Attachment 3

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

JOHN D. LEONARD, JR.
VICE PRESIDENT - NUCLEAR OPERATIONS

MAY 20 1988

SNRC-1464

Mr. William T. Russell
Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Post NRC Written Examination Review
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Russell:

LILCO's Operator Training Division, utilizing the expertise of several Shoreham senior reactor operators (SROs), has performed a review of the NRC written examinations and answer keys. This review included both reactor operator (RO) and SRO examinations and answer keys. The attached comments, along with supporting documentation, are provided to assist you and your staff in evaluating the responses to exam questions provided by our license candidates.

If clarification is needed on any of this material, please do not hesitate to contact Mr. L. Calone, Manager, Operator Training Division, at (516) 436-4046.

Very truly yours,

John D. Leonard, Jr.
Vice President - Nuclear Operations

DRH:ck

Attachments

cc: S. Brown
Document Control Desk
F. Crescenzo
D. Lange (Region 1)

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)

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].

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)

COMMENT

Should also accept for answer (a) part 2 that the power decrease was due to rod insertion (scram). Candidate was not asked specifically what signal caused the scram, therefore, TCV fast closure scram should not be required for full credit.

REFERENCE

No applicable reference.

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 SP 22.005.01, "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.)

ANSWER 1.07 (2.50)

a. The previous shift DID EXCEED the cooldown limit [+0.5] of 90 degrees F/hr [+0.5].

(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)

COMMENT

For part a - should also accept a cooldown limit of 100°F/hr as stated in 22.005.01 or 90°F/hr as stated in 22.001.01 which is more limiting.

REFERENCE

SP 22.005.01 Pages 3 and 9
SP 22.001.01 Page 4

QUESTION 6.08
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)

ANSWER 6.08
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)

COMMENT

Also accept alternate answer to part b, EDG output breaker closes onto Bus at 90% of rated voltage. This is when the timing sequence starts.

REFERENCE

HL-307-SH1 Page 11

QUESTION 2.04 (1.50)
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.

b. The "A" loop of RHR and the RCIC pump are not 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.

ANSWER 2.04 (1.50)
ANSWER 6.04 (1.75)

a. IS NOT

b. The RCIC pump WILL NOT cavitate. CST inventory will maintain RCIC pump suction pressure.

REFERENCE

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

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

COMMENT

For part b answer key should be expanded to accept the following alternate answer. The RCIC pump will not cavitate. Both PCV's if in AUTO will open to accommodate pressure/level in the heat exchanger and corresponding flow to RCIC.

REFERENCE

SP 23.121.01 Pages 17, 18, 19, 20, 21 STEP 8.1.7.

QUESTION 2.06 (3.00)
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)

ANSWER 2.06 (3.00)
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

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)

QUESTION 2.06
QUESTION 6.06 (Continued)

COMMENT

Answer to part a should be expanded to accept

(1) continuity amp meters in back of 603 should read 0
(2) reactor power decreases

REFERENCE

HL-123-SH1 Page 20
SP 23.123.01 Page 5

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)

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 s 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 on the *PNL-601 after the reason for the isolation has been determined and corrected) [+0.5].

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)

COMMENT

Part b answer change auto isolation signal reset pushbuttons to auto isolation signal keylock switches.

REFERENCE

SP 23.202.01 Page 11 Step 8.2.3

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)

ANSWER 2.11 (2.00)

a. local fuel damage (by generating a rod withdrawal block) [+1.0]

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

REFERENCE

1. Shoreham: SH-603,604,652-SH1, LO C.
   215002A402 215002K102 215002SG04 ...(KA's)

COMMENT

Answer key for part a should also accept the following alternate answer; overpowering local regions of core at > 30% power.

Part b answer in parenthesis is incorrect. Answer should be (by using 1 volt per operable input).

REFERENCE

HL-606-SH1 Page 4 (part a)
HL-606-SH1 Page 18 (part b)

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)

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 switches in BYPASS with mode switch in SHUTDOWN or REFUEL

[+0.75] each

REFERENCE

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

COMMENT

Part d answer should be changed to say manual bypass switch, not switches. There is only one bypass switch.

REFERENCE

HL-611-SH1 Page 14

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)

ANSWER 3.02 (3.00)

a. closure of RCIC steam supply stop valve (MOV-43) [+0.5] and closure of trip and throttle valve (MOV-44) [+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)

COMMENT

Part a answer incorrect, the trip and throttle valve (MOV-44) does not go closed on high level. (Answer could include injection valve (MOV-35) that goes closed when MOV 43 goes closed.)

REFERENCE

HL-119-SH1 Pages 13, 14, 18
GE DWG. 791E421TF of E51 System.

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)

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)

COMMENT

Also accept for b part that the discharge MOV does not have to be full open. The requirement is for the valve to be greater than or equal to 90% open. If candidate assumed loss of full open indication the answer should be Recirc. Pump 'A' unaffected. If assumes < 90% open Recirc. Pump 'A' trips.

REFERENCE

HL-120-SH1 Page 15 and 16
ESK 6B3103

QUESTION 4.01 (2.50)

Procedure AP 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)

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.
   21600SG1 241000SG1 ...(KA's)

COMMENT

For part b, should also accept every 30 minutes.

REFERENCE

SP 22.001.01 Page 8

QUESTION 7.10 (2.75)
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.

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.

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

ANSWER 7.10 (2.75)
ANSWER 4.04 (3.00)

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

(ALTERNATE ANSWER: to limit the total heat load placed upon the primary containment).

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

(ALTERNATE ANSWER: to reduce the load upon the Division I battery).

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)

QUESTION 7.10
QUESTION 4.04 (Continued)

COMMENT

Also accept for part c that the operator prevents automatic suction transfer so NPSH of the RCIC system is preserved or CST water is of higher quality than the suppression pool.

REFERENCE

HL-944-SH1 Page 20

QUESTION 7.09 (2.00)
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)

ANSWER 7.09 (2.00)
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)

COMMENT

For part b the answer may be that recirc MG have already been tripped as part of answer for the a part of the question. So depending on what was assumed that answer may be zero or 10 minutes.

REFERENCE

SP 29.017.01

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 SP 61.012.01, "Personnel Dose Limits" for

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

   2. whole body per quarter? (0.5)

ANSWER 4.12 (2.50)

a. 1. 3000 mrem [+0.5]
   2. 4000 mrem [+0.5]

b. 1250 mrem [+0.5]

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

REFERENCE

1. Shoreham: SP 62.012.01

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

QUESTION 4.12 (Continued)

Answer key should be changed to accept either 5000 mrem or 6750 mrem as correct for part a. 2.

IAW 10CFR20.101 there are no yearly limits. Part a provides quarterly limits, i.e., 1250 mrem and part b provides conditions which permit the individual to exceed that in part a. 3 rem/qtr is allowed exception to 1250 mrem/qtr as long as 5x(n-18) is not exceeded. If 5x(n-18) is exceeded then 1250 mrem/qtr is the limit, regardless of total exposure.

The above was properly applied in parts a. 1 and b. of the question. In part a. 2, the candidate could have assumed:

| | | | | |
|---|---|---|---|---|
| 1st qtr. | 3000 mrem | | 1st qtr. | 1250 mrem |
| 2nd " | 1250 mrem | | 2nd qtr. | 1250 mrem |
| 3rd " | 1250 mrem | or | 3rd qtr. | 1250 mrem |
| 4th " | 1250 mrem | | 4th qtr. | 1250 mrem |
| Total - - - - | 6750 mrem | | Total - - - - | 5000 mrem |

REFERENCE

10CFR20 Page 252

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)

ANSWER 5.03 (2.00)

SDM = 1-Keff; CR1 (1-Keff 1) = CR2 (1-Keff 2)
(200 cps) (0.04) = (1000 cps) SDM
SDM = 0.008 = 0.8 percent [+2.0]

REFERENCE

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

COMMENT

Also accept answer based on using the formula for SDM.

$$SDM = \frac{1 - Keff}{Keff}$$

The answer will be slightly different.

REFERENCE

NRC Equation Sheet

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)

ANSWER 5.10 (1.25)

a. (1) [+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)

COMMENT

Answer to part a is incorrect. Answer should be (2) the elevation where the Steam Separator is submerged.

REFERENCE

SP 23.121.01 Page 12

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)

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)

COMMENT

Also accept for part b that due to cold water from CRDH stratifying in bottom head region and being swept into the core this could cause a power excursion in the core.

REFERENCE

HL-120-SH1 Page 41

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 to be 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 29.008.02, "Fuel Cladding Failure." (1.0)

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 specific activity shall be limited to:

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

2. ≤ 100/E microcuries per gram [+0.5]

REFERENCE

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

COMMENT

Answer to part c should be changed to have less than and equal to signs changed. Answer should read as follows; primary coolant chemistry analysis indicates specific activity is greater than either:

a. .2 microcuries per gram dose equivalent I-131
b. 100 E microcuries per gram

REFERENCE

SP 29.008.01

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)

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)

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)

COMMENT

For part b - should also accept the STA as an answer. STA at Shoreham is a on shift representative of the Reactor Engineering Department.

REFERENCE

SP 12.002.01 Page 5

QUESTION 7.08 (3.00)

Concerning Primary Containment:

a. The attached figure PC/T-1 is an excerpt from emergency procedure SO 29.023.02, "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)

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)

COMMENT

For part b of answer, if candidate states that at a reduced drywell average temperature maximum allowable drywell pressure is less than 1.69 psig, full credit should be given. Without T.S. or graph provided specific temperature should not be required for full credit.

REFERENCE

Technical Specifications 3.6.1.6.1

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)

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)

COMMENT

1) Either answer is acceptable for full credit in part a. Normal watch at Shoreham is 8 hours and there are no 12 hour watches.
2) For part c the watch supervisor is required for the fire brigade and may not be included. Consideration should be given based on candidates assumption.

REFERENCE

Technical Specification Page 6-1

QUESTION 8.04 (2.50)

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

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 lifter lead requires an SECP hold-off tag in addition to a lifted lead/jumper tag. (0.5)

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.

QUESTION 8.04 (Continued)

COMMENT

For part e give credit for > 120 volts because the reason for 130 volts was chosen to exclude 120 volts system.

REFERENCE

No specific reference - operations engineer reviewed this question and stated the reason why 130 volts was chosen.

QUESTION 8.07 (2.00)

Using the Emergency Plan Implementing Procedure (EPIP 1-0) CLASSIFY the following events. INCLUDE the classification identifier number. (This in 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)

ANSWER 8.07 (2.00)

a. Alert No. 11 [+1.0]
b. Unusual Event No. 9 [+1.0]

REFERENCE

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

COMMENTS

For part b - should also accept Alert No. 10 based on the assumptions on the impact of loss of RBSW.

REFERENCE

Shoreham EPIP-1-0 Alert No. 10

Attachment 4

# RESOLUTION OF FACILITY COMMENTS

## SHOREHAM - WEEK OF MAY 16, 1988

RO EXAM

| Comment Number | Resolution |
|---|---|
| 1.06 | Comment accepted. In the answer key, "TCV fast closure" will be placed in parentheses. |
| 1.07 | Comment accepted. It should be noted that SP22.001.01 clearly states the administrative requirement for heatup and cooldown rates as "not to exceed 90°F/hr." SP22.005.01 is apparently deficient in that it quotes the Technical Specification limit on cooldown rate (100°F/hr) instead of the more restrictive Shoreham administrative requirement. |
| 2.02 | See comment 6.08. |
| 2.04 | See comment 6.04. |
| 2.06 | See comment 6.06. |
| 2.10 | Comment accepted. The answer key will be changed to reflect the correct type of switch ("keylock" instead of "pushbutton"). However, it should be noted that the type of switch is contained inside parenthesis and hence is not required for full credit. |
| 2.11 | Comment accepted. |
| 3.01 | Comment accepted. |
| 3.02 | Comment accepted. |
| 3.07 | Comment accepted. Candidate will be graded according to candidate's stated assumptions. |
| 4.01 | Comment not accepted. Even though the Technical Specification requirement is "at least once per 30 minutes," the more restrictive Shoreham administrative limit (every 15 minutes) must be used. |
| 4.04 | See comment 7.10. |
| 4.07 | See comment 7.09. |
| 4.12 | Comment partially accepted. The allowed exposure would be 6750 mrem. |

RESOLUTION OF FACILITY COMMENTS

SHOREHAM - WEEK OF MAY 16, 1988

SRO EXAM

| Comment Number | Resolution |
|---|---|
| 5.03 | Comment accepted. The answer key indicates the most simplistic method. It is general policy in grading to recognize acceptable variations in equations as well as in assumptions. |
| 5.10 | Comment accepted. |
| 6.02 | Comment partially accepted. The answer key reflects the Technical Specification basis, the minimum acceptable complete answer. If the candidate additionally includes the potential for cold water induced power excursions, he will not be penalized. |
| 6.04 | Comment not accepted. The correct answer is the availability of the CST to provide the RCIC pump additional suction flow upon demand. The pressure input from the RCIC pump suction will have no effect on the level control valve response. Refer to GEK 71261B, section 2-62, for additional information. |
| 6.06 | Comment partially accepted. The question specifically requires "SBLC system indications." Consequently, the answer key has been expanded to include the squib valve continuity amp meters, but not a reactor power decrease. |
| 6.08 | Comment partially accepted. The diesel generator output breaker will not close unless voltage is ≥90% of rated. Consequently, its actual closure is the reference for the timing sequence. If the candidate additionally includes that voltage be ≥90%, he will not be penalized. |
| 7.02 | Comment accepted. |
| 7.06 | Comment accepted. |
| 7.08 | Comment not accepted. Though the candidate should not be expected to memorize the Technical Specification graph for drywell temperature versus allowable drywell pressure, he should know the bulk average drywell temperature point below which the allowable drywell pressure is no longer a constant 1.[illegible] psig so as to know when to refer to the graph. |

| Comment Number | Resolution |
|---|---|
| 7.09 | Comment not accepted. The recirculation pumps must be fully runback before the recirculation pumps are tripped and so the amount of time the MG sets remain running is indeterminate. Additionally, all candidates during the exam were alerted to the fact that part (b) of the question wanted the time limit based upon the MG set limitations, independent of any operator actions required. |
| 7.10 | Comment partially accepted. The discussion section of SP29.015.02, "Loss of All AC," specifically discusses the reason the procedure directs the operator to defeat the suction transfer interlock. Water quality is not an included reason. The answer key reflects the minimum acceptable complete answer, based on this procedure. If the candidate <u>additionally</u> includes water quality, he will not be penalized. |
| 8.03 #1 | Comment not accepted. Both parts of the answer key answer constitute a complete answer. |
| #2 | Comment partially accepted. If the candidate indicates his assumption that the watch supervisor is required for the fire brigade, he will not be penalized for not including the watch supervisor in his answer. |
| 8.04 | Comment not accepted. 125 VDC systems also are subject to lifted lead/jumper and SECP requirements. 130 volts was chosen to exclude 120 VAC/125 VDC systems. |
| 8.07 | Comment partially accepted. Alert #10 is correct and supercedes Unusual Event #9. The answer key was revised to specify Alert #10 as the only acceptable answer. |

ATTACHMENT 5

SIMULATION FACILITY FIDELITY REPORT

| | |
|---|---|
| Facility Licensee: | Long Island Lighting Company<br>Post Office Box 618<br>Wading River, New York 11792 |
| Facility Licensee Docket No.: | 50-322 |
| Facility License No.: | NPF-36 |
| Operating Tests administered at: | Shoreham Simulator |
| Operating Tests Given On: | May 17 - 20, 1988 |

During the conduct of the simulator portion of the operating tests identified above, the following apparent performance and/or human factors discrepancies were observed:

1. The PAMS recorder read about 300 psig high on pressure and about 30 inches high on level. This was repaired on May 19, 1988.

2. The recirculation flow controller runback does not seal in.

3. With a small drywell leak in progress, the process radiation monitoring system causes alarms not associated with the drywell.

4. The turbine startup procedure requires the operator to monitor information that is not simulated.

5. RHR pump D failed to start on a LOCA signal.

6. Following a scram, the A RHR pump, which was running before the scram, tripped for no reason.

7. After vessel isolation, large level swells (30 to 35 inches) were observed in response to manual lifting of a Safety Relief Valve.

8. Malfunction TC03 (EHC pressure regulator oscillations) was inserted followed by TC02 (EHC pressure regulator fails low). This regulator was found in control after the reactor scram. The backup regulator should have been in control.

9. Malfunction RD03 (CRD flow control valve fails full open) should have produced a rod drift (according to the Simulator Cause and Effect Manual) and it did not.

Overall, the simulator fidelity was acceptable.