Attachment 2

GPU Nuclear

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Writer's Direct Dial Number

October 29, 1987

Mr. David Lange, Reactor Engineer
Lead BWR Examiner - NRC Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Lange:

Subject: Oyster Creek Nuclear Generating Station SRO Exam Comments

The purpose of this letter is to provide GPUN comments on several questions contained in the recently administered NRC SRO exam. Your willingness to consider Oyster Creek comments prior to the grading of the exam will help produce a well structured and accurate answer key, thus ensuring an effective evaluation tool.

If you should have any questions, please contact Mr. Rod Davidson of my staff at 609-971-4186.

Very truly yours,

P. B. Fiedler
Vice President/Director, OC

PBF:RDD:ms

Attachment

cc: Mr. William T. Russell, Administrator
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

NRC Question, Answer and Reference

Question 5.01

a. Reactor power has increased from 30 on IRM range 1 to 30 on IRM range 3 in 280 seconds. The point of adding heat is determined to be 30 on IRM range 8. How much longer will it take for reactor power to reach the point of adding heat if reactor period remains constant? Show all work.

Answer 5.01

a. $P1 = P0 \exp(t/T)$ where T = period, $t = 180$ seconds, $P1/P0 = 10$

$T = t/\mathrm{LN}\ (10)$
$T = 78.2$ seconds

Then $t' = T\ \mathrm{LN}\ (P2/P1)$ where $P2/P1 = 100$

$t' = (78.2)\ \mathrm{LN}\ (100)$
$= (78.2)\ (4.605)$
$= 360$ seconds (or 6 minutes)

Reference

1. Oyster Creek: Lesson Plan 300.11, 11.5, p. 14.

2. Oyster Creek: Lesson Plan 300.11, 11.2, p. 9.

Facility Comment

At Oyster Creek, 30 on IRM range 8 is the same as 30 on IRM range 9. Therefore, power increases by a factor of 1000, not 100.

$$\frac{P2}{P1} = 1000$$

Factoring this into the power formula results in a time of 540 seconds to point of adding heat (vice 360 seconds). This should be reflected in the answer key.

Supporting Documentation - see attachment #1.

NRC Question, Answer and Reference

Question 5.07

The reactor is operating at 65% power. Power is increased to rated (100%) using recirculation flow.

DESCRIBE HOW and WHY this transient would affect the magnitude of:

a. The void fraction

b. The void coefficient of reactivity

c. Control rod worth

Answer 5.07

a. As power is increased more boiling takes place (0.5) therefore the void fraction (amount of voids) increases (0.5).

b. As power increases, more voids are formed in the region of maximum thermal neutron flux (0.25), therefore, a small change in void fraction will have a large effect on reactivity (0.25). Hence, the void coefficient of reactivity becomes more negative (increases in magnitude) with increasing reactor power (0.5).

c. As power increases, moderate density decreases (0.25) and results in more fast neutrons and fewer thermal neutrons leaving the bundle next to the control rod (0.25). Since the control rod is a thermal neutron absorber, the worth of the control rod decreases (0.5).

-- OR --

Voiding allows neutrons to travel over longer distances (0.25) and thus there is more coupling or spreading of the reactivity of one core region with another (0.25), therefore, overall rod worth decreases (0.5).

Reference

1. Oyster Creek: Lesson Plan 300.08, 8,3, pp. 20, 21, and 22.

2. Oyster Creek: Lesson Plan 300.08, 8.12, p. 64.

Facility Comment

1. Part a. Once the 100% rod pattern is established, power increases via recirculation. Flow increase will actually decrease the void fractions slightly rather than increase it as the answer key indicates. This decrease compensates for the negative reactivity associated with the doppler coefficient and full temperature heatup.

NRC Question, Answer and Reference

Question 5.07

Facility Comment (Continued)

2. Part b. Since the void fraction decreases slightly, the void coefficient will actually become less negative with an increase in reactor power due to flow changes.

3. Part c. Control rod worth is a complex, multi-variable subject and is not easily discussed in terms of only one variable (i.e., void fraction). In this case, void fraction decreases and the effect on rod worth is to increase it. However, rod worth at low power conditions is definitely greater than at high power conditions due to the ratio of peak-to-average flux in the vicinity of the rod. There are competing variables involved which make this question very difficult to answer. Suggest you accept any reasonable discussion of rod worth or delete this part of the question entirely.

4. General: All three of these answers are linked together, i.e., if the candidate misses part a. (says voids increase), then his answer to part b. and c. will be wrong. Suggest candidate be given credit if he understands and explains the relationships between void fraction and void coefficient and void fraction and control rod worth even if his answer to part a. leads him in the wrong direction.

References

Oyster Creek: Lesson Plan 300.08

NRC Question, Answer and Reference

Question 5.09

State whether each of the following statements concerning core flow are TRUE or FALSE. If a statement is FALSE, EXPLAIN why it is false.

c. If core bypass flow is less than 10%, void fraction in the bypass region is calculated by the process computer and is used to adjust the LPRM input when determining core power distribution.

Answer 5.09

c. TRUE

Reference

1. Oyster Creek: Lesson Plan 81, A.1.e (3), p. 7.

2. Oyster Creek: Thermodynamics, Heat Transfer and Fluid Flow, Chapter 9, pgs. 9-51, 9-56, 9-58 and 9-59.

Facility Comment

1. Part c. only. Oyster Creek does not have/use the process computer referred to in the G.E. Thermodynamics, Heat Transfer & Fluid Flow manual. Recommend this part of question 5.09 be deleted.

NRC Question, Answer and Reference

Question 6.01

During your shift the instrument air-to-drywell auto isolation valve (V-6-395) failed closed without any operator action. WHAT are three (3) possible signals or conditions which could have caused the valve to close?

Answer 6.01

1. loss of instrument air pressure
2. loss of both AC and DC power sources
3. MSIV auto close signal

Reference

Facility Comment

1. There are five (5) signals which could cause an MSIV auto close condition. If a candidate answers with three (3) MSIV isolation signals, he should receive full credit, based on the way the question is worded.

Reference

OC Requal Manual - RPS handout (L.P.#46) pgs. 27 & 28.

Supporting Documentation - See Attachment #2

NRC Question, Answer and Reference

Question 6.02

The lead Control Room Operator has just placed the Liquid Poison System keylock switch on panel 4F in the "System 2" position.

b. What are four (4) indications available on Panel 4F that can be used to verify that Liquid Poison is injecting?

Answer 6.02

b. 1. pump on light is on (+0.25)
   2. SQUIB FIRED light is on (+0.25)
   3. pump discharge pressure (+0.25)
   4. tank level decreasing (+0.25)

Reference

1. Oyster Creek: Station Procedure 304, 5.3.3.2, P. 11.

2. Oyster Creek: Lesson Plan 53, Section V.D.

3. Oyster Creek: Liquid Control P&ID 148F723.

4. Oyster Creek: Lesson Plan 53, Knowledge Requirement 3.

Facility Comment

6.02 b. Should also accept neutron power decreasing (APRMs, IRMs, SRMs) period negative.

Reference

Response clearly understood.

NRC Question, Answer and Reference

Question 6.03

During a reactor startup with reactor power at 20 on Range 1 of the IRMs and the mode switch in STARTUP, the 24 volt DC power panel A output breaker fails open.

DESCRIBE HOW and WHY each of the following systems are affected by this failure?

a. Neutron Monitoring System

b. Reactor Building Ventilation System

c. Liquid Process Radiation Monitoring System

Answer 6.03

a. SRMs and IRMs (powered from 3R) (+0.4) fail downscale (rod block) (+0.4) due to loss of power to the instruments (+0.21).

b. reactor building ventilation system isolates (+0.5) due to loss of power to the area radiation monitoring system (+0.5).

c. RBCCW (+0.2), Service Water (+0.2) (and Radwaste Discharge - CAF-) fails downscale (+0.4) (-CAF) due to loss of power to the instruments (+0.2).

Reference

1. Oyster Creek: Lesson Plan 12, V.B.17.b, pp. 7 and 8.

2. Oyster Creek: Station Procedure 340.2, 3.2.2, p. 5.0.

3. Oyster Creek: Lesson Plan 12, Knowledge Requirement B.

4. Oyster Creek: Lesson Plan 69, Handout 623.03, pp. 3 and 4.

Facility Comment

Typically, the operators are not required to know how all system instrumentation fails on loss of power (i.e., upscale or downscale). There are System Diagnostic and Restoration procedures to assist the operator in evaluating the failure mode and restoring conditions to normal. Suggest this question be graded on system response (i.e., loss of power to the IRMs causes an INOP trip) rather than on which way the indications fail.

References

None

NRC Question, Answer and Reference

Question 6.04

Concerning the reactor recirculation flow control system:

c. WHAT are the two (2)functions of the low limiter portion of the dual limiters in the Individual Controller circuit?

d. WHAT are the inputs to the Master Controller when the Master Controller is in "MANUAL"?

Answer 6.04

c. Provide a position signal (40-50 percent) for the scoop tube during the MG Set start sequence (+0.51).

   Place an upper limit (less than 20 percent) on the setpoint signal from the output of the individual controller unit until the loop discharge valve is fully open (+0.5).

d. Demand signal from Turbine Control (+0.25) and speed feedback signal from the MG Set (Generator) tachometer (+0.25).

1. Oyster Creek: Lesson Plan 48, V.B. and Figure 9.

2. Oyster Creek: Lesson Plan 48, Knowledge Requirement 14.

3. Oyster Creek: Station Procedure 301, 5.0, p. 24.

Facility Comment

6.04 c. The low limiter portion of the dual limiter only has one function, not two. The only function of the low limiter is the first answer in the answer key.

The second answer in the answer key is the original purpose for the high limiter, but it is no longer used. Since you did not ask for a function of the high limiter, suggest you delete the second answer and give full credit for the first answer.

6.04 d. The only input into the Master Controller in "MANUAL" is the speed feedback signal from the MG set. The answer key should be revised to reflect this.

References

Lesson Plant #48

NRC Question, Answer and Reference

Question 6.05

The rod block display on Panel 4F indicates "REFUEL INTERLOCK".

a. What combination of conditions could have caused this annunciator?

Answer 6.05

a. refuel bridge over the reactor (+0.5), and refuel grapple hoist loaded more than 480 pounds (+0.5) or frame-mounted hoist loaded more than 400 pounds (+0.5)

b. refuel (+0.16), shutdown (+0.18), and startup

Reference

1. Oyster Creek: Station Procedure 302.2, Table 302.2A, p. E1-1.

2. Oyster Creek: Lesson Plan 81, 2.b., p. 16.

3. Oyster Creek: Lesson Plan 81, Knowledge Requirement 5.

4. Oyster Creek: Lesson Plan 45, Attachment 2, pp. 1 and 2.

Facility Comments

An acceptable answer should be that the refuel grapple hoist is loaded or the frame-mounted hoist is loaded. Setpoints are not specifically asked for and should not be required.

References

None

NRC Question, Answer and Reference

Question 6.06

OCNGS Procedure ABN-3200.30, "Control Room Evacuation" directs the operator to control reactor pressure using the isolation condensers and to maintain a cooldown rate of less than 100 F/hr.

a. WHAT valves and/or pumps can be controlled from the Remote Shutdown Panel that allow the operator to operate the isolation condensers?

c. Specifically, HOW is the specified cooldown rate determined and maintained when both isolation condensers are available?

Answer 6.06

a. IC B vent valves (V-14-2 and -19) (+0.25)
IC B isolation valves (V-14-32, -33, -35 and -37) (+0.5)
IC B shell water makeup valve (V-11-34) (+0.25)

c. plotted using reactor pressure (and converting to temperature)
secure IC A
cycle IC B condensate return valve (V-14-35)

Reference

1. Oyster Creek: Station Procedure ABN-3200.30, Attachment 2, p E2-1.

2. Oyster Creek: Station Procedure ABN-3200.30, pp. 6 and 7.

Facility Comments

6.06 a. There is some confusion as to what the word "controlled" means. The only two valves which can be "controlled", or positioned, by the operator at the RSD panel are V-14-35 (condensate return valve) and V-11-34 (condensate makeup valve). Refer to attached drawing, Attachment #3, of the remote shutdown panel indications and controls. The other valves (V-14-1 & -19, V-14-32, -33, & -37) will all interlock open when control is established at the RSD panel. Suggest full credit be given for V-14-35 and V-11-34 based on the definition of "controlled".

6.06 c. Suggest that securing the isolation condenser "A" be deleted from the answer key since, by procedure, this would have been done as part of the control room evacuation and not done locally.

References

ABN-3200.30

NRC Question, Answer and Reference

Question 6.08

STATE the effect of loss of instrument air on the following valves (i.e., fail open, fail closed, fail as is, no effect, etc.):

d. offgas system valves

Answer 6.08

d. fail open

Facility Comment

6.08 d. The offgas inlet valves to the SJAE (V-7-17-28) fail open while the offgas from the condensers (V-7-1-6) fail closed. The answer key should be revised to reflect this. (See attachment #4.)

References

AEOG Lesson Plan #68

NRC Question, Answer and Reference

Question 6.09

Standby Gas Treatment System I (preselected) automatically initiated on a spurious high drywell signal which subsequently cleared.

DESCRIBE the response of the SBGTS components (i.e., fans, dampers when each of the following events occur. Consider each event separately. Be specific.

a. The initiating signal (high drywell pressure)is reset.

Answer 6.09

a. no affect on SBGTS (SBGTS continues to operate)

Reference

1. Oyster Creek: Station Procedure 330, 3.2, p. 3.0.

2. Oyster Creek: Station Procedure 330, 5.2.1.8, pp. 8.0 and 9.0.

3. Oyster Creek: Lesson Plan 50, Section V, pp. 9 and 11.

4. Oyster Creek: Lesson Plan 50, Knowledge Requirement 6.

Facility Comment

When the initiating signal (high drywell pressure) is reset, the SBGTS will shut down automatically. The answer key should be revised accordingly. (See attachment #5.)

Reference

SBGTS Procedure #330, p. 10

NRC Question, Answer and Reference

Question 6.10

With the reactor operating at 50 percent power, the following sequence of events occur:

T = 0 minutes: earthquake

T = 1 minute: reactor lo lo water level signal

T = 5 minutes: reactor lo lo water level signal clears

T = 20 minutes: loss of all offsite power

WHAT is the expected status of the standby diesel generator EDG-1 (RUNNING AND LOADED), RUNNING AT RATED SPEED UNLOADED, IDLING, NOT RUNNING TRIPPED) at each of the following times? Assume no operator action and no other equipment is damaged.

a. T = 3 minutes

b. T = 6 minutes

c. T = 18 minutes

d. T = 22 minutes

e. T = 22 minutes, assuming a loss of all 125VDC power occurred in conjunction with the loss of all offsite power (at T = 20 minutes).

Answer 6.10

a. idling
b. idling
c. idling
d. running and loaded
e. idling (the bus undervoltage device requires DC for operation.)

Reference

1. Oyster Creek: Lesson Plan 65, pp. 26 - 29.

2. Oyster Creek: Lesson Plan 65, Knowledge Requirements 22 and 23.

Facility Comment

6.10 c. The correct answer should be "not running". It should have automatically shutdown at T = 5 minutes + 11.5 which is 16.5 minutes into scenario.

6.10 e. At least one candidate asked if the 125VDC D.G. batteries were available and was told "no" since the question states all 125 VDC batteries were lost. Therefore, the student answered that the D.G. would not start since no battery power existed to start them. Suggest that, based on the assumption, full credit be given either way (i.e., 125 VDC D.G. batteries available or not available).

NRC Question, Answer and Reference

Question 7.03

In accordance with OCNGS Procedure 202.1, "Power Operations", with reactor coolant temperature in the normal operating range, WHAT restrictions are placed on recirculation pump operation at speeds of 56.0 and 57.5 cps?

Answer 7.03

The RR pumps may be operated for short periods of time, not to exceed 10% of total operating time (+0.5) at 57.5 cps with D/W temperature at or below 135 degrees F (+0.5) or 56.0 cps with D/W temperature at 150 degrees F (+0.5)

Reference

1. Oyster Creek: Station Procedure 201.3, 3.5, p. 3.0.

2. Oyster Creek: Station Procedure 202.1, 3.11, p. 6.0.

Facility Comments

This is a very unusual plant condition and the operators are aware that limitations exist at these pump speeds but are not expected to memorize actual limitations. They should, however, know the basis for the precaution (as identified in objective 0 of TCR 2611.832.03. (Attachment #6)

Recommend accepting any reasonable answer which reflects an understanding of the reason for the precaution.

References

None

NRC Question, Answer and Reference

Question 7.06

In accordance with Station Procedure 2000-ABN-3200.30, "Control Room Evacuation":

b. WHICH of the required actions can be performed using backup methods from outside the control room?

Answer 7.06

b. all except confirming all rods are beyond position 02.

Reference

1. Oyster Creek: Station Procedure ABN-3200.30, 3.2, p. 3.

2. Oyster Creek: Station Procedure ABN-3200.30, Attachment 2, p. E1-1.

Facility Comment

7.06 b. Should also accept the following:

1. Rod position can be determined from outside the control room in two ways:

   a. Teletype printout from rod worth minimizer.

   b. Perform continuity check on each individual PIP cable at the 00 position.

References

None

NRC Question, Answer and Reference

Question 7.09

Step TOR/T-3 of Emergency Procedure EMG-3200.02, "Primary Containment Control-Torus Water Temperature, "directs the operator to enter EMG-3200.01, "RPV Control" and execute it concurrently with TOR/T before torus water temperature reaches the Boron Injection Initiation Temperature.

WHAT is the basis for this action?

Answer 7.09

If boron is not injected (to shutdown the reactor) before the torus water reaches a certain temperature (as determined by Boron Injection Initiation Temperature figure) (+0.5), the amount of energy produced by the reactor (+0.5) could exceed the amount of energy that can be dissipated by the torus (+0.5).

Reference

1. Oyster Creek: EMG-3200.02, Step TOR/T-3.

2. Oyster Creek: EMG-3200.01, Step RC/Q-4.

3. Oyster Creek: EOP Bases.

Facility Comment

The answer given in the answer key is the basis for the boron injection initiation temperature graph. Should accept the more general basis for the step as given in the attached section of the "Technical Basis for the OCNGS Emergency Operating Procedures"

Basis: Scramming the reactor before temperature reaches the Boron Injection Iniatiation Temperature ensures that, if possible, the reactor is shut down by control rod insertion before the requirement for boron injection is reached. See Attachment #7.

References

Technical basis for the OCNGS Emergency Operating Procedures.

NRC Question, Answer and Reference

Question 7.11

Using the attached IMP-1300.01, "Classification of Emergency Conditions", Attachment 1, STATE the emergency classifications for the following plant conditions or events. For each event INCLUDE the initiating condition category and STATE whether the Emergency Operations Facility (EOF) must be staffed in accordance with IMP-1300.25, "Emergency Operations Facility".

a. Reactor recirculation pump seal leakage is determined to be 60 gpm.

b. Smoke in the control room which causes a control room evacuation.

Answer 7.11

a. ALERT (+0.25), RCS Integrity (+0.25), No EOF staffing (+0.25)

b. ALERT (+0.25), Control Room Indication (+0.25), No EOF (+0.25)

Reference

1. Oyster Creek: Emergency Plan Implementing Procedure IMP-1300.01, Attachment 1.

2. Oyster Creek: Emergency Preparedness Implementing Document, 6430-IMP-1300.25, p. 2.0.

Facility Comments

7.11 a.& b. A recent change to EPIP-2 requires the EOF to be manned at the alert level.

Suggest either answer is acceptable due to the recent change to the procedure. See attachment #8.

Reference

Oyster Creek Emergency Preparedness Implementing Document 6430-IMP-1300.02, Rev. 4.

NRC Question, Answer and Reference

Question 8.03

A reactor startup is in progress at OCNGS. The mode switch is in STARTUP and reactor power level is presently at 20 on IRM Range 9.

b. WHAT Technical Specification requirement must be met prior to operation in IRM Range 10? WHAT is the bases for this requirement?

e. WHAT accident transient is the IRM scram function (Range 10) designed to mitigate?

Answer 8.03

b. must have greater than the minimum recirculation flow rate [+0.5] (39.65 E6 lb/hr) to ensure transient MCPR limits are not exceeded [+0.5].

e. improper startup of an idle recirculation loop [+0.51].

Reference

1. Oyster Creek: Technical Specification, 2.3.A.2, pp. 2.3-4 and 2.3-5.

2. Oyster Creek: Technical Specifications, 2.3.H, p. 3.3-4.

Facility Comment

8.03.b - Another requirement for Range 10 operation is Rx pressure 825 psig per Tech. Specs. Section 2.

8.03.e - Should also accept a continuous rod withdrawal accident based on Tech. Specs. Page 2.3-5 (2nd paragraph of basis).

Reference

NRC Question, Answer and Reference

Question 8.04

Technical Specification surveillance requirements for systems required to be operable must be performed within specified intervals.

c. For WHAT surveillance test are there no provisions for exceeding the surveillance interval?

Answer 8.04

c. containment leak rate test [+0.5].

Reference

1. Oyster Creek: Technical Specifications, 1.24, p. 1.0-5.

Facility Comment

8.04.c - Recommend this question be deleted based on the following logic:

1. This level of knowledge of surveillance requirements is not required to be memorized.

2. The question is somewhat misleading in that the containment integrated leak rate test does leave provisions for exceeding its surveillance interval (Technical Specification Section 4.5.D).

Reference

NRC Question, Answer a'd Reference

Question 8.08

Core reload is in progress during your shift. Requests have been made by the maintenance department and I&C shop to work on the following systems:

a. Standby Liquid Control
b. Reactor Recirculation System
c. Standby Gas Treatment System
d. Reactor Manual Control
e. Reactor Vessel Instrumentation
f. Standby Diesel Generator
g. Reactor Protection System
h. Reactor Water Cleanup System
i. Fuel Pool and Cooling System
j. Nuclear Instrumentation System

WHICH of the above systems would require prior authorization from the Manager Plant Operations before maintenance activities could begin?

Answer 8.08

Yes for a., b., d., g., and j. [+0.25] each (for each listed).
No for c., e., f., h., and i. [+0.25] each (for each not listed).

Reference

1. Oyster Creek: Station Procedure 205.5, 3.25, p. 6.0.

Facility Comment

8.08. - Should accept any reasonable answers for listing additional systems due to the following:

1. This is a procedure prerequisite which the candidates are not expected to know from memory.

2. During core reload evolutions, Procedure 205.5 would be routinely reviewed to verify that these systems would not be rendered inoperable.

3. The final check and balance rests with the MPO since he reviews all maintenance requests prior to issuing them to the G.S.S. Therefore, he is, in effect, authorizing outages on those selected systems.

Reference

None

NRC Question, Answer and Reference

Question 8.10

During a maintenance outage a 23 year old male worker with a lifetime exposure of 23 REM (NRC Form 4 on file) is assigned to work in a 200 mrem/hr radiation area. The worker has received 250 mrem so far this calendar quarter.

b. WHAT is the maximum extension (of exposure limits) this worker could receive without exceeding the 10CFR20 allowable whole body exposure limits?

Answer 8.10

b. Allowable extension is 5(N-18) R lifetime or 3 R/qtr [+0.5]

Allowable exposure = 5(23-18) = 25 R -23R = 2 R (+0.5)

Reference

1. 10CFR20.101.
2. Oyster Creek: Radiation Controls Policy and Procedure Manual, 9300-ADM-4000.01, 7.2, pp. 3.0 and 4.0.

Facility Comment

8.10.b - Should also accept 2250 mrem due to clarification by the examiner that the 250 mrem the worker had already received was included in the 23 REM lifetime exposure.

Reference

None

Megawatts vs Range

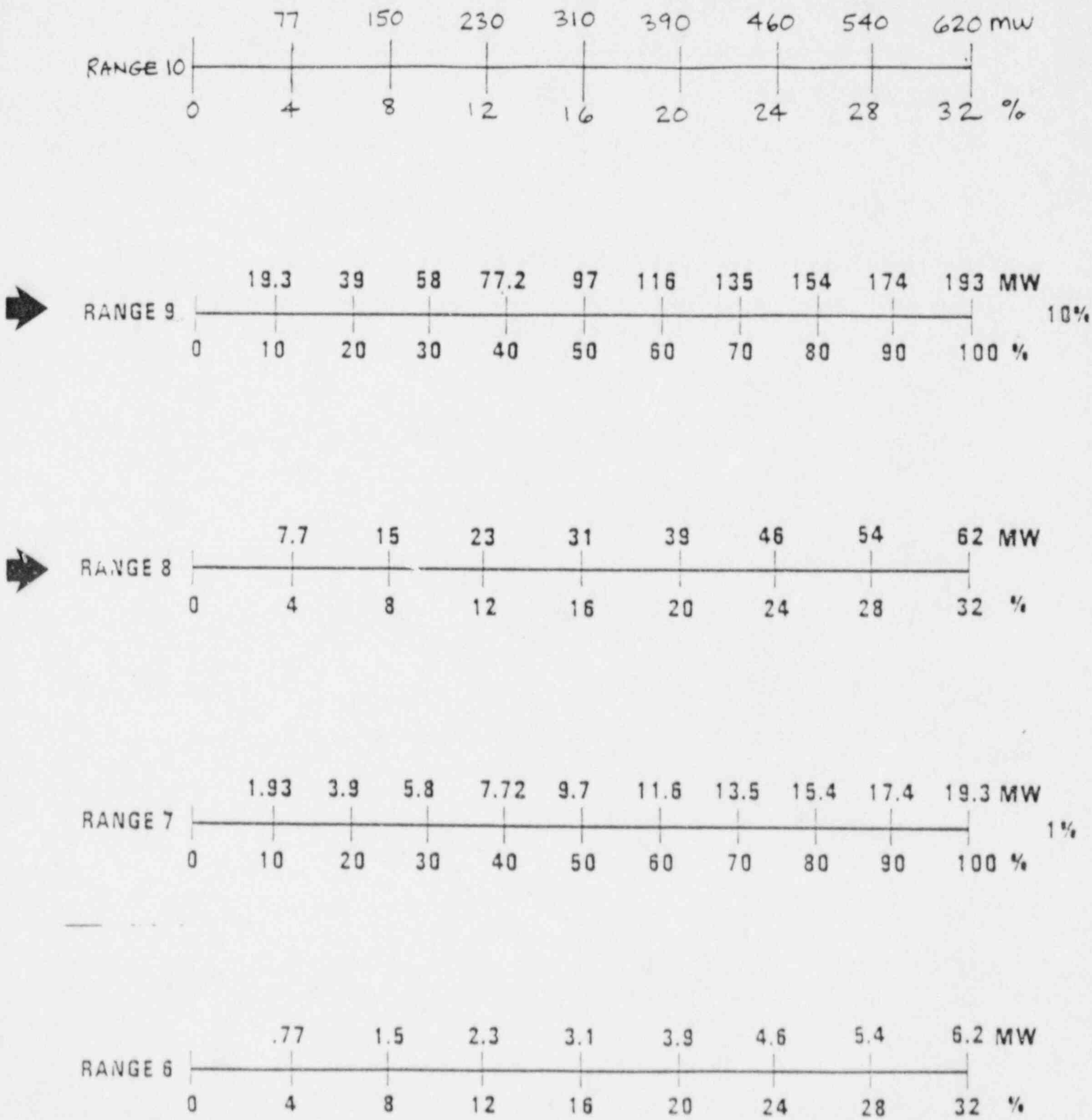


FIGURE 6B

2. Main Steam System:
(1K73 & 74, 2K73 & 74) (Figure #13)

a. Trip Signals

1) Steam Line High Rad: (1K13 & 14, 2K13 & 14)

a) 10 x Normal (Set: 600 units)

b) Same sensor and relays as scram

2) Main Steam Line Break (1K15 & 16, 2K15 & 16)

a) Steam Line High Flow

(1) 120% Rated Flow

(2) DP switches RE 22 A-H located on the North wall of the Drywell 23' elev.

b) Trunion Room High Temperature

(1) Ambient at power + 50°F (Set: 180°F)

(2) Sensors IB10A-R are located at four points along the steam tunnel.

3) Main Steam Line Low Pressure (1K117 & 118, 2K117 & 118)

a) 825 psig

b) Sensed on the two 24" headers before the 30" throttle.

c) Sensor RE23A-D located in the Feed Pump Room in cages.

d) Bypassed when mode switch is not in RUN.

4) Reactor Low Low Water Level Trip (1K19 & 20, 2K19 & 20)

a) 7'2" above the top of active fuel (0" Yarway)

b) Same instrument channel as D/W Isolation

2. Main Steam System:

   b. The following valves shut upon receipt of an isolation signal.

1)	MSIVs	V-1 - 7, 8, 9 & 10
2)	Iso. Cond. Vents	V-4 - 1, 5, 19 & 20
3)	MSL Drains	V-1 - 106, 107, 110 & 111
4)	Instrument Air/N_2 Supply	V-6 - 395
5)	Recirc Loop Sample Valves	V-24 - 28, 30

   c. Basis: Isolation of the reactor occurs if a steam line break outside the drywell occurs, a LOCA occurs, or a gross fuel clad failure occurs: This is done to limit the inventory loss from the core and the activity released to the environment.

   d. Reset Switch on 4F (353) resets the isolation trip relays. The operator must put all valve control switches in the shut position prior to resetting to prevent the valves auto opening.

3. Cleanup and Shutdown Cooling Isolation:
(1K75 & 76, 2K75 & 76)

   a. Trip Signals

      1) Reactor Low Low Level (1K19 & 20, 2K19 & 20)

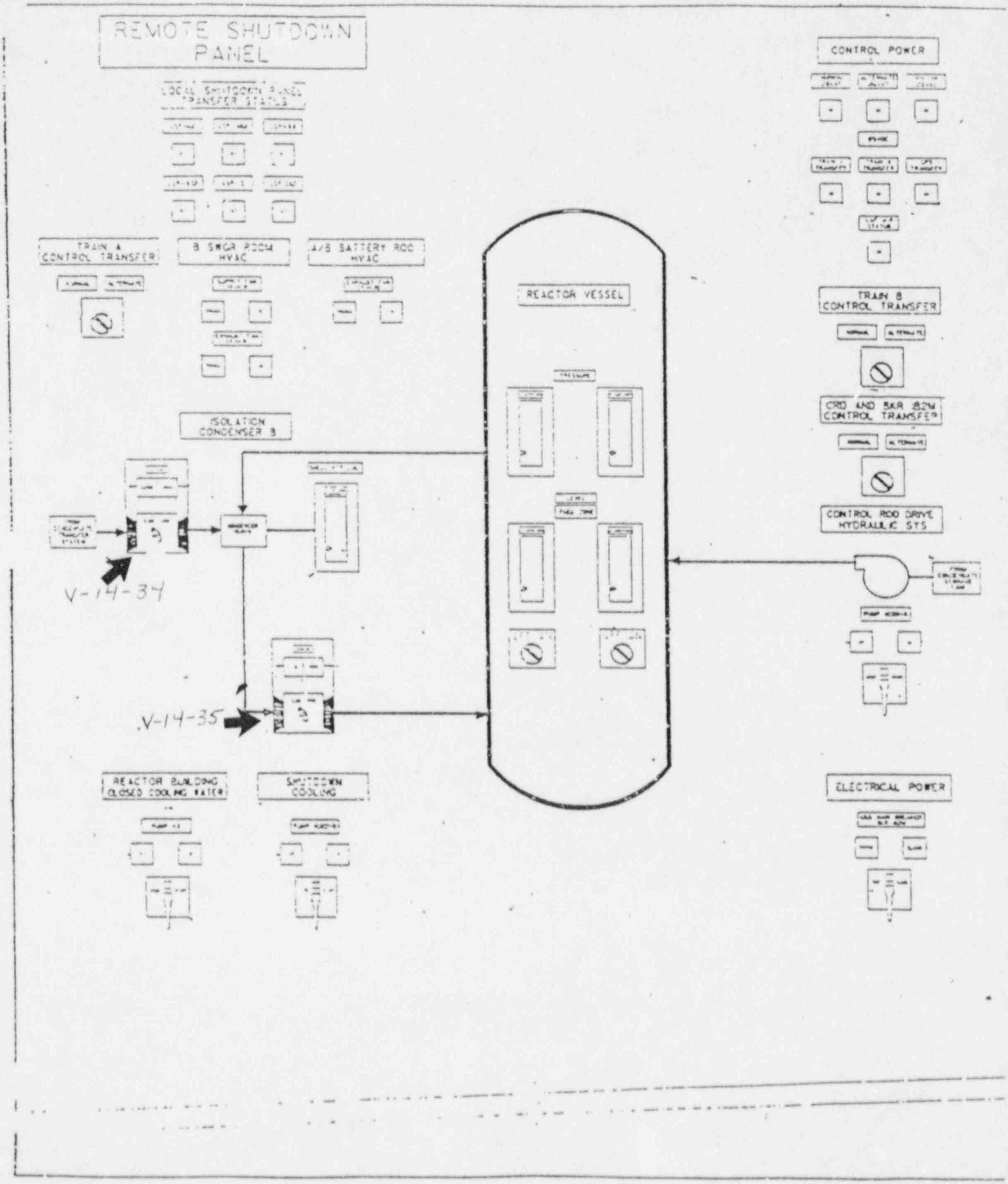
         a) Same Instrument Channel as Drywell Isolation

         b) Setpoint $\geq$ 7'2" above top of active fuel

      2) Drywell High Pressure (1K9 & 10, 2K9 & 10)

         a) Same Instrument Channel as Drywell Isolation

         b) Setpoint 2.0 psig


REMOTE SHUTDOWN PANEL
LOCAL SHUTDOWN PANEL TRANSFER STATUS
CONTROL POWER
TRAIN A CONTROL TRANSFER
B SWGR ROOM HVAC
A/B BATTERY RM HVAC
REACTOR VESSEL
TRAIN B CONTROL TRANSFER
CRD AND BKR 82M CONTROL TRANSFER
ISOLATION CONDENSER B
CONTROL ROD DRIVE HYDRAULIC SYS
V-14-34
V-14-35
REACTOR BUILDING CLOSED COOLING WATER
SHUTDOWN COOLING
ELECTRICAL POWER

(3) Off-Gas (from Condensers) -

Six Isolation Valves prior to mechanical vacuum pump tie in to piping (V-7-1, → -7-6).

- Air operated butterfly valves.
- AC power source to solenoid valves in air supply comes from Inst. Panel #48.

Loss of air on AC power - fail closed.

Four air operated valves at inlet of each SJAE, butterfly valves (V-7-17, → 7-28), two for each set.

Loss of air on DC power - fail open

DC power from Panel 7F

Each pair of off-gas inlets operates in conjunction with one steam inlet valve for each set of ejectors.

Valves are ganged together as follows:

1A1 - V-1-41	V-7-17 & 19
1A2 - V-1-42	V-7-18 & 20
1B1 - V-1-43	V-7-21 & 23
1B2 - V-1-44	V-7-22 & 24
1C1 - V-1-39	V-7-25 & 27
1C2 - V-1-40	V-7-26 & 28

Turning one switch at Panel 7F operates all 3 valves.

(3) System II (I) Inlet and Outlet Valves V-28-27 (V-28-23), V-28-30 (V-28-26) open.

(4) System I (II) Inlet and Outlet Valves V-28-23 (V-28-27), V-28-26 (V-28-30) close.

(5) System II (I) Orifice Valve V-28-28 (V-28-24) closes.

(6) System I (II) Orifice Valve V-28-24 (V-28-28) opens.

(7) Cross-tie Valve V-28-48 stays open.

(8) "TRAIN A (B) FLOW LO" Alarm (L-2(5)-b) Annunciates

CAUTION

If actuating signals no longer exist, reset of actuating signals will cause shut down of the SBGTS.

- If desired to cause shut down of SBGTS and if system initiated by the automatic startup utilizing either Reactor Building Vent Manifold CH. 1 or CH. 2 reset the Reactor Building Vent Rad Monitor.

- If desired to cause shutdown of SBGTS and if system was initiated by automatic startup as result of Reactor Low Low Water Level or high Drywell pressure, the Drywell Isolation Reset Button 3S2 must be reset. This must be accomplished in accordance with Section 10 of Procedure 312 "Reactor Containment Integrity and Atmosphere Control".

NOTE: EF 1-8 (EF 1-9) will continue to run after a low flow signal with its associated Inlet and Outlet Valves shut. Manual shutdown of the EF 1-8 (EF 1-9) is required.

5.3.2 Monitor stack release activities at Panel 10F as a verification of Standby Gas Treatment System operation.

TOR/T-3

BEFORE	Torus water temperature reaches Fig L, Boron Injection Initiation Temperature	THEN	Enter Procedure EMG-3200.01, RPV CONTROL at Step RC-1 and execute it concurrently with this procedure

A

DISCUSSION:

Scramming the reactor before torus temperature reaches the Boron Injection Initiation Temperature ensures that, if possible, the reactor is shut down by control rod insertion before the requirement for boron injection is reached. The requirement for Liquid Poison initiation is established in Step RC/Q-4 of procedure EMG-3200.01, RPV CONTROL. (Refer to Figure L for the discussion of the Boron Injection Initiation Temperature.)

The direction to enter the RPV Control procedure ensures that a reactor scram is only initiated once during any given event sequence that requires entry to the emergency operating procedures. This accomodates concurrent execution of the reactor power control section of the RPV control procedure and avoids unnecessary cycling of the control rod drive hydraulic system.

APPLICABLE CONDITIONAL STATEMENTS:

None

GPU Nuclear	OYSTER CREEK EMERGENCY PLAN IMPLEMENTING PROCEDURE	Number 6430-IMP-1300.02
Title Direction of Emergency Response		Revision No. 4

EMERGENCY DIRECTOR CHECKLIST

The Emergency Director should initial completed actions in the space provided below the appropriate emergency classification and provide the time of action and names as directed. If UE, ALERT, SAE or GE Checklist blanks have been marked for a previous classification (i.e., escalating or de-escalating) the area immediately under each blank may be used for reclassification annotations.

NOTE: Should it become necessary to evacuate the Control Room, complete Section 11.0 prior to any other.

UE ALERT SAE GE

SECTION 1.0 - EMERGENCY CLASSIFICATION & DECLARATION

____ ____ ____ ____ 1.1 Announce to CR personnel that ED duties have been assumed.

	Name	Time
On-duty GSS:	____________	______
On-call ED:	____________	______

____ ____ ____ ____ 1.2 Announce the emergency classification and the time declared.

Time: ____ UE ____ ALERT ____ SAE ____ GE

SECTION 2.0 - SECURITY NOTIFICATION

____ ____ ____ ____ 2.1 Notify the Security Shift Commander (4954, 4950) and provide him the current emergency classification.

____ 2.2 If in an Alert, direct the Shift Commander to activate tne Initial Response Emergency Organization and the Emergency Support Organization. *ESO utilizes EOF*

____ 2.3 If in a Site Area Emergency, direct the Shift Commander to conduct personnel accountability. Proceed to Section 6.0.

____ 2.4 If in a General Emergency, direct the Shift Commander to conduct site evacuation. Proceed to Section 6.0.

ATTACHMENT 3

NRC RESPONSE TO FACILITY COMMENTS

The following represents the NRC resolution to the facility comments (listed in Attachment 2) made as a result of the current examination review policy. Only those comments resulting in significant changes to the master answer key, or those that were "not accepted" by the NRC, are listed and explained below. Comments made that were insignificant in nature and resolved to the satisfaction of both the examiner and the licensee during the post examination review are not listed (i.e.: typographical errors, relative acceptable terms, minor set point changes).

Question 5.01a: Comment accepted. Error corrected.

Question 5.07: Comments accepted.

Question 5.09c: Comment accepted. Part c deleted.

Question 6.01: Alternate answer is acceptable.

Question 6.02b: Alternate answer is acceptable.

Question 6.03: Comment accepted.

Question 6.04c: Comment accepted. Referenced lesson plan did not contain clear information.

Question 6.04d: Comment accepted.

Question 6.05a: Comment accepted. Setpoints not required.

Question 6.06a: Comment accepted.

Question 6.06c: Comment not accepted. The question states that both isolation condensers are available and does not ask for locally performed operations only, therefore securing of the A Isolation Condenser is required for full credit.

Question 6.08d: Comment accepted. For full credit answer must state that the inlet valve fails open and the valve to the condenser fails closed.

Question 6.09a: Comment accepted.

Question 6.10c: Comment not accepted. The diesel will not automatically shutdown until the start signal is manually reset. The question states that no operator action is taken.

Question 6.10e: Comment accepted. If candidate answered that the diesel shutdown in part c of the question, "not running" will be accepted.

Question 7.03: Comment accepted.

Question 7.06b: Comment accepted. Answer was not contained in reference material and no reference was submitted with comment.

Question 7.09: Alternate answer is acceptable. Alternate answer was not contained in reference material submitted for exam preparation.

Question 7.11: Comment not accepted. The change to the procedure is dated 4/87. Operators are required to be trained on procedure revisions. Answer key was changed to reflect manning the EOF at the Alert level.

Question 8.03b: Comment accepted. Reactor pressure above 825 psig is required for full credit.

Question 8.03e: Comment not accepted. Reference does not support additional answer.

Question 8.04c: Comment accepted. Part c deleted.

Question 8.08: Comment not accepted. The G.S.S. in releasing the work has responsibility for determining if the work can be performed without MPO authorization.

Question 8.10b: Alternate answer is acceptable.

ATTACHMENT 4

PROCEDURE LIST

101 - Organization and Responsibility

108.4 - Control of Plant Modifications and Major Maintenance Work in Critical Plant Areas While the Plant is in Operation

112.1 - Technical Specification Supporting Installed Instrumentation

205.9 - Core/Pool Fuel Transfers

218 - Operation Below 10% Rated Power with the Rod Worth Minimizer Bypassed or Inoperable

312.1 - Bypassing Isolation Interlocks During Emergency Conditions

346 - Operation of the Remote and Local Shutdown Panels