U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No. 50-341 License No. CPPR-87

Licensee: The Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

Facility Name: Fermi Nuclear Plant

Examination Administered At: Fermi Nuclear Plant

Examination Conducted:

Examiners: T. Lang for
W. Thomas 9/25/84
Date

T. Lang for
K. Henry 9/25/84
Date

Approved By: J. I. McMillen
J. I. McMillen, Chief
Operating Licensing Section 9/25/84
Date

Examination Summary

Examination administered on July 10, 1984.
On July 10, 1984 Senior Operator License examinations were given to five contractor personnel at the Fermi Nuclear Plant. The five contractors are intended to fill the Senior on shift Advisors (SOA's) positions.
Results: Of the five candidates all five passed the examination, and will be issued a senior operator license.

REPORT DETAILS

1. Persons Examined

   SRO Candidates

   Five contractor candidates examined at the senior level to be used as SOA's.

2. Examiners

   W. Thomas, ORNL
   K. Henry, ORNL

3. Examination Review Meeting

   The examination was reviewed the the following utility personnel: Messrs. F. Abramson, R. Parmelee, G. Overbeck, and A. Wisniewski.

   During the review process only three questions were brought to the attention of the examiners as being bad questions. None of the three questions were deleted from the exam. The questions in contention were 5-11, 6-8, and 8-2b.

4. Exit Meeting

   The examination exit meeting was conducted on July 13, 1984. At the meeting the clear passes were made known to the utility (all five candidates). Also, at this meeting or immediately prior to it the utility furnished the examiners with their final exam comments.

Reviewed By:

U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Frederick E. Abramson
Gregg R. Overbeck
Anthony T Wisniewski
N.R.

MASTER COPY

Facility: ENRICO FERMI

Reactor Type: BWR/4

Date Administered: 7/10/84

Examiner: W.THOMAS

Applicant: ______

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple questions 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 Cat. Value	Category
25.0	25			5. Theory of Nuclear Power Plant Operation, Fluids & Thermodynamics
25.0	25			6. Plant Systems: Design, Control & Instrumentation
25.0	25			7. Procedures-Normal, Abnormal, Emergency & Radiological Control
25.0	25			8. Administrative Procedures, Condition and Limitations
100.0	100			TOTALS

Final Grade ______ %

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

______
Applicant's Signature

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS & THERMODYNAMICS (25.0)

5-1 a. In the course of a reactor startup, it is desirable to know how close the reactor is to criticality.

1. Briefly describe how the operator knows the reactor is close to critical, but still subcritical. (1.0)

2. How does an operator determine that the reactor "is critical"? (0.5)

b. What is the minimum acceptable SRM period during a reactor startup? (0.5)

c. The operator must take what immediate action, and notify whom, before resuming a control rod withdrawal following an unacceptable sustained reactor period? (0.5)

5-2 Assume the reactor is operating at 100% power in 3-element control and one recirculation pump trips. Indicate how each listed parameter would initially respond (increase, decrease, remain the same) and briefly explain the reason for the response.

a. Reactor power (neutron flux) (1.0)
b. Reactor water level (1.0)
c. Feedwater flow (1.0)

5-3 How does the MAGNITUDE of the VOID COEFFICIENT of REACTIVITY change (more negative, less negative, or unaffected) for each of the following changes in core condition? BRIEFLY EXPLAIN WHY.

a. Increase in core void fraction (1.0)
b. Decrease in fuel temperature (1.0)
c. Increase in core age (effective core size) (1.0)

5-4 Give the reason (basis) why chloride ion concentration limits are:

a. Lower during reactor startup (condition 2) than at full power operation. (1.0)

b. Lower during condition 2 than during refueling. (1.0)

(continued on next page)

5-5 For each of the following conditions, state whether individual control rod worth will <u>increase</u>, <u>decrease</u>, or <u>remain the same</u> and briefly <u>explain</u> why:

a. Moderator temperature <u>increase</u> (1.0)
b. Increase in control rod density (1.0)
c. Core void fraction <u>increase</u> (1.0)

5-6 a. Which isotope is produced at a faster rate from precursor decay, ^{135}Xe or ^{149}Sm? (0.5)

b. At what point in the operating cycle does ^{149}Sm reach its' peak? (0.5)

c. When would you expect the reactor to be Samarium-free? (0.5)

5-7 The reactor has been operated for several days at rated power and flow. An operator then makes a power reduction to 75% of rated by driving in control rods. If he does not change the speed of either recirculation pump, what will happen to indicated core flow? <u>Why?</u> (2.0)

5-8 List four (4) of the seven (7) heat gains and losses involved in a BWR reactor heat balance. Indicate whether each is a gain or a loss. (2.0)

5-9 A precaution in the condensate System Procedure (23.102) states. "Do not place the condensate minimum flow recirculation valve in service with less than two (2) condensate pumps operating". What is the reason for this caution statement? (1.0)

5-10 TRUE/FALSE. Air or gas entrained in water can cause pump cavitation even if NPSH > 0? (0.5)

5-11 TRUE/FALSE: During a loss of coolant accident, the fuel clad material (Zircaloy-2) will be <u>more</u> resistant to hydriding than the fuel channel material (Zircaloy-4). (0.5)

(continued on next page)

5-12 Shown in Figure 5.1 are the system curve and the head versus capacity (flow rate) curves for each individual pump, A and B. For each two-pump arrangement on the figure:

a. Construct the combined head versus capacity curve (both pumps running). (1.5)

b. Locate and label the operating point (both pumps running). (0.5)

5-13 Match the appropriate thermal limit (a-c), (2.0)

a. Linear heat generation rate (LHGR)
b. Average planar linear heat generation rate (APLHGR)
c. Minimum critical power ratio (MCPR)

to each FAILURE MECHANISM and to each LIMITING CONDITION given below:

FAILURE MECHANISM	LIMITING CONDITION
F1. Clad melting caused by decay heat & stored heat following a LOCA	L1. Coolant transition boiling
F2. Clad cracking resulting from the surface becoming vapor "blanketed"	L2. Clad plastic strain $\leq$ 1%
F3. Clad cracking caused by high stress from pellet expansion.	L3. Maximum clad temperature of 2200°F.

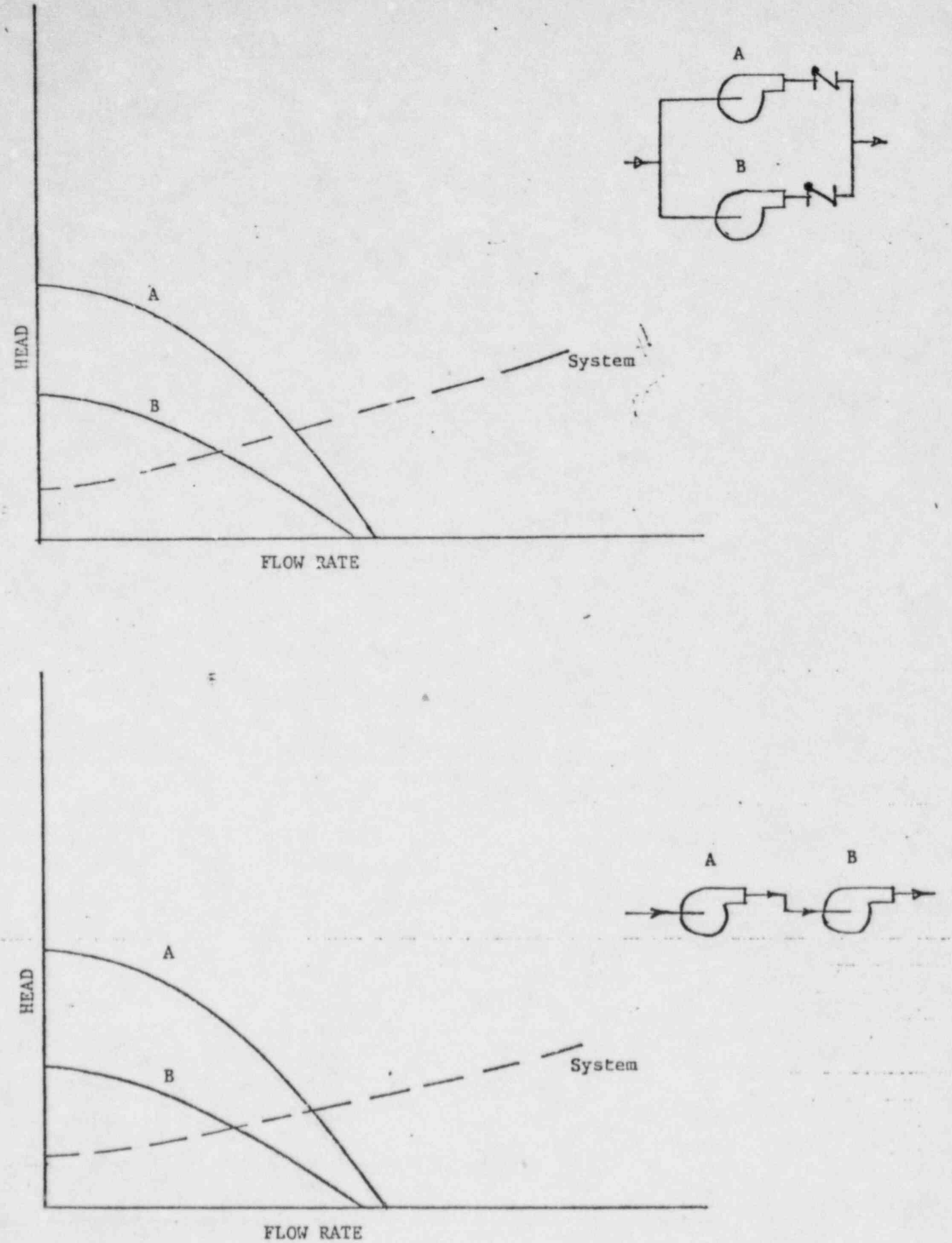

FIGURE 5.1

6. PLANT SYSTEMS: DESIGN, CONTROL & INSTRUMENTATION (25.0)

6-1 For the High Pressure Coolant Injection System:

a. What effect will each initial valve position (1-3) have on system operation and/or the position of HPCI valves when an auto-initiation signal is received?

1. Turbine exhaust valve (F021) is closed. (0.5)
2. Torus suction valves (F041 and F042) are open. (0.5)
3. Steam outboard isolation valve (F003) and its' bypass valve (F600) are each closed. (0.5)

b. Where does the barometric condenser vacuum tank condensate pump discharge to when:

1. the HPCI system has been initiated? (0.5)
2. the HPCI system is in the "standby mode"? (0.5)

6-2 Pertaining to the Rod Block Monitor (RBM) System:

a. What are three (3) ways to, or conditions that cause an RBM channel to be bypassed? (1.5)

b. Selecting a control rod surrounded by three (3) strings of LPRM's requires how many inputs to avoid an INOP trip? (0.5)

c. Which RBM channel receives an input from LPRM level C? (0.5)

d. With the reactor at rated power, what effect will manually bypassing the reference APRM have on the RBM system? (0.5)

6-3 a. Explain how the logic of the reactor protection system is altered by installing shorting links in the neutron monitoring system (i.e., links installed vs links removed). (2.0)

b. When is it a requirement that shorting links be removed? (1.0)

(continued on next page)

6-4 For the emergency diesel generator (EDG) system:

a. List three (3) signals that auto-start the EDG's. (1.5)

b. TRUE/FALSE: An emergency signal will not cause the diesels to start when the LOCAL/REMOTE switch is in the LOCAL position. (0.5)

c. Select from the list below three (3) trips that would cause an EDG to automatically shutdown when an emergency start signal is present. (1.0)

1- Jacket coolant pressure low (22 psig)
2- Failure to reach 250 RPM in 7 seconds
3- Low fuel oil pressure (7 psig)
4- High jacket coolant temperature (195°F)
5- Low lube oil pressure (23 psig)
6- High crankcase pressure (0.5 in water)
7- High lube oil temperature (225°F)

6-5 In the recirculation pump speed control system there are three (3) speed limiters--LIMITER #1, LIMITER #2, and LIMITER #3. What are the setpoints for these limiters and under what conditions will each limiter be bypassed? (3.0)

6-6 With regard to the reactor water cleanup system (RWCU):

a. What can happen if no operator action is taken following a Filter-Demineralizer-A high differential pressure alarm? Include setpoint. (0.5)

b. List three (3) independent system indications of a pipe break in the heat exchanger room. (1.5)

c. What two (2) conditions will cause the blowdown flow control valve to radwaste (or main condenser) to automatically close? Include setpoints. (1.0)

(continued on next page)

6-7 For the Reactor Core Isolation Cooling System:

a. Which RCIC components, if any, require AC power? (0.5)

b. Assume the following standby status of the RCIC system:

• CST suction valve (F010) closed.

• suppression pool suction valves (F029 and F031) each closed.

1. What will be the position of these valves following an automatic system initiation? (0.5)

2. What will be the position of these valves if Torus high water level occurs concurrent with the auto initiation signal? (0.5)

c. A LOCA condition exists and is of sufficient magnitude to cause RCIC injection. Subsequently RCIC trips on high water level, and then level begins decreasing. Without operator action, will RCIC automatically resume injection to the reactor once the trip signal has cleared? (1.0)

-----If yes, explain how the system functions to cause this result.

-----If no, explain the operator and system actions that must be accomplished to resume injection.

6-8 Explain the use of the 64T and 65T breakers. Include any interlocks associated with the closure of these breakers. (1.0)

6-9 List the following pumps in the order of decreasing design capacity (gpm). (0.5)

a. HPCI pumps (booster and main)
b. RCIC pump
c. Two (2) SBFW pumps
d. One (1) core spray pump

6-10 Control rods are pulled according to SEQUENCE-A and 94 of the rods have been withdrawn. Choose the proper RSCS switch positions below for this point in a reactor startup.

a. Mode switch position? (Insert/Normal/Withdraw) (0.5)
b. Sequence switch position? (A12/A34/Normal/B34/B12) (0.5)

6-11 List five (5) valve and/or component actions which occur on automatic LPCI initiation (which are related to LPCI operation). (2.5)

7. PROCEDURES-NORMAL, ABNORMAL, EMERGENCY & RADIOLOGICAL CONTROL (25.0)

7-1 For the reactivity control emergency procedure (29.000.08):

a. Give the three (3) procedure entry conditions. (1.5)

b. Immediate use of standby liquid control is required when what set of conditions exist? (1.0)

7-2 Assume an emergency exists which requires control room evacuation and plant shutdown from outside the control room (procedure 20.000.19):

a. If possible, what three (3) operator actions should be performed prior to leaving the control room? (1.5)

b. What is the procedure recommended method to scram the reactor from outside the control room? (0.5)

c. How is reactor pressure controlled both before and after MSIV closure? (1.0)

7-3 When is it necessary to commence a rapid plant shutdown because of a stuck open safety relief valve? (1.0)

7-4 List five (5) situations, or conditions, for which a standard Radiation Work Permit (RWP) is required. (2.5)

7-5 During full power operation a malfunction of the reactor pressure controller result in the pressure regulator signal failing high (which indicates a high reactor pressure). What is the effect of the malfunction on each parameter below? NOTE: Select one (1) underscored word for each part of the question.

a. Generator MWE output (will increase/will decrease) (0.5)
b. Reactor water level (increase/decrease/unaffected) (0.5)
c. Reactor pressure (increase/decrease) (0.5)
d. Main Steam Flow (increase/decrease/unaffected) (0.5)
e. Turbine bypass valve (open/remain closed) (0.5)

7-6 A supervisor having an SRO license (or SRO limited to fuel handling license) must be present on the refuel floor while "core alterations" are in progress. Define the term CORE ALTERATION. (1.5)

(continued on next page)

7-7 Pertaining to a reactor startup from cold condition:

a. On the reactor startup master check list, who (Job Title) decides if a system lineup is required? (0.5)

b. What three (3) conditions must be verified before placing the mode switch to run? (1.0)

c. List three (3) major components placed in-service at 60% reactor power. (1.0)

d. During heatup where is the reactor pressure regulator setpoint maintained, and why? (1.0)

e. What are the Tech. Specs. constraints on primary containment inerting? (1.0)

7-8 Give the <u>immediate operator actions</u> for each type of fuel cladding failure described below (procedure 20.000.07):

a. An indication of an abnormal high off-gas activity, but no main steam line high rad trip (small failure). (1.5)

b. A Main Steam Line high radiation trip (gross failure) (1.5)

7-9 With the reactor at rated power, what action is required if the low pressure CO_2 fire suppression system for the emergency diesel generator room becomes inoperable? (1.5)

7-10 Operation of the mechanical vacuum pumps should be limited to less than what thermal power level? Why? (1.5)

7-11 What are the three (3) conditions necessary for the reactor to be in a "hot standby" condition? (1.5)

8. ADMINISTRATIVE PROCEDURES, CONDITIONS & LIMITATIONS (25.0)

8-1 Give three (3) reasons why recirculation pump speed mismatch is administratively limited to a low value (less than 10%) during routine plant operation. (3.0)

8-2 Regarding changes to safety-related procedures (12.000.07):

a. Interim approval of procedure changes which appear on temporary change requests are granted by whom (Job Titles/qualifications)? (1.0)

b. Answer each part below TRUE/FALSE, and if FALSE briefly state why.

1----Changes in procedure format or procedure intent are initiated on temporary change requests. (0.5)

2----Temporary change requests are cancelled if final approval is not received within 14 calendar days after the interim approval date. (0.5)

3----Temporary change requests are not used to correct typographical, spelling or grammatical errors appearing in procedures. (0.5)

8-3 Describe the four (4) SAFETY LIMITS for the Fermi Plant. NOTE: Action statements are not required. (4.0)

8-4 Itemize the data required to be logged in the NSO log upon reaching criticality during a plant startup. (Five (5) required.) (2.5)

8-5 What administrative requirements must be met to continue a plant startup from 10% of rated power with a bypassed rod worth minimizer? (1.5)

8-6 a. Briefly describe the purpose and use of each of the following tags:

1. White tag with red lettering (0.75)
2. Safety tag. (0.75)

b. What does each of the following safety rope barriers protect against.

1. Solid orange rope barriers? (0.75)
2. Purple and yellow rope barriers? (0.75)

(continued on next page)

8-7 a. The Fire Brigade is composed of what personnel? (1.5)
b. Are there any personnel excluded from the Fire Brigade? Explain. (1.0)

8-8 A surveillance test for a safety related system is due, and the plant operating conditions will permit conducting the test, but testing is delayed because of scheduling difficulty. What are the Tech. Specs. provisions for extending the surveillance time interval (include in your answer any consideration to previous surveillance intervals)? (2.0)

8-9 A limiting condition for operation (LCO) exists because of low pressure on two (2) scram accumulators. If the inoperable accumulators are associated with withdrawn control rods and the reactor is at rated power, what must be immediately verified? How is the verification accomplished? (2.5)

8-10 When is an unlicensed person permitted to withdraw the control rods of an operating nuclear reactor? (1.0)

8-11 TRUE/FALSE: Reactor pressure and temperature should be recorded every 15 minutes when an RHR loop is operating in the shutdown cooling mode and reactor temperature is $\leq$ 170°F. (0.5)

5-1 a. 1. The period meter would show a trend of taking a longer time between rod pulls to return back to infinity. The count rate meter would show a trend of taking a longer time between rod pulls to reach an equilibrium count rate. The longer the time required for the period meter to return to infinity, and the longer it takes the count rate meter to reach equilibrium, the closer the reactor is to criticality. (1.0)

2. When a stable positive period is maintained with no control rod motion the reactor is announced "critical" (even though actually supercritical). (0.5)

b. Greater than 50 sec. (0.5)

c. Immediately insert control rods until a Rx period > 50 sec. is indicated. Contact the Nuclear Shift Supervisor prior to resuming withdrawal of control rods. (0.5)

REF.: Startup Procedure 22.000.03 p. 2 and 6

5-2 a. Decrease (0.5) due to increased core void content as recirc. flow decreases (0.5)

b. Increase (0.5) due to increased core voiding and recirc. pump no longer taking suction on the annualus. (0.5)

c. Decrease (0.5) due to steam flow decrease and level increase (0.5)

REF.: Decrease in core coolant flow events study guide KN/DW/R122.1

(continued on next page)

5-3 a. More negative. The increase in void fraction implies that more voids are being formed in the region of maximum thermal neutron flux (i.e., voids form lower in the core); therefore a small change in void fraction has a larger effect on reactivity with increased voiding. (9.8) (1.0)

b. Less negative. Decreasing fuel temp. decreases resonance capture (resonances not as broad) which results in an increase in resonance escape probability (i.e., for a given void fraction more neutrons can thermalize, interact with fuel and cause fission) - (9.3) (1.0)

c. Less negative. The more negative contribution caused by decreased resonance escape (increased resonance absorption in ^{240}Pu buildup with age) is overridden by the combined less negative contributions from 1) decreased thermal utilization (due to increased moderator-to-fuel ratio as fuel depletes) plus, 2) increased "core size" which decreases neutron leakage. (9.10) (1.0)

REF.: Reactor Theory Chapter 9, pages as given in () above

5-4 a. Disolved oxygen content in the coolant (water) is lower at power operation hence the effect of Cl ion concentration on stress corrosion cracking of SS is less at power operation. (1.0)

b. The water temperature necessary for stress corrosion to occur is not present during refueling operations. (1.0)

REF.: T.S. bases 3/4.4.4 Chemistry (p. B3/4.4.2)

5-5 a. Increase-Moderator density decrease results in increased leakage of thermal neutrons form fuel bundle into the control rod region or decreased moderator density results in an increase of migration length which allows rod capture of thermal neutrons formed more distant from the rod. (1.0)

b. Decrease-Increasing the number of notches in core decreases the region (zone) of control of a particular control rod. The rod is absorbing thermal neutrons leaked from fewer bundles because the additional inserted notches are "shadowing" it. (1.0)

c. Decrease-Increased voiding causes a significant decrease in thermal flux at the rod because of an increase in fuel resonance absorption i.e., fewer thermal neutron in the bundle are available for leakage to a control rod. (1.0)

REF.: Fermi Reactor Theory NT/R251/Chater 9.25-9.27

(continued on next page)

5-6 a. ^{135}Xe (0.5)

b. During a shutdown (following power operation) after all promethium has dec yed (0.5)

c. Initial core before startup (0.5)

REF.: Fermi Lesson Plan Chapter 10, Fission Product Poison

5-7 Core flow increases due to a reduction in two-phase losses in the core as power level decreases--or two-phase friction multiplier decrease caused by decreased void fraction resulting from decreased core heat flux. (2.0)

REF.: Thermo L.P. Chap. 14.2 (p 14.5, 14.6) and recirc. System Flow Control LP NT/R291/1.16

5-8 Any four (4) of the below for full credit: (2.0)

Heat gains	Heat losses
Reactor core thermal energy	Steam flow
CRD flow	Vessel heat losses to drywell
Recirc pump energy	RWCU (system net loss)
Feedwater flow	

REF.: Fermi Heat Transfer L.P. Chapter 9, Sec. 9.3 (Heat Balance) [and facility question 7, p. 9.15]

5-9 The valve flow setting (appx 9000 gpm) is too much for one pump and would cause the pump to "run out". (1.0)

REF.: Procedure 23.102 Rev. 3, p2 and p7

5-10 TRUE (0.5)

REF.: Fermi Thermo LP, Chap. 16 (PUMPS), p. 16.6

FALSE per facility supplied alternate reference

(continued on next page)

5-11 FALSE (0.5)

REF.: Fermi Core and fuel LP NT/R181/6.6

5-12 See attached graph

5-13 F1 = APLHGR L1 = MCPR (1/3 pts. each)
F2 = MCPR L2 = LHGR
F3 = LHGR L3 = APLHGR

REF.: Fermi Thermo L.P. Chap. 10 Reactor Thermal Limits

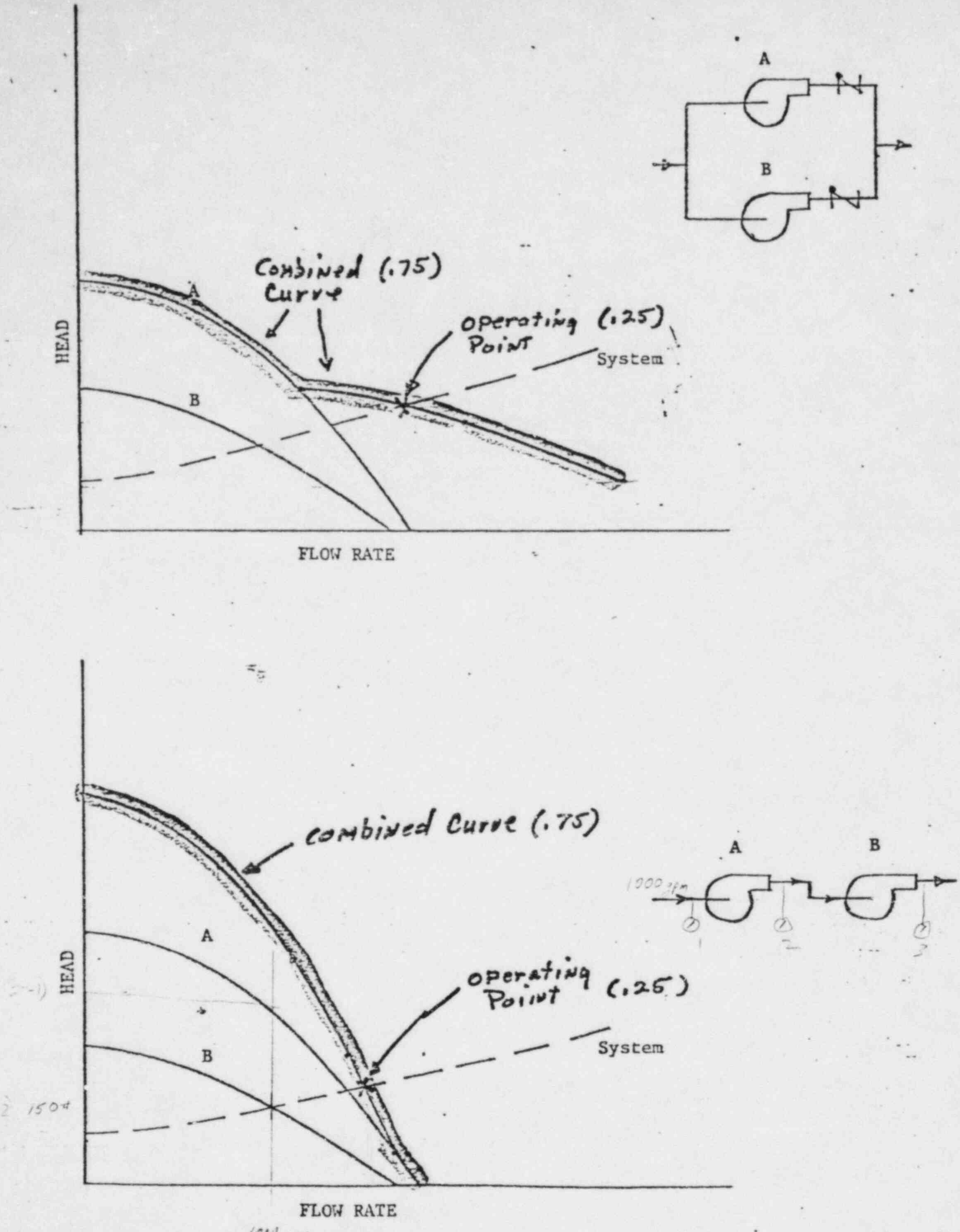
A
B
Combined (.75)
Curve
Operating (.25)
Point
System
A
B
HEAD
FLOW RATE
Combined Curve (.75)
A
B
Operating (.25)
Point
System
A
B
HEAD
FLOW RATE

FIGURE 5.1

Ref for Question 5-12

Copied from Fermi Fluid Mechanics LP

Chapter 16 (Page 16.20 text.)

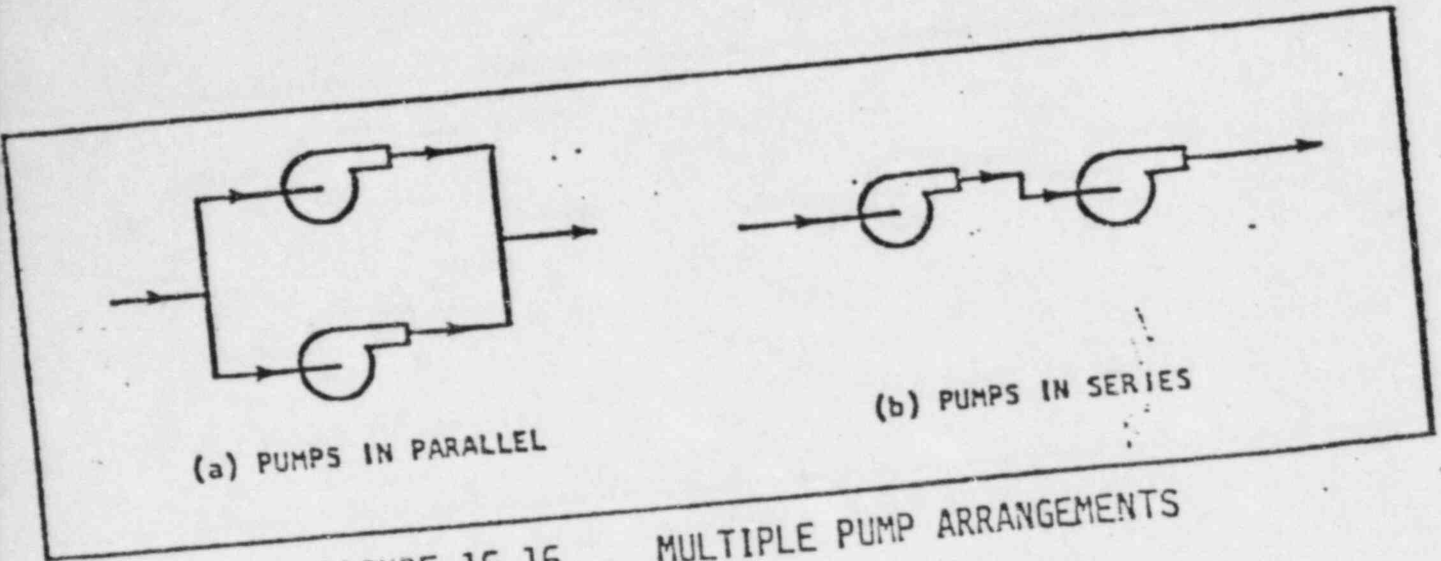

FIGURE 16.16 MULTIPLE PUMP ARRANGEMENTS

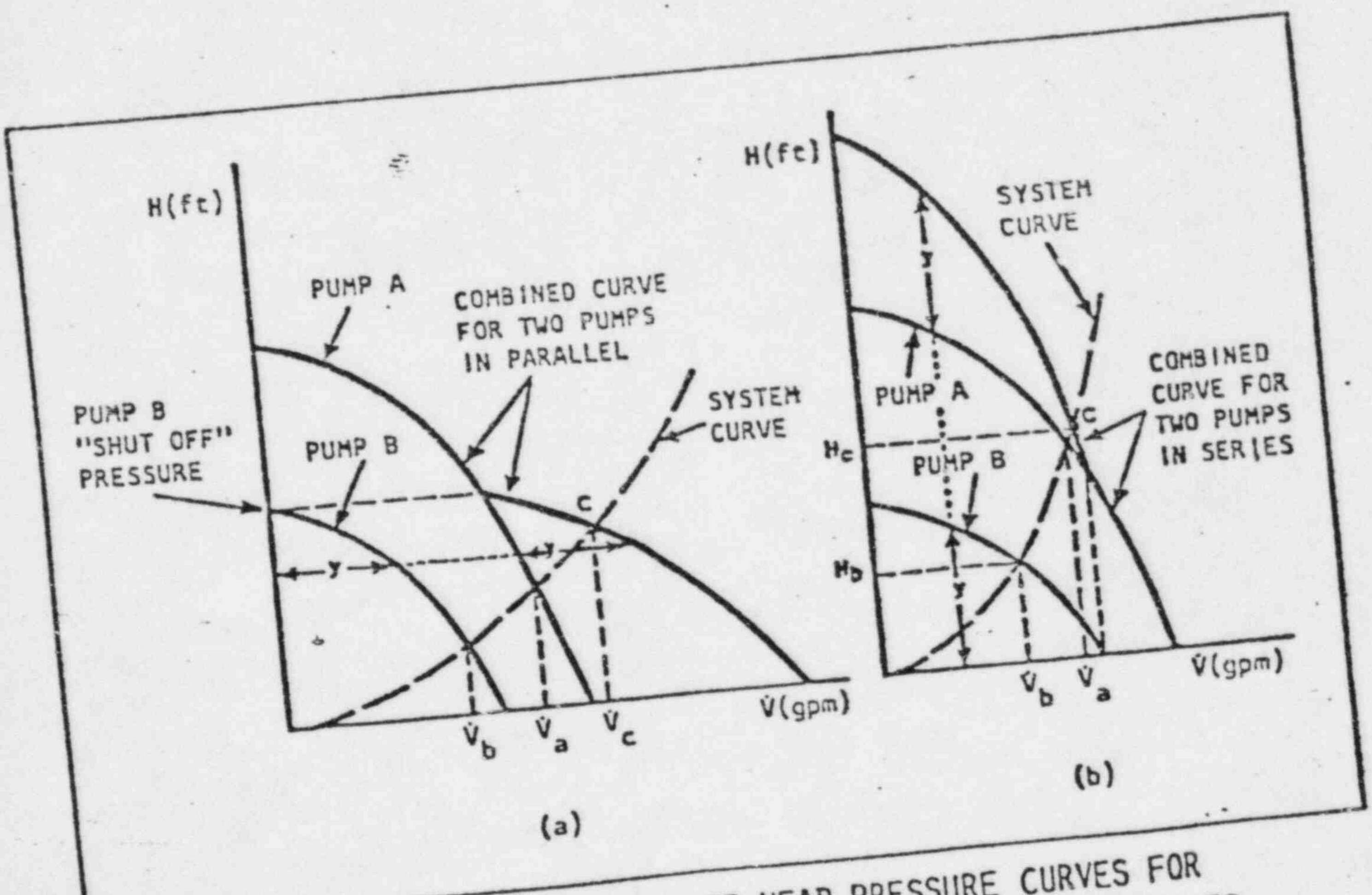

FIGURE 16.17 COMBINED HEAD PRESSURE CURVES FOR (a) TWO PUMPS IN PARALLEL AND (b) TWO PUMPS IN SERIES

6-1 a. 1. HPCI will not initiate. A closed F021 interlocks closed the steam inlet valve (F001). (0.5)

2. HPCI will auto-start and take suction from suppression pool. CST suction valve (F004) and both pump test return valves (F008 and F011) will be interlocked closed. (0.5)

3. HPCI will not initiate. The closed bypass valve (F600) prevents the outboard isolation valve (F003) from opening and supplying steam. (0.5)

b. 1. Suction of HPCI booster pump (0.5)

2. Clean radwaste (0.5)

REF.: HPCI LP NT/R287/1.18 to 1.23

6-2 a. Manual, using joystick. Less than 30% reactor power is sensed. An edge rod is selected. (0.5/ea.)

b. Inop occurs at <50% of the 6 (total) LPRM's inputs to the averaging circuit--hence, requires 3, or more, per channel. (0.5)

c. Channel A. (0.5)

d. Will cause the alternate APRM to be automatically selected. (0.5)

REF.: Rod Block Monitor L.P. NT/R283/1.10, /1.9. /1.13 and / 1.17

6-3 a. Links installed: NMS for RPS is "on-out-of-two, twice" logic and SRM Hi-Hi scram is bypassed. (1.0)

Links removed: Non-coincidence" logic, i.e., any one of 4 SRM's or any one of 8 IRM's or any one of 6 APRM's (which are operable) can insegt a trip when their respective trip setpoints are exceeded. (1.0)

b. During initial fuel loading, and during refueling operations when rods are withdrawn for shutdown margin demonstration. (1.0)

REF.: Intermediate Range Monitoring System L.P. NT/R275/2.11/2.12
Source Range Monitoring System L.P. NT/R213/3.17

(continued on next page)

6-4 a. Loss of power (undervoltage) to associated ESF bus. (0.5/ea.)
Low reactor water vessel level (Level 1)----Hi drywell pressure (1.63 psig) (setpoints not required).

b. FALSE (0.5)

c. #2, #5, and #6 (1/3-ea.)

REF,: Emergency Diesel Generator LP NT/R3v4/1.37, 1.66-1.68 (and confirmed by procedure 23.307 p. 3 & enclosure 2).

6-5 LIMITER #1--30% pump speed setpoint. Bypassed when recirc. pump discharge valve is fully open and total feedwater flow >20%. (1.0)

LIMITER #2--45% pump speed setpoint. Bypassed when both reactor feed pumps are running or reactor water level is above level 4 (192.5" TAF). (1.0)

LIMITER #3--80% pump speed setpoint. Bypassed if either or both check valves in the #5 heater drains to the flash tank, are not closed. (1.0)

REF.: Recirculation flow control system L.P. NT/R291/1.14/1.8

6-6 a. It goes into "Hold" at 30 psid. (Alarm is at 25-not required) (0.5)

b. 1----High area (room) temperature (Hx Room)
2----High differential temperature (air duct inlet/outlet) Hx Room. (0.5/ea.)
3----High differential flow after 60 sec. delay (i.e., flow downstream regen Hx plus blowdown flow minus pump discharge flow)

c. 1----Low pressure (5 psig) upstream of the valve (0.5/ea.)
2----High pressure (140 psig) downstream of valve

REF.: Reactor Water Cleanup System LP NT/R256/5.5,5.8,5.9, 5.13

(continued on next page)

6-7 a. Inboard Isolation Valve F007 (0.5)

b. 1. F010 opens. F029 and F031 remain closed. (0.5)
2. Their training material for valve interlocks indicate F010 is open and F029 and F031 are each closed. The printed matter says their auto swap occurs on low CST level but not Hi Torus level. System modifications at some plants do auto swap on torus Hi level, just like HPCI. ~~CAF~~ No auto-swap at Fermi II (0.5)

c. Yes. RCIC steam supply valve close on the Hi level trip instead of the Trip/Throttle valve. The trip will auto-reset and the steam supply valve auto-reopen at the initiation setpoint (Level 2) to allow restart of the turbine. (1.0)

REF.: RCIC L.P. NT/288/1.9. 1.20, 1.21, and 1.16

6-8 The 64T and 65T breakers (Maintenance bus ties) are used to establish a tie between Division I and Division II 4160 VAC busses. Interlocks prevent having the normal feed and the maintenance tie feed to a bus closed at the same time. (1.0)

REF.: Fermi Facility question & answer

6-9 (0.5)
1. HPCI (5000 gpm)
4. One CS (3175 gpm)
3. Both (2) Standby Feedwater Pumps (1300 gpm)
2. RCIC (625 gpm)

NOTE: Capacities are for ref. only not required for full credit

REF.: RCIC (NT/R288/1.5) • HPCI (NT/R287/1.5)
SBFW (NT/R257/1.2) • CS (NT/R291/2.6)

6-10 Normal Normal [Fermi Has 185 CR's – NT/R256/3.13] (0.5/ea

REF.: Rod Sequence Control System L.P. p. 2&3

6-11 Any five (5) of the 10 underlined items below for full credit. (0.5/ea.

a. Startup RHR pumps
b. Stop running RHR Service Water Pumps
c. Close RHR Hx service water outlet valves
d. Open RHR Hx bypass valve if closed
e. Close containment spray valves and full test valves (if open)
f. Actuate break detection circuitry to close recirc. pump discharge valve in broken loop, Trip recirc. pumps if running, interlock closed LPCI outboard and inboard injection valves to broken loop, open inboard and outboard injection valves to undamaged loop when the permissive pressure is received.

REF.: L.P. NT/R294/1.19-1.22

7-1 a. Receipt of a full scram and:

1) Sustained APRM indication ≥6% or (0.5)

2) More than one control rod does not insert to position 04 or less or (0.5)

3) Rx power cannot be determined (0.5)

b. Sustained Rx power ≥6% and torus water temperature reaches ~~100°F~~ 110 (0.67) or:

Anytime Rx power cannot be determined (0.33)

REF.: Procedure 29.000.08 Rev. 0 p. 1,2,3

7-2 a. 1-Place the reactor mode switch to SHUTDOWN (0.5/ea
2-Depress both scram pushbuttons
3-Arm and depress main turbine trip pushbutton

b. Scram the Rx from the relay room by taking one operable APRM's Mode Switch out of operate in Div. I (A,C,E) and one in Div. II (B,D,F). (NOTE: Lesson plan NT/R292/6.5 also include opening output breakers on feeders from RPS bus A and Bus B as a "backup means of scramming Rx"). (0.5)

A B

c. Turbine pressure regulator controlling the valves before MSIV closure. Operating the SRV's (F013M and N) from Shutdown panel (or allowing the valves to cycle) after MSIV closure. (1.0)

REF.: Procedure 20.000.19 pages 1-3 and Remote Shutdown System LP p. 5

7-3 When the SRV cannot be closed within 2 minutes or the torus temperature increases ≥95°F. (1.0)

REF.: Abnormal procedure 20.000.25, p.1 (Failed Open Safety relief valve)

(continued on next page)

7-4 Any five (5) below for full credit (2.5)

1) High radiation area (whole body dose > 100 MREM/hr)
2) For work being done in barriered and/or posted radiation or high radiation areas
3) Areas where reactor produced neutron radiation may exist in excess of 2 MREM/hr.
4) Areas with removable contamination > 10,000 DPM/100 centimeter square
5) Areas with airborne activity affecting work times or requiring the use of respiratory protection.
6) For work being done on contaminated equipment within a restricted area when determined necessary by health physics
7) For use of calibration sources when determined by health physics.

REF.: Radiation Work Permit procedure 12.000.13 (NOTE: Answer from Facility Question Bank

7-5 (0.5/ea.

	Short Term		Long Term
a.	~~decrease~~ Increase		Decrease
b.	increase	or	Decrease
c.	decrease		Increase
d.	increase		Increase
e.	open		Open

REF.: Reactor pressure controller failure (procedure 20.109.02 page 2--A3.0 Indications)

7-6 Core alterations shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. (1.5)

(Any 7 for full credit)

REF.: Refueling floor conduct of operations 12.000.46 and Tech. Specs. definition

(continued on next page)

7-7 a. Operations Engineer or his designee (22.000.01, p.1) (0.5)

b. All APRM's between 5%-10% and all APRM downscale alarms are cleared. Reactor pressure >850 psig and all main steam line low pressure alarms are cleared. (22.000.03, p.11) (1.0)

c. 3rd condensate pump, 3rd heater feedwater pump and 2nd reactor feedwater pump (22.000.03 p.16) (1/3/ea.)

d. 10 to 50 psig above reactor pressure to ensure the turbine bypass valves remain closed as not to affect reactor pressure, temperature, or power. (22.000.03, p. 9&10) (1.0)

e. Oxygen concentration <4% by volume within 24 hours after thermal power is >15% of rated (22.000.03, p.11; TS 3.6.6.3) (1.0)

REF.: Source procedure and page no ( ) above.

7-8 a. Section A--Fuel Cladding Failure (Small) (1.5)

Immediate Operator Actions

1 Notify the Nuclear Shift Supervisor(exclude from grading) of the event, actions taken, and that it may be required to classify the event in accordance with the Emergency Plan Implementing Procedure EP-101, "Classification of Emergencies."

2 Decrease load to maintain off-gas radiation within operating limits.

3 If the load decrease fails to reduce the gaseous release rate below the Technical Specification Limits, manually scram Reactor and perform Abnormal Operating Procedure #20.000.21, "Reactor Scram", concurrently with this procedure.

b. Section B--Fuel Cladding Failure (Gross) (1.5)

Immediate Operator Actions

1. Verify the following automatic actions have occurred:

1.1 Group I Isolation (MSIV's and drains)

1.2 Group 2 Isolation (Reactor water sample system).

1.3 Mechanical vacuum pumps trip, if running.

1.4 Reactor scram

(continued on next page)

2. Perform Abnormal Operating Procedure #20.000.21 "Reactor Scram" concurrently with this procedure.

3. Notify the Nuclear Shift Supervisor (exclude from grading) of the event, actions taken, and that it may be required to classify the event in accordance with the Emergency Plan Implementing Procedure EP-101 "Classification of Emergencies".

REF.: Fuel Cladding failure 20.000.07 Rev. 2,p.1

7-9 Within one (1) hour (0.5) establish a continuous fire watch (0.5) with backup fire suppression equipment (0.5) for the affected DG's.

REF.: Tech. Specs. 3.7.7.3 p. 3/4 7-26
Fire Protection System NT/R205/1.7

7-10 5% of rated (0.5)
The potential exists for forming an explosive hydrogen--oxygen mixture in the Mechanical Vac Pump and its discharge piping (1.0)

REF.: Op Procedure 23.125 p. 5 R2

7-11 Main steam lines isolated (0.5/ea.)
Reactor critical
Reactor pressure between 600 and 650 psig

REF.: Op Procedure 22.000.04 (Rev. 0)

8-1 1- Backflow through idle or reduced flow jet pumps can cause jet pump vibration and therefore unnecessary stress on jet pump components. (1.0)

2- To avoid the possibility of "fooling" LPCI loop selection logic (1.0)

3- Limiting mismatch provides some coastdown flow from the unbroken loop during a line break accident. (1.0)

REF.: Reactor Recirculation System L.P. NT/R258/1.8 (also Tech. Specs. basis)

8-2 a. 1) A knowledgeable member of the plant staff & (0.5/ea.)
2) A holder of an SRO license

b. 1- FALSE. These changes are initiated on Major Change requests. (0.5)

2- TRUE. (0.5)

3- TRUE. (Corrections are marked on a copy of the procedure and submitted to OSRO clerk for change and disposition-reason not required.) (0.5)

REF.: Plant Operations Manual Procedures (12.000.07, REv.9) Pages attachments 5, p.1 of 7; attachment 6 p.1 of 6; attachments 4, p.1 of 1

8-3 1- Thermal power shall not exceed 25% of rated thermal power with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow. (1.0)

2- The minimum critical power ratio shall not be less than 1.06 with the reactor steam dome pressure greater than 785 psig and the core flow greater than 10% of rated flow. (1.0)

3- The reactor coolant system pressure as measured in the reactor vessel steam dome, shall not exceed 1325 psig. (1.0)

4- The reactor vessel water level shall be above the top of the active irradiated fuel. (1.0)

REF.: Tech. Specs. 2.1.1, 2.1.2, 2.1.3, and 2.1.4

(continued on next page)

(Any 5 @ 0.5/ea.)

8-4 1) Time
2) Rod sequence
3) Rod group
4) Rod numbers
5) Rod position
6) Reactor coolant temperature
7) Reactor period (calculated)

REF.: OP. Procedure 22.000.03 Rev. 0

8-5 Requires a second licensed operator or other technically qualified member of the unit technical staff be present at the reactor control console (H11-P603) and verify compliance with the required rod sequence checkoff list. (i.e., serve as a human RWM). (1.5)

REF.: RWM Procedure 23.608 page 4 and Tech. Specs. 3.1.4.1

8-6 a. 1- The operators red tag is used to warn against the operation of electrical or mechnical equipment which could injure personnel. (0.75)

2- Warns against use of equipment that could be in an unsafe conditions (e.g., fire extinguisher, tools and ladders). (0.75)

b. 1- Energized electrical equipment - Not used for any purpose other than excluding people from approaching energized electrical equipment. (0.75)

2- Radiation and contamination hazards - Warns and excludes personnel from approaching sources of direct radiation and surface contamination. (0.75)

REF.: Tagging and Protective Barrier System — procedure 12.000.12 (R6)

8-7 Nuclear Supervising Operator (NSO) as brigade leader and four (4) additional members--excludes four (4) members of the min. shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency. (1.5) (1.0)

REF.: Shift Operators and Control Room Operators (21.000.01. p. 6)

(continued on next page)

8-8 1) Maximum allowable extension not to exceed 25% of the surveillance interval. But, 2) the combined time interval for any three (3) consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval. (2.0)

[Example: For a daily (24 hr.) surveillance test, such as SBLC solution temperature and volume, the current interval may be extended 1) for six (6) hours provided 2) the combined delay for the current plus two preceding intervals will not be exceeded by more than 6 hours total.]

REF.: Tech. Specs. 4.0.2 and 4.0.3

8-9 Verify at least one (1) CRD pump is operating (1.0). Verify by inserting at least one withdrawn control rod (1.0) at least one notch (0.5)

REF. T.S. 3.1. 3.5. a. 2a

8-10 If it is part of the training to qualify for an operator license and it is under the direction and in the presence of a licensed operator or senior operator. (1.0)

REF.: 10CFR55, ppg. 55 9b

8-11 FALSE (0.5)

REF.: 20.000.12 p. 1