Attachment 2

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

R D 1 BOX 208

DELTA, PENNSYLVANIA 17314

February 25, 1988

Mr. Allen G. Howe
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Howe:

The attachment to this letter documents the complete formal comment summary of the Reactor Operator License Examination administered on February 17, 1988.

All comments have been limited to those questions and/or answers which were specifically addressed during the post-examination review session conducted on February 17, 1988. The facility examination review team did not feel it necessary nor appropriate to comment on questions outside those discussed with the authors.

In the majority of cases, the referenced supporting documentation can be found in the materials forwarded to your office for exam preparation. In several cases specific references are not appropriate due to the nature of the reviewer's comments.

Sincerely,

Dickinson M. Smith
Vice President
Peach Bottom Atomic Power Station

DMS/RGA:lhd

Attachment

cc: J. F. Franz
L. G. Defferding
E. G. Firth
File-NTS

Attachment

Section 1

QUESTION 1.01 (2.50)

The reactor is brought critical at 60% on IRM range 2 with the MINIMUM permissible stable positive period allowed by procedure GP-2. Heating power is determined to be 40% on range 8 of IRM's.

a. WHAT is doubling time if period remains constant?

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

ANSWER 1.01

a. From GP-2, period equals 50 seconds. Thus doubling time equals 50/1.44 = 34.7 seconds.

b. 60% range 2 is equal to 0.06% on range 8
P(0) = 0.06 P(t) = 40 Period = 50 seconds
P(t) = P(0) e (t/period)
40 = 0.06 e (t/50 sec)
t = 50 ln 40/0.06 = 325 sec or 5 min 25 sec

REFERENCE

1. Peach Bottom: LOT-1430; 1530 LO #2.

29003K108 ...(KA'S)

FACILITY COMMENT

1.01 a. The 60% on IRM Range 2" can be interpretted as: 60 units on the 125 scale or 60% of the 125 scale or approx. 67 - 68 units on the 125 scale.

1.01 b. If an initial period for part 'A' is incorrectly assumed, credit should be taken off in part 'a' only. Since that same period will be used in this part of the question, if they initially assume an incorrect period they should lose credit only once. Period will be used in this part of the question, if they initially assume an incorrect period they should lose credit only once.

QUESTION 1.03 (3.00)

An EHC load reject occurs at 100% core thermal power with the EHC system aligned for normal 100% core thermal power with the EHC system aligned for normal 100% power generation. DESCRIBE and DISCUSS how

the following parameters respond during the first five minutes subsequent to the opening of the generator output breaker.

a. Reactor Power
b. Reactor Pressure
c. Reactor Water Level

ANSWER 1.03 (3.00)

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

b. Reactor pressure will rapidly increase due to the rapid closure of the TCV's. 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.

c. Reactor water level will initially drop as steam flow is abruptly interrupted. The feed control system will respond to increase level and level should then rise to the level controller setpoint.

REFERENCE

1. Peach Bottom: LOT 1600, LO #1 and #4.

241000K101 241000K102 241000K103 ...(KA'S)

FACILITY COMMENT

1.03 (General) The question implies we have one generator output breaker, we have two. The question does not have the proviso "Assume no operator action", therefore, any answers with operator actions specified should also be considered in addition to the NRC's response.

REFERENCE: LOT-0600, D.3.f. page 23

1.03 c. Assuming "no operator action", the initial low level signal will cause the RFP's to rapidly increase speed and overshoot the setpoint, potentially causing the pumps to trip at +45" Rx level.

QUESTION 1.04 (2.00)

PBAPS Unit 3 is taken critical during startup and a steady-state period is established. After the point if adding heat (POAH), the reactor period lengthens to infinity, and the reactor operator notes that the moderator temperature has changed from 240 degrees F to 260 degrees F.

a. WHAT reactivity coefficients turned reactor power? LIST them in order from the largest effect to the least effect.

b. HOW much positive reactivity was added to establish a stable positive period after criticality was obtained?

ANSWER 1.04 (2.00)

a. 1. mod temp coeff
2. fuel temp coeff
3. void coeff

b. Assume (1.) contribution from void and fuel temperature coefficient insignificant and (2.) moderator temperature coefficient = 1 x 10**-4 k/k/deg F.

(1 x 10**-4 (k/k)/deg F) x ((260 - 240) deg F) = 20 x 10**-4 k/k added

REFERENCE

1. Peach Bottom: Reactor Theory, Student Handout, Sections 26 through 30.

2. Peach Bottom: LOT 1440, LO #3 and 5.

292004K114 ...(KA'S)

FACILITY COMMENT

1.04 b. Should be receptive to any reactivity calculations that include both an α mod __and__ an α doppler contribution.

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 1000 psig at 0600. (It is not 0730 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 Technical Specification maximum allowable cooldown rate? (INCLUDE in your answer the PBAPS TECHNICAL SPECIFICATION COOLDOWN LIMIT and the assumptions and calculations used.)

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 of 100 deg F/hour.

(Tsat for 1000 psig = 546 deg F;
Tsat for 200 psig = 388 deg F;

cooldown rate = (546-388) deg F/1.5 hours
= 105 deg F/hr)

b. 68 +- 2 deg F (of cooldown required)
(Tsat for 200 psig = 388 deg F;
Tsat for 75 psig = 320 deg F;
(388-320) = 68 deg F)

REFERENCE

1. Peach Bottom: LOT 1150, LO #2.

2. Peach Bottom: LOT 1160, LO #2.

3. Peach Bottom: Technical Specifications, 2.2.2 and 3.6.A.1

205000K402 293003K123 ...(KA'S)

FACILITY COMMENT

1.07 a. In order to be absolutely correct, to solve for Tsat based on 1000 psig, interpolation in the steam tables must be done. Should allow for Tsats in the range of 548-544.

1.07 b. If the trainee improperly calculates Tsat for 200 psig in part 'a' of this question, it is possible that he would not arrive at the same answer given in the key for part 'b' of this question. The trainee should be penalized only once for the improper calculation of a Tsat for 200 psig in part 'a'.

QUESTION 1.08 (2.00)

Concerning the Bypass Flow in the reactor core.

a. DEFINE core bypass flow.

b. STATE the two most significant consequences that would occur if bypass flow were significantly reduced at full power.

ANSWER 1.08 (2.00)

a. (Core bypass flow is) that portion of total core flow that does not flow inside the fuel channels.

b. 1. Excessive voiding in bypass region resulting in unreliable LPRM readings.

2. Inadequate cooling of LPRm detectors resulting in premature LPRM detector failures.

REFERENCE 1. Peach Bottom: LOT 0010, IO #2.

293008K132 293008K133 ...(K/A'S)

FACILITY COMMENT

1.08 a. Instead of the clean definition given in the answer key, the trainees may try to list all the various types of core bypass flow to answer the questions. If given, this answer should also be considered for credit.

QUESTIONS 1.02, 1.05, 1.06, 1.09 and 1.10 - NO FACILITY COMMENT.

Section 2

QUESTION 2.02 (2.50)

The reactor water cleanup system is in operation with one pump and one filter demineralizer in service. A reactor startup and heatup is in progress with wide range reactor pressure indicating 400 psig. The RWCU dump valve is open, rejecting water to the main condenser to control reactor water level. Suddenly, the operator receives a RWCU low pump flow alarm and notes that system flow is 0 gpm and the previously running pump has stopped.

a. Given that the containment inlet and outlet isolation valves did not close, STATE four (4) possible causes of the pump trip.

b. STATE whether the RWCU dump valve (CV-55) will isolate CONCURRENTLY with any of the pump trips.

c. In the above example, if the operator also notices that an RWCU isolation has also occurred, STATE HOW the RWCU dump valve position at the time of the isolation can cause significant stress upon the RWCU system piping and components.

ANSWER 2.02 (2.50)

a. 1. low pump flow
   2. high pump vibration
   3. high pump flow
   4. pump motor supply breaker trip
   5. pump bearing cooling water outlet high temperature.
   6. pump motor thermal overload trip.

b. The RWCU dump valve (CV-55) WILL NOT concurrently isolate.

c. If the dump valve is open at the time of the RWCU system isolation, the system will rapidly depressurize, the water in the piping will flash to steam in the high temperature portions of the system, shocking (water hammering) the system piping and components.

REFERENCE

1. Peach Bottom: LOT 0110, LO #5 and #7.

204000K106 204000K401 204000K402 204000K407 ...(K/A'S)

FACILITY COMMENT

2.02 c. Examinees may take the answer to a point when control valve 55 auto isolates on upstream pressure of 5 psig. This would be correct. Do not count off.

QUESTION 2.08 (3.00)

The reactor is operating at 30% of rated core thermal power, 100% rod pattern, with the main generator on line. The "A" stator water cooling pump is tagged out (blocked).

a. DESCRIBE the response of the EHC system, reactor recirc pumps, and reactor power to a trip of the "B" stator water cooling pump, given the transient does not induce conditions requiring a turbine trip or RPS scram. INCLUDE in your discussion the time delay setpoints for major component trips and also final reactor power. ASSUME that no operator action is taken.

b. DESCRIBE the two (2) main turbine trips that are enabled on a total loss or stator water cooling. Include in your description all time delay setpoints.

ANSWER 2.08 (3.00)

a. 1. The EHC load set will run back to 25% causing all bypass valves to fully open.

2. The "A" reactor recirc pump will trip in 1.0 seconds.

3. The "B" reactor recirc pump will trip in 10.0 seconds.

4. Reactor power will stabilize at 50-55% of core rated thermal power.

b. The turbine will trip if generator amps are not less than 26,530 in 2 minutes or less than 7726 amps in 3.5 minutes.

REFERENCE

1. Peach Bottom: LOT 0630, LO #5 and #6.

202001K407 241000K123 241000K125 241000K405 ...(K/A'S)

FACILITY COMMENT

2.08.a.1 Question does not ask what happens to the bypass valves specifically. EHC runback should be sufficient to answer the question as taught in both EHC and Stator Water Coolant Lesson Plans.

QUESTION 2.10 (1.25)

The RCIC system is in operation on your shift to demonstrate operability for Technical Specifications.

a. DESCRIBE the RCIC system response if reactor water level exceeds +45 inches.

b. STATE whether operator action (WOULD/WOULD NOT) be required to permit the RCIC system to inject to the reactor if a reactor low-low water level condition occurs subsequent to the high level condition described in part "a" above.

ANSWER 2.10 (1.25)

a. The RCIC turbine steam supply valve (MO-131) closes (this is NOT a turbine trip).

b. (operator action) WOULD NOT (be required)

REFERENCE

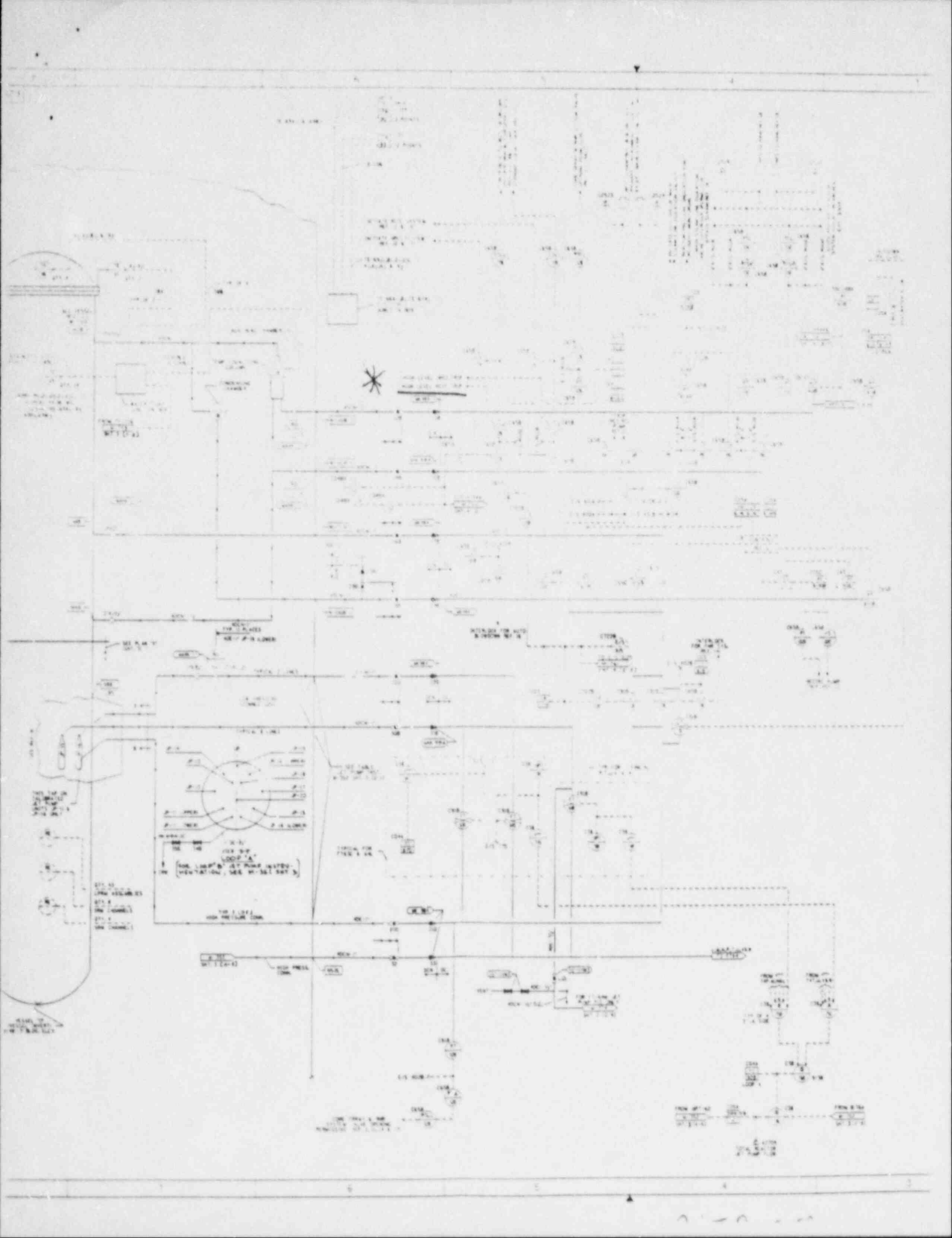
1. Peach Bottom: LOT 0380, LO #2 and #5.

217000K102 217000K402 217000SG7 ...(K/A'S)

FACILITY COMMENT

2.10.a. Answer key states "(this is NOT a turbine trip)". Students are taught this in the list of turbine trips, but it does not trip the trip throttle valve. It works through the MO-131 valve and is a self resetting trip at -48" reactor water level. P&ID 352 shows the instrument function at +45" as a high level RCIC trip. P&ID 352 is attached and marked.

QUESTIONS 2.01, 2.03, 2.04, 2.05, 2.06, 2.07 and 2.09 - NO FACILITY COMMENTS



Section 3

QUESTION 3.01 (3.00)

a. 1. The purpose is to limit peak fuel enthalpy in a postulated rod drop accident to 280 cal/gm.

2. Enabled at 25% (decreasing) Rx feedwater flow OR 25% (decreasing) Rx steam flow.

3. A bypass switch has been provided to manually bypass the RWM and requires a second licensed operator to verify that the operator at the reactor console is following the control rod program.

b. 1. The purpose is to limit peak fuel enthalpy in a postulated rod drop accident to 280 cal/gm.

2. Enabled at 21% (decreasing) power as measured by main turbine first stage shell pressure.

3. No provision to bypass.

REFERENCE

1. Peach Bottom: LOT 0090, LO #1, #2, and #4.

2. Peach Bottom: LOT 0100, LO #1 and #3.

3. Peach Bottom: LOT 0280, LO #1 and #5.

201002K104 201002K105 201002K106 201006K501 ...(KA'S)

FACILITY COMMENT

3.01 a. & b. 1. Incomplete answer for RWM and RSCS purpose (See LOT 0090 and 0100 for extensive answers.)

b.3. RSCS can only be bypassed in accordance with the TRIPS during an ATWS.

REFERENCE: TRIP T-101, Step RC/Q-45 and T-220.

QUESTION 3.04 (3.00)

The reactor is in cold shutdown with the "B" loop of RHR in shutdown cooling at a flow of 10,000 gpm using RHR pump "B". All other RHR pumps are secured and aligned for standby operation. The "D" high pressure service water pump is providing cooling water to the "B" loop RHR heat exchangers. RHR RPV head spray is not in service.

a. DESCRIBE WHAT automatic valve and pump actions should occur in the RHR system if reactor water level decreases to -10 inches with no

operator actions taken. LIMIT the description to only those components in the shutdown cooling flow path.

b. DESCRIBE WHAT operator actions, if any, are required to inject into the RPV with

1. the "A" loop of RHR

2. the "B" loop of RHR

all automatic actions properly occur.

ANSWER 3.04 (3.00)

a. 1. Shutdown cooling suction inboard and outboard containment isolation valves (MO-18 and MO-17) would auto close.

2. RHR pump "B" will trip.

3. Loop B LPCI injection valve (MO-25B) would auto close.

b. The operator must depress the shutdown cooling valve reset pushbutton for the "A" loop of RHR.

The operator must depress the shutdown cooling valve reset pushbutton for the "B" loop of RHR.

REFERENCE

1. Peach Bottom: LOT 0180, LO #2 and #6.

2. Peach Bottom: LOT 0370, LO #3, #4, #5, and #6.

203000K401 205000K404 216000K105 223002K108 ...(KA'S)

FACILITY COMMENT

Answer a.3. Loop A and B Injection Valves (MO-25A and B) would auto-close.

b.2. B Loop - Must open MO-13 (B-RHR Suction Valve) since this valve was originally closed in conformance to the normal shutdown cooling lineup which takes suction from the reactor vessel.

REFERENCE:

a.3. LOT 0180, page 8 (GROUP IIb) and LOT-0370, page 16 - IV.B.3.a. and page 17 - C.1.b.

b.2. LOT-0370 Handout, H-LOT-0370-3.

QUESTION 3.06 (3.00)

Concerning the Automatic Depressurization System (ADS):

a. Once ADS has commenced blowdown, STATE ALL the operator actions that could be taken in Unit 2 to reclose the relief valves prior to reactor pressure decreasing below 50 psig. b. STATE which signal input to ADS logic must in all cases be manually reset when the signal clears.

c. The ADS logic received a modification from its initial design that added a 9.5 minute timer and a keylock handswitch to each logic train. STATE the purpose(s) of this additional timer and handswitch. DESCRIBE in your statement how the timers and handswitches affect the logic. INCLUDE setpoints associated with these devices.

ANSWER 3.06 (3.00)

a. 1. Depress both the "A" and "B" timer reset pushbuttons to break the seal in.

2. Shutdown the RHR and core spray pumps.

3. Place the "A" and "B" keylock switches in "Inhibit".

b. high drywell pressure

c. The additional timer in each logic train automatically inserts a high drywell pressure permissive signal if reactor water level is not restored to a level greater than -130 inches within 9.5 minutes of level falling below -130 inches. The purpose of this feature is to make the logic responsive to a LOCA with the break physically outside the containment.

The keylock switches for each logic train of ADS disable their respective logic trains to prevent ADS relief valve actuation. The purpose of this feature is to provide a positive means to disable ADS when under certain accident conditions. ADS actuation would be highly undesirable. (ALTERNATE ANSWER: The purpose of this feature is to provide a positive means to disable ADS when directed by procedure.

REFERENCE

1. Peach Bottom: LOT 0330, LO #2, #3, and #5.

216000K107 218000K403 218000K501 ...(K/A'S)

FACILITY COMMENT

a. Operator actions are at the direction of the TRIPs only.

b. Answer should be HIGH DW PRESSURE/LOW RPV LEVEL RESET.

REFERENCE:

a. T-101 is only guidance for operator action; see also LOT-0330.

b. S.3.10.C and LOT-0330, page 8 and T-LOT-0330-4A.

QUESTIONS 3.02, 3.03, 3.05, 3.07, 3.08, 3.08 and 3.10 - NO FACILITY COMMENT.

Section 4

QUESTION 4.01 (3.00)

Regarding Administrative Procedure A-7, "Shift Operations:"

a. STATE which individual by title is required to authorize a startup subsequent to a shutdown or scram.

b. Appendix 5 of A-7 lists the specific duties of the control room operator. STATE the three (3) conditions under which the control room operator is responsible for and has the authority to shutdown the reactor.

c. Section 7.1 of A-7, "Shift Operations" contains guidance for "On-Duty" senior licensed operators and licensed operators concerning their PERSONAL CONDUCT while on shift. STATE two (2) of these guidelines that help ensure the units are operated as safely and as reliably as possible.

ANSWER 4.01 (3.00)

a. station superintendent

b. 1. The safety of the reactor is in jeopardy.

2. Operating parameters exceed reactor protection system setpoints and automatic shutdown does not occur.

3. When there is doubt as to whether safe conditions exist.

c. 1. (On duty SLO's and LO's) must be alert and attentive.

2. (On duty SLO's and LO's) must be aware of and responsible for the plant status at all times.

3. (On duty SLO's and LO's) must prohibit distracting activities in the control room.

REFERENCE

1. Peach Bottom: LOT 1570, LO #2a and #3b.

294001A103 ...(KA'S)

FACILITY COMMENT

a. May get PLANT MANAGER in lieu of STATION SUPERINTENDENT whose position has been eliminated.

QUESTION 4.02 (2.75)

The control room becomes uninhabitable because of a bomb threat and the decision has been made to immediately evacuate the control room.

a. LIST the seven (7) immediate actions the operator is to take PRIOR to exiting the control room as delineated by procedure SE-1, "Plant Shutdown from the Emergency Shutdown Panel - Procedure."

b. Once at the emergency shutdown panel, procedure SE-1 instructs the operator to place all the pistol grip hand switches on the emergency shutdown panel in the "pulled-out" position. STATE the purpose of this action.

ANSWER 4.02 (2.75)

a. 1. runback recirc flow to minimum

2. transfer house loads

3. manually scram and execute T-100

4. place the drywell instrument air in service.

5. close MSIV's

6. establish torus cooling

7. obtain master keys

b. Placing the switches in the "pulled-out" position transfers control of the associated components from the control room to the emergency shutdown panel.

REFERENCE

1. Peach Bottom: SE-1, "Plant Shutdown from the Emergency Shutdown Panel - Procedure."

(295016AK20 295016SG10 ...(KA'S)

FACILITY COMMENT

a. Examinee may use term D/W Instrument Nitrogen for D/W Instrument Air.

QUESTION 4.04 (2.00)

ON-105, "Control Rod Uncoupled-Procedure," provides instructions to follow in the event of an uncoupled control rod.

a. LIST three (3) indications of an uncoupled control rod.

b. HOW many recoupling attempts are allowed by ON-105?

ANSWER 4.04 (2.00)

a. 1. rod overtravel alarm when fully withdrawn.

2. control rod withdrawal with no apparent nuclear response.

3. no control rod drive water "stall flow" observed when performing an uncoupling check at position 48.

b. three

REFERENCE

1. Peach Bottom LOT 1550, LO #1 and #2.

201001SG15 ...(K/A'S)

FACILITY COMMENT

a.3. Unable to find reference for "No Stall Flow" statement in our materials. We would expect to get stall flow since CRD doesn't know if blade is coupled or stuck.

NO REFERENCE AVAILABLE.

QUESTION 4.11 (1.50)

Radiation work permits (RWPs) control work performed in the radiologically controlled area (RCA). Operations personnel have two "Operations RWP's" in effect at all times, one at each unit, allowing operators to perform certain functions. STATE three (3) operations functions these two RWPs together allow.

ANSWER 4.11 (1.50)

1. (operator) rounds
2. blocking
3. inspection

REFERENCE

1. Peach Bottom: LOT 1760, LO #4.
294001K103 ...(K/A'S)

FACILITY COMMENT

"Operation RWPS" have been replaced by "General RWPs" since July 1987. Examinees may be confused by the old term, but an explanation was provided by Examiner during examination for clarification.

QUESTIONS 4.03, 4.05, 4.06, 4.07, 4.08, 4.09, 4.10 and 4.12 - NO FACILITY COMMENT.

Attachment 3

FACILITY COMMENTS AND RESOLUTIONS

PEACH BOTTOM - WEEK OF FEBRUARY 16, 1988

Comment Number	Resolution
1.01a	Comment accepted
1.01b	Comment accepted. It is general policy in grading to award partial credit and to not "double jeopardize" incorrect answers.
1.03	Comment accepted.
1.03c	Comment accepted.
1.04b	Comment accepted.
1.07a	Comment partially accepted. Allowances are made for interpolation variances; however, the candidate should be able to convert a given saturation pressure to a saturation temperature within ±1°F. The ability to interpolate is required to use steam tables.
1.07b	Comment accepted. It is general policy in grading to award partial credit and to not "double jeopardize" incorrect answers.
1.08a	Comment partially accepted. If the candidate attempts to list the specific bypass flow paths, then all must be listed for full credit.
2.02c	Comment not accepted. Though CV-55 will isolate, the isolation setpoint is well below the pressure at which water in the high temperature portions of the RWCU system will flash to steam.
2.08a.1	Comment not accepted. The bypass valves are a part of the EHC system and their opening to prevent a high pressure scram are as significant as the EHC system throttling of control valves to limit turbine loud to 25%.
2.10a	Comment not accepted. The P&ID note referenced is an incomplete note and refers to a RCIC system trip. A RCIC system trip is not a turbine trip but a closure of MO-131 on high RPV water level. A turbine trip is a trip of the trip and throttle valve (TTV). When tripped, the TTV must always be reset by the operator. The distinction between a system and turbine trip is crucial in the response of the system to RPV water level transients.

Comment Number	Resolution
3.01a,b	Comment partially accepted. The answer in the answer key reflects the primary purposes of the RWM and RSCS and were the only responses required for full credit. The additional purposes in the PBAPS training material are truly secondary and not the reasons for including these systems in reactor design. If the candidate additionally lists any of the secondary purposes, he will not be penalized.
3.01b.3	Comment not accepted. The question clearly specifies bypassing the entire system.
3.04a.3	Comment partially accepted. The intent of the question was to address that portion of the RHR system where shutdown cooling flow existed prior to the isolation (the "B" loop). If the candidate additionally lists the "A" loop injection valve (MO-25A), he will not be penalized.
3.04b.2	Comment partially accepted. With the "D" RHR pump in standby, it will start regardless of the position of the B RHR pump suction valve. The B loop will inject with the "D" pump as a pressure source if the "B" shutdown cooling valve reset pushbutton is depressed. The intent of the question was to elicit the minimum required operator action to inject with the "B" loop. If the candidate additionally lists the actions necessary to get the "B" RHR pump started, he will not be penalized.
3.06a	Comment not accepted. The question clearly has the object to test knowledge of control logic.
3.06b	Comment not accepted. The question asks which "signal input" not which "reset pushbutton."
4.01a	Comment accepted.
4.02a	Comment accepted.
4.04a.3	Comment partially accepted. "Stall flow" refers to the lack of drive water flow when demanding the CRD mechanism to move. Rod position indication always indicates mechanism position. At position 48, if the rod is still coupled to the mechanism, the velocity limiter is backseated against the guide tube, and any attempt to withdraw the rod further will result in drive flow "stalling" towards 0 gpm. If the rod is not coupled, the mechanism will drive out beyond position 48 and drive flow will not stall. If the candidate adequately describes "stall flow," he will not be penalized.
4.11	Comment accepted.