copy at the job site. (0.5)

c. To perform his rounds, an equipment operator requires access to an unlocked room that has been designated as a "Radiation Area" by the health physics department. STATE whether an RWP WILL or WILL NOT have to be INITIATED. (ASSUME the health physics department has determined there is no loose surface or airborne contamination.) (0.5)

d. STATE the TYPE of RWP used by operators in performing surveillances during routine operating conditions. (0.5)

QUESTION 7.05 (2.00)

Adequate core cooling is one of the safety functions that the emergency procedures strive to maintain. If reactor water level cannot be maintained above the top of the active fuel, LIST in the order of preference the remaining two (2) forms (mechanisms) of core cooling that the emergency procedures will attempt to establish. (2.0)

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

QUESTION 7.06 (1.75)

Procedure SP 22.004.01, "Operation Between 20% and 100% Power," in the "Limitations and Actions" section, requires that PCIOMR (Preconditioning Interim Operating Management Recommendation) be followed.

a. STATE WHAT adverse condition can occur if PCIOMR is not followed at high power. (0.5)

b. STATE WHO is responsible for monitoring and supervising the details of the fuel preconditioning process. (0.5)

c. BRIEFLY DESCRIBE the fuel preconditioning process. (0.75)

QUESTION 7.07 (3.00)

a. The reactor is operating at rated core thermal power. The main condenser low vacuum alarm ("COND VACUUM LO") is received.

   1. STATE the immediate action(s) required by emergency procedure SP 29.012.01, "Loss of Condenser Vacuum." (0.6)

   2. LIST the four (4) automatic actions that will be initiated by a low vacuum condition. INCLUDE setpoints. (Assume a complete loss of condensor vacuum occurs.) (1.6)

b. LIST two (2) reasons WHY procedures do not allow condenser air removal pumps to maintain condenser vacuum when reactor power is above 4%. (0.8)

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

QUESTION 7.08 (3.00)

Concerning Primary Containment:

a. The attached figure PC/T-1 is an excerpt from emergency procedure SP 29.023.03, "Primary Containment Control." STATE WHICH primary containment design parameter could be exceeded (design value not required) if containment sprays were initiated with initial drywell temperature at 300 deg F and initial drywell pressure at 20 psig. INCLUDE in your discussion HOW such a condition could occur if containment sprays were initiated. (2.5)

b. For Operational Condition 1, under WHAT condition is the Technical Specifications maximum allowable drywell pressure less than 1.69 psig? (INCLUDE a numerical value for this condition.) (0.5)

QUESTION 7.09 (2.00)

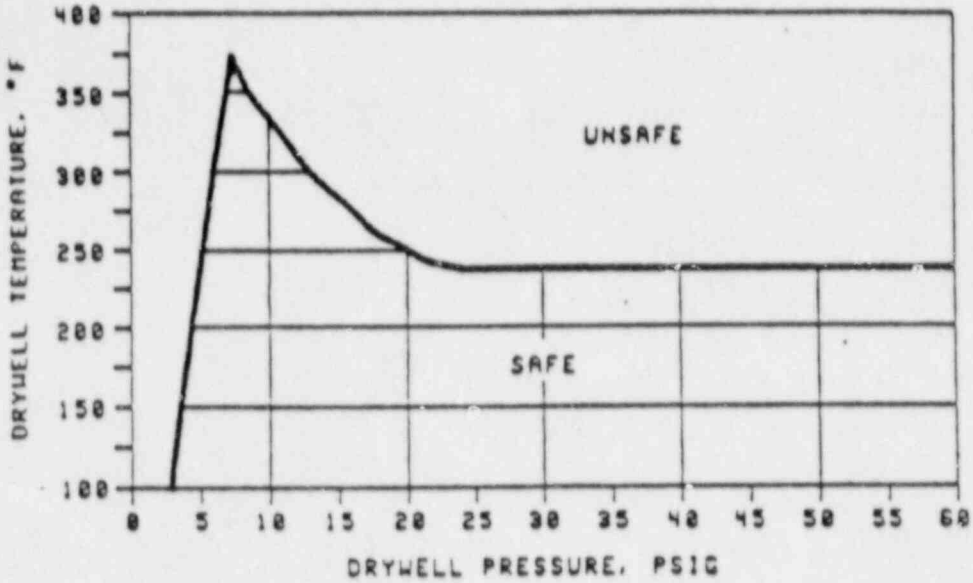
The reactor is operating at rated core thermal power. The reactor building closed loop cooling water (RBCLCW) head tank "low-low" level alarm ("RBCLCW HD TK A(B) LEV LO-LO") is received. RBCLCW has isolated from all nonsafety loads.

a. LIST ALL immediate actions that are required by SP29.017.01, "Loss of RBCLCW." (1.5)

b. STATE HOW LONG continued operation of the Reactor Recirc MG sets is allowed in this condition. (0.5)

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

FIGURE PC/T-1
DRYWELL SPRAY INITIATION LIMIT



QUESTION 7.10 (2.75)

A loss of ALL AC power has occurred. The "immediate actions" of emergency procedure SP 29.015.02, "Loss of All AC Power" are complete. The "subsequent actions" are now being performed. Reactor water level has been stabilized at +30" using the RCIC system alone (HPCI has been secured in accordance with the procedure).

a. STATE the reason WHY the procedure instructs the operator to depressurize the reactor as quickly as possible. (1.0)

b. STATE the reason WHY the procedure instructs the operator to secure the RCIC vacuum pump though RCIC is feeding the reactor pressure vessel to maintain water level. (1.0)

c. STATE one (1) reason WHY the procedure instructs the operator to PREVENT automatic suction transfer of the RCIC pump to the suppression pool. (0.75)

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

QUESTION 8.01 (3.00)

Standing Order Number 30 provides specific guidance concerning reactor operation while the 5% Low Power License is in effect. Concerning the requirements of this standing order:

a. STATE the MAXIMUM allowable "average core power" for ANY transient. (0.5)

b. DESCRIBE HOW "average core power" is to be determined from plant instrumentation. (1.0)

c. If the main turbine generator is being rolled for testing or is synchronized to the grid, STATE ALL restrictions on reactor power. SPECIFY WHICH indications are to be used for reactor power determination, INCLUDING the locations of the indications to be used. (1.5)

QUESTION 8.02 (1.00)

The Technical Specifications specify the maximum allowable mismatch between reactor recirculation pump speeds. STATE the Technical Specification BASIS for administratively controlling this mismatch in speeds. (1.0)

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

QUESTION 8.03 (3.00)

Procedure SP 21.001.01, "Shift Operations," requires that only "ACTIVE" licensed operators may assume the watch.

a. SPECIFY ALL quarterly watch standing requirements that must be met for a licensed operator to maintain his "active" status. (1.0)

b. Is a newly licensed individual considered "active" if he has not stood the required number of watches to maintain an "active" status? (YES/NO) (0.5)

c. When the reactor changes from Operational Condition 3 to Operational Condition 4, STATE which licensed individual(s), by title is/are no longer required to be a part of the shift complement. (1.0)

d. STATE which document must be signed by an individual if he is to receive credit towards maintaining his "active" status when he completes a watch in the capacity of his license. (0.5)

QUESTION 8.04 (2.50)

Concerning Procedure SP 12.011.01, "Station Equipment Clearance Permits (SECPs)":

a. STATE to which individual(s) the Watch Engineer can delegate his authority to APPROVE an SECP. (0.5)

b. STATE whether or not the second individual performing the independent verification of the placement of hold-off tags for an SECP is required to accompany the individual performing the SECP. (0.5)

c. DESCRIBE the condition under which the independent verification of an SECP on SAFETY related systems/components may be waived. (0.5)

d. STATE WHICH individual has the authority to WAIVE the requirement for an independent verification of an SECP involving SAFETY related systems/components. (0.5)

e. STATE the criterion for determining whether or not a lifted lead requires an SECP hold-off tag in addition to a lifted lead/jumper tag. (0.5)

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

QUESTION 8.05 (3.00)

Concerning Procedure SP 12.009.03, "Report of Abnormal Conditions (RAC) and Limiting Conditions for Operations (LCO)":

a. STATE WHICH individual(s) may INITIATE an RAC. (0.5)

b. STATE WHICH individual is to perform the shift technical advisor's (STA) review of an RAC if the STA is not required to be on shift and no STA is assigned to the shift. (0.5)

c. If a motor operated valve (MOV) governed by Technical Specifications is manually torqued and was consequently declared inoperable, STATE WHAT must be done to the valve to return it to an operable status. (0.5)

d. For the following situations, STATE whether a one-hour NRC notification (IS REQUIRED/IS NOT REQUIRED):

1. An inadvertent initiation of HPCI occurs that was NOT caused by a valid initiation signal. Some water was injected to the reactor pressure vessel but the transient did not require nor result in a scram. (0.5)

2. An EHC pressure regulator failure causes a high reactor pressure scram. Reactor pressure recorders indicate that reactor pressure peaked for several seconds at 1355 psig during the transient. (0.5)

3. An excessive primary leak rate leads the watch engineer to declare an "Unusual Event." (0.5)

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

QUESTION 8.06 (3.00)

Reactor power is 10%. The reactor mode switch is in "STARTUP." Control rod scram time testing has been completed for all control rods. A review of the testing results reveals that control rods 42-43, 34-35, and 22-27, all three of which are currently at position 48, had scram insertion times (as defined in the Technical Specifications) of 7.1 seconds.

ASSUME:
1. All 137 control rods were initially considered operable.
2. All control rods other than 42-43, 34-35, and 22-27 had scram insertion times of less than 5.0 seconds.
3. All Technical Specifications requirements for averaged scram times are satisfied.

Referring as necessary to the attached excerpt from the SNPS Technical Specifications and the attached diagram of the rod coordinate matrix, ANSWER the following questions:

a. DESCRIBE WHAT action(s) are required to be taken by Technical Specifications. REFERENCE the Technical Specification(s) (by number) that require(s) the action(s). (2.5)

b. Based solely upon the conditions above, STATE whether the reactor mode switch (CAN/CANNOT) be taken to the "RUN" position without violating Technical Specifications, once all required actions have been taken. (0.5)

QUESTION 8.07 (2.00)

Using the Emergency Plan Implementing Procedure (EPIP 1-0) CLASSIFY the following events. INCLUDE the classification identifier number. (This is NOT the event category number.)

a. The RPS system initiates a full scram. Reactor shutdown does not occur. The standby liquid control system is initiated and successfully terminates the transient. (1.0)

b. A total loss of service water occurs (both loops). The reactor is consequently shutdown to meet Technical Specifications requirements. (1.0)

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

QUESTION 8.08 (2.00)

Concerning Procedure SP 12.006.01, "Station Procedures - Preparation, Review, Approval, Change, Review, and Cancellation":

a. If a change is needed to correct a procedure deficiency to make the procedure conform to approved station design documents, is the resulting Station Procedure Change Notice (SPCN) classified as a "MAJOR" or "MINOR" change? (0.5)

b. STATE the maximum amount of time a Temporary Plant Change Note (TPCN) is valid once it is in effe (0.5)

c. STATE WHICH individual, by title, must give the final approval to make a TPCN involving safety related equipment a permanent procedure revision. (0.5)

d. STATE HOW MANY individuals holding SRO licenses MUST approve a TPCN BEFORE the TPCN can be placed into effect (i.e., used). (0.5)

QUESTION 8.09 (2.00)

As an Emergency Director for Shoreham's Emergency plan:

a. STATE three (3) duties that cannot be delegated. (1.5)

b. STATE the time limit for notifying New York State and Suffolk County officials once an emergency event has been declared. (0.5)

QUESTION 8.10 (1.50)

Concerning the basis for Technical Specifications limits upon primary containment parameters during normal operation:

a. STATE the Technical Specification basis for the maximum allowable drywell air temperature. (0.75)

b. STATE the Technical Specification basis for the maximum allowable suppression pool water level. (0.75)

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

QUESTION 8.11 (2.00)

Concerning Procedure SP 21.003.01, "Operations Reports":

a. DOES a reactor trip report have to be initiated for a MANUAL UNPLANNED scram (YES/NO)? (0.5)

b. Prior to restart, MUST the ROC review the reactor trip report if the cause of the scram has not been positively identified (YES/NO)? (0.5)

c. STATE WHAT the watch engineer's final signature on a Reactor Trip Report indicates. (0.5)

d. STATE WHICH individual, by title, must perform the FINAL review of this report as one of the prerequisites for commencing the subsequent reactor startup. (0.5)

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

ANSWER 5.01 (3.00)

a. 1. void coefficient [+0.25]
   2. Doppler coefficient [+0.25]

b. Reactor power wil' first increase greater than 4.5% [+0.25] as xenon is initially depleted faster than it is produced [+0.25]. Reactor power will subsequently stabilize at less than 4.5% [+0.25] as the production of xenon from the decay of iodine restores xenon concentration to a new higher equilibrium value [+0.25].

c. (Reactor period would) LESS THAN [+0.5]. Increased core age results in a higher fraction of core power being produced by Pu-239 [+0.25] which reduces the value of BETA-EFFECTIVE [+0.5]. From the period equation, as BETA-EFFECTIVE decreases, for a given reactivity insertion, the resultant reactor period decreases [+0.25].

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 11, LO CB and Lesson 15, LO CD, CH.

2. GE: Reactor Theory, Chapters 3, 4, and 6.

292003K106 292004K114 292006K105 292006K106 ...(KA'S)

ANSWER 5.02 (2.50)

a. (Control rod) 18-35 [+0.5]

b. (Control rod) 22-35 [+0.33]. The rod shadowing effects of adjacent rods [+0.33] will be more significant in dampening radial power because of the higher local control rod density relative to the control rod tip (of control rod 22-35) [+0.34].

c. When a shallow rod is withdrawn a notch, the void fraction in the adjacent fuel channels increases throughout the entire boiling length [+0.33]. Though local power will increase where the rod tip was withdrawn [+0.33], it is possible for the negative reactivity of the increased voiding to be the dominate effect (causing net total power to decrease) [+0.34].

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 13, LO CB, CC.

2. GE: Reactor Theory, Chapter 5.
292005K104 292005K112 292008K119 ...(KA'S)

ANSWER 5.03 (2.00)

DM = 1-Keff; CR1 (1-Keff 1) = CR2 (1-Keff 2)
cps) (0.04) = (1000 cps) SDM
.1 = 0.008 = 0.8 percent [+2.0]
(Note : SDM = 1 - Keff / Keff ~~may also~~ is an alternate equation for SDM)

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 16, LO CB.
292002K113 ...(KA'S)

ANSWER 5.04 (2.00)

a. Doubling time = 100/1.44 = 69.4 seconds [+0.5]

b. Rod block will occur at 34/40 of full scale on range 3 (also allow 35/40 of full scale) [+0.5]

P(o) = 10; P(t) = 34
P(t) = 10e**(t/period)
34 = 10e**(t/100 sec)
t = 100 ln(34/10) = 122 sec (2 min 2 sec) [+1.0]
(ALTERNATE ANSWER: 125 sec or 2 min 5 sec IF P(t) = 35 was used)

REFERENCE

1. Shoreham: HL-602-SH1, LO B.2.

2. Shoreham: HL-900-SH1, Lesson 15, LO CA.
215003K401 292003K108 292003K109 ...(KA'S)

ANSWER 5.05 (2.50)

a. more
b. decrease
c. increase
d. less
e. increase

[+0.5] each

REFERENCE

1. Shoreham: HL-900-SH1, Lesson 12, LO CG.
292005K109 ...(KA'S)

ANSWER 5.06 (3.00)

a. Reactor power will rapidly increase due to the pressure increase [+0.5]. Power will then decrease due to the TCV fast closure scram [+0.5].

b. Reactor pressure will rapidly increase due to the rapid closure of the TCVs [+0.5]. Pressure will then decrease due to the scram and the opening of the bypass valves which will then attempt to maintain reactor pressure at 920 psig [+0.5].

c. Reactor water level will initially drop due to collapsing of voids [+0.5]. The feed control system will respond to increase level and level should then rise to the level controller setpoint (level may overshoot causing feed pumps to trip) [+0.5].

REFERENCE

1. Shoreham: HL-900-SH1; HL-657-SH1, LO C3.
241000K101 241000K102 241000K103 ...(KA'S)

ANSWER 5.07 (2.00)

Assumptions:

1. HPCI isolation setpoint = 100 psig [+0.5]

2. maximum allowable heatup rate = 90 deg F/hr [+0.5]

3. 40 psig = 54.7 psia; Tsat = 287 deg F
100 psig = 114.7 psia; Tsat = 338 deg F
[+0.5]

Time to clear setpoint = ((338 - 287)deg F)/(90 deg F/hr)
= 51 deg F/(90 deg F/hr)
= 0.57 hrs (34 minutes +/- 3 minutes)
[+0.5]

REFERENCE

1. Shoreham: HL-901-SH1, Lesson 2, LO CB; HL-202-SH1, LO CG.
206000K402 293003K123 ...(KA'S)

ANSWER 5.08 (2.25)

a. The basis is to ensure the reactor coolant pressure boundary integrity is maintained [+0.5]. ASME code allows 110% (1375 psig) overpressurization of design [+0.5]. Because reactor pressure is measured at the steam dome, 50 psi is subtracted from 110% design value to account for the head of water above the lowest point in the vessel [+0.5].

b. minimum critical power ratio (MCPR) [+0.75]

REFERENCE

1. Shoreham: HL-904-SH1, Lesson 2, LO CB, CC.

2. Shoreham: Technical Specifications, Section 2.0.

293009K105 ...(KA'S)

ANSWER 5.09 (3.00)

a. 1. fractional increase in reactor recirc pump speed: 45/30 = 1.5 [+0.25]

2. (pump laws state power increases by the cube of the fractional increase in speed): (1.5)**3 = 3.375 [+0.75]

3. power consumed by 2 pumps at 45% speed = (0.18 MW x 2)(3.375) = 1.21 MW [+1.0]

b. The process computer takes steam flow to be the sum of feedwater flow [+0.7] and control rod drive cooling water flow [+0.3].

REFERENCE

1. Shoreham: HL-901-SH1, Lesson 3, LO CB.

2. GE: HTFF, Chapter 7.

293006K108 293007K111 293007K113 ...(KA'S)

ANSWER 5.10 (1.25)

a. (2) [+0.75]

b. The reactor pressure vessel could generate steam due to boiling in the upper region though coolant temperature indications indicate adequate subcooling. [+0.5]

REFERENCE

1. Shoreham: HL-901-SH1, Lesson 5, LO CB.

2. GE: HTFF, Chapter 8.
293008K134 293008K135 ...(KA'S)

ANSWER 5.11 (1.50)

a. The assembly (bundle) power that would cause the onset of transition boiling [+0.75].

b. (1) [+0.75]

REFERENCE

1. Shoreham: HL-904-SH1, Lesson 1, LO CA, CF.
293009K117 293009K122 293009K123 293009K124 ...(KA'S)

ANSWER 6.01 (3.00)

a. no [+0.5]

b. 1. overspeed [+0.5]
   2. generator phase differential [+0.5]
   3. generator overcurrent [+0.5]

c. 1. will not [+0.5]
   2. will not [+0.5]

REFERENCE

1. Shoreham: HL-307-SH1, LO A, C.4, D.
264000K106 264000K601 ...(KA'S)

ANSWER 6.02 (1.50)

a. approximately 48% [+0.5] (allow 43% - 51%)

b. To prevent undue thermal stress [+0.34] on the vessel nozzles [+0.33] and bottom head region [+0.33].

REFERENCE

1. Shoreham: HL-120-SH1, LO F.
202001K102 202001SG1 202001SG5 ...(KA'S)

ANSWER 6.03 (2.00)

a. upscale [+0.5]
b. decrease [+0.75]
c. increases [+0.75]

REFERENCE

1. Shoreham: HL-621-SH1, LO I.D.5.
216000K105 ...(KA'S)

ANSWER 6.04 (1.75)

a. IS NOT [+0.5]

b. The RCIC pump WILL NOT cavitate [+0.25]. CST inventory will maintain RCIC pump suction pressure [+1.0].
(Note. RHR heat exchanger flow will not increase, nor will RCIC suction pressure affect heat exchanger level control valve position)

REFERENCE

1. Shoreham: HL-121-SH1, LO CC.

2. Shoreham: 23.121.01.
217000A101 217000K101 217000K105 217000SG1 ...(KA'S)

ANSWER 6.05 (3.00)

a. Hi drywell pressure 1.69 psig OR [+0.5]

Lo Lo Rx water level -132.5" [+0.5]

(NOTE: Do not penalize if RPV low pressure permissive of 338 psig is also listed.)

b. 1. containment spray valve manual override keyswitch in "MANUAL" [+0.5]

2. containment spray valve accident control switch in "MANUAL" [+0.5] (until seal-in status light is lit)

c. 1. shutdown cooling mode [+0.5]

2. fuel pool cooling mode [+0.5]

REFERENCE

1. Shoreham: HL-204-SH1, LO CC.
226001K403 ...(KA'S)

ANSWER 6.06 (3.00)

a. 1. squib valve continuity circuit indicator lamp extinguishes

2. squib valve loss of continuity annunciator

3. SBLC pump discharge pressure greater than reactor pressure

4. SBLC pump running indication (ON)

5. SBLC storage tank level decreasing

6. squib valve continuity ~~ampmeters~~ amp meters.

Any four (4) [+0.25] each, +1.0 maximum.

b. A too rapid injection rate could cause insufficient mixing and uneven concentrations of boron circulating in the core [+0.5] leading to power oscillations ("chugging") [+0.5].

c. below (the core plate) [+0.5]

d. No [+0.5]

REFERENCE

1. Shoreham: HL-123-SH1, LO B.1, F.
211000K106 211000K403 211000K405 211000K506 ...(KA'S)

ANSWER 6.07 (2.75)

a. (1) (rod block) [+1.0]

b. (3) (rod block and half scram) [+1.0]

c. (2) (recirculation loop "Driving" flow) [+0.75]

REFERENCE

1. Shoreham: HL-603-SH1, LO I.C.
215005K110 215005K116 215005K607 ...(KA'S)

ANSWER 6.08 (2.50)

a.

1.	RHR pump (3)	2 second T.D.	[+0.2]
2.	core spray pump (1)	7 second T.D.	[+0.2]
3.	service water pump (4)	12 second T.D.	[+0.2]
	RBSVS/CRAC water chillers (2)	12 second T.D.	[+0.2]

(Point awards above are for T.D. values only. [+0.7] for correct order, no partial credit)

b. The timing sequence is initiated by the closing of the diesel generator output breaker. [+1.0]

REFERENCE

1. Shoreham: HL-307-SH1, LO B.1.b.
262001A304 262001K301 264000K506 ...(KA'S)

ANSWER 6.09 (2.50)

a. The exhaust damper [+0.5] of the operating supply fan [+0.5] will modulate further open [+0.5].

b. (1) trip [+0.5]
(2) remain running [+0.5]

REFERENCE

1. Shoreham: HL-405/418-SH1, LO I.D.5.
261000K101 261000K401 ...(KA'S)

ANSWER 6.10 (1.50)

a. Primary containment could be overpressurized [+0.5] because of steam bypassing the suppression pool, pressurizing containment [+0.5].

b. Yes [+0.5]

REFERENCE

1. Shoreham: HL-654-SH1, LO CB.
223001K405 223001K501 223001K503 ...(KA'S)

ANSWER 6.11 (1.50)

a. 1. RWCU nonregenerative heat exchanger outlet (filter demineralizer inlet) temperature high [+0.25]

2. standby liquid control (SLC) system initiation [+0.25]

b. (In-line) conductivity must be sampled at least once every four hours[+1.0]. (continuous conductivity indication has been lost)

REFERENCE

1. Shoreham: Technical Specifications 3.4.4.

2. Shoreham: HL-709-SH1, LO CE, CF, CG.
204000K404 204000K507 204000SG11 204000SG5 ...(KA'S)

ANSWER 7.01 (2.50)

a. 60 seconds [+0.5]

b. To prevent control rod drive mechanism damage [+0.5] during a scram [+0.5].

c. 15 minutes [+0.5]

d. To prevent inadvertent bypass valve operation [+0.5].

REFERENCE

1. Shoreham: SP 22.001.01; HL-106-SH1 LO J.
216000SG1 241000SG1 ...(KA'S)

ANSWER 7.02 (3.00)

a. 1. increase RWCU flow to maximum [+0.5]

2. increase primary coolant sampling frequency to at least once every four hours [+0.25] until specific activity is below Technical Specifications limits [+0.25]

b. Prevents release of activity [+0.5] should a steam line rupture occur [+0.5].

c. Primary Coolant activity greater than either: ~~Primary coolant specific activity shall be limited to:~~

1. or 0.2 microcuries per gram dose equivalent I-131 [+0.5]

2. 100/$\bar{E}$ microcuries per gram [+0.5]

REFERENCE

1. Shoreham: SP 29.008.01.
295017AK20 295017SG10 ...(KA'S)

ANSWER 7.03 (2.50)

1. the entry condition was high drywell pressure >1.69 psig [+1.0]

2. the emergency procedures which should have been entered were:

   a. "Emergency Shutdown" (SP 29.010.01) [+0.5]
   b. "Reactor Pressure Vessel (RPV) Control" (SP 29.023.01) [+0.5]

   c. "Primary Containment Control" (SP 29.023.03) [+0.5]

REFERENCE

1. Shoreham: HL-944-SH1, LO I.B.

2. Shoreham: HL-944-SH2, LO I.B.

295024SG11 ...(KA'S)

ANSWER 7.04 (2.50)

a. 1. watch engineer [+0.5]
   2. watch supervisor [+0.5]

b. white [+0.5]

c. will not [+0.5]

d. extended (RWP) [+0.5]

REFERENCE

1. Shoreham: SP 12.012.01

294001K103 ...(KA'S)

ANSWER 7.05 (2.00)

1. spray cooling (ALTERNATE ANSWERS: steam cooling with injection of make up water to the RPV --OR-- one core spray pump spraying at rated flow) [+1.0]

2. steam cooling [+1.0]

REFERENCE

1. Shoreham: HL-944-SH3.
295031EK10 295031EK30 ...(KA'S)

ANSWER 7.06 (1.75)

a. fuel clad cracking [+0.5] (ALTERNATE ANSWERS: pellet-clad interaction --OR-- fuel failure)

b. the reactor engineer [+0.5] (ALTERNATE ANSWER: the Reactor Engineering Department) or the Shift Technical Advisor (STA)

c. the rate of increase in reactor power (LHGR) is limited (controlled) [+0.75]

REFERENCE

1. Shoreham: HL-904-SH1, Lesson 4, LO CB, CC.

2. Shoreham: SP 22.004.01.
239009K136 294001A103 ...(KA'S)

ANSWER 7.07 (3.00)

a. 1. commence power reduction (in accordance with SP22.004.01, "Operation Between 20% and 100% Power") [+0.6]

2. a. main turbine trip [+0.2] at 22.5" Hg vac [+0.2]

b. reactor feed pump (turbine) trip [+0.2] at 20" Hg vac [+0.2]

c. MSIV (/main steam drain) isolation [+0.2] at 8.5" Hg vac [+0.2]

d. turbine bypass valve (TBV) isolation [+0.2] at 7" Hg vac [+0.2]

b. 1. to avoid hydrogen explosion (above 4% power) [+0.4]

2. the level of radioactivity in the noncondensable condenser gases is significant (above 4% power) [+0.4] (ALTERNATE ANSWER: condenser air removal pump exhaust is not treated prior to release)

REFERENCE

1. Shoreham: SP 29.012.01.

2. Shoreham: HL-701/714-SH1, LO E.4.
295002AK20 ...(KA'S)

ANSWER 7.08 (3.00)

a. The design maximum suppression pool to drywell differential pressure (could be exceeded) [+1.25]. The drywell could depressurize at a rate faster than the rate at which the suppression chamber to drywell vacuum relief system could equalize the resulting differential pressure [+1.25].

b. (When) drywell bulk average temperature is < 110 deg F. [+0.5]

REFERENCE

1. Shoreham: HL-944-SH3, LO I.E.
295024EK30 295028EK30 ...(KA'S)

ANSWER 7.09 (2.00)

a. 1. trip the operating RWCU pump [+0.25]

2. isolate the RWCU system from containment (close MOV's 033 and 034) [+0.25]

3. reduce reactor recirc pump speed to minimum [+0.25]

4. trip both reactor recirc pumps [+0.25]

5. initiate emergency shutdown procedure (SP 29.010.01) [+0.25]

6. trip CRD pumps after all control rods are verified inserted [+0.25]

b. 10 minutes [+0.5]

REFERENCE

1. Shoreham: SP 29.017.01.

295018SG10 295018SG11 ...(KA'S)

ANSWER 7.10 (2.75)

a. Early RPV depressurization will result in suppression pool and drywell (containment) temperatures [+0.25] and pressures [+0.25] remaining below designed limitations [+0.5].

(ALTERNATE ANSWER: to limit the total heat load [+0.5] placed upon the primary containment [+0.5])

b. (The RCIC vacuum pump is secured) to prolong [+0.5] the use of the division I battery [+0.5].

(ALTERNATE ANSWER: to reduce the load [+0.5] upon the Division I battery [+0.5])

c. 1. to slow the rate of containment temperature and pressure rise

2. to avoid failure of the RCIC turbine due to high lube oil temperatures

(Either 1. or 2. for [+0.75])

REFERENCE

1. Shoreham: SP 29.015.02.

295003AK20 295003AK30 ...(KA'S)

ANSWER 8.01 (3.00)

a. 5% [+0.5]

b. (Average core power is) the average of all APRM readings. [+1.0]

c. The highest reading APRM [+0.25] is not to exceed 5% [+0.25]. The average APRM reading (average core power) [+0.25] is not to exceed 4.75% [+0.25]. APRM readings are to be taken at panel 608 [+0.5] (also allow "backpanel").

REFERENCE

1. Shoreham: Standing Order No. 30.
294001A103 ...(KA'S)

ANSWER 8.02 (1.00)

The limits ensure an adequate core flow coastdown [+0.5] from either recirculation loop following a LOCA [+0.5].

(ALTERNATE ANSWER: to comply with ECCS LOCA design criteria [+1.0])

REFERENCE

1. Shoreham: Technical Specifications, Basis 3/4.4.1.
202002SG6 ...(KA'S)

ANSWER 8.03 (3.00)

a. A licensed operator must perform his duties (of his license - RO/SRO) for seven 8-hour shifts [+0.5] or five 12-hour shifts [+0.5].

b. yes [+0.5]

c. the watch supervisor [+0.34] and either the nuclear station operator [+0.33] or the assistant nuclear station operator [+0.33]

d. the shift turnover sheet [+0.5]

REFERENCE

1. Shoreham: SP 21.001.01.

2. Shoreham: SP 21.002.01.
294001A103 ...(KA'S)

ANSWER 8.04 (2.50)

a. the (on-duty) watch supervisor [+0.5]

b. is not required [+0.5]

c. (Independent verification may be waived) when significant radiation exposure could result (to the individual performing the verification) [+0.5].

d. The watch engineer (WE) has the authority (to waive the independent verification) [+0.5].

(ALTERNATE ANSWER: the watch supervisor (WS) has the authority if the WE has delegated to the WS SECP approval authority)

e. Any lead that in normal service could be exposed to voltages in excess of 130 volts (requires a hold-off tag). [+0.5]

REFERENCE

1. Shoreham: SP 12.011.01.

2. Shoreham: SP 12.035.01.

294000K102 ...(KA'S)

ANSWER 8.05 (3.00)

a. ANY station employee [+0.5]

b. the "management on call individual" [+0.5]

c. the valve must be electrically stroked [+0.5]

d. 1. is not required [+0.5]
   2. is required [+0.5]
   3. is required [+0.5]

REFERENCE

1. Shoreham: SP 12.009.03.

294000A103 ...(KA'S)

ANSWER 8.06 (3.00)

a. 1. control rods 42-43, 34-35, and 22-27 must all be declared INOPERABLE [+0.2] per Technical Specification 3.1.3.2 [+0.1]

   2. control rods 42-43 and 34-35 [+0.5] must be fully inserted [+0.25] and their directional control valves disarmed (either electrically or hydraulically) [+0.25] per Technical Specification 3.1.3.1 [+0.1]

   3. control rod 22-27 must be inserted at least one notch [+0.25] by drive water pressure [+0.25] or else be fully inserted [+0.25] and disarmed [+0.25] per Technical Specification 3.1.3.1 [+0.1]

b. can [+0.5]

REFERENCE

1. Shoreham: Technical Specifications, 3/4.1.3.

201003SG5 ...(KA'S)

ANSWER 8.07 (2.00)

a. Alert No. 11 [+1.0]
b. ~~Unusual Event~~ Alert No. ~~9~~ 10 [+1.0] (note: total loss of service water induces total loss of RBCCW)

REFERENCE

1. Shoreham: EPIP 1-0.
294000A116 ...(KA'S)

ANSWER 8.08 (2.00)

a. minor [+0.5]
b. 31 days [+0.5]
c. plant manager [+0.5]
d. one [+0.5]

REFERENCE

1. Shoreham: SP 12.006.01.
294000A103 ...(KA'S)

ANSWER 8.09 (2.00)

a. 1. classification of the emergency

2. directing the notification of offsite officials

3. making offsite protective action recommendations

4. making the decision to evacuate the site

5. authorization for workers to exceed 10CFR20 emergency radiation exposure limits

6. approve proposed press releases

Any three (3) [+0.5] each, +1.5 maximum.

b. 15 minutes [+0.5]

REFERENCE

1. Shoreham: SP 12.002.01.

2. Shoreham: EPIP 1-2.
294001A116 ...(KA'S)

ANSWER 8.10 (1.50)

a. Ensures that in the event of a LOCA [+0.25], the peak containment air temperature will not exceed the design basis limit of 340 deg F [+0.5].

b. Ensures that in the event of a LOCA [+0.25], the peak containment pressure will not exceed the design basis limit of 48 psig [+0.5].

REFERENCE

1. Shoreham: Technical Specifications, Basis 3/4.6.1.7 and 3/4.6.2.
223001SG6 ...(KA'S)

ANSWER 8.11 (2.00)

a. yes [+0.5]

b. yes [+0.5]

c. (The watch engineer's signature indicates) concurrence with all sections of the report [+0.5].

(ALTERNATE ANSWER: All information in the report is correct.)

d. the plant manager [+0.5]

REFERENCE

1. Shoreham: SP 21.003.01.
294001A103 ...(KA'S)