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

EXAMINATION REPORT NO. 88-07 (OL)

FACILITY DOCKET NO. 50-219

FACILITY LICENSE NO. DPR-16

LICENSEE: GPU Nuclear Corporation
P. O. Box 388
Forked River, New Jersey 08371

FACILITY: Oyster Creek Nuclear Generating Station

EXAMINATION DATES: April 11 - 14, 1988

CHIEF EXAMINER: Terry E. Walker 7/5/88
T. Lumb, Senior Operations Engineer Date

APPROVED BY: David J. Lange 7-14-88
David J. Lange, Chief, BWR Section Date
Operations Branch, Division of Reactor Safety

SUMMARY: Written examinations and operating tests were administered to one (1) senior reactor operator (SRO) candidate and three (3) reactor operator (RO) candidates. All of the candidates passed the examinations.

DETAILS

TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written	3 / 0	1 / 0
Operating	3 / 0	1 / 0
Overall	3 / 0	1 / 0

1. CHIEF EXAMINER AT SITE: T. Lumb, Senior Operations Engineer

2. OTHER EXAMINERS: M. Daniels, Examiner (Sonalysts)
   S. Pullani, Senior Operations Engineer (Examiner in Training)
   T. Fish, Operations Engineer (Examiner in Training)

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

   STRENGTHS

   a. Security awareness

   b. Radiological Control procedures

   c. Knowledge of remote shutdown equipment and procedures

   DEFICIENCIES

   a. Understanding of diesel generator component operations

   b. Reactor Operator ability to locate items in Technical Specifications

4. The following is a summary of generic strengths and deficiencies noted from the grading of the Reactor Operator written examinations. (No generic information is available for the Senior Reactor Operator written examination due to the single candidate.) This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

STRENGTHS

a. Knowledge of precautions for operating centrifugal pumps - Question 1.06

b. Knowledge of thermal limit failure mechanisms, limiting conditions and limiting parameters - Question 1.08

c. Understanding of the effects of increasing power on core flow and recirc pump NPSH - Question 1.09

d. Understanding of overall plant system design - Section 2

e. Understanding of Core Spray system initiation - Question 3.05

f. Knowledge of startup procedure requirements affecting control rod movement - Question 4.03

g. Understanding of the procedure for RBCCW failure response - Question 4.04

h. Knowledge of the verification requirements for operation of Standby Liquid Control - Question 4.05

i. Understanding of Shutdown Cooling system operational precautions - Question 4.06

j. Understanding of recirculation pump operational precautions - Question 4.09

DEFICIENCIES

a. Understanding of xenon effects on reactor power following a transient - Question 1.01

b. Understanding of plant response to a rod withdrawal - Question 1.10

c. Understanding of Containment Spray system initiation - Question 3.05

d. Understanding of ADS operation - Question 3.09

e. Knowledge of the bases behind various main turbine trip signals - Question 3.10

f. Knowledge of 10CFR20 and Oyster Creek administrative radiation exposure limits - Question 4.10

5. Personnel Present at Exit Interview:

NRC Personnel

T. Lumb, Senior Operations Engineer
S. Pullani, Senior Operations Engineer
T. Fish, Operations Engineer
M. Daniels, Examiner (Sonalysts)
E. Collins, Resident Inspector

Facility Personnel

P. Fiedler, Vice President and Director, Oyster Creek
E. Fitzpatrick, GPU Nuclear
J. Sullivan, Director, Plant Operations
H. J. Lapp, Jr., Manager, Plant Training
R. Davidson, Manager, Operator Training
H. Tritt, Supervisor, License/Non-License Operator Training
M. Heller, Oyster Creek Licensing Engineer

6. Summary of NRC Comments Made at Exit Interview:

During the administration of the written examination there was distraction from technicians working in the building. This was resolved following discussion with the technicians. The comments on the written examinations should be sent to the NRC within five working days.

There were some problems with access to vital areas, but the problems were resolved promptly during the operating examiantions. Health Physics and Operations personnel were cooperative. The generic strengths or weaknesses noted on the operating examinations were presented (see section 3 of this report). The results of the examinations would not be discussed at the exit meeting but would be contained in the Examination Report. Every effort would be made to send the candidate's results in approximately 30 working days.

The examiners noted a problem with utilizing the Emergency Operating Procedure (EOP) flowcharts in their present form. The flowcharts are stored rolled up and the candidates had difficulty reading them in the unrolled position.

7. Summary of Facility Comments Made at Exit Interview:

The facility training personnel commented that the operating examinations were lengthy and that there were fewer questions on the written examinations than on past examinations.

Attachments:
1. Reactor Operator Written Examination and Answer Key
2. Senior Reactor Operator Written Examination and Answer Key
3. Facility Comments on Written Examination after Facility Review
4. NRC Response to Facility Comments

REACTOR OPERATOR EXAM

NRC Question, Answer and Reference

Question 1.02

While the plant is operating at 100% power, a malfunction occurs in a Recirculation Pump MG Set Speed Controller, causing it to attempt to attain full speed at maximum acceleration.

DESCRIBE ANY CHANGES THAT TAKE PLACE in the PERCENT VOIDS, REACTOR POWER and MODERATOR TEMPERATURE in the twenty (20) seconds following the malfunction. EXPLAIN the reasons for the changes.

Answer 1.02

a. The percent voids decreases due to the increased recirculation flow moving the void boundary higher in the core.

b. Reactor power rapidly increases to the scram setpoint due to the increased moderation in the volume where the voids had been displaced.

c. Moderator temperature will increase due to less sub-cooling for the higher recirculation flow.

Reference

1. Oyster Creek Nuclear Generating Station
2. Lesson Plan 201, BWR Operating Characteristics, p. 42.
3. Learning Objective 823.03 1-C-12

Facility Comment

Moderator Temperature may be seen to decrease due to increased feed flow leading to increased sub-cooling. Should also accept an answer of moderator temperature remaining unchanged due to averaging effects of transient and the fact that once voids are formed, the saturated system effect maintains moderator temperature relatively constant.

Reference: None

NRC Question, Answer and Reference

Question 1.04

For each of the following conditions, EXPLAIN WHICH coefficient of reactivity acts FIRST to change the power. Indicate if power INCREASES or DECREASES due to the coefficient. (Assume no RPS action).

a. The MSIV's close at 100% power.

b. Feedwater injection temperature decreases 30 F in three (3) minutes.

c. A fully inserted rod drops out of the core at power.

Answer 1.04

a. Void coefficient power increases.

b. Moderator temperature coefficient power decreases.

c. Doppler (Fuel temperature) coefficient power decreases.

Reference

1. Oyster Creek Nuclear Generating Station
2. Lesson Plan 300.08, Reactivity Coefficients and Control, Rod Worth, pp. 17 and 28.
3. Learning Objective 842.08

Facility Comment

Misspelling in answer of Part b. A decreasing moderator temperature will cause a power increase, not decrease. Examiner agreed.

Reference: LP 300.08, pp. 32-40

NRC Question, Answer and Reference

Question 1.05

Steam enters the turbine control valves at 935 psig with a st*am quality of 88%.

DETERMINE THE FOLLOWING

a. The steam temperature.

b. The specific enthalpy of the steam.

Answer 1.05

a. From the Steam Tables 935 psig = 950 psia (0.25) = 538.4 F

b. hf = 534.7, hfg = 550.4, hg = 1192.9

hm = hf + .88 hfg

hm = 534.7 + .88 (650.4)

hm = 534.7 + 572.4

hm = 1107.1 btu/lb

Reference

1. General El ctric BWR Academics Series 1985
2. Heat Transfer and Fluid Flow., pp. 5-56 and 5-57
3. Learning Objective 852.06 1-C

Facility Comment

Sh accept reasonable values using the Mollier Diagram also. Additionally, we . .int value for the calculations in this section are too high and shou consider reducing the point va'ue.

Reference: Mollier Diagram

NRC Question, Answer and Reference

Question 1.06

Running a pump at full discharge pressure and no flow is referred to as "dead heading".

Answer each of the questions listed below concerning the operation of pumps.

a. A centrifugal pump should not be run for extended periods of time "dead headed". WHAT IS THE REASON for this precaution?.

b. WHAT design feature of the Core Spray pumps is used to minimize the possibility of "dead heading" the pump?

c. WHAT design feature is used on Standby Liquid Control pump to minimize "dead heading" the pump?

d. When starting a large centrifugal pump, the downstream system should be filled and the discharge valve closed. WHAT ARE two (2) reasons for this precaution?.

Answer 1.06

a. If the pump is run for extended periods of time at shutoff head, sufficient heat would be added to the fluid to cause cavitation and resulting internal pump damage.

b.1 A recirculation valve is installed between the pump and the discharge valve to open on low system flow.

2 A relief valve is installed between the pump and the discharge valve, set to lift below the shutoff head of the pump.

NRC Question, Answer and Reference

Answer 1.06 (Continued)

c.1 With the discharge valve open, a la ge mass of fluid woulu be moved causing a high torque and excessive starting currents, which could damag the motor windings.

2 If the downstream system were not filled (not providing back pressure) the pump could go to "runout" conditions resulting in damage to the motor windings.

Reference

1. General Electric BWR Academics Series 1985
2. Heat Transfer and Fluid Flow., pp. 6-108 and 6-109
3. Learning Objective 817.02 1-A-1, 1-A-6, 1-A-8 and 1-J

Facility Comment

In Part d of this question, another problem with starting a pump with the downstream ping not filled is the damage that can be done by water hammer. Should consider water hammer as an acceptable answer.
Answer Key is misnumbered also. Answer b.2 is in reality c, and c.1, c.2 are d.1, d.2

Reference: None

NRC Question, Answer and Reference

Question 1.07

The Oyster Creek Main Condenser maintains a sub-cooled condition for condensate depression.

a. WHAT is one (1) advantage and one (1) disadvantage of having condensate depression in the condenser.

b. For a condensate depression of 10 F, what would be the TEMPERATURE at Condensate Pump suction with a condenser vacuum of 27" Hg?

Answer 1.07

a.1 Assure net positive suction head to the condensate pumps.

2 Reduces plant overall efficiency.

b. 27" Hg = 1.47 psia
Sat temperature for 1.47 psia = 109 F
10 condensate depression = 109-10 = 99 F (+/- 1 F)

Reference

1. General Electric BWR Academics Series 1985
2. Heat Transfer and Fluid Flow., p. 7-45
3. Learning Objective 852.06 1-G

Facility Comment

Should consider use of values in the temperature portion of the Steam Tables in addition to the pressure portion. 1.43# = 114 F and 1.51# = 116 F. Should accept answer based on these numbers. Additionally, should consider reducing the point value of the calculation.

Reference: Table 1: Saturated Steam; Temperature Table

NRC Question, Answer and Reference

Question 1.08

On the attached chart (Figure 2), IDENTIFY the areas indicated by letters A through E.

Answer 1.08

a. Fuel cladding cracking due high stress.

b. MLHGR

c. Clad temperature of 22 F

d. Fuel Cladding Cracking due to lack of cooling

e. CPR

Reference

1. General Electric BWR Academics Series 1985
2. Heat Transfer and Fluid Flow., p. 9-15
3. Learning Objective 852.09 1-K and 1-0

Facility Comment

In Part b of question LHGR was misspelled MLHGR.

Reference: GE HTFF, Chapter 9, p. 9-69

NRC Question, Answer and Reference

Question 2.01

An automatic initiation signal for the Isolation Condenser System has been received (High Drywell pressure or Low RPV level), and the condensers are in operation, when a steam line break occurs BETWEEN the pressure vessel and the differential pressure detectors for the "A" condenser.

a. WILL the "A" Isolation Condenser isolate? EXPLAIN.

b. On an isolation signal, WILL a LOSS of 480 volt AC power PREVENT the ISOLATION of the condenser? EXPLAIN.

c. WILL a LOSS of Instrument and Service Air PREVENT AUTOMATIC MAKEUP to the Isolation Condenser? EXPLAIN.

Answer 2.01

a. Yes. The differential pressure detectors respond to a differential pressure (with steam flow passing in either direction).

b. No. Each steam supply and condensate return line has two (2) MOV isolation valves in series. One valve is 480 volts AC and the other is 125 volts DC. On loss of 480 volt AC power the DC valves will still close.

c. No. An air accumulator provides air to operate the makeup valves at least six (6) times with the complete loss of Instrument and Service Air.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 23; Isolation Condenser System, pp. 9, 10 and 13
3. Learning Objective 823.23 I

NRC Question, Answer and Reference

2.01 (Continued)

Facility Comment

There is no automatic makeup to the isolation condensers. (Part c). OC has remotely operated makeup valves instead of automatic. The accumulators will allow continued operation from the remote location up to six times.

Reference: OPM Module 23, p. 23-17

NRC Question, Answer and Reference

Question 2.02

During the plant startup the Main Steam Isolation Valves (MSIV's) are being opened.

a. 125 volt DC power is lost to MSIV-04A Air Supply Solenoid #1. WILL this loss of power prevent the MSIV from being opened? EXPLAIN.

b. On a loss of Instrument and Service Air (including the depressurization of the accumulators), WILL the MSIV's close on an isolation signal? EXPLAIN.

c. The MSIV's have a minimum and maximum closing time. STATE the minimum and maximum closing time and EXPLAIN why each of these limitations is required.

Answer 2.02

a. No. Loss of power to the DC solenoid #1 allows it to remain in the deenergized position. Flow from the #2 AC solenoid valve will pass through the #1 DC solenoid valve and pressurize the header.

b. Yes. Spring pressure alone is sufficient to close the MSIV, with no pneumatic supply.

c.1 Minimum 3 seconds. Minimizes pressure buildup in the reactor vessel due to the determination of steam flow.

2 Maximum 10 seconds. Limits Off Site Dose Rates in the event of a steam line break outside the Drywell.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 26; Steam Systems, pp. 23 to 25
3. Learning Objective 828.26 B-1

NRC Question, Answer and Reference

2.02 (Continued)

Facility Comment

MSIV's will close on a loss of air to the valve, with or without an isolation signal (Part b). Should accept answer stating that the MSIV's will already be closed.

Reference: OPM Module 26, p. 26-24

NRC Question, Answer and Reference

Question 2.03

The Core Spray System has initiated on Low Low Reactor Water Level.

DESCRIBE THE EFFECTS which occur on the following components as result of the initiation signal. Include any time delays and/or equipment failures.

a. Emergency Diesels

b. Priority Core Spray Pumps (A and B)

c. Core Spray Pumps C and D

d. Priority Booster Pumps (A and B)

e. Booster Pump C and D

f. The parallel isolation valves

Answer 2.03

a. The Emergency Diesels start.

b. Priority Core Spray Pumps (A and B) start.

c. Core Spray Pumps C and D start in ten (10) seconds if Priority Core Spray Pumps A and B fail to start.

d. Priority Booster Pumps (A and B) start.

e. Booster Pumps C and D start five (5) seconds after Booster Pump A or B fails to start.

f. The parallel isolation valves open when pressure reduces to 285 psig.

NRC Question, Answer and Reference

2.03 (Continued)

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 10; Core Spray System, pp. 33 to 34
3. Learning Objective 828.10 D

Facility Comment

Should also accept the actual opening setpoint for the parallel valves in the Core Spray System (Part F) which is 300#.

Reference: Standing Orders #1

NRC Question, Answer and Reference

Question 2.04

An ELECTRICAL FAILURE has caused an Electromatic Relief Valve (EMRV) to open while at 100% power.

a. WHAT would be the FIRST indication received in the Control Room that a relief valve was open?

b. LIST two (2) indications, available to the Control Room Operator, that will determine WHICH relief valve has opened.

c. HOW can the position of a relief valve in an intermediate position be determined?

Answer 2.04

a. Control Room Alarms "EMRV Open" and/or "SV/EMRV Not Closed" would annunciate.

b.1 The Valve Monitoring System (VMS) provides individual valve position indication on Panel 1F/2F.

2 Red and Green solenoid indicating lamps on Panel 1F/2F indicate OPEN and CLOSED for individual valves.

c. Panel 15R has meters which indicate the position of the valve based on VMS amplifier outputs.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 05; pp. 11, 12, 13, 14 and 27
3. Learning Objective 828.05 I

NRC Question, Answer and Reference

2.04 (Continued)

Facility Comment

Part a. Operators are not required to know the actual names for the alarms they will receive. Should accept reference to an alarm for the EMRV's.

Part b. Can also get indication for which valves are open from the VMS Panel 15R.

Part c. The VMS indication on 1F/2F also shows the postulated position of the valve. However, since this is based on noise levels, the VMS cannot determine the actual position of an intermediate valve. There is no accurate mechansism for determining actual valve position. Should accept answers referring to above.

Reference: OCNGS Learning Objectives 828.05
OPM Module 5, pp. 12, 14, 15

NRC Question, Answer and Reference

Question 2.07

The Reactor Feedwater Pumps may operate at greater than design flow conditions, in response to level control demand signals, resulting in pump runout conditions.

a. What PROTECTION IS PROVIDED to prevent the pump/motor from being damaged due to pump runout. (Include setpoints if applicable)

b. WHAT two (2) methods/conditions which will reset the pump runout PROTECTIVE FUNCTION?

c. HOW DOES THE SYSTEM RESPOND if the runout is reset while the valve controller output is reading upscale?

Answer 2.07

a. The flow control valve locks up at 2.67 x EE6 lbs/hour.

b.1 The Valve Monitoring System (VMS) provides individual valve position indication on Panel 1F/2F.

2 Red and Green solenoid indicating lamps on Panel 1F/2F indicate OPEN and CLOSED for individual valves.

c. Panel 15R has meters which indicate the position of the valve based on VMS amplifier outputs.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 05; pp. 11, 12, 13, 14 and 27
3. Learning Objective 828.05 I

NRC Question, Answer and Reference

2.07 (Continued)

Facility Comment

The runout protection setpoint (Part A) can vary depending on Condensate Demineralizer differential pressure. Should not require setpoint because it does vary.

Reference: General Operating Procedure 230.0

NRC Question, Answer and Reference

Question 2.08

Reactor Recirculation Pump seal pressures are monitored to determine proper opэration of the seals.

a. STATE what type of seal failure is indicated by the following conditions, if they occur while operating at 100% power.

1) The number 2 seal pressure is 850 psig.

2) The number 2 seal pressure is 150 psig.

b. Under what two (2) conditions MUST a Recirculation Pump be shutdown due to seal failure?

Answer 2.08

a.1 Failure of the number one (1) seal.

2 Failure of the number two (2) seal.

b.1 If the number one (1) seal temperature increases to 180 F.

2 If the number two (2) seal temperature increases to 160 F.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 38A; Reactor Recirculation System, pp. 34 and 35
3. Learning Objective 828.38 E

NRC Question, Answer and Reference

2.08 (Continued)

Facility Comment

Part a. Also, accept for #2, the fact that the #1 seal restrictive oriface could be plugged.

Part b. Per OC Learning Objectives 828.38, the operator is not required to memorize the actual trip points based on seal failure. Before reaching the required temperatures the CRO will receive an alarm condition and the alarm response procedure will direct his action on tripping the pump. Suggest consideration of answers based on this fact.

Reference: OPM Module 38A, p. 38A-10
OC Learning Objectives 828.38
RAP 3024.01 E-7-B

NRC Question, Answer and Reference

Question 2.09

The Standby Gas Treatment System has initiated on Lo Lo Reactor Water Level.

a. At what water level does the system initiate?

b. LIST the four (4) additional initiating signals that will start the Standby Gas Treatment System. (Include setpoints if applicable)

c. How do the following system/components RESPOND when the Standby Gas Treatment System initiates?

1) Outside air supply dampers.

2) Reactor Building Ventilation.

3) Drywell coolers.

Answer 2.09

a. 86 inches above the top of the active fuel (TAF).

b.1 Reactor Building Rad Monitor High; 13 mr/hr.

2 North Wall High Vent Monitor Trip; 70 mr/hr.

3 Operating Floor High Radiation Trip; 70 mr/hr.

4 High Drywell Pressure; 3.5 psig.

c.1 No automatic action.

2 Isolation

3 No automatic action.

NRC Que-'..on, Answer and Reference

2.09 (Continued)

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 42; Secondary Containment, pp. 47, 48
3. Learning Objective 828.42 F and M

Facility Comment

Part b. Should accept either Tech Spec or actual system setpoints for this question.

b.1 13 mr/hr (actual) or 17 mr/hr (T.S.)

b.2 70 mr/hr (actual) or 100 mr/hr (T.S.)

b.3 70 mr/hr (actual) or 100 mr/hr (T.S.)

b.4 3.0 # (actual) or 3.5 # (T.S.)

Part c. There are two outside air supply damper sets associated with the Rx Building supply fans. One set is manually adjusted to set the pressure in the building. The other set are on the discharge of each supply fan and close on a trip of the supply fan to prevent back flow through an idle fan. Should consider answers based on either or both sets of dampers.

Reference: Standing Order #1

NRC Question, Answer and Reference

Question 3.01

a. WHICH three (3) scrams are defined as part of the MANUAL scram circuit?

b. During initial core loading, additional reactor protection is provided by removing the non-coincident jumpers.
WHAT effect does this have on the RPS system?

c. DESCRIBE the EFFECT on the RPS system if the Mode Switch is in STARTUP, reactor pressure is set at 700 psig and an IRM Switch is placed in range ten (10).

Answer 3.01

a.1 Manual

2 Mode Switch to SHUTDOWN

3 Non-coincident scram (Initial Fuel Loading)

b. Changes the RPS response to Nuclear Instrumentation from a one-out-of-two-taken-twice logic to a non-coincident one-out-of-twenty logic for a reactor scram.

c.1 Placing an IRM in range ten (10) with the Mode Switch not in RUN causes a Main Steam Line Isolation.

2 With the Reactor pressure below 825 psig, the Main Steam Line Isolation results in a reactor scram.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 37; Reactor Protection System, pp. 26, 42 and 44
3. Learning Objectives 828.37 1-A and 1-D

NRC Question, Answer and Reference

3.01 (Continued)

Facility Comment

Non-coincidence jumpers (Part a) are not covered as part of the scram circuitry because they have been installed since the initial fuel load and this scram probably will never be used again. Suggest deleting this portion of question.

Reference: OC Lesson Plan 2610.828.37

NRC Question, Answer and Reference

Question 3.02

The Oyster Creek plant is operating at 80% power when the reactor level signal to the Feedwater Level Control System is lost.

WHAT is EFFECT, if any, on the following components/parameters due to the loss of the level signal? (Assume no operator action for two (2) minutes.)

a. Actual level

b. Feed flow

c. Feedwater Regulating Valves

d. Main Turbine

Answer 3.02

a. Level increases

b. Feed flow increases

c. Feedwater Regulating Valves lockup (on Feed Pump Turbine runout)

d. Trips (on high reactor vessel level).

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 28; Feedwater Control, pp. 19, 20
3. Learning Objectives 828.18 F-c

Facility Comment

Answer Key references feed pump turbines. OC has motor driven feed pumps..

Reference: OPM Module 17

NRC Question, Answer and Reference

Question 3.03

Core Spray Pipe Break Detection is provided for the Oyster Creek Core Spray System in the form of differential pressure instrumentation.

Answer the following questions concerning the break detection system.

a. What is the NORMAL DIFFERENTIAL PRESSURE indicated by the system?

b. HOW would it be possible to determine if a line break were BETWEEN the CORE SHROUD and the VESSEL WALL, OR in the DOWNCOMER ANNULUS?

Answer 3.03

a. -2.0 inches of water (+/- 0.5 inches)

b. A break between the core shroud and the vessel wall would indicate a delta P of approximately 6 psid while a break in the downcomer would indicate approximately 8-10 psid (due to the differential pressure across the steam dryer and separator).

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 55; Reactor Vessel Instrumentation System, pp. 23, 24

Facility Comment

Part b of this question caused much confusion as to actual areas in RPV being discussed. Suggest deleting Part b of this question.

Reference: None.

NRC Question, Answer and Reference

Question 3.05

a. LIST the intiating signals for the Containment Spray System.

b. LIST the equipment that automatically starts on initiation of the Containment Spray System. (Include the time frame when equipment starts.)

Answer 3.05

a.1 Low Low Reactor Water Level 86" above the top of active fuel TAF AND

2 High Drywell Pressure 1.85 psig.

b.1 Containment Spray Pumps A nd C start 34-46 seconds after auto initiation signal is received.

2 Corner Room Cooling Fans start with the spray pump start.

3 Emergency Service Water Pumps A and C start 48-52 seconds after the Containment Spray Pumps start.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 9; Containment Spray and Emergency Service Water Systems, pp. 24, 15 and Table 09-2

NRC Question, Answer and Reference

3.05 (Continued)

Facility Comment

Part a. Accept actual value or Tech Spec value for drywell high pressure signal. 3# (actual) and 3.5# (T.S.).

Part b. The 5% valve also opens on an initiation of the Containment Spray System 20 seconds after sensing 1500 gpm flow rate. Suggest do not take credit off for discussing this valve or auto start of corner room fans.

Reference: Standing Order #1
OPM Module 9, pp. 12, 14 and 22

NRC Question, Answer and Reference

Question 3.08

The Oyster Creek reactor is at 100% power.

For each of the valve conditions listed below, STATE if a (FULL SCRAM, HALF SCRAM or NO RPS ACTION) results.

a. TSV 2 is less than 90% open.

b. TSV 1 and 4 are less than 90% open.

c. TSV 2 and 4 are less than 90% open.

d. Emergency trip oil pressure drops to 150 psig on TCV 1 (A).

e. Emergency trip oil pressure drops to 150 psig on TCV 1 (A) AND TCV 1 (C).

Answer 3.08

a. NO RPS ACTION

b. NO RPS ACTION

c. ONE HALF SCRAM

d. ONE HALF SCRAM

e. FULL SCRAM

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 37; pp. 26 through 30
3. Learning Objective 828.37 I-N

NRC Question, Answer and Reference

3.08 (Continued)

Facility Comment

Oyster Creek has no trip funtions off of TCV position, or ETO pressure as sensed on the TCV (Parts d and e). The ETO pressure for the generator load reject scram is sensed off of 4 acceleration relays in the turbine controls system. Should accept "No Action" answers for these questions or answers involving ETO and the acceleration relays.

Reference. OPM Module 37, p. 29

NRC Question, Answer and Reference

Question 3.10

For the turbine trips listed below, LIST the SETPOINT, and WHAT PROTECTION is provided by the trip.

a. Emergency Governor.

b. Vacuum Trip #1.

c. Reactor High Level.

d. Turbine No Load With Second Stage RSCV Not Shut.

Answer 3.10

a. 110% Prevents turbine damage due to overspeed.

b. 22" Hg Prevents overpressurizing the Main Condenser.

c. 175" TAF Protects against water damage to the main turbine blades.

d. 130 MW Protects the reheaters from high differential temperatures between the tube and the shell.

Reference

1. Oyster Creek Nuclear Generating Station
2. Operations Plant Manual Module 50; Main Turbine Table 2
3. Learning Objective 828.50 1-2-C

Facility Comment

Should also accept 20% load for Part d.

Reference: OPM Module 50, Table 50-2, Page 1 of 2

NRC Question, Answer and Reference

Question 4.01

In accordance with Standing Order 39, due to a design deficiency of the containment isolation relay, on the receipt of a containment isolation signal, the operator is directed to "place the Torus sample valves and the pump selector switch to the OFF position.

EXPLAIN WHY this action is required, and what the CONSEQUENCES could be if this action were not taken, and the containment isolation relay failed.

Answer 4.01

Failure of the containment isolation relay can prevent the automatic closure of the Torus O2 analyzer valves (V-38-22 and V-38-23). Failure of the valves to go closed could result in an off site radiation release.

Reference

1. Oyster Creek Nuclear Generating Station
2. Standing Order 39

Facility Comment

Failure to close these valves may breach primary containment and cause a release into the Secondary Containment but may not cause a release to off site as the SBGTS is also placed in service at the same time. Accept answers discussing a release to the Secondary Containment only.

Reference: Standing Order #1 and #39, Page 2

NRC Question, Answer and Reference

Question 4.02

In accordance with Oyster Creek Nuclear Generating Station Procedure 106, there are circumstances when the Shift Control Room Operator has the authority to shutdown the reactor.

INDICATE YES OR NO if the Shift Control Room Operator has the authority to shut down the reactor for the following circumstances.

a. Verified operating parameters should have initiated a scram and no scram has occurred.

b. In the Control Room Operators, judgement a situation exists which jeopardizes or threatens to jeopardize public or plant safety.

c. When verified operating parameters should have initiated a safeguard system and no initiation occurred.

d. When in the Control Room Operators opinion a violation of the Technical Specifications has occurred.

Answer 4.02

a. Yes

b. No

c. Yes

d. No

Reference

1. Oyster Creek Nuclear Generating Station Procedure 106, p. 11

NRC Question, Answer and Reference

4.02 (Continued)

Facility Comment

Part b. Accept yes as this is an entry condition into RPV control which directs the CRO to scram the Rx.

Part c. Accept yes if Tech Spec violation discussed is an LSSS.

Reference: RPV Control Procedure EMZ-3200.01
Administrative Procedure 106, p. 4

NRC Question, Answer and Reference

Question 4.03

A reactor startup is in progress at Oyster Creek.

Answer the following questions in accordance with Oyster Creek Operating Procedure 201.1 Approach to Criticality.

a. WHAT ACTION(S) is (are) required by the Reactor Operator if the Rod Worth Minimizer (RWM) becomes inoperable before the first 12 rods are withdrawn? WHAT NOTIFICATIONS are required to be made?

b. WHAT LIMITATION is placed on control rod movement, when the SRM's indicate three (3) count rate doublings of the initial count rate?

Answer 4.03

a.1 Cease rod withdrawal operations.

2 Notify the Manager, Plant Operations and the Core Manager.

b. All further control rod movements must be "notched" from position 06 to 48.

Reference

1. Oyster Creek Nuclear Generating Station Procedure 201.1, pp. 6 and 8

Facility Comment

Part a. Answer may deal with the calendar year startup requirements as spelled out in Tech Specs. Should accept this answer a' c s operators are not quired to memorize procedures.

Reference: General Operating Procedure 201.1
Tech Spec 3.2.B

NRC Question, Answer and Reference

Question 4.04

The plant is operating at approximately 100% power with both RBCCW pumps running, when RBCCW pump 1-1 trips. Consequently RBCCW return temperatures begin increasing.

a. Per ABN-3200.19, "RBCCW Failure Response", WHAT are three (3) methods available to reduce the heat load on the RBCCW system?

b. While carrying out steps to reduce the heat load, RBCCW pump 1-2 trips. Neither pump can be restarted. WHY does ABN-3200.19 require the recirculation pumps to be tripped? (2 reasons)

Answer 4.04

a.1 Shutdown and isolate the cleanup system.

2 Reduce power

3 Reduce circulation flow to minimum.

4 Transfer the TBCCW heat exchangers to the circulating water system.

b. Seal and bearing cooling and recirculating pump motor cooling are lost.

Reference

1. ABN-3200.19, RBCCW Failure Response, pp. 4-7
2. LO TCR 828.35.4, 801.01 A.7

NRC Question, Answer and Reference

4.04 (Continued)

Facility Comment

Part a. Should accept anything that will act to reduce the heat load on RBCCW system or increase cooling water flow. Operators are not required to memorize steps or sections of the procedures. This was discussed with the Examiner.

Reference: OC Learning Objectives 828.35 and 801.01

NRC Question, Answer and Reference

Question 4.05

Plant conditions have made it necessary to inject Standby Liquid Control into the Oyster Creek reactor vessel.

Station Operating Procedure 304, Standby Liquid Control System Operation, lists seven (7) verifications required to be made by the Control Operator to assure proper Standby Liquid Control operation.

WHAT ARE five (5) of the verifications that are required to be made?

Answer 4.05

a. The PUMP ON indicating light becomes illuminated.

b. The SQUIBB LIGHT for the selected system becomes illuminated.

c. An upscale reading on the pump discharge pressure is observed.

d. The FLOW ON alarm is annunciated.

e. The SQUIBB VALVE OPEN alarm is annunciated.

f. The Cleanup System inlet valves close. (Valves V-16-1, V-16-2 and V-16-4 close)

g. The Standby Liquid Control Tank level indication is decreasing.

Reference

1. Oyster Creek Nuclear Generating Station Operating Procedure 304, pp. 10
2. Learning Objective 828.46 D

NRC Question, Answer and Reference

4.05 (Continued)

Facility Comment

Part F. Change V-16-4 to V-16-14.
Should accept power decreasing by APRM's which are also on 4F.

Reference: DWG: GE 148F444
OPM Module 39, pp. 31 and 45
Operating Procedure 304, Step 5.3.2, p. 10

NRC Question, Answer and Reference

Question 4.06

Operating Procedure 305, Shutdown Cooling System Operation identifies the following as a prerequisite for operation during shutdown.

"Loop E" Recirculation Pump is running, or the "E" Loop Discharge Valve is Closed".

a. WHAT IS THE REASON for this prerequisite and WHAT would be the CONSEQUENCES of operating with the pump not running and the discharge valve open?

Answer 4.06

a. The Shutdown Cooling System is connected to the "E" Recirculation loop, the reactor vessel will be "short circuited" unless the "E" pump is running, or the "E" loop discharge valve is closed. Inadequate cooling for the core and possible core damage could result.

Reference

1. Oyster Creek Nuclear Generating Station

2. Operating Procedure 305, Shutdown Cooling System Operation, pp. 8 & 17

Facility Comment

Initial problem is repressurization of RPV. Core damage could result at a later time.

Reference: GE Thermo Test, Chapt. 9, pp. 117-118

NRC Question, Answer and Reference

Question 4.07

A Control Room fire has made it necessary to evacuate Oyster Creek Control Room. In accordance with Procedure ABN-3200.30, Control Room Evacuation:

a. WHAT ARE the four (4) actions which MUST BE PERFORMED prior to leaving the Control Room.

b. WHAT ARE four (4) of the five (5) additional actions which SHOULD BE PERFORMED prior to leaving the Control Room.

Answer 4.07

a.1 Manually scram the reactor and verify all rods inserted to or beyond 02.

2 Trip all five Recirculation Pumps.

3 Close the MSIV's.

4 Trip all three Reactor Feedwater Pumps.

b.1 Trip the main turbine.

2 Check the electrical distribution system.

3 Confirm transfer of the house loads to Startup Transformers SA and SB.

4 Confirm that EDG 1 and EDG 2 start and idle.

5 Trip all three Condensate Pumps.

NRC Question, Answer and Reference

4.07 (Continued)

Reference

1. Oyster Creek Nuclear Generating Station
2. ABN-3200.30 Control Room Evacuation, pp. 3 & 4

Facility Comment

Part b. Should accept "Initiate B Isolation Condenser" and "Place Page System to General".

Reference: ABN-3200.30, pp. 4 & 5

NRC Question, Answer and Reference

Question 4.08

A primary system LOCA has occurred and entry into the Emergency Procedures is required.

a. For the following conditions, STATE which Emergency Procedures would be entered. (More than one procedure may be required for each condition)

1) Drywell pressure 3 psig

2) Drywell temperature 150 F

3) Reactor Pressure Vessel level 138 inches

b. LIST the three (3) Coolant Injection Sub-systems which could be utilized to inject, if the Reactor Pressure Vessel level is between 61 and 180 inches, and coolant injection is required.

c. If ONLY ONE (1) of the above Coolant Injection Sub-systems can be lined up for injection, LIST the four (4) Alternate Injection Sub-systems which may be utilized to provide injection.

Answer 4.08

a.1 EMG-3200-01 (RPV Control), EMG-3200-02 (Primary Containment Control)

2 EMG-3200-02 (Primary Containment Control)

3 EMG-3200-01 (RPV Control)

NRC Question, Answer and Reference

Answer 4.08 (Continued)

b.1 Condensate

2 Core Spray 1

3 Core Spray 2

c.1 Fire Water

2 Core Spray Keep Full System

3 Liquid Poison Test Tank

4 Liquid Poison Boron Tank

Reference

1. Oyster Creek Nuclear Generating Station

2. Emergency Procedure EMG-3200.03, p. 5

3. Learning Objective 845.04 D, 845.10 D

Facility Comment

Part b and c. With level between 61 and 180 inches there is no need to go to Level Restoration Procedure. Operators may not be familiar enough with "Injection Sub-System" and "Alternate Injection Sub-Systems" terms to relate them to the various systems called out in the procedure. Should consider all systems that could be listed (i.e., CRD, Core Spray, Feed and Condensate). Discussed with Examiner.

Reference: RPV Control Procedure EMG-3200.01
Level Restoration Procedure EMG-3200.03

NRC Question, Answer and Reference

Question 4.09

According to Station Procedure 2000-ABN-3200.02, "Recirculation Pump Trip", if all five recirculation pumps trip during power operation, the operator is to confirm that all pump suction and discharge valves are open, maintain water level and scram the reactor.

a. WHY is it necessary to scram the reactor if all the recirculation pumps trip?

b. WHY do Technical Specifications require at least two (2) recirculation loop suction valves and their associated discharge valves to be in the full open position whenever the reactor head is installed on the vessel?

c. SP 2000-ABN-3200.02 cautions that if all the recirculation pumps have tripped, no recirculation pump should be restarted until the reactor has been depressurized to atmospheric pressure. WHAT is the REASON for this limitation?

Answer 4.09

a. Tech Specs do not allow operation with less than four recirculation loops in service.

b. The requirement assures an adequate flow path exists from the annular space between the pressure vessel wall and the core shroud to the core region. This assures that reactor water level instrumentation readings are indicative of the water level in the core region.

c. To prevent thermal over-stress in the bottom head region of the vessel.

NRC Question, Answer and Reference

4.09 (Continued)

Reference

1. Oyster Creek: Station Procedure ABN-3200.02, 3.3.1, p. 8.0
2. Oyster Creek: Technical Specifications, Section 3.3.F, p. 3.3-3
3. Oyster Creek: Station Procedure 204.1, 3.1, p. 2.0

Facility Comment

Tech Specs only require a shutdown of the reactor of less than four (4) recirc loops are in service (Part a). OC has a scram at this point because plant operation is outside of design limits. Should accept answer discussing design limits also.

Reference: Tech Specs Section 3.3.F

SENIOR REACTOR OPERATOR EXAM

NRC Question, Answer and Reference

Question 5.03

If reactor power is increased from 60% to 70% by pulling rods, explain HOW and WHY the following parameters are affected? (Assume that no other operator action is taken.)

a. Recirc. pump NPSH.
b. Core flow.

Answer 5.03

a. As power increases, recirc. ratio decreases, which decreases annulus temperature, which increases pump NPSH.
b. As power increases, voiding increases, core dp increases, which decreases core flow.

Reference

GE thermodynamics, Heat Transfer and Fluid Flow, Pages 7-93 thru 7-96 and 9-45 thru 9-48.
OC Lic. Op. Annual Exam. Bank, Item Code 5-7.
Hot Lic. Training Content Record 823.02, LO A.3.

Facility Comment

Part a. Should also accept discussion on recirc. ratio decrease due to increase in feedwater flow which increases subcooling and NPSH.
Part b. Also accept references to 2 phase flow causing core dp to increase and decrease core flow.

Reference OPM Module 18, Pages 4, 20-28 and
GE HTFF Manual, Chapter 9, Pages 45-50.

NRC Question, Answer and Reference

Question 5.07

a. DEFINE Critical Power Ratio (CPR).
b. HOW would Critical Power change (INCREASE, DECREASE, or NO CHANGE) for each of the following:

   1. Reactor pressure decrease.
   2. Inlet subcooling increase.

c. DESCRIBE the mechanism by which the fuel cladding could be damaged if the MCPR safety limit is exceeded.

Answer 5.07

a. Bundle power required to produce the onset of transition boiling divided by the actual bundle power.
b. 1. INCREASES
   2. INCREASES
c. Transition boiling causes rapid wetting and drying of the clad surface which results in large temperature oscillations and cyclic stress resulting in cladding perforations.

Reference

OC Lic. Op. Annual Exam. Bank, Item Code 5-3.
Hot Lic. Training Content Record 852.09, LOs P., Q. & R.
GE Thermodynamics, Heat Transfer and Fluid Flow, Pages 9-19 & 9-68.

Facility Comment

Should also accept critical power over actual bundle power for Part a.

Reference

GE HTFF Manual, Chapter 9, Pages 43 and 92.

NRC Question, Answer and Reference

Question 5.09

Briefly DESCRIBE the three (3) mechanisms which can be used to assure adequate core cooling. STATE the order of preference for the mechanisms and EXPLAIN why they are preferred in the stated order.

Answer 5.09

1. Core submergence is established by maintaining RPV water level at or above the top of active fuel (TAF). It is the most preferred mechanism of heat removal because indication of level above TAF provides confirmation that adequate core cooling exists.

2. Spray cooling is established (if level cannot be maintained above TAF) by one core spray subsystem operating at or above design conditions. It is less attractive than core submergence because of the inability to confirm proper in-core flow distribution.

3. Steam cooling is established (when no source of makeup is available) by maintaining a steam updraft through the core with the isolation condensers or the EMRVs. It is the least attractive mechanism for core cooling due to the high differential temperature between the fuel and steam (limited time duration over which steam cooling can be maintained) required and the lack of instrumentation to directly confirm steam flow.

Reference

Hot Lic. Training Content Record 845.01, LO A.
Emergency Operating Procedure Fundamentals, Page 6.

Facility Comment

Only two methods are considered to actually assure adequate core cooling. Spray cooling does not assure adequate core cooling because it cannot be verified as having the proper flow distribution to assure cooling. As such, OC does not have a spray cooling procedure.

Reference

OC Handout 87.03, Page 5
Tech Basis for OCNGS EOPs, Pages 1-8 to 1-11.

NRC Question, Answer and Reference

Question 5.10

a. Following a LOCA, high drywell temperature can affect reactor level indications.

1. EXPLAIN HOW and WHY indicated reactor level on a Rosemount level instrument would deviate from actual reactor water level following an increase in drywell temperature. STATE whether indicated level would be HIGHER or LOWER than actual level.

2. WHAT is the major concern associated with the false indication?

b. During a rapid depressurization below 500 psig, Yarway level instruments are not used to monitor reactor water level.

1. EXPLAIN HOW and WHY indicated reactor level on a Yarway level instrument would deviate from actual reactor water level during a rapid depressurization below 500 psig. STATE whether indicated level would be HIGHER or LOWER than actual level.

2. WHAT is the major concern associated with the false indication?

Answer 5.10

a. 1. As the reference leg water temperature increases, the reference leg water density decreases causing the sensed differential pressure to decrease. The lower differential pressure registers as an increased vessel level. Indicated level is HIGHER than actual.

2. Automatic Actions (initiated on Lo or Lo-Lo reactor level) could be delayed or prevented.

b. 1. Water in the heated reference legs will flash below 500 psig causing water in the reference leg to decrease causing the sensed differential pressure to decrease. The lower differential pressure registers as an increased vessel level. Indicated level is HIGHER than actual.

2. Accurate water level indication is necessary to ensure adequate core cooling is maintained.

Reference

OC Operations Plant Manual, Module 55, Pages 11, 21 & Table 55-1.
Symptom Based Emergency Operating Procedure Basis Review, Page 7.
Hot Lic. Training Content Record 815.04, LO A.

NRC Question, Answer and Reference

5.10 Continued

Facility Comment

Part b. refers to Yarway level instruments. Actual instruments are Rosemounts but are referred to as Yarways for human factor reasons. Should accept a discussion based on Yarway or Rosemount.

Reference

OPM Module 55, Page 11 and Table 55-1.

NRC Question, Answer and Reference

Question 6.01

The lead Control Room Operator has just placed the Liquid Poison System keylock switch on Panel 4F in the "System 1" position.

a. WHAT two (2) actions occur as a direct result of placing the switch in this position? (within the Standby Liquid Control System).

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

Answer 6.01

a. 1. Liquid Poison Pump 1 starts.
   2. Squib Valve 1 Fires.

b. 1. PUMP light is on.
   2. SQUIBS light is on.
   3. Pump discharge pressure.
   4. Tank level decreasing.
   5. Poison inlet valve OPEN light is on.

Reference

OPM-Module 46:SLO, Pages 14,16, Fig. 46-4.
LO TCR 828.46 E,F,O.

Facility Comment

Add "power decreasing" as an answer to part b. because the Nuclear Instrumentation System reads out on Panel 4F.

Reference

None.

NRC Question, Answer and Reference

Question 6.02

The Oyster Creek reactor is at 100% power.

a. A turbine trip occurs. HOW will each of the following sets of valves respond (OPEN, CLOSE or REMAIN AS IS)?

1. Turbine Stop Valves.
2. Turbine Control Valves.
3. Turbine Bypass Valves.

b. For each of the valve conditions listed below, STATE if a (FULL SCRAM, HALF SCRAM or NO RPS ACTION) results with the plant at 100% power.

1. TSV 1 is less than 90% open.
2. TSV 2 and 4 are less than 90% open.
3. Emergency trip oil pressure drops to 150 psig on TCV 1 (A) and TCV 1 (C).

Answer 6.02

a. 1. TSV's close.
2. TCV's close.
3. Turbine bypass valves open.

b. 1. NO RPS ACTION.
2. ONE HALF SCRAM.
3. FULL SCRAM.

Reference

Oyster Creek Nuclear Generating Station.
Operations Plant Manual Module 37, Pages 26 through 30.
OPM-Module 51: Turbine Control System, Pages 9,20,35,41.
LO TCR 828.51 D,F and 828.37 I-N.

Facility Comment

ETO is sensed from 4 acceleration relays, not from the TCV (Part b.3). Should accept any discussion of ETO with reference to the acceleration relays, or "No Action".

Reference

OPM Module 37, Page 29.

NRC Question, Answer and Reference

Question 6.03

The reactor is operating at 100% and a full flow surveillance on the Core Spray System is in progress. The CS system is recirculating water back to the Torus when a CS initiation signal is received.

a. WHAT is/are the initiation signal(s)? Setpoints required.

b. HOW does the test-to-return valve respond to the initiation signal?

c. Initially, DOES the CS discharge valve automatically open? WHY or WHY NOT?

Answer 6.03

a. Low Low vessel water level of 86 inches above TAF high D/W pressure of 3.5 psig.

b. The valve will automatically close.

c. No.
The Valve will remain closed as long as reactor pressure is 285 psig.

Reference

OPM-Module 10-Core Sprav System, Pages 19-23.
LO TCR 828.10 C,D.

Facility Comment

Should accept either Actual Setpoints or Tech Spec Setpoints for Part a.

Low Low Level 90" TAF (Actual) 86" TAF (T.S.).
High Dw. Pres. 3# (Actual) 3.5# (T.S.).

Reference

Standing Order #1.

NRC Question, Answer and Reference

Question 6.04

Reactor Recirculation Pump seal pressures are monitored to determine proper operation of the seals.

a. STATE what type of seal failure, if any, is indicated by each of the following conditions with the plant operating at 100% power.

   1. The number 2 seal pressure is 850 psig.
   2. The number 2 seal pressure is 150 psig.

b. Under WHAT two (2) conditions MUST a Recirculation Pump be shutdown due to seal failure?

Answer 6.04

a. 1. Failure of the number one (1) seal.
   2. Failure of the number two (2) seal.

b. 1. If the number one (1) seal temperature increases to 180°F.
   2. If the number two (2) seal temperature increases to 160°F.

Reference

Oys ter Creek Nuclear Generating Station.
Operations Plant Manual Module 38A Reactor Recirculation System, Pages 34 and 35.
Learning Objective 828.38 H.

Facility Comment

Also accept plugging of #1 seal restrictive orifice as an answer for Part a.2.

Reference

OPM Module 38A, Pages 9 and 28.

NCR Question, Answer and Reference

Question 6.05

An automatic initiation signal for the Isolation Condenser System has been received and the condensers are in operation, when a steam line break occurs BETWEEN the pressure vessel and the differential pressure detectors for the "A" condenser.

a. WILL the "A" Isolation Condenser isolate? EXPLAIN.

b. On an isolation signal, WILL a LOSS of 480 volt AC power PREVENT the ISOLATION of the condenser? EXPLAIN.

c. WILL a LOSS of Instrument and Service Air PREVENT automatic makeup to the Isolation Condenser? EXPLAIN.

Answer 6.05

a. Yes. The differential pressure detectors respond to a differential pressure (with steam flow passing in either direction).

b. No. Each steam supply and condensate return line has two (2) MOV isolation valves in series. One valve is 480 volts AC and the other is 125 volts DC. On loss of 480 volt AC power the DC valves will still close.

c. No. An air accumulator provides air to operate the makeup valves at least six (6) times with the complete loss of Instrument and Service Air.

Reference

Oyster Creek Nuclear Generating Station.
Operations Plant Manual Module 23 Isolation Condenser System, Pages 9, 10 and 13.
Learning Objective 828.23 I.

Facility Comment

In Part c., the question refers to automatic makeup to the Isolation Condenser. OC does not have an automatic makeup to the IC's. Makeup is a remotely controlled function. Loss of air does not prevent remote operation of the makeup valve for up to six lines.

Reference

OPM Module 23, Page 23-17.

NRC Question, Answer and Reference

Question 7.04

The plant is shutdown for refueling. The reactor head is removed and the reactor cavity is flooded above the main steam nozzles. In preparation for fuel offload and maintenance on the recirculation pumps, your shift has been instructed to secure and isolate all five recirculation pumps in accordance with Procedure 301, "Nuclear Steam Supply System".

a. A temporary change to Procedure 301 has been written to allow isolation of all five recirculation loops simultaneously. EXPLAIN why this can be done without violating a safety limit.

b. Procedure 301 allows operation of the recirculation pumps under two conditions when fuel is removed from the core. STATE the two (2) conditions.

Answer 7.04

a. The safety limit does not apply since the head is removed and the cavity is flooded above the main steam nozzles.

b. 1. If it is necessary to initiate Standby Liquid Control.
   2. All LPRMs, IRMs and SRMs in the vessel are surrounded on all sides by fuel assemblies and/or guide blades.

Reference

Procedure 301, "Nuclear Steam Supply", Page 22.
LO TCR 828.38 G.

Facility Comment

Answer requires operator to have memorized a section of the procedure. This is not required per our Learning Objectives. Should allow leeway on answer in light of this. Comment discussed with examiner.

Reference

OC Learning Objectives 828.38.

NRC Question, Answer and Reference

Question 7.06

A containment isolation signal has been received.

Due to a design deficiency in the torus oxygen sample line isolation logic, Standing Order 39 requires that the Torus sample valves and the pump selector switch are placed in the OFF position on the receipt of a containment isolation signal.

a. EXPLAIN WHY this action is required and WHAT the CONSEQUENCES could be if this action were not taken and the containment isolation relay failed.

b. An emergency condition exists which necessitates bypassing of the isolation signal.

   1. WHO (by title) must approve bypassing of the isolation signal?

   2. TRUE or FALSE? Procedure 312.1, "Bypassing Isolation Interlocks During Emergency Conditions", controls the installation and removal of any jumpers required to bypass the isolation signal.

Answer 7.06

a. Failure of the containment isolation relay can prevent the automatic closure of the Torus O2 analyzer valves (V-38-22 and V-38-23). Failure of the valves to go closed could result in an off site radiation release.

b. 1. The Emergency Director or the Group Shift Supervisor.
   2. FALSE (prior to removal control must be transferred to Procedure 108, "Equipment Control").

Reference

OCNGS Standing Order 39.
License Event Report (LER) 87-040.
Procedure 312.1, "Bypassing Isolation Interlocks During Emergency Conditions", Page 3.
LO TCR 830.05 B.

Facility Comment

Part b.1. This requires memorization knowledge of Procedure 312.1 and is not required by our Learning Objectives.

NRC Question, Answer and Reference

Question 7.06 Continued

Facility Comment Continued

Part b.2. Sign off pages for jumpers does control the removal of the jumpers in that there is a sign off block verifying that control has been transferred to Procedure 108. Therefore, should consider True as a correct answer. Also this requires intimate knowledge of the procedure which is not required by our Learning Objectives. Comment discussed with examiner.

Reference

Operating Procedure 312.1, Page E9-2.
OC Learning Objectives 845.02.

NRC Question, Answer and Reference

Question 7.07

The plant is operating at approximately 100% power with both RBCCW pumps running, when RBCCW pump 1-1 trips. Consequently RBCCW return temperatures begin increasing.

a. Per ABN-3200.19, "RBCCW Failure Response", WHAT are three (3) methods available to reduce the heat load on the RBCCW system?

b. While carrying out steps to reduce the heat load, RBCCW pump 1-2 trips. Neither pump can be restarted. WHY does ABN-3200.19 require the recirculation pumps to be tripped? (2 reasons)

Answer 7.07

a. 1. Shutdown and isolate the cleanup system.
   2. Reduce power.
   3. Reduce recirculation flow to minimum.
   4. Transfer the TBCCW heat exchangers to the circulating water system.

b. Seal and bearing cooling and recirculating pump motor cooling are lost.

Reference

ABN-3200.19, RBCCW Failure Response, Pages 4-7.
LO TCR 828.35 4, 801.01 A.7

Facility Comment

Should accept anything that will act to reduce the heat load on the RBCCW system or increase cooling water flow. Operators are not required to memorize steps or sections of the procedures. This was discussed with the examiner.

Reference

OC Learning Objectives 828.35 and 801.01.

NRC Question, Answer and Reference

Question 7.08

For each condition/situation, STATE which Emergency Operating Procedures, if any, would be entered. If none, state NONE.

a. Drywell pressure of 2.3 psig.
b. Rx. Building ventilation exhaust radiation of 15 mr/hr.
c. Bulk Drywell temperature of 153 degrees F.
d. Torus water level of 162 inches.
e. Load Reject has occurred and power is 40%.
f. RPV water level is 125 inches above TAF.

Answer 7.08

a. None.
b. EMG-3200.11, Secondary Containment Control.
c. EMG-3200.02, Primary Containment Control.
d. EMG-3200.02, Primary Containment Control.
e. EMG-3200.01, RPV Control.
f. EMG-3200.01, RPV Control.

Reference

Emergency Operating Procedures: EMG-3200.01, 3200.02, 3200.11, Page 3.
LO TCR 845.03 B, 8450.11 B, 845.06 B.

Facility Comment

Part e. Should be changed to "None" in the answer key as 35% is within the capacity of our bypass valves to handle for a finite time period (40%) and the load reject scram is bypassed below this point - therefore, no scram is required. Discussed with examiner.

Reference

RPV Control Procedure EMG-3200.01.
OPM Module 37, Pages 29-31.

NRC Question, Answer and Reference

Question 7.09

Emergency Procedure EMG-3200.09, "Level/Power Control" requires that reactor water level be lowered by securing injection systems if:

- Power is above 2%, AND
- Torus water temperature is above the Boron injection temperature, AND
- Either an EMRV is open or drywell pressure is above 3.0 psig.

a. WHICH of the injection systems WOULD NOT be secured in order to lower water level?

b. Reactor water level will continue to be lowered until one of three (3) conditions is met. WHAT ARE the three (3) conditions?

Answer 7.09

a. 1. Boron injection.
   2. CRD.

b. 1. Reactor power drops below 2%.
   2. Reactor water level reaches 0 inches.
   3. All EMRV's remain closed and drywell pressure remains below 3.0 psig.

Reference

EMG-3200.09, Page 5.
LO TCR 845.19 B.

Facility Comment

To answer this question requires that the procedure be memorized which is not required by our Learning Objectives. Should be very broad in viewing the answers received on the questions, also consider conditional statements. Discussed with examiner.

Reference

OC Learning Objectives 845.19.

NRC Question, Answer and Reference

Question 7.10

According to Procedures IMP-1300.02, "Direction of Emergency Response" and IMP-1300.03, "Emergency Notification":

a. (TRUE/FALSE) The Emergency Director (ED) is responsible for classifying the event unless overruled by the Emergency Support Director (ESD).

b. (TRUE/FALSE) Once an event has been classified and the NRC has been notified of its classification, NRC permission is required to change the emergency's classification.

c. (TRUE/FALSE) Even with the ESD function activated, the ED is still responsible for approving and directing information releases to the media.

d. (TRUE/FALSE) The ED shall follow the NRC's advice to deviate from established operating procedures during attempts to control the emergency.

e. (TRUE/FALSE) In the event that the NRC and the ED/ESD have differing recommendations for courses of protective action, the Plant Manager will resolve the conflict.

Answer 7.10

a. True
b. False
c. False
d. False
e. False

Reference

IMP-1300.02, Pages 2-5.
IMP-1300.03, Page 5.
LO.

Facility Comment

The answer to Part a. is in reality false because the only time the ESD can override the ED is in situations where the ESD wants to classify an event at a higher level than the ED. The ESD cannot prevent the ED from escalating an emergency condition.

Reference

OC Procedure IMP-1300.02, Page 4.

NRC Question, Answer and Reference

Question 8.02

Procedure 126, Procedure for Notification of Station Events, defines six categories of events and specifies appropriate notification requirements.

a. DEFINE Category I events.
b. HOW will you determine what notifications must be made for Category I events?
c. HOW LONG do you have to notify the NRC after a Category II event occurs?

Answer 8.02

a. This category includes all events which result in declaration of an Unusual Event, Alert, Site Area Emergency, or a General Emergency.

b. Procedure 126 directs you to appropriate EPIPs which specify the notifications that must be made.

c. One hour.

Reference

OC Procedure 126, Procedure for Notification of Events, Page 4.
OC Lic. Op. Annual Exam. Bank, Item Code 8-56.
Hot License Training Content Record 820.05, LO: XX.

Facility Comment

Should also accept the general definition of "all events classified by the EPIP's".

Reference

None.

NRC Question, Answer and Reference

Question 8.04

a. WHAT is required to maintain a Senior Reactor Operator (SRO) license in an ACTIVE status in accordance with 10 CFR Part 55?

b. If a person's SRO license is in an INACTIVE status, what is required before he or she can resume duties that require an active SRO license?

c. An unlicensed operator wishes to operate the controls to insert a rod during a rod pattern exchange operation under the direction of the lead Control Room Operator. Is he allowed to do this? WHY or WHY NOT?

Answer 8.04

a. The license holder must perform the functions of an SRO on a minimum of 7 eight hour shifts or 5 twelve hour shifts per calendar quarter.

b. The license holder must perform a minimum of 40 hours of shift functions under the direction of an SRO. These functions must include a tour of the plant and all shift turnover procedures.

c. (Yes or No) He may perform this function only if it is part of a licensed operator training program.

Reference

10 CFR Part 55, Sections 13(a) (2), 53(e) and 53(f).

Facility Comment

Should not weigh the time requirements for SRO license requirements as heavily as they are. Should have most credit based on functions necessary to maintain a license.

Reference

None.

NRC Question, Answer and Reference

Question 8.09

A startup is in progress with the mode switch in STARTUP. The 'A' Offgas Process Radiation Monitor has been inoperable for one week. The replacement parts are on order. On your shift the 'B' Offgas Process Radiation Monitor failed to meet the surveillance test acceptance criteria. Repairs and recalibration is expected to take until the end of the shift.

Can you, as the Group Shift Supervisor, direct the Startup to continue? If it can continue, can the mode switch be taken to RUN when plant conditions permit? EXPLAIN.

NOTE: USE THE ATTACHED SECTIONS OF THE TECHNICAL SPECIFICATIONS TO ANSWER THIS QUESTION. FULLY REFERENCE ALL APPLICABLE SECTIONS OF THE TECHNICAL SPECIFICATIONS.

Answer 8.09

Yes the startup can continue and the mode switch can be taken to RUN.

TS Table 3.1.1 requires one trip system operable in STARTUP and 2 channels within the trip system operable. Note ii allows removal of one channel within the trip system for maintenance. With no channels operable note jj allows continued operations for 72 hrs. provided that the stack noble gas monitor is operable.

Reference

Oyster Creek Technical Specifications Table 3.1.1
Hot Lic. Training Content Record 850.90 LO J.1 and J.2.

Facility Comment

Note ii of Table 3.1.1 is not applicable, but overridden by Note jj. Credit should not be taken off for not referring to Note ii.

Reference

Tech. Specs. Table 3.1.1.