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

EXAMINATION REPORT NO. 85-12 (OL)

FACILITY DOCKET NO. 50-219

FACILITY LICENSE NO. DPR-16

LICENSEE: GPU Nuclear Corporation
P. O. Box 388
Forked River, New Jersey 08731

FACILITY: Oyster Creek Nuclear Generating Station

EXAMINATION DATES: April 8-12, 1985

CHIEF EXAMINER: D. J. Lange 5/24/85
D. J. Lange, Reactor Engineer (Examiner) Date

REVIEWED BY: John Berry 5/30/85
J. A. Berry, Lead Reactor Engineer (Examiner) Date

RM Keller 6/3/85
R. M. Keller, Chief, Project Section 1C Date

APPROVED BY: HB Kister 6/3/85
H. B. Kister, Chief, Project Branch No. 1 Date

SUMMARY: Operator Licensing Examinations were conducted at the Oyster Creek Nuclear Generating Station during the period of April 9-12, 1985. Eight Reactor Operator candidates and three Senior Reactor Operator candidates were administered written and oral examinations. All candidates successfully passed both the written and oral examinations.

During the oral examinations, the Senior Reactor Operator candidates demonstrated an overall strength in their ability to use Technical Specifications and Emergency Plan procedures. During grading of the written examinations, an overall strength was noted in the area of Plant Design/Instrument Controls, for both Reactor and Senior Reactor candidates. No generic weaknesses were noted for either the Reactor or Senior Reactor Operator oral or written examinations.

REPORT DETAILS

1. Type of Exams: Replacement

2. Exam Results:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	8/0	3/0
Oral Exam	8/0	3/0
Overall	8/0	3/0

3. Chief Examiner at Site: D. Lange, USNRC, Reactor Engineer Examiner

4. Other Examiners: B. Hajek, NRC Consultant Examiner
   D. Hill, EG&G Idaho, Inc.
   A. Mendiola, USNRC (OLB) Examiner Trainee

5. Oyster Creek Entrance Meeting:

   NRC Attendees

   D. Lange, USNRC Region I, Reactor Engineer Examiner
   B. Hajek, NRC Consultant Examiner

   Facility Attendees

   Rod Davidson, Training Supervisor
   Derrick Wilson, Training Instructor

   An entrance meeting was conducted immediately following the start of the RO/SRO written exam. A tentative schedule for oral exam assignments was discussed. The two hour exam review was scheduled at the training department following completion of the exam. A tentative exit meeting was set up for 8:00 AM Friday morning, April 12, 1985

6. Summary of strengths noted on oral exams:

An overall strength was noted in all candidates ability to use piping and instrument drawings and their ability to use normal and emergency procedures.

The SRO candidates demonstrated a strong safety awareness to overall plant conditions and abnormal events.

Candidates did very well using the control room plant reference material.

The SRO candidates did very well using Technical Specifications and emergency plan procedures.

7. Exit Interview Details

Personnel Present at Exit Interview:

NRC Personnel

D. Lange, Chief Examiner
J. Berry, Lead BWR Reactor Engineer (Examiner)
A. Mendiola, OLB, Examiner Trainee
W. Bateman, Senior Resident Inspector

NRC Contractor Personnel

B. Hajek, NRC Consultant Examiner

Facility Personnel

P. B. Feidler, Vice President and Director, Oyster Creek
R. Davidson, Operations Training Manager
D. Wilson, Training Instructor (Initial Operator Licensing)
W. Stewart, Operations Manager
D. Holland, Supervisor of Licensing

Summary of Comments made at exit interview:

The Chief Examiner advised the facility of the preliminary results of the oral examinations.

The Chief Examiner noted the generic strengths observed during the SRO Oral Exams.

The Chief Examiner noted a problem area, with the noise level of facility personnel in the control room, during the oral exams. This created a problem, for both candidates and examiners, to a point where examinations had to be stopped and the problem brought to the Senior Shift Supervisor's attention.

The Chief Examiner also noted that certain control room piping and instrument diagrams needed to be replaced due to their worn condition.

The Chief Examiner thanked the facility for the use of the conference room and commented on the cleanliness of the plant.

The Training Department submitted the facility comments for both the RO and SRO written exams. The facility wanted an indication of when the next round of Operator Licensing Certificate presentations would be conducted. The NRC Lead Examiner explained how this would be accomplished.

The facility noted that the problem areas identified during the Oral Exams would be addressed.

Attachments:
1. Written Examination and Answer Key (SRO/RO)
2. Facility Comments on Written Examinations Made After Exam Review and NRC Resolutions to Those Comments

GPU Nuclear

Memorandum

Subject: Exam Comments

Date: April 11, 1985

R Davidson

From: R. Davidson
Operator Training Manager

Location: Forked River

To: D. Lange
NRC Examiner

General Comments

Oyster Creek recognizes the difficulty of writing a valid license exam, especially in the area of procedures. We have recently spent about 400 manhours developing the six exams for the Licensed Operator Requalification program and experienced many difficulties in this area.

However, we have some concerns about Section 7 of the SRO exam. Generally, our concerns lie in the selection of the procedures to be examined. For example, the questions on EPIP-7 asked detailed specifics of how to treat injured workers and also guidelines for extremities and skin exposure limits during emergency conditions. The SRO candidate should be aware of this EPIP, but do not need to memorize its contents or reproduce parts of it.

A question on ABN-7 asked what plant parameters must be checked when an unexplained reactivity change occurred at power. To our knowledge, there have been very few, if any, unexplained reactivity changes and if one were experienced, the Reactor Engineers would be notified and appropriate action taken in accordance with the procedure. To expect the SRO's to respond to this question was unreasonable.

Some questions required brute memorization of cautions and limitations vice knowing why cautions and limitations are there.

For example, one question asked them "to state the two precautions and limitations prior to putting the static charger in service." This is a relative obscure procedure and to expect regurgitation is unreasonable.

Some questions required immediate operator actions where none are defined by procedure. Our training program stresses the use of procedures but not rote memorization.

OYSTER CREEK NUCLEAR GENERATING STATION

NRC EXAMINATION OF APRIL 9, 1985

EXAMINERS: RO - BRIAN HAJEK

SRO - DAVID LANGE

FACILITY COMMENTS

Specific Examination Question Comments

RO Examination

Section 1

1.01 Major part of answer should be towards the periphery and not require towards the top since the peak shifts axially with core age/rod position.

1.03 a. part of answer - should accept inadequate volume in the Accumulator to give proper scram insertion.

b. expand band to 1150 (Proc. 302.1)

1.04 a. Also acceptable - Negative reactivity due to doppler coefficient.

1.05 b. Answer key should be changed to reflect the fact that to increase flow it is less efficient to change pump speed.

1.09 a. Should accept any transient which may impact pressure and/or temp. The judgement as to which is most severe is open for discussion.

Section 2

2.07 c. 50 psid due to modification

2.11 b. Answer key changes to

1. Block of Low flow/High temp trip

2. Manually start the backup ESW pump.

Section 3

3.10 Should also accept valve being pinned as reason for not moving.

Section 4

4.02 Other possible answers such as loss of feedwater heaters.

4.05 Ans. Breakers remain closed
Breakers may trip on regaining DC

4.06 a. Answers should be for turbine trip not generator trip.

b. Feedwater heaters may not be on at 30%
(Step 4.19 Procedure 201.3)
Possible answer - Reactivity increase due to pressure spike from stop valve closure.

4.07 c. Answer should not include 100#

4.08 b. First part of answer is not necessarily an immediate action. Suggest it be deleted.

4.09 Additional answer is that the oncoming operator must be alert and coherent (step 4.3.1 Procedure 106)

(1466H)

SRO EXAMINATION

Section 5

5.07 Voids don't change during this transient. Presence of void is the answer.

Rod worth at power lower due to voids X_D will also be the primary effect of turning power.

5.08 a & b One heat source is the control rods & NI's.

Section 6

6.02 a. High drywell pressure setpoint by tech specs is 2.4#.

6.06 - Unit substations are only 460V breakers supplied by 125V DC for control.

- Should not grade on position indication.

Section 7

7.02 a. Pressure greater than 850# prior to going to range 10

b. Tech spec answer should be acceptable

c. Procedure and alarm

7.03 b. Also vibration induced failure of tubes

c. Secondary is Fire Brigade

7.04 b. Other loads on 24V DC should be acceptable

c. Other cautions in procedure

7.06 a. Qualified first aid member only required answer.

b. Coordinator should be acceptable

c. Hospital should be acceptable

7.07 a. Level restored should be acceptable

7.08 a. Other reasonable answers

c. 80% should not be required

7.09 b. More answers than listed

7.10 More answers than listed

7.11 Alert and coherent should also be acceptable (Procedure 106 Step 4.3.1, Pg. 24)

Section 8

8.04 b. Exact limits should not be required to be memorized.

8.05 a. When fire watch required

b. Communications

8.06 This 15 day requirement is not asked for in the question and should not be required.

8.11 a. Question asks for the temperature, not a definition, therefore there could be confussion on this part.

b. Should also accept 50°F T for recirc pump start and 100°F Heatup/Cooldown

NRC Resolution to Facility Comments (RO Exam)

Category 1

1.01	Near the Top of the Core is not required for 1.1(2).
1.03 a, b	Comment accepted.
1.04 (a)	Add, negative reactivity due to dopler coefficient during normal operating conditions.
1.05 (b)	Not accepted. This is true for decreasing flow, but not for increasing flow. Will consider candidates answer on pump laws.
1.09 (a)	Considered during grading.

Category 2

2.07 (c)	Accepted with documentation provided. Due to NOV. 1984 modification.
2.11 (b)	Comment accepted.

Category 3

3.10	Comment accepted. Will accept, inadvertently left pinned.

Category 4

4.02	Alternate reasonable answers accepted.
4.05	Comment accepted if candidate describes correct logic sequence.
4.06 (a)	Turbine Trip accepted.
(b)	Considered during grading if candidate assumes that the F.W. Heaters are not in service at 30% power.
4.07 (c)	Comment accepted if candidate says that the MPR is kept higher than Rx. Pressure.
4.08 (b)	Comment accepted and point value re-distributed for remainder of answer.
4.09	Alert and coherent accepted as an additional answer.

NRC RESOLUTION TO FACILITY COMMENTS (SRO EXAM)

Category 5

5.07 - The presence of voids, during this transient is the acceptable answer.

5.08 - The heat source, of control rods and N.I., will be accepted in addition to gamma and beta heating.

Category 6

6.02(a) - 2.0# is accepted, 2.4# will be accepted if candidate identifies this change from 2.0# → 2.4#.

6.06 - Unit substations - accepted.
- Will not deduct points for improper position indication.

Category 7

7.02 a) Correct answer is, pressure > 850# prior to gaining to range 10 on IRM's.
b) Considered during grading if candidate explains responsibilities.
c) Caution in procedures accepted in addition to (2) alarm conditions.

7.03 b) Maintenance of any ht. exch. assoc. prob. accepted if candidate explains prob with isolation valves.
c) Fire brigade response accepted.

7.06 a) Reasonable answers accepted if candidate identifies that personnel must be qualified in first aid.
b) Accepted.
c) Accepted.

7.07 a) Considered during grading.

7.08 a) Reasonable answers, affecting reactivity change, considered during grading.
b) First immediate action of a rapid reduction in Recirculation flow accepted.

7.09 b) Additional answers, only if correct as to actions performed before leaving control room, accepted.

7.10 - Considered during grading.

7.11 - Alert and Coherent is an acceptable answer. Any (2) of the following three answers are correct.

1. Alert and coherent.
2. Fully qualified for that shift position.
3. Fully aware of plant conditions.

Category 8

8.04 (b) Question asked for Limits, ±10% of Federal and Admin. Limits considered during grading.

8.05 (a) Not accepted.
(b) Not accepted.

8.06 - Candidate is not required to say 15 day requirement. The T.S. section, and most Limiting LCO is required.

8.11 (a) Considered during grading.
(b) Temp. thermal units for heatup and cooldown accepted as examples of temp requirements for various operating conditions.

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: OYSTER CREEK

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/04/09

EXAMINER: LANGE, D.

APPLICANT: MASTER

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00	________	________	5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	________	________	6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	________	________	7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00	________	________		TOTALS

FINAL GRADE ________________%

All work done on this examination is my own. I have neither given nor received aid.

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (2.00)

NOTE: Answer the following questions from a theoretical standpoint, not from an Oyster Creek system design standpoint.

a. With the plant operating at 90 % power , extraction steam to one string of low press. feedwater heaters is removed. An engineer, observing that turbine load increased by 15 MWe after the extraction steam removal, concludes that this action has improved the plant's thermodynamic efficiency (NOT heat rate). Do you agree with this conclusion? Explain your answer fully. (2.00)

QUESTION 5.02 (2.25)

For each of the events listed below, state which reactivity coefficient will respond first and if it adds positive or negative reactivity.

a. Relief Valve opening at 100 % power . (0.75)
b. Rod drop at 100 % power. (0.75)
c. Isolation of a feedwater heater string at 75 % power. (0.75)

QUESTION 5.03 (1.50)

If the Main Condenser and associated systems were absolutely AIR TIGHT would there be any need for the Steam Jet Air Ejectors during full power operation ? (Explain your answer). (1.50)

QUESTION 5.04 (3.00)

Indicate whether the following will INCREASE or DECREASE reactivity during operation AND briefly EXPLAIN why.

a. Moderator temperature increases while below satruation temperature. (.75)
b. Fuel temperature increases. (.75)
c. Loss of a feedwater heater. (.75)
d. A sudden reduction in reactor primary system pressure. (.75)

QUESTION 5.05 (2.00)

Concerning the term NPSH ; How would you account for the fact that the NPSH is different for a recirculation pump with the plant operating at 100 % power versus 4 % power ? Include in your answer at which power level will the NPSH be higher, the operating conditions that are affecting the change, and the reason why. (2.00)

QUESTION 5.06 (2.25)

a. Following a Standby Liquid Control system initiation, from rated power, list five (5) REACTIVITIES that the system must overcome to provide its intended function. (1.25)
b. What is the bases for having a minimum and maximum injection time requirement for the SBLCS. (1.00)

QUESTION 5.07 (2.00)

What is the operational requirement and main purpose for having the Rod Worth Minimizer ? What is one of the principle reasons it is not required above 10 % power operation ? (2.00)

QUESTION 5.08 (1.50)

Concerning Reactor Core Flow ;
a. How is Core flow heated in the bypass region of the core. (0.50)
b. Why is a certain amount of bybass flow required. (0.50)
c. List the factors affecting flow resistance in a fuel bundle for both single and two phase flow. (0.50)

QUESTION 5.09 (2.00)

During a cooldown of the reactor vessel ~~from outside the control room~~, reactor pressure decreased from 885 psig. to 595 psig. in one half hour.Has your reactor cooldown limit been exceeded ? { show all work } (2.00)

QUESTION 5.10 (2.25)

Oyster Creek's reactor is operated within three (3) specified Thermal Limits. List each of the three limits and explain the specific heat transfer related problem that the limit protects against. (2.25)

QUESTION 5.11 (2.00)

The steady state MCPR limit given in Tech. Specs. is multiplied by a flow biasing correction factor Kf. Explain the bases for this correction factor including the events associated with it. (2.00)

QUESTION 5.12 (2.25)

For the following changes in plant parameters will control rod worth increase, decrease, or not be affected? Briefly explain why?

a. An increase in moderator temperature. (0.75)

b. An increase in void content. (0.75)

c. An increase in fuel temperature. (0.75)

QUESTION 6.01 (2.25)

Concerning the Main Turbine Control System ;

a. Following a Main Turbine (lo- vacuum) Trip from rated power, what prevents the turbine control valves and stop valves from popping open once the turbine is reset ? (0.75)

b. Briefly explain how a main Turbine Generator runback condition will occur. Include in your answer, the actuation signal, the response of the control and bypass valves, and when or if a turbine trip will occur. (1.50)

QUESTION 6.02 (2.75)

Concerning the Standby Gas Treatment System (SBGTS) ;

a. List four automatic signals that will place the SBGTS in service. NOTE: Setpoints required. (1.00)

b. With the SBGTS operating at rated flow, which way will the inlet and outlet valves fail upon a loss of instrument air . Can the valves still be operated. (briefly explain). (1.00)

c. If the SBGTS was automatically initiated by a reactor Low water level signal, what must be done to shut the system down once the Low -level signal clears ? (0.75)

QUESTION 6.03 (2.25)

The EMRV 's have a three (3) position switch station on panel 1F/2F. (Automatic / Off / and Manual) . List the functions provided by each Switch Position. (2.25)

QUESTION 6.04 (2.00)

Concerning the Isolation Condenser ;

a. Why would it be necessary to manually remove A and E Recirculation pumps from service after a manual initiation of the Isol. Cond. (0.75)

b. All automatic isolation valves for the Isol. Cond. can be manually overriden, (opened or closed), regardless of what is being called for by the logic circuitry . (TRUE or FALSE) (0.50)

c. Why is there a caution that directs you to maintain Reactor water level below ~~95"~~ Yarway indication. (0.75)
185 TAF

QUESTION 6.05 (2.50)

You are in the process of using the Alternate Shutdown Cooling method, (cleanup system, condensate, and main condenser) ;

a. List two (2) methods you can use to decrease the temperature of the non-regenative heat exchanger outlet as it approach's 130 deg.F. (1.00)

b. During this shutdown cooling method why might you want to increase hot-well level ? (0.50)

c. Why might you want to place the mechanical vacuum pump in service, and what precaution should you be aware of when placing a negative pressure on the reactor. (1.00)

QUESTION 6.06 (2.50)

During the 4:00PM-12:00 MID shift you experience a total loss of all 125 VDC power;

a. For each of the following components listed below, indicate whether there is a loss of position indication, control power, and/or reset capability. NOTE; If more than one condition applies, indicate so. (2.50)

1. Shutdown cooling system isolation valves.
2. Main Steam Isolation Valves.
3. Isolation Condenser, isolation valves.
4. All 4160 V and 460 V breakers .
5. EMRV's .

QUESTION 6.07 (3.00)

Concerning the Reactor Level and Feedwater Control System ; (Reactor Power 100 %, steady state)

For the following, Loss of Signal failures to the Feedwater Control Sys., indicate in which direction reactor water level will respond,(inc. or dec.) the reason for the change, and the probable automatic action that will occur with no operator action. (3.00)

a. Feedwater Flow
b. Steam Flow
c. Reactor Water Level
d. Steam Pressure
e. Feedwater Temperature

QUESTION 6.08 (1.50)

During a Reactor Isolation, with the Isolation Condensers in service, you receive a triple -low water level alarm. If all control room level indicators appear to be normal;

a. What caused this alarm condition ? (0.75)
b. Why the apparent contiidiction between this alarm and indicated level ? (0.75)

QUESTION 6.09 (3.00)

a. For the following Reactor Level instruments, indicate where the water level is being sensed and if there is any compensation involved. (1.50)

1. Wide Range GE-MAC.
2. Narrow Range GE-MAC
3. Lo-Level Yarway
4. Lo-Lo-Level Yarway
5. Lo-Lo-Lo- Bartons
6. Fuel Zone Rosemounts & GE-MAC's

b. Concerning the Fuel Zone Level Instrumentation ;

1. What signal(s) is required to turn the system on ? (0.75)
2. Why won't the HIGH PRECISSION requirement of this system work if the Core Spray pumps are running ? (0.75)

QUESTION 6.10 (2.25)

Concerning the Standby Diesel Generators ;

a. Explain the automatic sequential actions of the Diesel Generator for the following two conditions ; Limit answer to operation of D/G only.

1. Inadvertant LOCA signal, immediately clearing, with no loss of off-site power. (0.75)
2. Loss of off-site power followed by a LOCA signal, with the Core Spray system loads already picked up, and then additional loading causes the Diesel Generator to stall and output breaker to trip. (0.75)

b. If an inadvertant, false, start signal is received by the D/G , how can it be manually shut down ? Would you want to do this immediately ? (yes or no and WHY) (0.75)

QUESTION 6.11 (1.00)

Operating Procedure # 308 , (Emergency Core Cooling Operation), provides direction on controlling Core Spray injection to maintain reactor water level.

a. What action must be taken to manually take control of the Core Spray System. (0.50)

b. Once manual control of the Core Spray system has been taken, what C/S components are used to control injection to the vessel. (0.50)

QUESTION 7.01 (2.00)

According to procedure ABN-3200.13,(Response to Loss of All 125 VDC), a caution exists, that directs you to either maintain the plant in a steady state condition or enter a prompt reactor scram, irrespective of the Technical Specification requirements. What two (2) specific criteria are used to make this determination ? (2.00)

QUESTION 7.02 (2.25)

According to procedure 201.1 (Approach to Criticality) ;

a. Why is there a caution in this procedure that instructs you not to switch to range 10 too early while in the startup mode. Briefly explain the bases for this caution. (0.75)
b. If the RWM becomes inoperable before the first twelve (12) control rods have been withdrawn , can the startup ~~commence~~ CONTINUE ? (If not why ? If so under what conditions ?) (0.75)
c. What two (2) warning indications are available to assure that the minimum permissable positive period of 30 sec. is not exceeded. (0.~~50~~ .75) 28%

QUESTION 7.03 (2.50)

Concerning the 1983 Outage ' Control Room Modifications .';

a. Where are the new,' Loss of Power Alarms ', for the Vital Instrument Panel power supplies located ? (0.75)
b. Why were modifications made to the Shutdown Cooling System and what equipment/components had to be replaced and/or installed as a result of this modification. (1.00)
c. What is the PRIMARY and SECONDARY fire protection system for the New Cable Spreading Room. (0.75)

QUESTION 7.04 (2.25)

Concerning Procedure 340.2 ,(24 VDC Distribution System) ;

a. Can you commence a startup with the 24 VDC power panel 'A' inoperable ? (24 VDC power panel 'B' is operable) (0.50)
b. A loss of all 24 VDC will cause what three (3) systems to be lost or activated ? (1.00)
c. What two precautions/ limitations exist concerning the operation and/or removal from service of the STATIC BATTERY CHARGERS . (0.75)

QUESTION 7.05 (2.50)

According to procedure EPIP-2 (Emergency Direction) , the Emergency Director has certain responsibilities and/or authorities that may NOT be delegated to a subordinate during emergency conditions. List five (5) of these actions that may not be delegated. (2.50)

QUESTION 7.06 (1.75)

Concerning Procedure EPIP-7, (Personnel Injury);

a. What plant personnel are to be used to make up the FIRST AID TEAM.(0.50

b. During an emergency condition, (activation of the OSC), who would be responsible for coordinating the off-site medical transport for injured personnel. (0.75

c. If unsuccessful attempts have been made to de-contaminate an injured person requiring off-site medical attention, the Emergency Director should be communicating with what off-site personnel ? Where ? (0.50

QUESTION 7.07 (3.00)

Concerning EMG-3200.01 (RPV Control/ RC/L Level Control) ;

a. Under what conditions can the automatic controls of the Core Spray System be placed in its manual mode of operation. (1.00)

b. List the Entry Conditions for this procedure. (2.00)

QUESTION 7.08 (3.00)

Concerning procedure ABN-3200.07 (Unexplained Reactivity Change) ;

a. List ~~six~~ five (5) plant parameters/indications that should be checked if an unexplained reactivity change should occur at rated power. (1.50)

b. Depending on the magnitude of the reactivity change list four alarms that may be initiated. (prior to a reactor scram) (0.80)

c. If this reactivity change is a result of decreased temperature, due to a loss of a feedwater heater string , and a reactor scram has not occured, what would be your next immediate action. (.70)

QUESTION 7.09 (2.75)

.30

Concerning procedure ABN-3200.01,(Control Room Evacuation) ;

a. Following a control room evacuation how is reactor level controlled ? (include location and components) (0.75)

b. Being forced to evacuate the control room, an attempt should made to bring the plant to a safe shutdown condition before leaving. List, eight (8) operator actions / verifications to be attempted prior to leaving. (2.00)

QUESTION 7.10 (2.00)

List six (6) automatic actions you would want to verify upon receiving a reactor level Lo-Lo condition during a small pipe break LOCA. Include those actions that may have been initiated from other signals as well. (2.00)

QUESTION 7.11 (1.00)

According to Administrative Procedure 106,(Conduct of operations), what two criteria, concerning the on-comming operators, must the off-going operators be satisfied with before relinquishing their responsibilities. (1.00)

QUESTION 8.01 (2.00)

Concerning the Limiting Condition For Operation (Operability Requirements) as described in Technical Specifications. When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation only if two (2) conditions are satisfied . LIST THESE TWO CONDITIONS. (2.00)

QUESTION 8.02 (2.50)

During your shift, 12:00mid - 8:00AM , with the plant at 100 % steady state power, one of your operators informs you that the ~~closed~~ BLUE position indication light for one EMRV is not indicating. The problem is determined not to be a burned out light bulb but a possible logic circuit problem .
This problem cannot be worked on until 8:00 am, when maintenance personnel are available.
Using the attached sections of Tech. Specs , can the plant continue to operate under this condition ? Fully reference all sections of T.S related to this situation, giving a brief description and bases for any actions you would take. (2.50)

QUESTION 8.03 (2.50)

Concerning the use of Radiation Work Permits and the Control of Locked High Radiation Areas ;

a. List three conditions that must be adhered to prior to performing maintenance activities under the use of an Extended RWP. (1.50)
b. List two operational conditions that must be satisfied prior to entry into the TIP SHIELD room. (1.00)

QUESTION 8.04 (2.75)

Concerning Administrative/Federal RAD. dose limits and EMERG. Exposure's.;

a. List the Federal Occupational Dose Limits and the administrative dose limits normally provided to site workers for; WHOLE BODY, SKIN, and EXTRIMITY. (1.00)
b List the emergency exposure guidelines for whole body, extremeties and thyroid that apply for CORRECTIVE ACTION and LIFE SAVING ACTION. (1.00)
c. Who , by title, is responsible for authorizing emergency exposures and can this responsibility be delegated ? (0.75)

QUESTION 8.05 (2.00)

a. What criteria is used to determine if a continuous FIRE WATCH is required ? (0.50)
b. What communication and /or documentation would be required on your shift for a continuous FIRE WATCH ? (0.75)
c. If a fire were to occur within the station, what caution is required concerning the automatic fire protection system on elevation 119 '? (briefly explain why) (0.75)

QUESTION 8.06 (2.50)

* NOTE: USE THE ATTACHED SECTION OF THE TECHNTCAL SPECIFICATIONS TO *
* ANSWER THE FOLLOWING QUESTION. FULLY REFERENCE ALL SECTIONS YOU USE. *

During a shift turnover, with the plant operating at 75% power, you are informed that the Quarterly MSIV closure Surveillance test has exceeded the maximum allowable extension interval,and must be performed on your shift. Halfway through the test, ' ONE' Outboard MSIV FAILS to meet the specified closing time. In accordance with the Tech Specs:

a. What situation exists due to the surveillance test being outside of the test frequency schedule. (1.00)
b. What actions must be taken due to the fact that the MSIV has failed it's closing time test ? (1.50)

QUESTION 8.07 (2.00)

In accordance with the Technical Specifications, the reactor was scrammed due to Suppression Pool water temperature being greater than 110 degrees F. The reactor is now in HOT SHUTDOWN, Suppression Pool Cooling is ON, and Suppression Pool water temperature is 95 degrees F. Using the attached section of Technical Specfications, can you commence a startup ?? (Fully Explain) (2.00

QUESTION 8.08 (2.50)

During the beginning of your shift it is determined that one of the Containment Spray pumps in Loop 'A' is inoperable. Half way through your shift the Diesel Generator associated with Containment Spray Loop ' A ' fails its operability Surv. Test.---Using the attached sections of Tech. Spec's, what action is required due to this situation ?

(2.50)

QUESTION 8.09 (1.75)

With the plant in cold shutdown, conditions in the reactor shiuld be maintained to prevent thermal stratification of the reactor water level and inadvertent repressurization. List two (2) operating methods, briefly explaining how they are being used, to maintain this cold stable condition. (1.75)

QUESTION 8.10 (3.00)

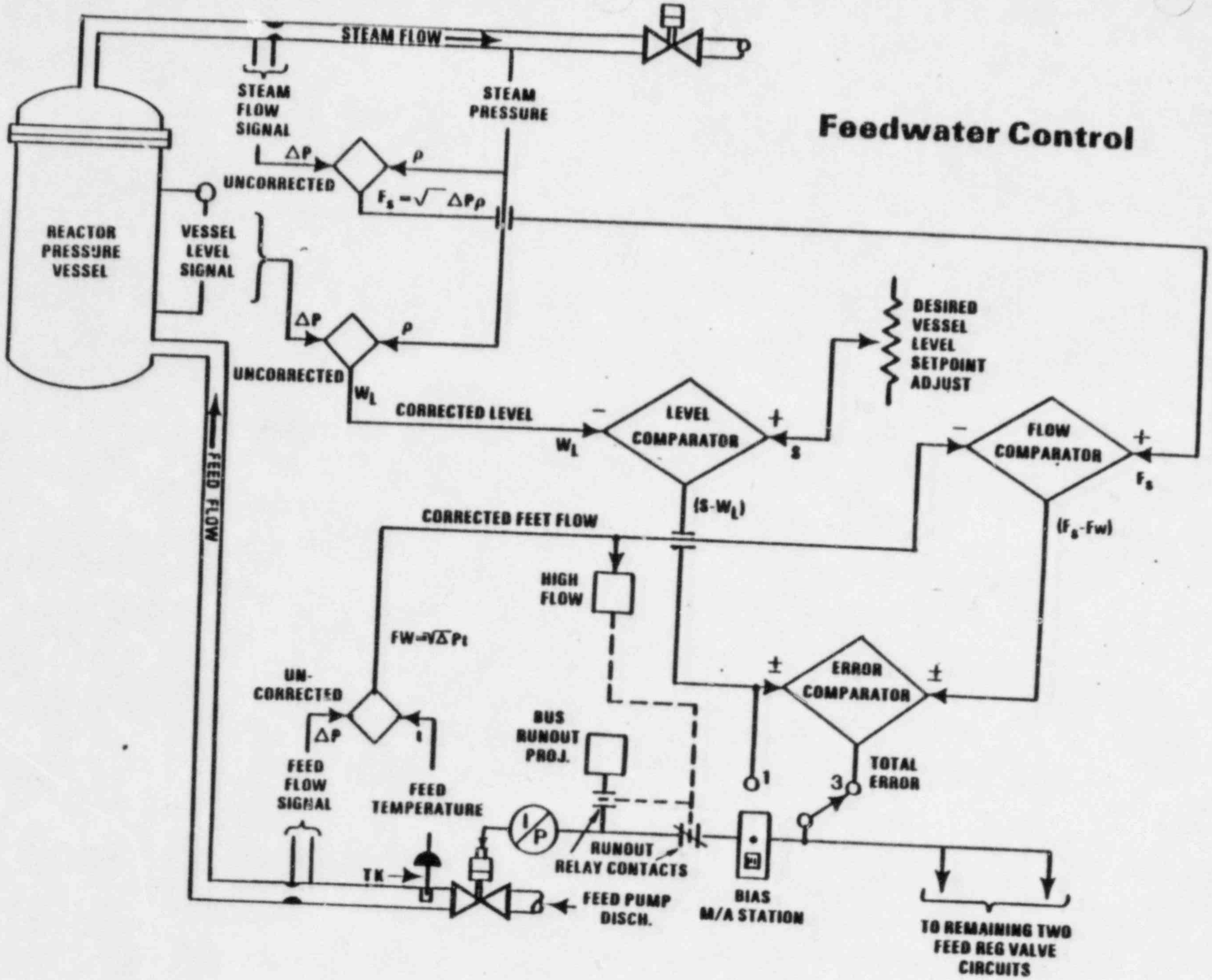
Concerning Technical Specifications---> Safety Limits and Specifications ;

a. What is the bases for keeping a minimum of two (2) Recirculation Loop suction and discharge valves open at all times ? (1.00)
b. During all modes of reactor operation with irradiated fuel in the vessel what is the minimum water level to be maintained, and how does this level compare to the lowest point at which water level can be presently monitored. (1.00)
c. What are the Tech. Spec. requirements for SRM operability during a core alteration involving the removal of one (1) control rod ? (1.00)

QUESTION 8.11 (1.50)

a. What is the NIL- DUCTILITY Transition Temperature ? (0.50)
b. What Limitations are placed on plant operations, at Oyster Creek, to minimize the possibility of Brittle Fracture. (three required) (1.00)

Feedwater Control



44

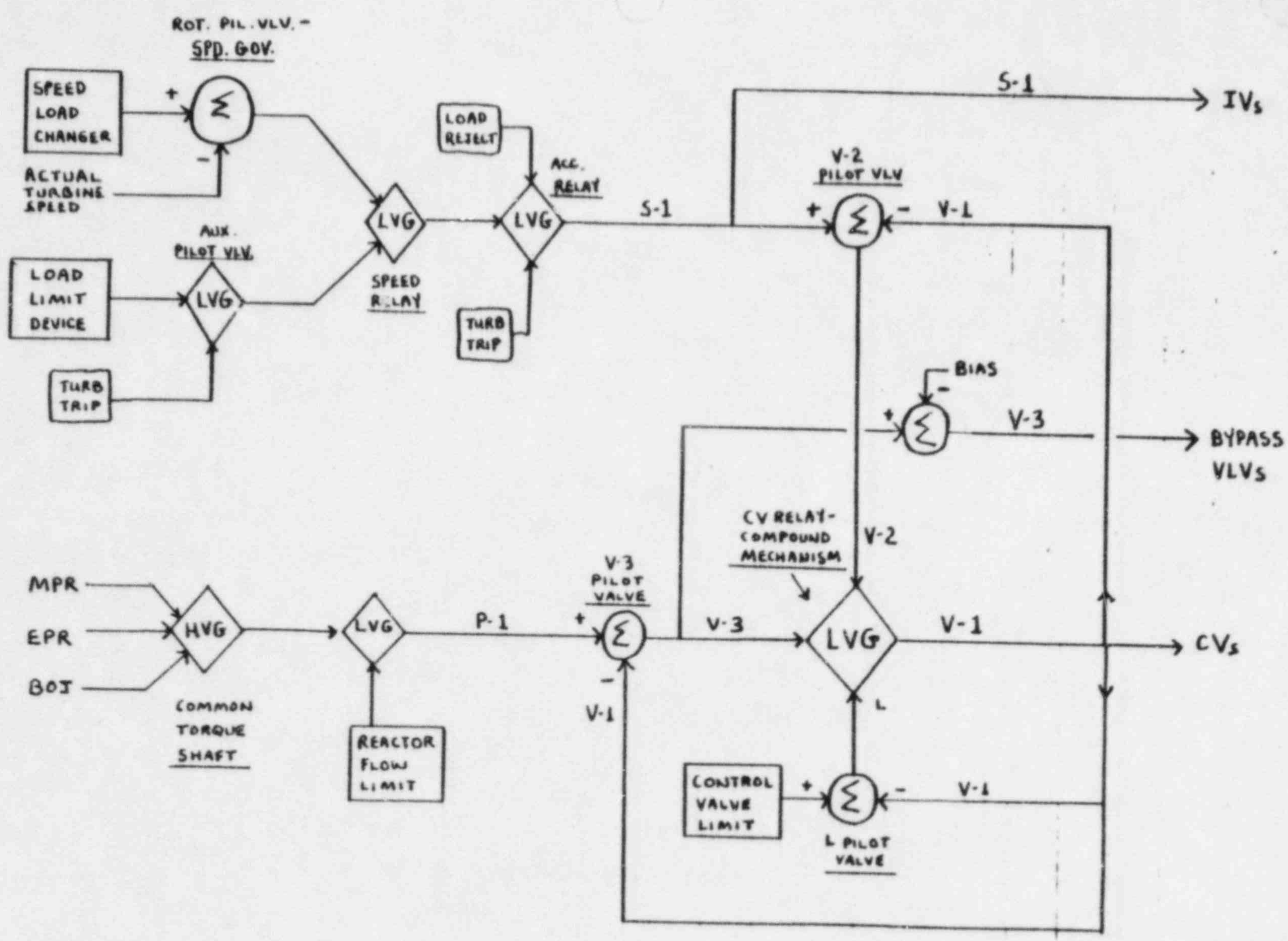


FIG. 11.

FUNCTIONAL BLOCK DIAGRAM

TABLE II-3-1

PROPERTIES OF SATURATED STEAM AND SATURATED WATER (TEMPERATURE)

Temp F	Press. psia	Volume, ft³/lb Water v_f	Volume, ft³/lb Evap v_{fg}	Volume, ft³/lb Steam v_g	Enthalpy, Btu/lb Water h_f	Enthalpy, Btu/lb Evap h_{fg}	Enthalpy, Btu/lb Steam h_g	Entropy, Btu/lb x F Water s_f	Entropy, Btu/lb x F Evap s_{fg}	Entropy, Btu/lb x F Steam s_g	Temp F
32	0.08859	0.01602	3305	3305	−0.02	1075.5	1075.5	0.0000	2.1873	2.1873	32
35	0.09991	0.01602	2948	2948	3.00	1073.8	1076.8	0.0061	2.1706	2.1767	35
40	0.12163	0.01602	2446	2446	8.03	1071.0	1079.0	0.0162	2.1432	2.1594	40
45	0.14744	0.01602	2037.7	2037.8	13.04	1068.1	1081.2	0.0262	2.1164	2.1426	45
50	0.17796	0.01602	1704.8	1704.8	18.05	1065.3	1083.4	0.0361	2.0901	2.1262	50
60	0.2561	0.01603	1207.6	1207.6	28.06	1059.7	1087.7	0.0555	2.0391	2.0946	60
70	0.3629	0.01605	865.3	865.4	38.05	1054.0	1092.1	0.0745	1.9900	2.0645	70
80	0.5068	0.01607	633.3	633.3	48.04	1048.4	1096.4	0.0932	1.9426	2.0359	80
90	0.6981	0.01610	468.1	468.1	58.02	1042.7	1100.8	0.1115	1.8970	2.0086	90
100	0.9492	0.01613	350.4	350.4	68.00	1037.1	1105.1	0.1295	1.8530	1.9825	100
110	1.2750	0.01617	265.4	265.4	77.98	1031.4	1109.3	0.1472	1.8105	1.9577	110
120	1.6927	0.01620	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120
130	2.2230	0.01625	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130
140	2.8892	0.01629	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140
150	3.718	0.01634	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150
160	4.741	0.01640	77.27	77.29	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160
170	5.993	0.01645	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170
180	7.511	0.01651	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180
190	9.340	0.01657	40.94	40.96	158.04	984.1	1142.1	0.2787	1.5147	1.7934	190
200	11.526	0.01664	33.62	33.64	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200
210	14.123	0.01671	27.80	27.82	178.15	971.6	1149.7	0.3091	1.4509	1.7600	210
212	14.696	0.01672	26.78	26.80	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212
220	17.186	0.01678	23.13	23.15	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220
230	20.779	0.01685	19.364	19.381	198.33	958.7	1157.1	0.3388	1.3902	1.7290	230
240	24.968	0.01693	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240
250	29.825	0.01701	13.802	13.819	218.59	945.4	1164.0	0.3677	1.3323	1.7000	250
260	35.427	0.01709	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	260
270	41.856	0.01718	10.042	10.060	238.95	931.7	1170.6	0.3960	1.2769	1.6729	270
280	49.200	0.01726	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280
290	57.550	0.01736	7.443	7.460	259.4	917.4	1176.8	0.4236	1.2238	1.6473	290
300	67.005	0.01745	6.448	6.466	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300
310	77.67	0.01755	5.609	5.626	280.0	902.5	1182.5	0.4506	1.1726	1.6232	310
320	89.64	0.01766	4.896	4.914	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320
340	117.99	0.01787	3.770	3.788	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340
360	153.01	0.01811	2.939	2.957	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360
380	195.73	0.01836	2.317	2.335	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380
400	247.26	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400
420	308.78	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420
440	381.54	0.01926	1.1976	1.2169	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440
460	466.9	0.0196	0.9746	0.9942	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460
480	566.2	0.0200	0.7972	0.8172	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480
500	680.9	0.0204	0.6545	0.6749	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500
520	812.5	0.0209	0.5386	0.5596	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520
540	962.8	0.0215	0.4437	0.4651	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540
560	1133.4	0.0221	0.3651	0.3871	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560
580	1326.2	0.0228	0.2994	0.3222	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580
600	1543.2	0.0236	0.2438	0.2675	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600
620	1786.9	0.0247	0.1962	0.2208	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620
640	2059.9	0.0260	0.1543	0.1802	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640
660	2365.7	0.0277	0.1166	0.1443	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660
680	2708.6	0.0304	0.0808	0.1112	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680
700	3094.3	0.0366	0.0386	0.0752	822.4	172.7	995.2	0.9901	0.1490	1.1390	700
705.5	3208.2	0.0508	0	0.0508	906.0	0	906.0	1.0612	0	1.0612	705.5

MASTER ANSWER KEY

ANSWERS -- OYSTER CREEK -85/04/09-LANGE, D.

ANSWER 5.01 (2.00)

A. NO ,
Thermodynamic efficiency is a comparison of energy in versus energy out. [0.5] The increase in generator output resulted from decreasing the amount of steam diverted to the LP FW heaters.[0.5] This condition requires additional energy output from the reactor to raise FW temp to the same saturation temp as before [0.5] Thus, thermodynamic efficiency of the plant has gone down. [0.5] More delta T across the heater would have caused more extraction steam to have been removed from the turbine.

B. ~~Reactor Power (CMWT), increases (0.25), due to the core inlet temp-decreasing thus causing more heat to be added to reach the same core exit enthalpy . (0.50)~~ No part B, Computer Typo. D[?]

REFERENCE
Oyster Creek LP. objective # 27, FW heater failure.
Basic Turbine/Plant thermo. effeciency.

ANSWER 5.02 (2.25)

a. VOID COEFFICIENT (0.50), adds negative reactivity (0.25)
b. FUEL TEMP. COEFFICIENT (0.50), adds negative reactivity (0.25)
c. MODERATOR TEMPERATURE COEFFICIENT (0.50), adds positive reactivity (0.25

REFERENCE
Oyster Creek LP.# 300.08, Rx.Coefficients and Control Rod Worth.

ANSWER 5.03 (1.50)

YES. (0.50) To maintain the removal of non-condensable gasses produced from the decomposition of water , activation products and noble gasses produced in the fuel and leaking into the coolant via. cladding cracks. (1.00)

REFERENCE
Oyster Creek LP # 68, pg. # 5 , Design Bases for SJAE.

ANSWER 5.04 (3.00)

a. Adds negative reactivity [0.25] due to the increase in neutron leakage - Moderator temperature coefficient. [0.50]
b. Adds negative reactivity [0.25] due to the increase in neutron capture in the fuel - Doppler coefficient. [0.50]
c. Adds positive reactivity [0.25] due to the decrease in neutron leakage - Moderator temperature coefficient. [0.50]
d. Adds negative reactivity [0.25] due to the increase in neutron leakage - Void coefficient. [0.50

REFERENCE
Oyster Creek LP. Reactor Theory CH. #8 (8.12) Plant Parameter Effects on Control Rod Worth and Reactivity Coefficients.

ANSWER 5.05 (2.00)

At 100 % power. (0.75) At 4 % power, you are at operating pressure but at low feedwater flow rate. NPSH is low due to T inlet being high. As power increases, pump inlet temperature is reduced due to mixing in the downcomer. T-inlet is lower so P-sat. at inlet is lower, therefor NPSH is higher. (1.25)

REFERENCE
Oyster Creek Fluid Flow CH #7.

ANSWER 5.06 (2.25)

a. 1. Rated void collapse.
   2. Fuel doppler effect decrease.
   3. Xenon decay.
   4. Temperature decrease to 125 deg.F.
   5. Shutdown margin 4 % delta K. (5 correct ans.@ 0.25 each)
b. Minimum- To prevent power chugging caused by uneven mixing. (0.50)
   Maximum- To compensate for cooldown following Xenon Peak. (0.50)

REFERENCE
Oyster Creek LP.# 53,SBLCS.

ANSWER 5.07 (2.00)

Purpose & Req.-> Serve as a backup to procedural controls, limiting control rod worths, to assure that in the event of a control rod drop the reactivity addition rate would not lead to significant fuel damage. (1.00)

One of the principal reasons that it is not required is due to the presence of coolant voids. The rod drop accident would be less severe. [As the voids increased during the transient, the power excursion would be greatly limited.] (1.00)

REFERENCE
Oyster Creek LP # 49, RWM.

ANSWER 5.08 (1.50)

a. This flow is heated by neutron and gamma heating. + CONTROL HEATING Rod Dj (0.50)
b. Core bypass flow is needed for cooling of components in the bypass region and to prevent excessive voiding in the region which could affect the accuracy of the nuclear instrumentation. (0.50)
c. Single phase flow- friction, acceleration, and local restrictions. (.25)
Two phase flow- same factors as single phase determined by a multiplier looking at, coolant quality, bundle flow rate and sys.press. (0.25)

REFERENCE
Oyster Creek LP. Thermo. Ht. Transfer, Fluid Flow CH.# 9.

ANSWER 5.09 (2.00)

First, convert psig. to psia. by adding 14.7 psi. Then, refering to the steam tables:
900 psia. = 532 deg.F
610 psia. = 488 deg.F
532 deg.F - 488 deg.F =44 deg.F / half hour, or 88 deg./hr (1.50)

NO. The cooldown limit of 100 deg.F/hr has not been exceeded. (0.50)

REFERENCE
Oyster Creek LP . Thermodynamics and Heat Transfer (Steam Tables)

ANSWERS -- OYSTER CREEK -85/04/09-LANGE, D.

ANSWER 5.10 (2.25)

1. MCPR- protects against the onset of transition boiling. (.75)
2. LHGR- protects against exceeding 1% plastic strain on the clad due to excessive heat generation in the fuel. (.75)
3. MAPLHGR- ensures that peak fuel clad temperature will not exceed 2200 degrees F during a DBA-LOCA. (.75)

REFERENCE
Oyster Creek Heat Transfer and Thermo. CH.#9, Learning Objective 9-1b .

ANSWER 5.11 (2.00)

This flow adjustment factor increases the MCPR limit at core flows less than rated. Events such as LOSS OF FW.HEATING and TURBINE TRIP without bypass become less severe when initiated from power levels less than the design value. This is due to decreased steam flow. But events such as inadvertant start up of an idle recirc. pump, recirc. flow controller failure (increased flow) and FW flow controller failure (max) can become more severe than transients which are limiting at design conditions. (2.00)

REFERENCE
Heat Transfer and Thermal Limits, CH # 9 , Oyster Creek .

ANSWER 5.12 (2.25)

a. increase (0.25) As moderator temperature inc. so does thermal diffusion length which increases the leakage of thermal neutrons from the fuel bundles and into the control rod regions.Thus rod worth inc. (0.5).
b. decrease (0.25) Moderator density decreases resulting in more fast neutrons and fewer thermal neutrons leaving the bundle. Since control rods are thermal absorbers, overall control rod worth decreases (.5).
c. no effect (0.25) Since fuel temp. affects primiraly fast neutrons, which are resonantly captured, and control rods are thermal neutron absorbers, fuel temp. and rod worth are essentially independent of each other. (.5).

REFERENCE
Oyster Creek RX.Theory CH #8, pg. #64 &65.

ANSWER 6.01 (2.25)

a. The Load Limit handwheel trip prevents this action.
(Trip oil pressure must first be restored and the handwheel moved to the closed position, relatching the handwheel spindle.) (0.75)
b. 1. Runback- Caused by loss of stator cooling. (0.50)
2. Control Valves move in the closed direction. (0.25)
Bypass Valves move in the open direction. (0.25)
3. Turbine Trip- If, after three (3) minutes, the stator current is greater than (4800 amps.) (0.50)

REFERENCE
Oyster Creek LP # 73, Turbine Control System, Rev. #1 Pg.-27.

ANSWER 6.02 (2.75)

a. Reactor building vent High Radiation. > 13 mr/hr. (0.25)
Reactor building operating floor (elev.119) > 70 mr/hr after a 2 minute time delay. (0.25)
High drywell pressure > 2.~~00~~ 40 psig. (0.25)
Lo-Lo reactor water level (86" above the TAF) (0.25)
b. Valves fail as-is. Accumulators on both inlet and outlet valves allow for five (5) valve cycles, after a loss of inst. air. (0.75)
c. The drywell isolation reset button must first be reset. (3-S-2) (0.50)

REFERENCE
Oyster Creek Oper. Procd. # 330 rev.14 (SBGTS)

ANSWER 6.03 (2.25)

1. Auto. (0.25), allows auto operation of EMRV's for ADS and OVER-PRESSURE relief function. (0.50)
2. Off. (0.25) , allows auto action of ADS but defeats pressure relief function. (0.50)
3. Manual. (0.25) energizes the solenoid valves to open the EMRV's. (0.50)

REFERENCE
Oyster Creek LP # 5 , ADS.

ANSWERS -- OYSTER CREEK -85/04/09-LANGE, D.

ANSWER 6.04 (2.00)

a. To prevent an inadvertant system isolation caused by induced high flow through the condensers. (0.75)
b. TRUE (0.50)
c. The steam lines to the isol.cond. come off directly at an elevation corresponding to approx. 95" Yarway indication. Operating above this water level will cause a potential for damage due to water hammer.(0.75)

REFERENCE
Oyster Creek LP # 21, Isol. Cond.

ANSWER 6.05 (2.50)

a. Increase RBCCW flow by opening the NRHX outlet valve, (V-5-122). (0.50)
Reduce letdown flow by using FCV-ND-22 . (0.50)
b. To minimize the Oxygen addition to the condensate system. (0.50)
c. This is an attempt to reduce the Oxygen level content in the reactor coolant water. (0.50). With a negative pressure in the reactor, boiling will occur at a lower temperature than 212 deg. F.(0.50) (1.00)

REFERENCE
Oyster Creek , Procd. # 203.3, Alternate S/D Cooling.

ANSWER 6.06 (2.50)

a. 1. Loss of position indication.
2. Loss of position ind. and reset capability.
3. Loss of position ind. and reset capability.
4. Loss of control power. - 460 MCC - AC
5. Loss of control power. 460 Unit S.B. 4160 - all. (0.50 for each correct ans.)

REFERENCE
Oyster Creek, 2000-ABN-3200.13. Response to loss of All 125 VDC. pg. 1-8.

ANSWER 6.07 (3.00)

a. FW-Flow > Level increases (0.20), due to large flow error signal causing FRV to open fully (0.20). Probable Turbine Trip on high level.(0.20)
b. STM-Flow> Level decrease (0.20), due to large flow error signal causing the FRV to close (0.20). Possible Low level Scram. (0.20)
c. Rx.Water Level> Level increases (0.20), due to large level error signal causing FRV to open fully (0.20). Probable Turb.Trip on hi-level. (0.20)
d. STM-Pressure > Indicated level will decrease (0.20), due to the decrease in indicated steam flow there will be a resulting overfeeding and under-feeding signal being sent to the FRV (level vs steam flow signal)(0.20). Depending on the power level,(steam flow), and level being maintained at the time of the incident, actual level may increase or decrease. (0.2 Possible Turbine Trip or Reactor Scram (level inc. or dec),but in either case less significant than a loss of the basic three elements. (0.20)
e. Feedwater Temp.> Level increase (0.20), due to the temp. signal failing high causing an indicated decrease in FW-flow and opening the FRV.(0.20) Probable Turb. Trip on high level. (0.20)

REFERENCE
Oyster Creek LP. #44, Attachment #2 Loss of Input Signal to FW-Control.

ANSWER 6.08 (1.50)

a. No return flow path VIA Recirculation loop piping exists. (0.75)
b. This is because the Tripple low sensor monitors level within the Shroud while all other sensors monitor annulus level. (0.75)

REFERENCE
Oyster Creek Procd. # 307, pg. 4. Isolation Condenser System.

ANSWER 6.09 (3.00)

a. For instruments # 1,2,3,4, water level is sensed in the annulus.
For " # 5,6, level is sensed in the core shroud.

Compensation;

1. none
2. density compensated (using MSL pressure.)
3. & 4. compensated (using heated reference leg.)
5. none (calibrated for operating conditions.)
6. density compensated (using reactor pressure.)

(0.25 for each full correct answer)

b.1. System turns on only when all five recirc. pumps trip. (0.~~50~~ 75)
~~2. With the recirc. pumps running a false level delta P signal is generated due to forced circulation and resulting pressure drop across the steam separator.~~ (0.50) ~~8~~ J.
3. The variable leg of the transmitter is sensed at the Core Spray sparger, therefor a false high level signal is generated due to the C/S pump discharge head. (0.~~50~~ 75)

REFERENCE
Oyster Creek LP. # 64, pg.7-16, and pg.37.

ANSWER 6.10 (2.25)

a. 1. The D/G immediately starts, goes to idle, the automatically shuts down after a time delay of (11.5 Min.) (0.75)
2. The persistant undervoltage condition will cause an automatic restart of the D/G and begin loading in a timed sequence. (0.75)

b. The Control Room start/stop switch or the fuel oil cut-off switch in the D/G engine compartment.(0.25)
NO-, The 11.5 min. idle feature allows engine cooldown before shutting down. (0.50)

REFERENCE

Oyster Creek Oper. Procd. # 341, pg.1-26.

ANSWER 6.11 (1.00)

a. Must have cleared or overridden all initiating and auto start signals. (0.50)

b. Controlled by opening and closing the C/S parallel valves. (0.50)

REFERENCE
Oyster Creek, Oper. Procd. # 308, sec.5.3.5, and LP. # 10, C/S logic.

ANSWER 7.01 (2.00)

1. If a reactor limiting safety system setpoint has/has not, been exceeded. (1.00)
2. If reactor vessel pressure boundary integrity or fuel integrity is threatened. (1.00)

REFERENCE
Oyster Creek Procd. ABN-3200.13 , Rev. 0, pg.4&10.

ANSWER 7.02 (2.25)

a. Switching the range switch too early may result in an automatic closure of the MSIV's and MSL drain valves. (Technical Specifications require a minimum recirculation flow of ~~39.65 x 10^6 lb/hr~~ while in the startup mode before switching any IRM range switch to position 10.) (0.75)
b. Yes. (0.25) Only with the permission of the Manager of plant operations, or his designe, in accordance with the approved procedure (#218). (0.50)
c. 1. SRM PERIOD SHORT alarm on panel 3F. (0.375)
2. Amber period alarm on panel 4F. (0.375)
3. By Procedure (CAUTIONS). DJ B- T.S.

REFERENCE
Oyster Creek Procd. # 201.1, pg. 1-12.

ANSWER 7.03 (2.50)

a. Annunciation for these new alarms are on panel 9-x-F in the Main Control room. (0.75)

b. The system could not be properly isolated to perform maintenance of the SDC heat exchanger and the inside containment isolation valves and pump inlet valves needed repair. vibration prob. DJ (0.25)
1. Replacement of inside containment isolation valves. (0.25)
2. Replacement of pump inlet valves for each loop. (0.25)
3. Installation of flow ind. (local) for each of the (3) SDC loops. (.25)
c. Primary- Automatic wet pipe sprinkler system. (0.50)
Secondary- Manual wet pipe hose station. (Fire Brigade DJ) (0.25)

REFERENCE
Oyster Creek LP. # 57, Handout 1210.03., Learning Objectives.

ANSWER 7.04 (2.25)

a. NO, (Both A & B 24 VDC panels must be energized before the RX. is made Critical). (0.50)

b. 1. Loss of neutron monitoring system. (0.33)
   2. Loss of the area radiation monitoring system. (0.33)
   3. Initiation of the Reactor building ventillation Isolation. (0.33)

SRM IRM APRM / accepted 2/d

c. 1. Do not operate the Static Chargers without its battery being connected. (0.375)
   2. Do not remove from service unless its respective battery is supplying power to the affected DC bus. (0.375)

REFERENCE
Oyster Creek Procd. 340.2, pg.# 1-6.

ANSWER 7.05 (2.50)

1. Classification of the event.
2. Approving and directing official notification to off-site agencies.
3. " " " information releases to the media.
4. Approving and conveying protective action recommendation to the N.J. office of Emergency Mgnt.
5. Directing on-site evacuation at the Alert or lower level to non-assigned personnel.
6. Authorizing emergency workers to exceed 10 CFR 20 Rad. exposure limits.

(any 5 at 0.50 each)

REFERENCE
Oyster Creek EPIP-2, pg# 3

ANSWER 7.06 (1.75)

a. A member of the plant shift, qualified in first aid, and a qualified Radiation Control Tech. (0.50)

b. The Operation Support Coordinator,(The Emerg. Dir. would have this responsibility during non-OSC activation). (0.75)

c. The Nursing Service Supervisor at Community Memorial Hospital. (0.50)

REFERENCE
Oyster Creek EPIP-7, Personnel Injury.

ANSWER 7.07 (3.00)

a. 1. Misoperation in automatic is confirmed by at least two independent process parameter indications. (0.50)
2. Core cooling is assured AND this procedure, (~~SOP 24~~ EMG-3200.01), directs you. (0.50

b. 1. RPV Level below + 138 TAF.
2. Drywell Pressure above 2.00 psig.
3. A condition that requires a reactor scram and power above 2 %.
4. RPV pressure above 1050 psig.
5. MSIV closure.
6. A condition requiring a scram , to conserve inventory or reduce release of radioactivity to the enviroment, determined by the operator. (0.50

REFERENCE
Oyster Creek EMG-3200.01 RPV Control.

ANSWER 7.08 (3.00)

a. 1. Main Gen. Output.
2. LPRM readings.
3. Reactor power.
4. Steam flow or temp.
5. Feedwater flow or temp.

(5 correct ans at .3 ea.)

b. 1. LPRM - HI , , 2. APRM - HI, (3) IRM - HI , (4) SRM -HI, inop, short P.
(four correct at 0.20 each)

c. Reduce power to 80 % of the power level prior to the change using Recirc flow. (0.70)

REFERENCE
Oyster Creek, 3200.07 pg 1-10

Part c - 3200.16

ANSWER 7.09 (2.75)

a. Using the Control Rod Drive System , elev.23' in the reactor bldg. by confirming both CRD pumps are running and throttleing the by-pass valve. (0.75)

b. 1. Scram the Reactor
2. Verify all rods in.
3. Trip all Recirc. pumps.
4. Trip the Main Turbine.
5. Close the MSIV's
6. Trip all FW pumps
7. Trip all condensate pumps.
8. Initiate both isolation condensers.

(eight correct answers at 0.25 each)

REFERENCE
Oyster Creek , ABN 3200.30 (Control Room Evacuation)

ANSWER 7.10 (2.00)

1. Core Spray system initiation.
2. Diesel generator start and idle.
3. Recirc. pumps trip.
4. Primary Containment Isolation.
5. Secondary Containment Isol. and start of SBGTS.
6. Reactor Isolation.
7. Isolation Condenser initiation. (any 6 at 0.33 each)

REFERENCE
Oyster Creek Q & A bank , and LP. for Reactor Level Instrumentation.

ANSWER 7.11 (1.00)

1. Is fully qualified and/or licensed, to assume the shift position. (0.50)
2. Is fully aware of existing plant / equipment conditions. (0.50)
3. Be alert / coherent. DJ.

REFERENCE
A.P-106, Sec. 4.3.1.1

Three answers accepted (.33 ea)

Pg. 24 4.3.1

ANSWER 8.01 (2.00)

1. Its Corresponding NORMAL or EMERGENCY power source is operable. (1.00)
2. All of its redundent system(s), subsystem(s), train(s), component(s) and device(s) are operable. (1.00)

REFERENCE
Oyster Creek, TS.Sec. 3.0, Operability Requirements. Specification B.

ANSWER 8.02 (2.50)

With no indication of valve position and possible logic or maintance problem existing and unable to be corrected, the valve has to be considered inoperable according to the definition of operability. (0.50)
The LCO for pressure relief systems-Solenoid Actuated Relief Valves , as stated in section 3.4-4 , requires all (5) valves to be operable when you are at operating temperature and pressure.(0.50) The primary bases is for depressurization to allow for full flow core spray operation in the event of a small line break when the feedwater system is not active. (0.50)
Section 3.4.B-2 allows for continued operation for up to (3) days provided the motor operated isolation and condensate makeup valves in both isolation condensers are demonstrated to be operable. (0.50)
If specification 3.4.B 1 & 2 are not met the reactor press shall be reduced to less than 110 psig. in 24 hours. (0.50)

REFERENCE
Oyster Creek T.S. sec. 3.4-4 , LCO for ADS.

ANSWER 8.03 (2.50)

a. No contamination or potentially contaminated system opening. (0.50)
   Must be a job of short duration. (0.50)
   The work is not expected to cause any significant change in the static radiological condition in the area. (0.50)
b. 1. TIP detectors are either in their chamber shields or inserted to greater than 400 ". (0.50)
   2. Caution tags are placed on the drive control drawers,(panel 4R)(0.50)

REFERENCE
Oyster Creek Procd. 915.12 (RWP) and ADM-4110.06 (Control of Locked Hi-RAD Areas.)

ANSWER 8.04 (2.75)

			Administrative:	
a. Federal:				
Whole Body-	1.25 R/qt (3R/qt max.NRC-4)		1R/qt-4R/yr	(0.33)
Skin -	7.5 R/qt		5R/qt	(0.33)
Extremenity-	18.75 R/qt		15R/qt	(0.33)
b. Whole Body-	25 R	75 R		(0.33)
Exrtemenity-	100 R	300 R		(0.33)
Thyroid-	125 R	No Limit		(0.33)
	(protective)	(life saving)		

c. Emergency Director (0.50). NO, (0.25) (0.75)

REFERENCE
Oyster Creek ADM-4000.01, pg # 3-5 and EPIP-7 pg# 3.

ANSWER 8.05 (2.00)

a. A continuous fire watch is required when the fire supression system in that area has been isolated. (0.50)
b. One hour communication to the control room operator. (0.50), who documents conformation in the control room log. (0.25)
c. This system shall be valved out of service and only used as a manual system when the new fuel storage vault contains new fuel and is uncovered. Evacuation of the area is necessary before manual action.(0.75)

REFERENCE
Oyster Creek Procd. # 120.4 Fires pg #2 and # 120.2 Cont. Fire Watch.

ANSWER 8.06 (2.50)

a. The LCO for the MSIVs are stated in ~~Table 3.1.1~~ Spec. 3.5.A.3, Table 3.5.2 DJ/. Two main steam line isolation valves per main steam line shall be operable (with closing times greater than or equal to 3 secs and less than or equal to 10 sec.) Operating outside of this specification is a violation of T.S. and therefore is a reportable occurance. (1.00)
b. Action req.of ~~3.1.1~~ 3.5.A.3 states that with the one MSIV INOP due to exceeding the allowable closing time, the affected steam line shall be isolated. If the problem is not corrected in ~4 hrs., initiate an orderly shutdown. Table 3.1.1 also allows one hour to complete the test without having to trip the inoperable trip system. (1.50)

REFERENCE
Oyster Creek Tech. Spec. Table 3.1.1 and LP # 23, pg. 33.

ANSWER 8.07 (2.00)

NO.(.75) Operation shall not be resumed until the pool temperature is reduced to below the power operation limit specified in section 3.5.A.1.c. This section only allows power operation to be resumed if pool temperature is BELOW the 95 deg.F . (1.25)

REFERENCE
Technical Specification 3.5.A.1, specfication c.3

ANSWER 8.08 (2.50)

With the Diesel failing its surv. oper.test the "A" Containment Spray Loop becomes inoperable. (0.75). T.S. 3.4.c.3 requires that plant operation can only continue for 7 days and that the C/S loop "B" be demonstrated operable daily.(1.00) This LCO is different for only one pump being inoperable as in the beginning of the shift where plant operation could continue for a period of 15 days providing the other pump in "A" loop be demonstrated operable daily. T.S 3.4.c.4 (0.75)

REFERENCE
Oyster Creek T.S. Sec. 3.4.c.3,4,5.

ANSWER 8.09 (1.75)

1. By increasing Reactor Water Level---> This helps maintain and promote natural circulation. (0.875)
2. Maintaining forced circulation ---> By using one recirculation pump or un-throttled shutdown cooling flow. (0.875)

REFERENCE
Oyster Creek Thermo.& Heat Transfer, CH.# 9, pg.125, Chapter Summary.

ANSWER 8.10 (3.00)

a. This assures that an adaquate flow path and communication exist from the annular space and the core shroud to core region. (0.50) This is needed to be assured that reactor water level instrument readings are indicative of water level in the core region. (0.50)

b. 1. 4 ft, 8" above the TAF. (0.~~50~~ .75), Water level can presently be monitored ~~4 ft, 8" below the TAF. (0.50)~~ → Fuel Zone. Barton (.25)
T.S. is wrong
DJ

c. The SRM nearest the core alteration must be operable. (0.50
Two SRM's must be operable; one in the core quadrant and one in the adjacent quadrant. (0.50)

REFERENCE
Oyster Creek, Tech. Spec bases and spec. sec.2.1, Safety Limits and Fuel Clad Integrity.

ANSWER 8.11 (1.50)

a. The temperature at which metal suffers brittle fracture without exhibiting ductile yeild. (0.50)

b. 1. The temperature at which the head closure studs can be tensioned. (.33
2. The minimum temperature for a given reactor pressure under various operating conditions. (.33)
3. The minimum temperature at which the reactor can be made critical (.33

REFERENCE
Oyster Creek Thermo. and Heat Transfer, Learning Goals and Chapter Summary for LP. # 10.

- Heat-up + cooldown rate
- Recirc PP. Temp limits. DJ

QUESTION	VALUE	REFERENCE
05.01	2.00	DJL0000185
05.02	2.25	DJL0000186
05.03	1.50	DJL0000187
05.04	3.00	DJL0000188
05.05	2.00	DJL0000196
05.06	2.25	DJL0000197
05.07	2.00	DJL0000198
05.08	1.50	DJL0000211
05.09	2.00	DJL0000212
05.10	2.25	DJL0000213
05.11	2.00	DJL0000214
05.12	2.25	DJL0000215
	25.00	
06.01	2.25	DJL0000199
06.02	2.75	DJL0000200
06.03	2.25	DJL0000201
06.04	2.00	DJL0000202
06.05	2.50	DJL0000203
06.06	2.50	DJL0000204
06.07	3.00	DJL0000208
06.08	1.50	DJL0000209
06.09	3.00	DJL0000210
06.10	2.25	DJL0000218
06.11	1.00	DJL0000235
	25.00	
07.01	2.00	DJL0000205
07.02	2.25	DJL0000206
07.03	2.50	DJL0000207
07.04	2.25	DJL0000219
07.05	2.50	DJL0000230
07.06	1.75	DJL0000231
07.07	3.00	DJL0000232
07.08	3.00	DJL0000233
07.09	2.75	DJL0000234
07.10	2.00	DJL0000236
07.11	1.00	DJL0000237
	25.00	
08.01	2.00	DJL0000216
08.02	2.50	DJL0000217
08.03	2.50	DJL0000221
08.04	2.75	DJL0000222
08.05	2.00	DJL0000223
08.06	2.50	DJL0000224
08.07	2.00	DJL0000225

QUESTION	VALUE	REFERENCE
08.08	2.50	DJL0000226
08.09	1.75	DJL0000227
08.10	3.00	DJL0000228
08.11	1.50	DJL0000229

	25.00	

	100.00	

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility: Oyster Creek

Reactor Type: BWR

Date Administered: April , 1985

Examiner: Brian K. Hajek

Candidate: (Print) MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value		Category
25	25			1.	Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25			2.	Plant Design Including Safety and Emergency Systems
25	25			3.	Instruments and Controls
25	25			4.	Procedures - Normal, Abnormal, Emergency, and Radiological Control
100					TOTALS

Final Grade ________%

All work done on this examination is my own. I have neither given nor received aid.

Candidate's Signature

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, AND FLUID FLOW (25)

1.1 O.P. 201.1, Approach to Critical, cautions "under conditions of high Xenon concentration, the source range count rate may be quite small as criticality is approached. It is, therefore, necessary to monitor source range instrument responsiveness to control rods with extreme care. . ."
Explain the reason for this caution, and how the conditions of high Xenon concentration can affect individual control rod notch worths. (3.0)

1.2 Core orificing is used to assure uniform flow through all fuel elements, somewhat independent of the power variations in individual fuel bundles.

a. Explain how the resistance to flow changes as the power increases in a fuel bundle. (1.0)

b. How does core orificing assure relatively uniform flow independent of plant operating conditions and location in the core? (2.0)

1.3 The HCU accumulators are charged with nitrogen to assure rapid insertion of the control rods.

a. When the accumulators are precharged, instructions in O.P. 302.1, Control Rod Drive Hydraulic System, state that the accumulator charging pressure be set only when the temperature of the nitrogen has reached equilibrium. What is the purpose of this instruction, and what would result if the instruction was not followed? (2.0)

b. What is the approximate accumulator nitrogen pressure during normal reactor operation? Why is it so different from the precharge pressure? (1.0)

1.4 During a refueling outage, the core is loaded with sufficient fuel to achieve an effective multiplication factor (Keff) of about 1.257.

a. Since Keff in a critical reactor is 1.000, give three reasons why the core is loaded with so much excess Keff. (1.5)

b. What are the two means of compensating for this excess reactivity in the reactor? (1.0)

Category Continued on Next Page

Flow can be controlled by throttle valve or by changing pump speed.

1.5 a. For the centrifugal pump characteristic curve shown in Figure 1.1, show how the pump operating ~~curve~~ point will change, and explain why the change occurs, if a valve in the system is throttled one-half closed, such as might be done in the Feedwater Control System. Be sure to label all points and lines. (1.5)

b. The Recirc System uses changes in pump speed to control flow rate. Using the same figure, explain which flow control system is more efficient from the standpoint of the amount of power required. (1.5)

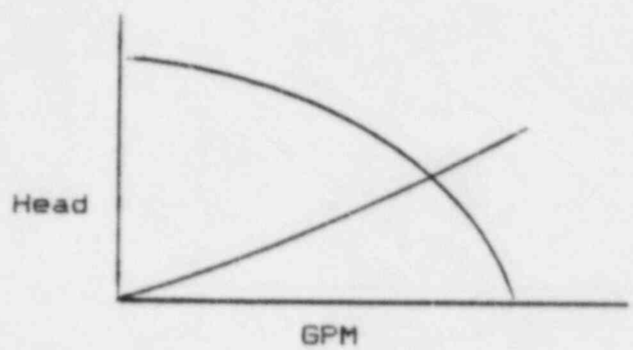


Figure 1.1. Pump Characteristic Curve

1.6 Consider a single control rod positioned near the radial center of the core. During a reactor startup near the end of life, this control rod will be the first control rod to be withdrawn. This full withdrawal of the control ROD will have a certain reactivity worth. After the reactor has been made critical and 100 percent power has been achieved, if this control rod is fully inserted while reactor power is maintained at 100 percent, will the reactivity worth of its full insertion be greater than, the same as, or less than it was at the time of reactor startup? Fully explain your answer. (3.0)

1.7 Describe the initial reactivity response (positive or negative) and the resultant effect on core power (increase, decrease, no change) for each of the following changes in plant parameters. Include which reactivity coefficient is responsible for the change.

a. Closure of one MSIV at full power. (1.5)

b. Loss of feedwater heating. (1.5)

Category Continued on Next Page

1.8 The reactor is critical at "50" on Range ~~2~~ 3 of the IRMs. A control rod is withdrawn three notches, resulting in a power increase with a stable reactor period equal to the minimum permissible sustained positive period permitted in OP-201.1, Approach to Critical. Heating power is estimated to be at "30" on Range ~~2~~ 3. Show all your assumptions and work for the following calculations.

a. What is the doubling time? (1.25)

b. How long will it take to reach heating power? (1.25)

1.9 The MCPR operating limit is established to ensure that the fuel cladding integrity safety limit is not exceeded for any moderate frequency transient. For operation of the Oyster Creek reactor, give two examples of postulated pressure transients, and two examples of postulated temperature transients, and indicate which transient will have the most severe effect on MCPR. (2.0)

End of Category

2. PLANT DESIGN, INCLUDING SAFETY AND EMERGENCY SYSTEMS (25)

2.1 Each EMRV discharge line is equipped with two vacuum breakers.

a. When and under what plant conditions will the vacuum breakers open? (1.5)

b. What adverse condition could occur if the vacuum breakers did not function properly, and why must this condition be avoided? (1.5)

2.2 The Shutdown Cooling System is used to complete plant cooldown after the main condensers are no longer acting as a heat sink.

a. When Shutdown Cooling is started, how must operation of the RBCCW System be changed? (1.0)

b. What requirements are placed on the lineup and/or operation of the "E" Recirc loop? Why? (1.5)

2.3 The combined Standby Gas Treatment System contains seven automatic isolation and purge valves.

a. If the SBGT System is operating with its normal valve lineup, how will these valves (dampers) position on a loss of (1) solenoid power, and (2) on a loss of instrument air? (1.5)

b. What system operational capabilities are provided in case of a loss of instrument air? (0.5)

2.4 Service Water normally provides cooling water to the RBCCW heat exchangers. What ~~three~~ two alternate functions may be provided by the Service Water System? Include the operating conditions under which these functions are provided. (1.5)

2.5 The Standby Liquid Control System requires heaters in the storage tank.

a. Why must an elevated tank temperature be maintained? (0.5)

b. What will cause an automatic trip of the heaters? (0.5)

Category Continued on Next Page

2.6 The function of the vacuum priming system is to remove air from the Circulating Water System. The system includes two vacuum pumps, one of which (the "preferred" pump) normally runs continuously.

a. Under what conditions will the reserve pump start? (0.5)

b. Explain how and why condenser vacuum may be affected if the vacuum priming system fails to operate properly, such as if a float valve should fail in the closed position. (1.5)

2.7 Each Core Spray System contains main and booster pumps to provide water to the reactor under accident conditions.

a. What will cause the start of the priority main Core Spray pump? (0.5)

b. Under what conditions will the backup main Core Spray pump start? (0.5)

c. When will a permissive to ADS be provided? (0.5)

d. When will the parallel isolation valves open? (0.5)

e. O.P. 308, Emergency Core Cooling Operation, directs that Core Spray injection be controlled to maintain water level. What must be done to take manual control, and what Core Spray component will be used to control injection to the vessel? (1.0)

2.8 The Reactor Cleanup System has two types of system isolations. For each isolation type, (1) list the signals that initiate the isolation, (2) the actions that occur within the Cleanup System, and (3) how the isolations must be reset. (3.0)

Category Continued on Next Page

2.9 The Isolation Condensers are available to depressurize the reactor and to remove decay heat if the main condenser is unavailable.

a. What is the normal valve lineup for standby operation of the Isolation Condensers? Why is this lineup required? (1.5)

b. When the Isolation Condensers auto initiate, all five Recirculation System pumps trip. If the Isolation Condensers are manually initiated, only one or two Recirculation System pumps may need to be tripped. Briefly describe the required Recirculation System valve lineups for each of these operating conditions, and explain the reason for these lineups. (1.5)

2.10 Three independent power supplies are provided for the four 4160 vac buses.

a. With the plant operating at 100 percent power, what is the normal power source for each of the four 4160 vac buses? (1.0)

b. With a loss of the normal power source, how may each bus be powered? Be sure to consider all alternatives. (1.5)

2.11 The Containment Spray System provides for Containment cooldown following a loss of coolant accident.

a. What initiation signals must be present for Containment Spray to start automatically? (0.5)

b. What will occur if the Emergency Service Water pump for either System I or II fails to start, and what must you do to assure system operation? (1.0)

End of Category

3. INSTRUMENTS AND CONTROLS (25)

3.1 High temperature in the drywell can affect reactor level indication.

a. Explain why the indicated level will not read true and in what direction it will read under conditions of high drywell temperatures. (1.0)

b. Which level detectors (Yarway, Barton, or GEMACS) will be most affected by high drywell temperatures? Why? (0.5)

c. How much level error can be introduced by high drywell temperatures, and what adverse effects can this have on required system actions? (1.0)

3.2 Instrument Air is required to position or maintain the position of a large number of valves in the plant. For a loss of instrument air header pressure, how will each of the following be affected? (2.5)

a. Hotwell level control valve (V-2-16) - Spill valve
b. Drywell Equipment Drain Tank valves
c. CRD flow control valves
d. Isolation Condenser vent valves
e. Cleanup System letdown valves.

3.3 Each APRM channel contains six trip units that provide rod blocks and reactor scram signals. For each trip unit, with the reactor operating in the RUN mode, indicate whether a rod block or a scram signal is generated, and whether that signal may be reset. Setpoints are not required. (3.0)

3.4 The reactor, initially operating at 100 percent power, has just scrammed due to a load reject. The reactor protection system operates as expected and the reactor shutdown occurs normally. Explain in detail what you must do to reset <u>this particular</u> scram, including any time delays that occur, as well as the reasons for these time delays, and any interlocks that may be involved. (2.5)

Category Continued on Next Page

3.5 An Anticipated Transient Without Scram (ATWS) pump trip was added to the Reactor Recirculation System to reduce reactor power if the reactor does not scram when it should.

a. What parameters or plant conditions initiate the trip logic, and what sensors are used? (1.5)

b. Briefly explain how a spurious trip is precluded on a loss of 125 VDC power to the Channel A trip circuitry. (1.5)

3.6 List two alternate instruments or instrumentation channels that could be used to determine reactor pressure in the event that normal reactor pressure instrumentation (wide and narrow range reactor pressure instruments) was inoperable. (1.0)

3.7 You are withdrawing control rods during a power increase from hot standby. The reactor power is about 15 percent.

a. What are three indications you would have that a control rod might be uncoupled? (Note: These indications might not occur simultaneously.) (1.0)

b. What action would you take to recouple the control rod, and what action is required if the control rod does not recouple (assuming it can be moved with normal drive pressure). (2.0)

3.8 Area Radiation Monitors are located throughout the plant. However, automatic protective actions are initiated by only ~~two~~ some (4) of the monitors.

a. ~~Which~~ Name two Area Radiation Monitors which initiate automatic protective actions? (1.0)

b. What protective action will occur if the alarm trip point is reached on either of these instruments? (2.0)

Category Continued on Next Page

3.9 The RBCCW pump breakers are equipped with a two position control switch (Normal/Bypass).

a. What is the function of this switch, and where is it positioned during normal plant operations? (1.0)

b. If a loss of power and/or a LOCA occurs, how will the pumps respond for each of the two switch positions? Be sure to consider the response of the pumps for the switch initially being positioned in both Normal and Bypass, as well as the response of the pumps should the operator manually change the switch position after the event has occurred. (2.0)

3.10 The reactor has been operating at 80 percent power. You are in the process of increasing power to 100 percent, and are currently holding at about 90 percent. All three feed pumps are in operation with the Master Controller in Auto. You notice the flow for the 1C string is lower than the flows for the other two strings, which are well matched.

a. How can you balance the flows for the three strings? (0.5)

b. If the flow in the 1C string does not respond to the action you have taken in Part (a) of this question, and it is determined that the lack of response is due to a Feed Reg Valve lockup, what two conditions could have caused the lockup? (1.5)

End of Category

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25)

4.1 O.P. 341, Standby Diesel Generator Operation, cautions that when testing is performed, the DG shall not be restarted in a fast start mode during a period of 15 minutes to three hours after shutdown.

a. What is the reason for this caution? (1.0)

b. How does this caution affect the release of the second Diesel Generator for maintenance? Can the second Diesel Generator be started and/or operated? Explain your answer. (1.5)

c. If a Diesel Generator is required to supply power to its respective bus during this three hour period, are you permitted to make it available? (0.5)

4.2 With the reactor operating at 90 percent power, according to ABN-3200.07, Unexplained Reactivity Change,

a. List four indications you would have if a reactivity increase had occurred. (1.0)

b. What three systems should be checked, and what conditions should be looked for, to determine the cause of the reactivity addition? (1.5)

4.3 According to O.P.324, Thermal Dilution Pumps, the Dilution Pump "Auto-Bypass" switches should always be in the "Auto" position, except that there are four exceptions when the switches should be in the "Bypass" position.

a. Why should the pumps normally be operated with the switches in the Auto position? (0.5)

b. What are two of the four times that the switches should be in the Bypass position, and what is the reason for having the switch in the Bypass position for each of these times? (2.0)

4.4 What are the six entry conditions for the RPV Control Procedure, EMG-3200.01? (3.0)

Category Continued on Next Page

Should set this up that the Rx was ... 1/2 @ > 50% & was brought down to 30% because bpvs weren't brought in until 200MW on a power increase

4.5 Operating Procedure 340.1, 125 VDC Distribution Systems "A" and "B", cautions against interrupting 125 VDC to certain 4160 VAC switchgear during normal operation. Explain this caution and the consequences of interrupting this power. (1.5)

4.6 With the reactor operating at 30 percent power, a turbine trip occurs due to a Main Generator fault. According to ABN-3200.10,

a. List five automatic actions that will occur as a result of the turbine trip? (1.5)

b. What effect will this transient have on reactivity? Why? (1.0)

4.7 During plant heatup after reaching critical, according to O.P. 201.2,

a. What are two methods of verifying that recirc flow is sufficient for operation in Range 10 of the IRMs? (1.0)

b. By what pressure must the Cleanup Auxiliary Pump be removed from service? (0.5)

c. How are the bypass valves kept closed prior to the reactor pressure reaching 900 psig? (0.5)

4.8 The reactor is operating at 100 percent power when a single relief valve inadvertently lifts.

a. To which Emergency Operating Procedure will you be directed, and what entry condition(s) specific to this event will direct you there? (1.0)

b. What immediate actions are you required to take according to this procedure? Do not include any actions that may be required after a reactor scram may occur. (2.0)

4.9 According to Administrative Procedure 106, Conduct of Operations, what two statements complete the following sentence regarding Shift Turnover:

"All Off-Going Operators shall not relinquish their responsibilities until he/she is satisfied that the On-Coming Operator: . . . ." (2.0)

Category Continued on Next Page

4.10 According to ABN-3200.01, Reactor Scram, one of the primary concerns after a scram has occurred is to maintain the RPV pressure below 1050 psig. Section 3.7 states that if the Main Condenser is not available, or if the Turbine Bypass Valves are inadequate, four other systems or components may be used to augment RPV pressure control. List these four systems or components along with any restrictions on their use or availability. (3.0)

End of Examination

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$$t_{1/2}eff = \frac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$$

$\Delta E = 931\ \Delta m$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m}Cp\Delta t$

$\dot{Q} = UA\Delta T$

$Pwr = W_f \Delta n$

$I = I_o e^{-\mu x}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\lambda\rho)$

$\rho = (K_{eff}-1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)

MASTER RO
ANSWER KEY

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, AND FLUID FLOW (25) - ANSWERS

1.1 (1) Under conditions of high Xenon concentration, such as following a shutdown, the Xenon will be at the highest levels where the neutron flux was the highest. These locations will be in the central core regions where the SRMs are located. This will tend to suppress the flux around the SRMs, while the flux in regions of the core where Xenon is lower in concentration will have higher flux. (1.5)

(2) Relatively higher notch worths will be found in core regions that had relatively low neutron flux levels during the previous reactor operation because under conditions of a startup with high Xenon, the flux is now relatively higher in these regions. This would be in the peripheral areas of the core, and particularly near the top of the core where boiling had been occuring. (Other reasonable explanations also acceptable.) (1.5)

.75

.75

not required

Note that axial flux profile changes as a function of core life

REFERENCE: O.P. 201.1, Section 3.7
Reactor Theory, Chapter 8,
ppg. 60 - 62.

1.2 a. As power increases, the amount of boiling in a flow channel increases, and causes an increase in the resistance to flow - that is, core dP increases.

b. (1) Two regions are used for core orificing. The central region consists of all but the outermost ring of perimeter fuel with the smaller or fewer orifices in the outer fuel positions.

(2) The orifices installed in the fuel supports add a relatively large dP to the total dP across the core. The dP added by boiling is then insignificant compared to the dP of the orifices, and so the dP is nearly the same (as is flow) to all fuel cells under any condition of power.

REFERENCE: Lesson Plan, NSSS, ppg. 13 - 14.
OC Exam Bank, Question NS-1.

1.3 a. (1) As the high pressure gas in the supply bottle is released into the accumulator, it expands rapidly and thus cools. As it warms to ambient temperature, the pressure in the accumulator will increase. [By filling the accumulator slowly, the gas will tend to reach equilibrium temperature and establish the proper pressure more rapidly.] Therefore, you must wait until equilibrium is reached before the called for pressure can be set. (1.5)

(2) If it were permitted to fill rapidly to the pressure given in the procedure [540 - 620 psig] , as it warmed to ambient conditions the gas pressure would increase to a value higher than the procedure calls for, and an overpressure condition might result. (0.5) ⇒ leads to possibility of not compressing piston and then not having sufficient water volume. (0.2)

b. Normal pressure ~~at~~ 1150 per procedure operating conditions is about 1000 to ~~1050~~ psig. (0.25)
The charging water decreases the accumulator gas volume by about half. Note that the gas pressure is not the same as the charging water pressure. (0.75)

REFERENCE: O.P. 302.1, Sections 3.2.1 and 3.3.

1.4 a. (1) Negative reactivity added from moderator coefficient during heatup.
(2) Negative reactivity added due to fission product poison buildup.
(3) Negative reactivity added during power operation due to void coefficient.
(4) Negative reactivity added during power operation due to fuel burnup.
(5) Negative reactivity due to doppler during normal operating condition
Any three for credit.

b. (1) Control rods
(2) Burnable poison

REFERENCE: Reactor Theory, Chapters 7 and 8.

1.5

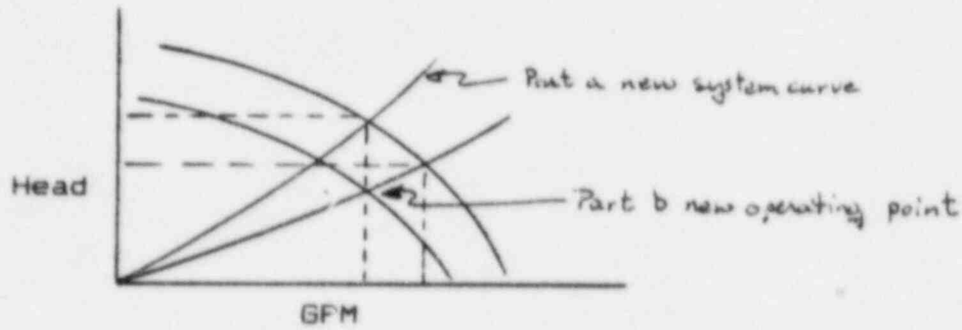


Figure 1.1. Pump Characteristic Curve

This is true for decreasing flow, but not for increasing flow. May need to consider answer on pump laws.

a. Throttling the valve causes the friction losses in the system to increase, and the system curve to move to the left.

b. Changing pump speed is more efficient since a lower flow rate can be achieved without increasing the pump head. Increasing the pump head would require more power.

REFERENCE: GE Thermal Sciences Text, ppg. 7-114 - 7-121.

1.6 Its reactivity worth will be less. (0.5)

Rod worth is proportional to the (local flux divided by the average flux) squared. With all rods inserted, the average flux is low. Then when the rod is withdrawn, it causes the local flux to be relatively high, resulting in a relatively high reactivity worth.

If the same rod is inserted from the fully withdrawn position when all other rods are mostly withdrawn, the flux depression caused by the inserted rod will result in a small value of (local flux/avg flux)^2. The worth of the rod in this case is thus much less than in the above case.

REFERENCE: Reactor Theory, Chapter 8.

1.7 a. Reactivity increases
Power increases
Void coefficient

b. Reactivity increases
Power increases
Moderator temperature coefficient

REFERENCE: Reactor Theory, Chapter 8.

1.8 a. Min permissible stable period = 30 sec (0.5)

DT = period/1.445 or 1.443

= 30/1.445 = 20.8 sec 0.75 (~~1.0~~)

b. $P = Po\ e^{t/T}$ $P/Po = 3000/5 = 600$

$\ln 600 = t/T$

$t = 30 * \ln 600 = 192$ sec = 3.2 min

REFERENCE: Reactor Theory, Chapter 11.
O.P. 201.1, Section 3.2.

1.9 Pressure Transients
(1) Turbine trip without bypass above 40 percent power
(2) Turbine trip with bypass above or below 40 percent
(3) Turbine trip without bypass below 40 percent power
(4) Generator load reject
(5) MSIV closure
(6) Loss of vacuum
(7) Pressure regulator failure
(8) EMRV closure

Temperature Transients
(1) Feedwater Controller failing at maximum demand
(2) Loss of Feedwater heaters
(3) Inadvertent Isolation Condenser initiation

0.45 for any two of each type of transient
0.2 for indicating that turbine trip without bypass above 40 percent is the most limiting transient

REFERENCE: GE Thermal Sciences Text,
ppg. 9-94 - 9-96.
Lesson Plan # 201, ppg. 19 - 33.

End of Category

2. PLANT DESIGN, INCLUDING SAFETY AND EMERGENCY SYSTEMS (25) - ANSWERS

2.1 a. The vacuum breakers (check valves) will open after the EMRVs close (0.75)
when the discharge pipe cools and the pressure inside the pipe falls below that in the drywell (torus). (0.75)

b. If the vacuum breakers did not function properly, the low pressure condition would cause a long water slug to rise in the discharge pipe. (0.75)

This condition must be avoided because of the steam vent clearing phenomena. If the water slug is too long, structural damage may occur. (0.75)

REFERENCE: Lesson Plan # 5, ppg. 8 and 18.
Exam Bank Qs # AD 7 and 9.

2.2 a. Normally RBCCW is operated with one pump running and two heat exchangers in service. (0.5)
For Shutdown Cooling operation, a second pump must be started. (0.5)

b. The discharge valve must be shut (0.5)
or the pump must be running (0.5)
to prevent short circuiting the vessel. (0.5)

REFERENCE: a. Procedure 309.2, Section 3.0.2.5
b. Procedure 305, Section 3.3.15 .

2.3 a. (1) On a loss of solenoid power, the inlet and outlet~~, and crosstie~~ valves will fail closed, (0.6)
and the inlet purge and crosstie valves will fail open. (0.36)

(2) On a loss of instrument air, all valves will fail as is. (0.3)

b. If instrument air is lost, the inlet and outlet valves only may be operated [through five cycles] with air in air accumulators.

REFERENCE: Lesson Plan # 50, Secondary Containment and SBGT, ppg. 11 - 13.

2.4 (1) Cooling water to the TBCCW when Circ Water is shutdown (0.75)
~~(2) Maintains ESW side of the Containment Spray heat exchangers full and vented during normal operations~~
(3) Provides a manual backup of sealing water to the Circ Water pumps when normal Fire Protection Water System Supply is unavailable. (0.75)

REFERENCE: Lesson Plan # 51, Service Water System, pg. 2.

2.5 a. To preclude precipitation of the sodium pentaborate.

b. Low Water level in the tank [6"] above the heaters.

REFERENCE: OC Exam Bank, Question # LP-5.

2.6 a. The reserve pump will start on low vacuum [18 inches] in the vacuum tank.

b. If a float valve should fail closed, air entrained in the circ water will collect in the high points of the system - primarily at the top of the affected water box. (0.5)

This will result in circ water not entering the upper tubes, (0.5)

Resulting in decreased condensation of the turbine exhaust and reduced condenser efficiency. (0.5)

REFERENCE: Lesson Plan # 14, Main Condenser Operation, ppg. 9 - 11.

2.7 a. Low Low reactor water level [7' 2"] (1 of 4)
or
High Drywell pressure [2 psig] (1 of 4)

b. If the priority pump discharge pressure doesn't reach 100 psig within 10 seconds

c. After 230 psig system pressure is achieved
or 50 psid across the pump (per Non modification)

d. When reactor pressure falls to 285 psig

e. Must have cleared or overridden all initiating and auto start signals (0.5)
Control by opening and closing the parallel valves. (0.5)

REFERENCE: Lesson Plan # 10, Core Spray System, Figure - Core Spray Logic. O.P. 308, Section 5.3.5.

2.8 (1) Type I
1. Low flow or outlet valve shut in service filter [80 gpm]
2. Aux pump cooling water high outlet temp [130 F]
3. Non-regen Hx outlet temp high [140 F]
4. High system pressure [140 psig]
5. SBLC on [15 gpm]

Type II
1. High Drywell pressure [2 psig] changed to 2.4 psig as Tech Spec amendment change
2. LoLo Rx water level [7'2"]

(2) Type I
1. Isolates V-16-1, 2, and 14
2. Trips system recirc pump
3. Aux pump trips on two valves closing

Type II
1. Isolates V-16-1, 2, 14, and 61
2. Trips system recirc pump
3. Aux pump trips on two valves closing

(3) The isolation must be reset using the Drywell Isolation reset switch for both types since the signal seals in. (0.4)

(0.2) per item in parts (1) and (2). Setpoints not required.

REFERENCE: OC Exam Bank, Question # CU-6 and Lesson Plan # 43, Reactor Cleanup System, pg. 8.

2.9 a. (1) Vent valves [1, 19 and 5, 20] are normallv open (0.25)
(2) Condensate return valves [34 and 35] normally closed (0.25)
(3) Condensate AC valves and Steam valves [30,31,36, and 32, 33, 37] normally open (0.25)

The valve lineup provides for the venting of non-condensibles to the main steam lines (0.25)
and for keeping the ICs warm. (0.25)
The AC condensate return valve is kept open for IC reliability. (0.25)

b. In both cases, the suction and main discharge valves of at least two loops must be kept open (0.5)
to assure a flowpath for natural circulation. (0.25)

If two or more recirc pumps are operating, the main discharge valves for the non-operating loops must be closed, and the suction and discharge bypass valves must be open. (0.5)
This is to assure forced circulation through the core. (0.25)

REFERENCE: O.P. 307, Sections 2.2 and 2.3
Lesson Plan # 21, Isolation Condensers, ppg. 6 and 10.

2.10 a. The normal power source is the main generator through the aux transformer. (0.5)
1C and 1D are fed from 1A and 1B, respectively. (0.5)

b. If power is lost from the aux transformer, all four buses will be powered from the startup transformer if it is available. (0.5)

If the startup transformer is not available, 1C and 1D will be powered from their respective diesels. No power will be available for 1A and 1B. (1.0)

REFERENCE: Lesson Plan # 39, Plant Electrical Distribution, ppg. 5 - 6, and
P & IDs 3001 and 3002.

2.11 a. High drywell pressure
low low water level

b. ~~The CS pump will trip.~~ Trip block initiated ✓ (0.5)

~~Start the backup CS pump and~~ the backup ESW ✓ pump should be started by the operator (0.5)

REFERENCE: O. P. 310, Sections 3.3 and 4.3.

End of Category

T=0 CS pump starts 40 sec after init signal

T=45 ESW pump starts - If it doesn't start, it gives a block to the lo flow & Hi T trips on the CS pump (forever) to permit an op. to respond by starting the B/U ESW pump only

3.3

Trip Unit	Rod Block	Scram	Resetable
Inop	Y	Y	
Downscale	Y	*	
Int. Slope	Y		Y
Normal Slope	Y		Y
High	Y		
High-High		Y	

* Get a reactor scram with the companion IRM upscale at High-High

0.3 per named trip unit
0.12 per correct Y or *

REFERENCE: Lesson Plan # 37, Nuclear Instrumentation, ppg. 21 - 25.

3.4 (1) The load reject scram will clear (be bypassed) when steam flow falls to less than 40 percent of rated flow. (0.5)

(2) The mode switch must be taken to shutdown by procedure 3200.01 to permit SDV Hi bypass. (0.5)
[This inserts a scram that is bypassed after 20 sec.]

(3) The high discharge volume trip may then be and must be bypassed to vent and drain the SDV. (0.5)

This can only be done after a 10 second time delay to give the scram time to complete the rod insertion. (0.5)

(4) The scram reset switches may then be depressed to reset the scram [reenergizing the scram relays and pilot solenoid valves, and to supply air to the scram valves.] (0.5)

REFERENCE: Lesson Plan # 46, Reactor Protection System, ppg. 22 - 25.
OC Exam Bank, Question # RP-8.

[illegible]

3.5 a. High pressure [1060 psig] (0.25)
sensed by the isolation condenser pressure
sensors (0.5)
Low low reactor water level (0.25)
sensed by the core spray level sensors (0.5) Same as lolo Yarway
or as sensed by the ISO cond logic

b. There are four sensor channels, and it takes
two channels to cause a trip. (0.5)

The trip must be caused either by Channels A
and B together, or by Channels C and D
together. (0.5)

125 VDC power is thus provided to Channels A
and C from one source [Power Panel D], and to
Channels B and D from a second source [Power
Panel F]. (0.5)

REFERENCE: Lesson Plan # 48, Recirc Flow
Control, Attachment 2.

3.6 (1) Isolation Condenser pressure
(2) RCP # 1 seal pressure
(3) Cleanup System pressure

Any two of three.

REFERENCE: OC Exam Bank, Question VI-6.

3.7 a. (1) Change in flux
(2) Rod Overtravel Alarm
(3) The control rod position display goes
dark.

b. (1) Drive the rod in until a power response is
observed, or until it reaches "00".
(2) Withdraw the rod to its programmed
position, observing power response.
(3) If "48" is reached, perform a coupling
check.
(4) If it does not recouple, insert to "00"
and isolate the HCU.

REFERENCE: Lesson Plan # 45, Reactor Manual
Controls, pg. 11.
OC Exam Bank, Question # RM-5
ABN-3200.06, Abnormal Control Rod
Motion, Section 3.5.

3.8 a. Fuel Pool low range (C-9)
Reactor Operating floor equipment hatch area (B-9)
Vent Monitors - #1 and #2
Accept any 2 for credit

b. Two min timer starts
If the timer times out (0.5)
Isolates the reactor building normal ventilation (0.75)
Initiates the SBGT System (0.75)

REFERENCE: Lesson Plan # 4, pg. 6.

3.9 a. This switch will determine the response of the pumps during the sequential loading of the DGs after a loss of electrical power. (0.75)

It is normally in the Normal position. (0.25)

b. When in Normal, if a loss of power occurs with no LOCA signal, the pumps will auto start in sequence [166 sec]. If a LOCA signal is present, the pumps will not start [since they aren't needed]. (1.0)

When in Bypass, the pumps will auto start in sequence whether a LOCA signal is present or not on a loss of power. (0.5)

If the switch was initially in Normal when a loss of power occured with a LOCA signal, it must be taken to Bypass by the operator to start a pump if the operator has determined the need for a pump. (0.5)

REFERENCE: Lesson Plan # 41, RBCCW, pg. 7.

3.10 a. By adjusting the bias adjust knobs on each of the individual M/A stations.

b. The feed reg valve lockup would have occured due to
a loss of instrument air pressure [<70 psig] ~~(0.5)~~
or due to a loss of electrical signal. ~~(0.5)~~
Could have been inadvertently left pinned.
(0.75) [illegible]

REFERENCE: Lesson Plan # 44, Reactor Level and Feedwater Control, ppg. 4 - 9.
OC Exam Bank, Questions FW-2 and 5.

End of Category

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25) - ANSWERS

4.1 a. If a fast start is initiated during this period, a potential failure of the turbocharger bearing may result. This may cause the DG to fail to start when it is needed.

b. The testing must have have been completed at least three hours prior to releasing the second DG for maintenance. (0.75)

The second DG may be started and operated as long as it is not started or cycled in the fast start mode. (0.75)

c. Yes

REFERENCE: O.P.341, Sections 2.2.4, 2.2.11, and 2.2.12, and Standing Order # 28.

4.2 a. (1) MW electric increase
(2) Reactor thermal power increases
(3) LPRM readings increase
(4) Steam flow increases
(5) Feed flow increases
(6) Other reasonable parameter changes not listed in the procedure may be accepted

b. (1) CRD for abnormal control rod motion
(2) Recirc for a flow abnormality
(3) Turbine Generator for a pressure regulator malfunction

May be other reasonable answers e.g. loss of FW htrs

REFERENCE: ABN-3200.07, Sections 2.2 and 3.3.1.

4.3 a. To insure the pumps will trip whenever a scram occurs.

b. (1) When the intake temperature is above 60 degrees to insure dilution in the discharge canal to preclude thermal shock during a scram.

(2) When thw weekly CRD scram brush recorder assurance check is being performed to preclude a pump trip when the recorder is tested.

(3) When the pump is required to provide dilution flow during flow during an overboard (radioactive discharge) release. [If a scram occurs during this time, the release should be terminated and the pump tripped.]

(4) When the reactor is shut down because they aren't needed. Also following a scram, they should not operate to minimize the rate of temperature decrease in the discharge canal.

0.5 for condition and 0.5 for reason.

REFERENCE: O.P. 324, Section 3.4.

4.4 (1) RPV level below the low level scram setpoint [138 inches]
(2) Drywell pressure above scram setpoint [2.0 psig]
(3) A condition that requires a reactor scram AND power is above 2 percent (APRM downscale trip)
(4) RPV pressure above high reactor pressure scram setpoint [1050 psig]
(5) MSIV closure
(6) A condition which requires a reactor scram in the judgement of the operator to either conserve RPV inventory or to reduce the release of radioactivity into the environment.

REFERENCE: EMG-3200.01, pg. 3.

4.5 After interrupting the DC control power, the breakers will continue to function as required. However, when re-applying DC, the DC trip coils on certain breakers may trip. [Examples are the feedpump, recirc pump, and RWCU pump breakers.]

REFERENCE: O. P. 340.1, Section 2.2.5, and OC Exam Bank, Question # DC-2.

1. Breakers remain closed
2. Lose auto & remote manual control.
3. On reapply, they may trip depending on how they are wired

Also from Table 1, individual valves are listed — a. Main Stop ...
b. Control vlvs cl
c. Rate/Intercept ...
d. low load ... vlvs open
e. Ext. ... clo
f. Ext. ... (main removed)
g. Gen ...

4.6 a. (1) 230 KV breakers open
(2) Turbine steam valves close
(3) Bypass valves open
(4) Loads transfer to startup transformer
(5) DGs start and idle
(6) Main turbine aux oil pump starts
(7) Turning gear oil pump starts

b. Reactivity will increase because of a loss of feedwater heating.
[FW htrs may not be in operation because they ... @ 200 MWe]
REFERENCE: ABN-3200.10, Sections 2.3, 3.2, 3.6, 3.7.

4.7 a. (1) Monitor the plant computer which provides a readout in lbm/hr.
(2) Use curve in procedure to convert gpm reading as a function of reactor pressure to minimum required flow.

b. Prior to reaching 125 psig [Bearings and mechanical seals designed for only 150 psig.]

c. By setting the MPR 100 psig above reactor pressure. Keep MPR higher than Rx Pressure in reality. - ok answer

REFERENCE: O.P. 201.2, Sections 4.1, 4.9, and 4.10.1.

4.8 a. Direction will be to EMG-3200.02, Containment Control, and specifically into TOR/T. (0.5)

The specific entry condition is Torus water temperature above 90 degrees [possibly just rising and approaching 90 degrees - CAF] (0.5)

b. (1) Execute TOR/T, DW/T, PC/P, and TOR/L concurrently. ~~(0.5)~~ not required
Note that no action will be required in in the latter three.
(2) Attempt to close the valve by
(a) cycling it (0.67)
(b) removing control power to it by pulling the fuses (0.67)
(3) If the valve cannot be reseated, scram the reactor. (0.67)

REFERENCE: EMG-3200.02, Containment Control and above indicated sub parts.

4.9 a. "Is fully qualified, and/or licensed, to assume the shift position."

b. "Is fully aware of existing plant/equipment conditions."

Exact wording not required. - Also - if there is doubt that he is alert, coherent, & fully capable.

REFERENCE: A.P. 106, Section 4.3.1.1. and 4.3.1

4.10 (1) Isolation Condensers
Must trip A and E recirc pumps
(2) EMRVs only if Torus Water Level is [above 90 inches] sufficient
(3) Cleanup only if boron injection is not required
(4) Main Steam Line Drains only if the Main Condenser is available.

0.25 for each system, 0.5 for each restriction

REFERENCE: ABN-3200.01, Section 3.7 and Table 2.

End of Examination