ENCLOSURE 3

CP&L

Carolina Power & Light Company

Shearon Harris Energy & Environmental Center
Route 1, Box 327
New Hill, North Carolina 27562

January 24, 1991

File: NTS-3501

Serial: 91NTS061

Mr. Chuck Casto
US NRC--Region II
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Subject: SRO NRC Written Exam Comments

Dear Mr. Casto:

On January 22, 1991, Shearon Harris Nuclear Power Plant received NRC SRO examinations. The examination comments are submitted by CP&L. Copies of reference material are included where indicated.

Should you need any explanations or additional reference material, please do not hesitate to contact the SHNPP Manager - Training, Mr. A. W. Powell, at (919) 362-3219.

R. B. Richey
Vice President - Harris Nuclear Project

ATB/ljw
Enclosures
c: Mr. J. E. Tedrow (NRC-SHNPP)

SRO EXAM GENERAL COMMENTS

1. We would like to take this opportunity to express our appreciation to NRC Region II for the initial review of the written exam. We especially appreciate the effort by the NRC representative to open the NRC office on the weekend to continue the review of CP&L's comments. During the written exam review conducted in the NRC Region II office, numerous discrepancies affecting approximately 70% of the exam questions were identified and corrected prior to administration of the exam.

2. The walk-through and simulator exams were reviewed on site before they were administered to the candidates. A number of changes recommended to the JPM questions were incorporated in the walk-through portion of the exam.

3. Written exam comments:

   a. Several questions require memorization that would not normally be expected of an SRO candidate. These questions, along with comments, can be located on Attachment I. These comments are provided for information only.

   b. Technical comments are on Attachment II. It was our intent to identify and correct technical inaccuracies on the preview. However, due to the extent of changes made and limited time available for review, six questions still remain technically incorrect.

ATTACHMENT I

MEMORIZATION COMMENTS

Question Number	Comment
16	Question required memorization of steps contained within PLP-500. All of the titles listed are indeed notified by this procedure. The question asks which one is notified immediately. This was commented on during exam preview.
47	Question requires the operator to have the main steam isolation signal logic diagram memorized in detail.
56	Question requires the operator to have pressure switch setpoints committed to memory. This was commented on during exam preview.
29, 64, and 75	Questions require the operator to have Tech Spec LCO text as well as applicable action statements committed to memory.
79	This question requires memorization of Tech Spec Interpretation 89-004. Interpretations are not normally committed to memory.

ATTACHMENT II

DETAILED COMMENTS ON SPECIFIC QUESTIONS

NRC EXAM QUESTION 40

QUESTION: 040 (1.00)

What operator action is required to regain use of pressurizer heater group "C" following a loss of pressurizer level (level less than 17%)

a. With level less than 12% cycle the heater breaker at MCC.

b. With level less than 17% cycle the heater control switch.

c. With level less than 12% cycle the heater control switch.

d. With level less than 17% cycle the heater breaker at MCC.

ANSWER:

b

REFERENCE:

PZRLC-LP-3.0, L.O. 1.1.6, page 15

CP&L COMMENT

Page 15, Section 2.5.3 of PZRLC-LP-3.0 states that pressurizer level must be greater than 17 percent to allow htr Group "C" to be reenergized manually. Since all choices state level is less than 17 percent, no answer is correct (see attached).

RECOMMENDATION

Delete question because no answer is correct.

KEY AIDS

2. Hagan type controller
   a. Receives input from
      1. Auctioneered high Tavg (program level)
      2. Reference pressurizer level (actual level)

B. Pressurizer Level Controller Selector
   1. CHAN 460/461, CHAN 459/460, CHAN 459/461
   2. Normally selected to CHAN 459/460

2.5.3 Interlocks

A. Heaters
   1. Interlocked with PZR level (decreasing)
      a. De-energize when 1 of 2 selected channels fails below 17% (may be overridden at local control station)
      b. Backup heaters automatically re-energize when 2 of 2 channels greater than 17% if control switches in ON or AUTO
      c. Variable control heaters must be re-energized manually when level requirements satisfied

i.e. >17%

   2. Interlocked with PZR level (increasing)
      a. Backup heaters energize at +5% above program level
         - Heatup of insurge water ensured

p. 5a

NRC EXAM QUESTION 46

QUESTION: 046 (1.00)

Given the following:

- The plant is operating normally at 50% power.
- All systems are in automatic.

WHICH one of the following correctly describes the effect on the listed parameters resulting from a sudden closure of the "B" loop main steam line isolation valve?

a. Total steam flow remains the same, loops "A" and "C" Tavg decreases, loop "B" Tavg increases.

b. Total steam flow and loops "A" and "C" Tavg remain the same, loop "B" Tavg increases.

c. Total steam flow and loops "A" and "C" Tavg increases, loop "B" Tavg decreases.

d. Total steam flow remains the same, loops "A" and "C" Tavg increases, loop "B" Tavg decreases.

ANSWER:

b

REFERENCE:

T&AA-LP-2.25, pages 19 and 20, L.O. 1.1.3

CP&L COMMENT

T&AA-LP-2.25, pages 18-20, calculate the expected response of an MSIV closure assuming rod control was initially in manual. The lesson plan states that Loop "A" and "C" Tavg will decrease vice remain the same. Since rod control is given in the initial conditions as being in automatic, the absolute value of Tavg will decrease; however, the trend is still the same. The actual answer is therefore "a" vice "b."

RECOMMENDATION

Change answer key to "a" vice "b."

KEY AIDS

2. Feed flow increases attempting to recover level
2. Overpressure reactor trip is initiated
   a. Causes a turbine trip
      1. Steam flow drops to nearly zero
      2. ΔT across SGs is nearly zero
   b. Steam pressure increases even more
      1. Follows T_{avg} after trip
      2. Increases to PORV setpoint (~ 1106 psig)
   c. PORVs relieve steam at a rate matching reactor heat input
      1. From decay heat
      2. From RCPs
   d. T_{avg} will stabilize around T_{sat} for 1106 psig (~ 559°F)
      1. Pressurizer level follows T_{avg}
      2. Stabilizes at the program level for new T_{avg}
      3. Pressure will eventually return to normal

2.5.3 Closure of Main Steam Isolation Valve

A. Given: the reactor is operating at 50% power with turbine control in automatic and rod control in manual when a main steam isolation valve shuts. What would happen to the following plant parameters? — TAA-TP-160, TAA-TP-161
   1. Turbine power
   2. Reactor power
   3. T_{avg} (all loops)

p. 6a

KEY AIDS

4. SG pressures
5. Core T_{avg}, ΔT
6. T_H, T_C (all loops)

B. Assumptions
1. No plant trip
2. No SG safety or relief valve actuation
3. No operator action

C. Parameters
1. Turbine power
   - Remains at 50% (EHC in auto.)
2. Reactor power
   a. Remains constant at 50%
   b. Determined by turbine power
3. T_{avg} of affected loop
   a. $\dot{Q}SG$ of affected ~ 0%
   b. ΔT of affected ~ 0°F
   c. T_C goes to T_H
   d. T_{avg} of affected = T_H
4. T_{avg} of unaffected loops
   a. 50% power removed by 2 SGs
   b. Larger primary to secondary ΔT across the unaffected SG
      $\dot{Q}_{SG} = UA(T_{avg} - T_{stm})$
      $(T_{avg} - T_{stm})$ after
      $= 3/2(T_{avg} - T_{stm})$ before
      $T_{avg} - T_{stm} \approx$ 48°F at 100% or
      24°F at 50%
      $(T_{avg} - T_{stm})$ after = 3/2(24°F)
      = 36°F
   c. Loop ΔT increases by 3/2 for the unaffected loops.
      New loop ΔT = (1.5)(31.8) = 47.7°F

3/2 TAVG - TSTM = 36°F ΔT

At 100%
T_C = 557
T_{avg} = 588.8
T_H = 620.6
ΔT = 63.6
T_{stm} = 540.8

At 50%
T_C = 557
T_{avg} = 572.9
T_H = 588.8
ΔT = 31.8
T_{stm} = 548.9

p 6b

KEY AIDS

d. $T_{avg}(\text{core}) = \dfrac{2T_{avg}(\text{unaffected}) + T_{avg}\ (\text{aff})}{3}$

T_{avg} (core) must be constant since power constant, no rod motion, no change in boron

$$572.9°F = \frac{2\left(\frac{T_H + T_H - 47.7}{2}\right) + T_H}{3}$$

T_H = 588.8°F (did not change)

T_{avg} (unaffected) = 588.8 −.5(47.7)

T_{avg} (unaffected) = 564.95°F (or 565°F)

5. SG pressure of affected SG
   a. $T_{avg} \approx T_{stm}$
   b. T_{avg} = 588.8°F = T_{stm}
   c. P_{stm} at 588.8°F ~ 1418.7 psia
6. SG pressure of unaffected SGs

   $SG\Delta T = T_{avg} - T_{stm}$

   T_{stm} = 565°F − 36°F

   = 529°F

   P_{stm} at 528°F = 878 psia
7. T_H in all loops remains at 588.8°F
8. T_C in affected loop goes to T_H
9. T_C in unaffected loops

   T_H is 588.8°F

   Tavg is 564.95°F (or 565°F)

   572.9 INITIALLY
   565° FINAL

   T_C = 541.1°F
10. Note that if rods were in auto control, they would step in to reduce auctioneered high Tavg from 588.8°F to 572.9°F

shows change in absolute value.

p 6c

NRC EXAM QUESTION 59

QUESTION: 059 (1.00)

Technical Specifications state that "All shutdown and control bank rods shall be OPERABLE and within + or - 12 steps of their group step counter demand position. WHICH one of the following will cause entry into the associated Technical Specification action statement? (Assume only one rod in control bank D is effected and all other bank D rods move in response to a 3 degree Tavg - Tref mismatch.)

a. The CRDM lift coil energizes and the movable gripper de-energizes before the stationary gripper energizes during rod insertion.

b. The CRDM lift coil and the movable gripper coil remain de-energized while the stationary gripper functions normally during rod withdrawal.

c. The CRDM lift coil and the movable gripper coil remain energized while the stationary gripper functions normally during rod withdrawal.

d. The CRDM lift coil and the movable gripper energize at the same time while the stationary gripper functions normally during rod insertion.

ANSWER:

b

REFERENCE:

RODCS-LP-3.0, pages 16 through 21, L.O. 1.1.6

CP&L COMMENT

RODCS-LP-3.0, pages 16 through 20, explain the sequence of events necessary to move control rods. Answer "b" is a correct answer as it will result in an eventual misalignment due to simultaneous deenergization of both the movable and stationary gripper. Answer "a" is also correct as it too results in deenergizing both the movable and stationary grippers simultaneously.

RECOMMENDATION

Accept either answer a. or b as correct.

KEY AIDS

F. CRDM operation
   1. Hold
      a. Lower coil (stationary gripper coil) energized at lower current
      b. Stationary gripper latches engaged
   2. Reactor trip
      a. All coil current interrupted by
         1. Choice (manual)
         2. RPS
         3. Electrical malfunction
      b. All gripper latches disengaged under spring force
      c. Drive rod drops
      d. Rods inserted into core
   3. Rod repositioning — CRDM Model
      a. Withdrawal
         1. Initial conditions
            - Hold
            - Stationary gripper coil energized
            - Low current (4.4 amps)
            - Movable gripper coil deenergized
            - Lift coil deenergized
         2. Stationary gripper coil current increases to eight amps lifting rod 1/16"

NOTE: CRDM model does not lift rod this 1/16" with stationary grippers because of design limitations.

3. Movable gripper coil energized
4. Movable gripper latches engage drive rod groove 1/16th inch below ridge
   - Allows movable grippers to freely engage without trying to simultaneously lift rod

NOTE: CRDM model lifts rod 1/16" as movable grippers engage as evidenced by chipped paint on rod. (This is <u>not</u> the way the real device functions)! This step occurs almost simultaneously with the stationary gripper coil current increases.

5. Stationary gripper coil current decreases to zero
   - Weight transferred to movable gripper as stationary coil disengages during current decreases
   - Rod lowered 1/16th inch by this action

NOTE: CRDM model does not demonstrate this action because of design limitations

6. Lift coil energizes by high current (40 amps)
   - Movable gripper assembly rises 5/8th inch
   - Drive rod raised 5/8th inch
7. Stationary gripper coil reenergized at high current (8.0 amps)
   - Engages drive rod
   - Raises drive rod 1/16th inch

NOTE: CRDM model does not demonstrate this action because of design limitations

   - Load transfers to stationary grippers
   - No load on movable grippers
8. Lift coil current profiled (reduced) to low current (16 amps) to reduce coil heating
   - Drive rod still held by stationary grippers
9. Movable gripper deenergizes
   - Latches disengage and swing out freely

NOTE: CRDM model drops 1/16" inch as movable latches disengage. (This is not the way the real device functions).

10. Lift coil deenergizes - movable gripper assembly lowers 5/8th inch
11. Stationary gripper coil current reduced to 4.4 amps
    - Rod drops 1/16"

NOTE: CRDM model will not demonstrate the 1/16" inch drop because of design limitations

12. Rod raised 5/8 inch in 780 milliseconds

b. Rod insertion

NOTE: CRDM model exhibits similar design limitations as in the withdraw sequence

1. Hold position
   - Stationary gripper coil at low current
   - Movable gripper coil deenergized
   - Lift coil deenergized
2. Stationary gripper coil current increases to 8.0 amps
   - Rod lifted 1/16"
3. Lift coil energized
   - High current (40 amps)
   - Movable gripper assembly raised 5/8th inch

KEY AIDS

4. Movable gripper coil energized.
   - Movable gripper engages
   - Movable gripper latches 1/16th inch below ridge
5. Stationary gripper coil current decreases to zero
   - Weight transferred to movable gripper as stationary coil current decreases and stationary gripper disengages
   - Rod lowered 1/16th inch by this action
6. Lift coil deenergized
   - Movable gripper assembly lowers rod 5/8th inch
7. Stationary gripper coil reenergized at high current (8 amps)
   - Stationary grippers engage drive rod
   - Stationary gripper latches raise drive rod 1/16th inch
   - Weight transferred to stationary gripper
8. Movable gripper coil deenergizes
   - Latches freely disengage

KEY AIDS

3. Logic error: zero current to both stationary and movable gripper coils simultaneously (could cause rod drop)
4. Multiplexing error: more than one group in a power cabinet attempts to move at the same time
5. Loose card

b. Detection of errors
   1. Feedback from sampling resistors compared to demand current
   2. Logic Coincidence - Error must be sensed on all rods within group (4 out of 4)
      - Allows movement of single rods if Lift Coil Disconnect Switches open

c. Energizes both stationary and movable gripper coils simultaneously
   1. Low current
   2. Locks rods

d. "Rod Control Urgent Failure" annunciator
   1. ALB-13
   2. Power cabinet

e. Inhibit signal prevents all rod motion
   1. All rods in that cabinet
      - "Automatic" mode
      - "Manual" mode

NRC EXAM QUESTION 70

QUESTION: 070 (1.00)

WHICH one of the following describes the parameters the ERFIS computer uses to derive the RCS subcooling margin during a loss of all AC?

a. RCS narrow range pressure to calculate T-sat, highest one of the core exit thermocouples or loop wide range T-hot.

b. RCS narrow range pressure to calculate T-sat, highest of the average core exit thermocouples in the hottest quadrant or loop wide range T-hot.

c. RCS wide range pressure to calculate T-sat, highest one of the core exit thermocouples or loop wide range T-hot.

d. RCS wide range pressure to calculate T-sat, highest of the average core exit thermocouples in the hottest quadrant or loop wide range T-hot.

ANSWER:

d

REFERENCE:

RCTEMP-LP-3.0, page 7, L.O. 1.1.11

CP&L COMMENT

Although this lesson plan implies "d" as the correct answer, a more recently developed lesson plan (COMP-LP-3.2, Section 2.3) explains the correct way subcooling is determined.

Lesson plan COMP-LP-3.2 was not provided with the exam reference materials. These are RO level lesson plans and only selected ones are reviewed during the SRO training program.

CP&L will update Lesson Plan RCTEMP-LP-3.0.

RECOMMENDATION

Delete the question because no answer is correct.

KEY AIDS

2. When started, previous eight minutes of data plotted from a historical buffer
3. Accessed by using paging keys on the second level display
4. Contain SPDS status boxes at bottom of each display

2.3 INSTRUMENTATION AND SIGNAL USAGE

A. Top Level Display
   1. Subcooling — COMP-TP-58.0
      a. Average of five hottest incore thermocouples
      b. Pressurizer pressure > 1700 psig and PT 402, PT 403 when < 1700 psig

actual method of subcooling determination

B. Subcriticality
   1. Power Range ≥5% — COMP-TP-46.0
      a. Takes 2 of 4 channels — COMP-TP-59.0
      b. Current value display averages those power ranges that are ≥5%
   2. Intermediate Range Startup Rate
      a. Takes average of 5 consecutive readings over 10 seconds
      b. Compared to next block of 5 reading to calculate SUR
      c. Startup rates calculated for both channels then combined in 1 of 2 logic
   3. Source Range Energized
      a. Looks at P-6 setpoint
      b. 1 of 2 logic
   4. Source range startup rate - calculated same way as intermediate range

p 8a

[illegible]

KEY AIDS

C. Loop Temperature Difference (ΔT) — RCTEMP-TP-3.0, RCTEMP-TP-5.0

1. Generated from narrow range Tc and Th RTD's by the following calculation

$$\Delta T = Th - Tc$$

2. Used to generate inputs for control and protection instrumentation
   a. Protective functions derived from temperature elements described in 2.3.B.2.a
   b. Control functions derived from temperature elements described in 2.3.B.2.b
   c. ΔT display
      1. Two separate ΔT's for each loop
      2. Indication for both channels in all three loops on MCB
      3. ΔT provides inputs to RPS and control systems

D. RCS Subcooling

1. Calculated by one of two 100% redundant ERFIS computers
2. ERFIS calculate saturation temperature (T_{SAT}) corresponding to the RCS wide range pressure
3. ERFIS screens all 51 core exits thermocouples and calculates the average core exit temperature in the hottest core quadrant
4. The core exit temprature is auctioneered high against the RCS loop wide range T_H RTD's
5. The auctioneered T_H is compared to

implied answer

[illegible]

KEY AIDS

the calculated T_{SAT} to derive the subcooling margin

6. Calculations are updated at a minimum of every 30 seconds

2.4 SYSTEM ARRANGEMENT/MAJOR COMPONENTS — RCTEMP-TP-2.0, RCTEMP-TP-3.0

A. Protection Circuits
   1. Computed Tavg and ΔT output signals from each loop provide inputs to reactor protection
      a. Three loop channels independent of each other
      b. Coincidence is two-out-of-three (2/3)
   2. Each loop inputs to four protection circuits (Tavg)
      a. Overtemperature ΔT (OTΔT) - setpoint development
      b. Overpower ΔT (OPΔT) - setpoint development
      c. Low Tavg (feedwater isolation with reactor trip)
      d. Low-Low Tavg (steam dump block P-12)
   3. ΔT inputs for two protection circuits
      a. OTΔT - provides reactor power input
      b. OPΔT - provides reactor power input

B. Control Circuits — RCTEMP-TP-4.0
   1. Operation of Tavg in control circuitry — RCTEMP-TP-5.0
      a. Three Tavg signals from control Tavg signals are auctioneered

NRC EXAM QUESTION 81

QUESTION: 081 (1.00)

Below PATH-1, entry point C, the path is designed to initiate recovery from WHICH one of the following events?

a. Steam Generator tube rupture.

b. Stuck open PORV and associated block valve.

c. Faulted Steam Generator.

d. Loss of coolant flow.

ANSWER:

b

REFERENCE:

EOP-LP-3.1, page 8, L.O. 1.A.1

CP&L COMMENT

EOP-LP-3.1, pages 8 and 9, state that this point of PATH-1 is for both a loss of reactor or secondary coolant. EPP-14 also directs a transition to this point. Therefore, both answer "b" and "c" are correct.

RECOMMENDATION

Accept both answers "b" and "c" as correct.

A2 If there are no intact SGs available, the actions of EPP-015 can be performed out of sequence to enhance plant control

Q3 What are recent changes to PATH-1, revision 1A?

A3
1. Added check for RHR in shutdown cooling mode
2. Moved step to reset SI to early in PATH-1
3. Modify FRP-H.1 transition to check SG level

C. PATH-1 (from entry point C)

1. Use associated WOG guideline to facilitate discussion — WOG E-1
   a. Purpose--provides actions to start recovery from loss of reactor or secondary coolant
   b. Reference events
      (1) Reactor Coolant leak (< 3/8 inches)
         (a) Within capacity of charging
         (b) Normal plant shutdown
      (2) Small break (3/8 inch to 1 inch)
         (a) RCS will depressurize
         (b) Reaches equilibrium pressure when break flow equals injection flow
         (c) Containment indicates break but pressure probably remains below 3 psig
         (d) RCS subcooling increases throughout transient due to cold injection water
         (e) RCS repressurization possible if
            1) Injection flow >> break flow or
            2) Heat sink not maintained (reach saturation)
      (3) Medium break LOCA (1 inch to 1 foot^2)
         (a) RCS pressure drops rapidly to saturation
         (b) Injection flow < break flow initially

p. 9a

(c) RCS drains until break flow switches from liquid to steam
1) Draining of SG tubes reduces heat removal
2) Core uncovery possible for cold-leg break
(d) When break flow becomes steam
1) RCS pressure decreases
2) Injection flow increases
3) Heat removal increases
(4) Large break LOCA (> 1 foot^2) has four stages
(a) Blowdown of RCS to CNMT pressure
(b) Refill when cooling water fills vessel lower plenum
(c) Reflood as downcomer annulus fills providing a static head for pushing water into the core
(d) Long term recirc when ECCS suction is swapped to the CNMT sumps
(5) Loss of secondary coolant
(a) RCS pressure decreases with cooldown
(b) When faulted SG drys out and AFW is isolated, decay heat begins to restore RCS temperature and pressure
(c) Since there is no break flow, injection increases RCS pressure and level
(d) If injection is not terminated, the PRZ becomes water solid and PORVs lift
c. Major action categories
(1) Monitor plant equipment for optimal mode of operation — EOP-TP-158.01
(a) RCP seal injection
(b) SG levels

Instructions	Response Not Obtained
6. Check CST Level - GREATER THAN 10%	Switch to alternate AFW water supply.

****

CAUTION

A maximum of 1600 PSID between the RCS and SG enhances SG tube integrity.

****

Instructions	Response Not Obtained
7. Check Faulted SG(s):	
a. Pressure - STABLE	a. WHEN pressure has stabilized, THEN dump steam from intact SGs to stabilize RCS temperature. GO TO Step 8.
b. Dump steam from intact SGs to stabilize RCS temperature.	
8. Check Secondary Radiation:	
a. Periodically sample SGs for activity.	
b. Secondary radiation - NORMAL	b. GO TO PATH-2, entry point J.
9. GO TO PATH-1, Entry Point C.	

- END -

DIRECTION TO PROCEED TO PATH-1, ENTRY POINT "C"
FOR RECOVERY ACTIONS

NRC EXAM QUESTION 90

QUESTION: 090 (1.00)

The reactor has tripped from a 100 day run at 100% power. The "C" Steam Generator PORV is not functioning. WHICH one of the following describes the plant response to this event shortly after the trip? (Assume no operator action is taken.)

a. RCS T-Cold would be at the saturation temperature of SG safeties setpoint and RCS delta-T would be about 2-3 degrees F.

b. RCS T-Cold would be at the saturation temperature of SG safeties setpoint and RCS delta-T would be about 30 degrees F.

c. RCS T-Cold would be at the no-load value of T-Avg and RCS delta-T would be about 2-3 degrees F.

d. RCS T-Cold would be at the no-load value of T-Avg and RCS delta-T would be about 30 degrees F.

ANSWER:

b

REFERENCE:

T&AA-LP-2.17, page 12, L.O. 1.1.2

CP&L COMMENT

The listed reference deals with an analysis of an SGTR event vice a Rx trip without a significant problem. In CP&L's recommended rewrite of the question during the preview visit, a loss of AC power was also included in the question. This would result in a loss of all RCPs, both "A" and "B" SG PORVs, and condenser steam dumps. This scenario results in the answer key response of "b" as correct. However, since no loss of all AC has been stated, answer "b" is not correct due to the functioning of all RCPs and steam dumps.

RECOMMENDATION

Change the correct answer to "c."

ENCLOSURE 4

NRC Resolution of Facility Comments

SRO Examination

Question 040

NRC Resolution:

This question was provided by the facility during the pre-exam review. Reference material (PZRLC-LP-3.0, section 2.5.3c), states that "variable control heaters must be re-energized manually when level requirements satisfied" and section 2.5.3a states that the interlock that de-energizes the heaters when 1 of 2 selected channels falls below 17 percent may be overridden at the local control station. "Manual" operation may be construed as "local" station operation or "manual" operation at the control boards. In either case, it is not clear in the question stem or destructors. The facility's recommendation is accepted. The question was deleted.

Question 046

NRC Resolution:

Both destructors "a" and "b" may be evaluated as correct depending on the instantaneous event time the parameters are evaluated. Shearon Harris' automatic rod control system uses auctioneered high Tavg. At the instant of "B" loop MSIV closure, one of the 3 Tavg instruments is providing input to automatic rod control. At this point "A" and "C" loop Tavg are essentially the same until the automatic rod control dead band is overcome by the increasing "B" loop Tavg and rods commence driving in. Once rod motion commences, the Tavg in all loops begin to decrease until the Tavg-Tref and nuclear power inputs to automatic rod control is nulled within the summing amplifier. Since the evaluation period was not made clear in the stem of the question the facility's recommendation is accepted. The answer key was changed from "b" to "a".

Question 059

NRC Resolution:

The facility's comments states that both distractors "a" and "b" will <u>eventually</u> cause entry into Technical Specifications for control rod OPERABILITY. That is in fact true. Distractor "a" will cause the rod to "ratchet" inward, while distractor "b" will result in a dropped rod. The question

stem clearly states that the response is to be evaluated for only a three degree Tavg-Tref mismatch. That implies a finite period. Since the "ratchet" effect is unpredictable and may or may not result in a deviation of greater than 12 steps, the facility recommendation is accepted. The answer key was changed to accept either "a" or "b".

Question 070

NRC Resolution:

The reference material provided to the examiner for examination development supported the question and its correct response. During the post exam review the facility presented updated and revised reference material that resulted in making all distractors incorrect. Therefore, the facility's recommendation is accepted. The question was deleted.

Question 081

NRC Resolution:

The question and distractors as originally written provided only one correct response. During the facility's pre-exam review, distractor "c" was changed making both distractors "b" and "c" correct. The facility recommendation was not accepted. The answer key remains as is.

Question 090

NRC Resolution:

The original question stem is quoted as follows:

"The reactor tripped and main condenser vacuum was lost due to a loss of all AC power, a Steam generator Tube Rupture (SGTR) has also occurred. Which one of the following describes the plant response to this event?"

During the pre-exam review the stem content of the question was changed such that it yielded a complete subject change. No subsequent changes were made to the distractors including failure to define the correct response. This question stem change deleted all reference to the loss of AC power and the SGTR making the correct response "c". Therefore, the facility's recommendation is accepted. The answer key was changed from "b" to "c".

ENCLOSURE 5

SIMULATOR FACILITY REPORT

Facility Licensee: Carolina Power and Light Company

Facility Docket No.: 50-400

Operating Tests Administered on: January 23-25, 1991

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, no simulator fidelity items were identified.