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

EXAMINATION REPORT NO. 50-470/84-6

FACILITY DOCKET NO. 50-470

LICENSEE: Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

FACILITY: Combustion Engineering Training Center

DATES: June 17-18, 1984

CHIEF EXAMINER: Original Signed By: JUL 1 5 1984

Noel F. Dudley — Date

APPROVED BY: Original Signed By: JUL 2 6 1984

Chief, Project Section 1D — Date

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PDR ADOCK 05000470
Q PDR

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07/10/84

REPORT DETAILS

TYPE OF EXAMS: Initial ____ Replacement _X_ Requalification ____

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail	Inst. Cert Pass/Fail	Fuel Handler Pass/Fail
Written Exam	/	/	2/1	/
Oral Exam	/	/	1/1	/
Simulator Exam	/	/	1/1	/
Overall	/	/	2/2	/

1. CHIEF EXAMINER AT SITE: N. Dudley, NRC
2. OTHER EXAMINERS: R. Keller, NRC
3. PERSONS EXAMINED

Dieli, Armand G.	Instructor Certification
Finnerty, Wayne M.	Instructor Certification
Sundal, Harold W.	Instructor Certification
Webber, Ronald C.	Instructor Certification

1. Summary of generic strengths or deficiencies noted on oral exams:

   None

2. Summary of generic strengths or deficiencies noted from grading of written exams:

   Weaknesses identified in a previous examination had not been corrected. Candidates answered correctly 86% to 44% of the questions repeated from a previously administered examination.

   Technical Specification 3.2.2.2.1 is incorrect and should read:

   FTxy = Fxy(1+Tq) vice FTxy = Fxy(1-Tq).

3. Comments on availability and candidate familiarization with plant reference material:

   None

4. Comments on availability and candidate familiarization with plant design, procedure, T. S. changes and LERs:

   None

5. Comments on interface effectiveness with plant training staff and plant operations staff during exam period.

   Staff was very effective in implementing planned transients on the simulator and interfacing with candidates. One instructor was used as an operator and provided the correct amount of support to the crew. His proficiency as the secondary operator increased the difficulty of tripping the plant on a transient.

6. Improvements noted in training programs as a result of prior operator licensing examinations/suggestions, etc:

   Prints were available in the main control room. A new administrative procedure for updating and tracking the applicability of operating procedures has been drafted and will adequately insure proper maintenance of main control room procedures. (#50-470/84-05-01 closed)

7. Personnel Present at Exit Meeting:

NRC Personnel

N. Dudley
R. Keller

Facility Personnel

W. Burchill
D. Foley
W. Souder

8. Summary of NRC Comments made at exit interview:

The name of the clear pass on the operating examination was given.

The draft procedure for insuring applicability of main control room procedures was discussed.

The following problems with the simulator were noted:

1. Tc on Loop 12 read high during natural circulation.

2. No indication of a lifted relief valve on the HPI Header was observed.

3. Annunciator audibles are not consistent with actuation and clearing of alarms.

The efforts of the simulator staff in improving procedures was noted and additional areas needing improvement were discussed.

9. Summary of facility comments and commitments made at exit interview:

Facility will implement the draft administrative procedure for controlling operating procedures in the near future.

10. CHANGES MADE TO WRITTEN EXAM

Question No.	Change	Reason
5.03.b (answer)	Change	Changed to "remains the same" since pressurizer level program does not begin to increase until 15% power.
5.04 (answer)	Change	Figure 14 change to reflect the fact that water/fuel ratio does not change substantially over core life.
5.05 (answer)	No Change	"Effects generator output, system voltage" was not included in answer key since answers will be evaluated as to candidates understanding of reactive load.
5.08	No Change	Candidates were verbally informed that emergency buses were available.
6.02a. (answer)	Addition	Accept also "overpressurization of RCS by Design", which is the purpose of pressurizer safety valves.
6.05a. (answer)	Addition	Accept also "in cores" since incore nuclear instruments (NI) do provide indication of nuclear power.
6.06a. (answer)	Addition	Accept also "APD-ASI indication", since a failed NI detector will cause skewed ASI indications.
	Change	Delete word "light" from "Level 1 bistable energized" since level 1 bistable effects trip signals in addition to energizing a light.
6.06b. (answer)	Addition	Accept also "Tave-Tref mismatch" since failed NI channel will effect Tave reading.
6.10c. (answer)	Change	Changed to "SW needed to provide cooling to equipment (auxiliary building or containment air cooler) during plant operations" to reflect Technical Specifications Design Basis for Service Water System.

Question No.	Change	Reason
7.1(4) (answer)	Change	Correct answer is D in accordance with EOP 7D figure 5.21.
7.08 (Question)	No Change	Question is applicable since it is allowed by procedure at the simulator to use automatic rod control, even though most operating CE plants do not allow operations in automatic rod control.
7.10b (Question)	No Change	Question is applicable to SRO level of knowledge. It requires knowledge of abnormal system alignments during emergency situations which are allowed by procedures.
8.2 (Answer)	Addition	Accept also "does not address unreviewed safety question", in accordance with AP-10 Page 5.
8.03 (Answer)	Change	Delete "and then lower the fuel" since interpretation of AP-12 Page 3 may allow fuel load to be completed, or left in the fuel handling machine.
8.06 (Answer)	Change	Corrected answer to reflect FTxy out of limits.
8.10b (Answer)	Change	Corrected to reflect fact Technical Specifications requires operations of two trains of HPI.
8.10c (Answer)	Change	Technical Specification general statement 3.0.3 is acceptable in place of detailed shutdown requirements.

Attachment:

Written Examination and Answer Key (SRO)

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

FACILITY: C-E SIMULATOR

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 84/06/18

EXAMINER: DUDLEY, N.

APPLICANT: ________________________

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	PETE DELARCO	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	R. PRICE	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	W. SONDER	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	D. FOLEY	________	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)

Choose the correct response to each of the following. (2.0)

1. Moderator temperature coefficient becomes more negative from BOL to EOL primarily because of:
   A. The larger change in resonant escape probability per degree change in moderator temperature.
   B. The larger change in core leakage per degree change in moderator temperature.
   C. More thermal neutrons are available for absorption in the moderator.
   D. The smaller change in thermal utilization factor per degree change in moderator temperature.

2. Doppler coefficien ($\Delta\rho$ /degree F fuel) becomes more negative from BOL to EOL because of.
   A. An increase in effective fuel temperature.
   B. Clad creep and fuel pellet swell.
   C. The production of plutonium-240.
   D. The overlapping of resonant peaks.

3. Control rod worth is greatest.
   A. At higher boron concentrations
   B. At higher moderator temperatures
   C. At low boron concentrations
   D. At iower moderator temperatures

4. Beta effective decreases over core life because:
   A. The fraction of neutrons which are born delayed as a result of fission of U235 changes as the fuel is depleted.
   B. The resonance absorption region of U238 broadens as the core ages.
   C. The response of the reactor to changes in reactivity becomes faster as the core ages.
   D. The percentage of fissionable isotopes vary as the core ages.

QUESTION 5.02 (1.50)

After criticality is achieved, the startup procedure requires the operator to level power at 10 -4% power to record critical data. The operator establishes a ZERO DPM SUR and verifies that the wide indication is steady with no rod motion. One minute later, the operator notices that power is increasing.
Explain why?

QUESTION 5.03 (2.00)

Would reaching the point of adding heat INCREASE, DECREASE, or have NO EFFECT on each of the following parameters if a constant reactivity addition continued? (assume all controls in automatic)

a. Start Up Rate (SUR)
b. Pressurizer level
c. T cold
d. Turbine bypass controller output

QUESTION 5.04 (1.50)

Indicate on the attached Keff vs Water to Fuel Ratio figure the point where the reactor would be operating at (A) the beginning of life, and (B) the end of life. What is the major reason for the operating point to vary with core age?

QUESTION 5.05 (1.50)

a. When the turbine generator is loaded, how is reactive load (KVARS) measured, and how does the amount of reactive load effect generator operations? (1.0)

b. How is reactance adjusted? (0.5)

QUESTION 5.06 (1.50)

Explain how an air ejector reduces the pressure in the main condenser. (1.5)

QUESTION 5.07 (2.00)

a. Describe how the cooldown rate would be controlled while on natural circulation. (1.0)

b. HOW and WHY would delta T and flow rate observed during natural circulation compare 10 minutes and 1 hour after a trip from 100% power accompanied by a loss of all Reactor Coolant Pumps. (1.0)

K_{EFF} VS WATER TO FUEL RATIO

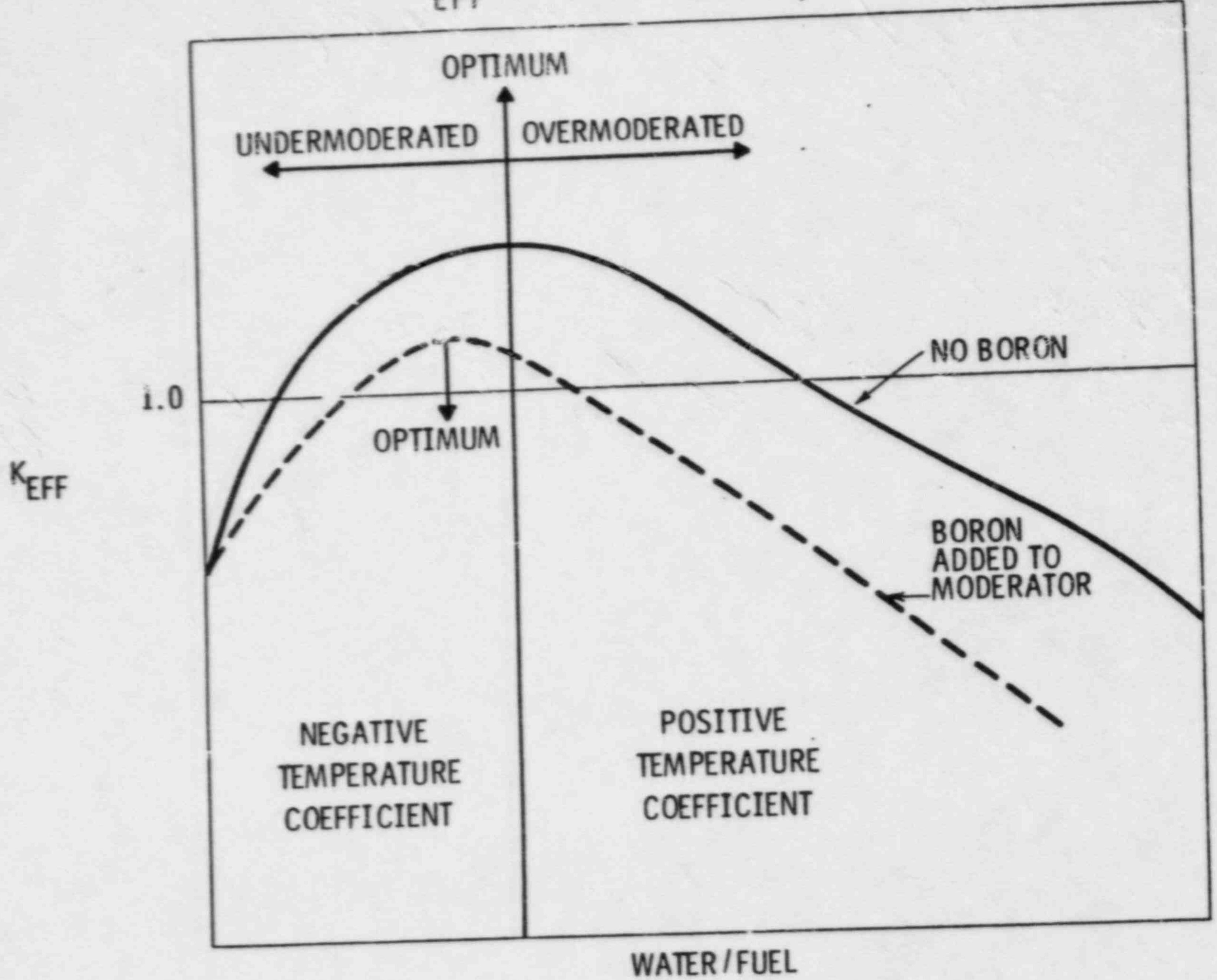


Figure 14

QUESTION 5.08 (2.00)

If all site power was lost and pressurizer heaters could not be energized, could the plant be maintained greater than 40 degrees subcooled if all equipment operated properly? Justify your answer.

QUESTION 5.09 (2.00)

Figure 1-1 depicts a closed loop system with two parallel centrifugal pumps and four parallel heat exchangers. Initially, both pumps are running and all four(4) heat exchangers are in service. What would the flow rate be through the #2 heat exchanger if the second pump is stopped (#1, 2 and 3 heat exchangers and pumps are identical). Show all work.

QUESTION 5.10 (3.00)

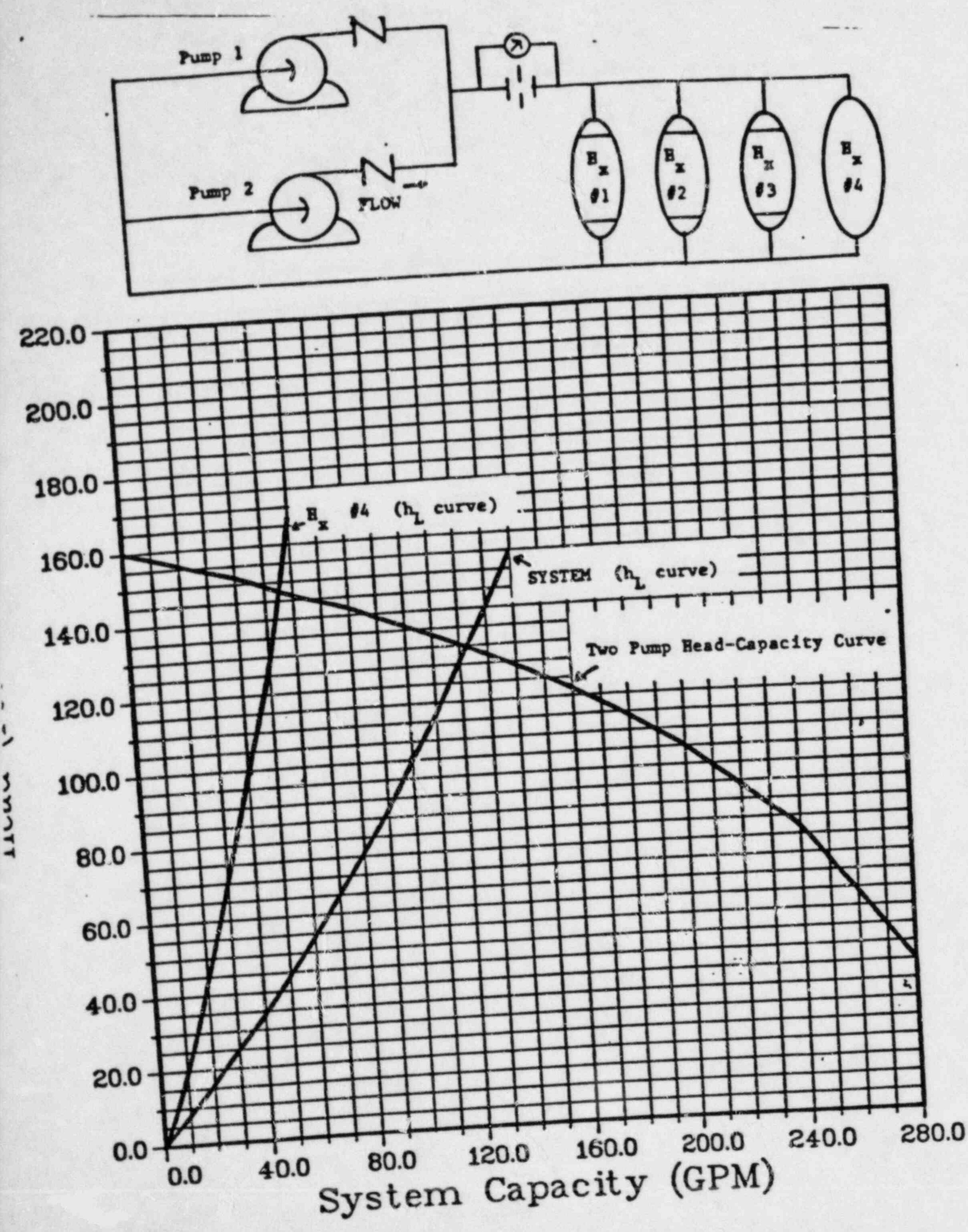
Would fuel center line temperature INCREASE, DECREASE, or REMAIN THE SAME in each of the following situations? Briefly explain.

a. Power decreases with constant Tave.
b. Tave increases with constant power.
c. Core age increases with constant power
d. Pressurizer pressure increases with constant power.

QUESTION 5.11 (3.00)

a. With an initial count rate of 20 cps, reactivity is added to increase the count rate to 40 cps. What would be the effect of adding the same amount of reactivity again? Explain your answer. (1.5)

b. With an initial power level of 10%, reactivity is added to raise power to 20%. What would be the effect of adding the same amount of reactivity again? Explain your answer. (1.5)


Pump 1
Pump 2
FLOW
Hx #1
Hx #2
Hx #3
Hx #4
Hx #4 (hL curve)
SYSTEM (hL curve)
Two Pump Head-Capacity Curve
220.0
200.0
180.0
160.0
140.0
120.0
100.0
80.0
60.0
40.0
20.0
0.0
0.0
40.0
80.0
120.0
160.0
200.0
240.0
280.0
System Capacity (GPM)

QUESTION 5.12 (3.00)

If at BOL power is reduced from 100% to 50% and stabilized, breifly explain HOW and WHY each of the following plant parameters will be affected over the next 5 hours. Assume all systems are in automatic, rod control is in manual sequential, and no operator action is taken.

a. RCS temperature
b. RCS pressure
c. S/G pressure
d. Turbine Generator control valve position
e. Feed flow

QUESTION 6.01 (2.50)

What effect will an SIAS have on each of the following components?

a. SALT ~~Sea~~ Water System outlet valves on the Component Cooling Water heat exchanger.
b. SALT ~~Sea~~ Water System outlet valves on the Service Water heat exchanger.
c. The three (3) High Pressure Safety Injection Pumps.
d. The three (3) Salt Water Pumps.
e. Service Water supply valves to the containment air coolers.

QUESTION 6.02 (2.50)

a. If the RCS pressure is 350 psia, RCS temperature is 340 F, and warming of the Shutdown Cooling System (SDC) has NOT started, HOW and WHY is the RCS protected against an overpressurization? (1.0)

b. Would resin damage occur if the SDC System was lined up to the CVCS purification ion exchangers and the RCS temperature was above 140 F? Assume any automatic protective features function correctly. EXPLAIN your answer. (1.5)

QUESTION 6.03 (2.00)

Answer the following questions in terms of system components, and flow paths to be used. A detailed step by step procedural description is not required.

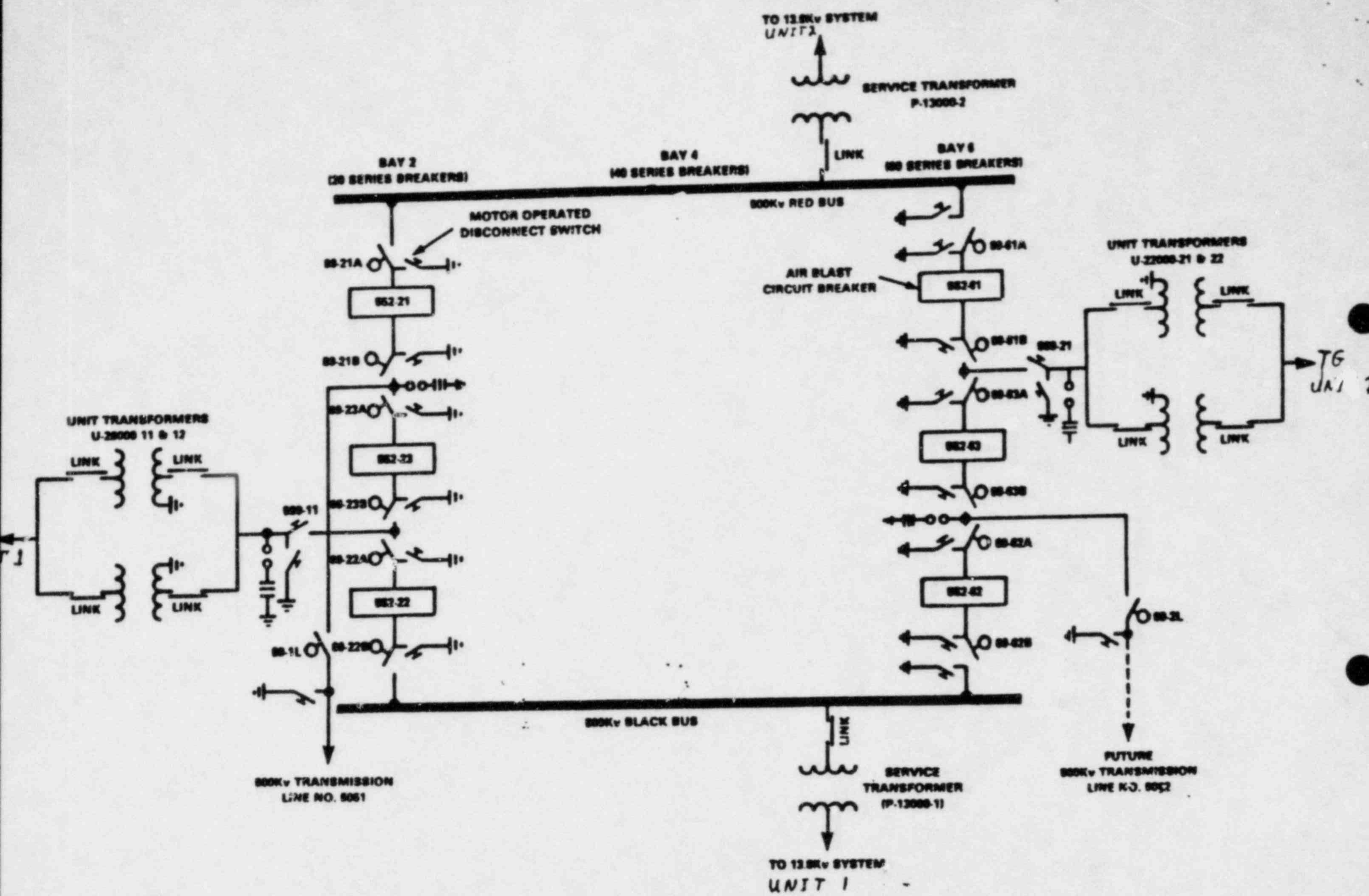
a. If level is increasing in a Safety Injection Tank (SIT) due to leakage from the RCS, how is level controlled? (0.7)

b. How would proper boron concentration be maintained in the SIT if RCS inleakage continued? (1.3)

QUESTION 6.04 (2.00)

a. Indicate on the figure of the ring bus the normal breaker lineup for Unit 1's main generator in operation and Unit 2's main generator secured.

b. Indicate on the figure of the ring bus the expected breaker lineup one minute after a reactor trip on Unit 1 if Unit 2's main generator is secured.


TO 13.8Kv SYSTEM
UNIT 2
SERVICE TRANSFORMER
P-13000-2
LINK
BAY 2
(20 SERIES BREAKERS)
BAY 4
(40 SERIES BREAKERS)
BAY 6
(60 SERIES BREAKERS)
500Kv RED BUS
MOTOR OPERATED
DISCONNECT SWITCH
AIR BLAST
CIRCUIT BREAKER
UNIT TRANSFORMERS
U-22000-21 & 22
LINK
TG
UNIT 2
UNIT TRANSFORMERS
U-26000 11 & 12
LINK
500Kv BLACK BUS
500Kv TRANSMISSION
LINE NO. 5051
SERVICE
TRANSFORMER
(P-13000-1)
FUTURE
500Kv TRANSMISSION
LINE NO. 5052
TO 13.8Kv SYSTEM
UNIT 1

QUESTION 6.05 (2.50)

What alternate instrumentation can be used to verify each of the following indications:

a. Nuclear excore instrumentation (TWO sets required) (1.0)
b. Subcooled margin meter (ONE set required) (0.5)
c. Tavg (TWO sets required) (1.0)

QUESTION 6.06 (2.50)

a. What FOUR alarms or indications would be received if the lower detector of a linear power safety channel failed low when reactor power was at 20%?

b. What FOUR alarms or indications would be received if the upper detector of a linear power safety channel failed high when reactor power was at 100%?

QUESTION 6.07 (3.00)

Will the plant trip as a result of the following situations. Explain your answers. Consider each situation separately. (FOUR figures are provided)

a. 120 volt vital bus A is deenergized and channel B pressurizer pressure indication fails high.
b. Loop 1 Tc channel A fails high and loop 2 Th channel B fails high while at 80% power.
c. The Flow Dependent Setpoint Selector Switch is placed in the 3 pump position while at 70% power and Axial Shape Index is -0.3.
d. SG 1 pressure channel A fails to 400 psia and SG 2 pressure channel B fails to 450 psia while at 60% power.

QUESTION 6.08 (2.00)

What FOUR ways do Engineered Safety Features function to mitigate the consequences of an incident and thereby enhance the protection of the public against the accidental release of fission products?

QUESTION 6.09 (3.00)

Explain how the plant would respond to a loss of both main feed pumps if all systems are in automatic and no manual action is taken. Follow the transient until the plant stabilizes. Include reactor, feedwater, turbine, steam generator, and pressurizer.

QUESTION 6.10 (3.00)

Why must each of the following components be operable prior to initiating shutdown cooling?

a. Low pressure safety injection pumps
b. Component cooling water pumps
c. Service water pumps
d. Both engineered safeguards buses

QUESTION 7.01 (2.00)

For each of the following questions choose the correct response.

1. Which RCS Inventory Control Success Path should be followed if subsequent to a reactor trip the RCS pressure is 1500 psia and pressurizer level is decreasing.

   A. Via Containment Sump Recirculation
   B. Via LPSI
   C. Via HPSI
   D. Via CVCS

2. Which Loss of Heat Removal Success Path should be followed if after a reactor trip from 100% power all RCP's are secured, RCS pressure is 1200 psia, both the Steam Bypass Control System and Feedwater Regulating System are operating properly, RCS temperature is 450 F, and pressurizer level is 150"?

   A. Normal RCS heat removal
   B. RCS heat removal via HPSI and SG
   C. RCS heat removal via once through cooling
   D. RCS heat removal via shutdown cooling

3. Which RCS Pressure Control Success Path should be followed if after a reactor trip and subsequent Safety Injection Actuation, RCS temperature is 532 F, RCS pressure is 2100 psia, SG pressure is 900 psia, and pressurizer level is 200"?

   A. Via heaters and sprays
   B. Via CVCS
   C. Via HPSI
   D. Via PORV(s)

4. Which Containment Integrity Success Path should be followed if 30 hours after a LOCA containment pressure is 3.5 psig, hydrogen concentration is .5%, and high radiation levels exist in the containment?

   A. RAS actuation and verification
   B. CIAS/CSAS actuation and verification
   C. Containment H2 Control actuation and verification
   D. CRS actuation and verification

AMOUNT OF LOAD CHANGE
vs
TIME TO MAKE LOAD CHANGE

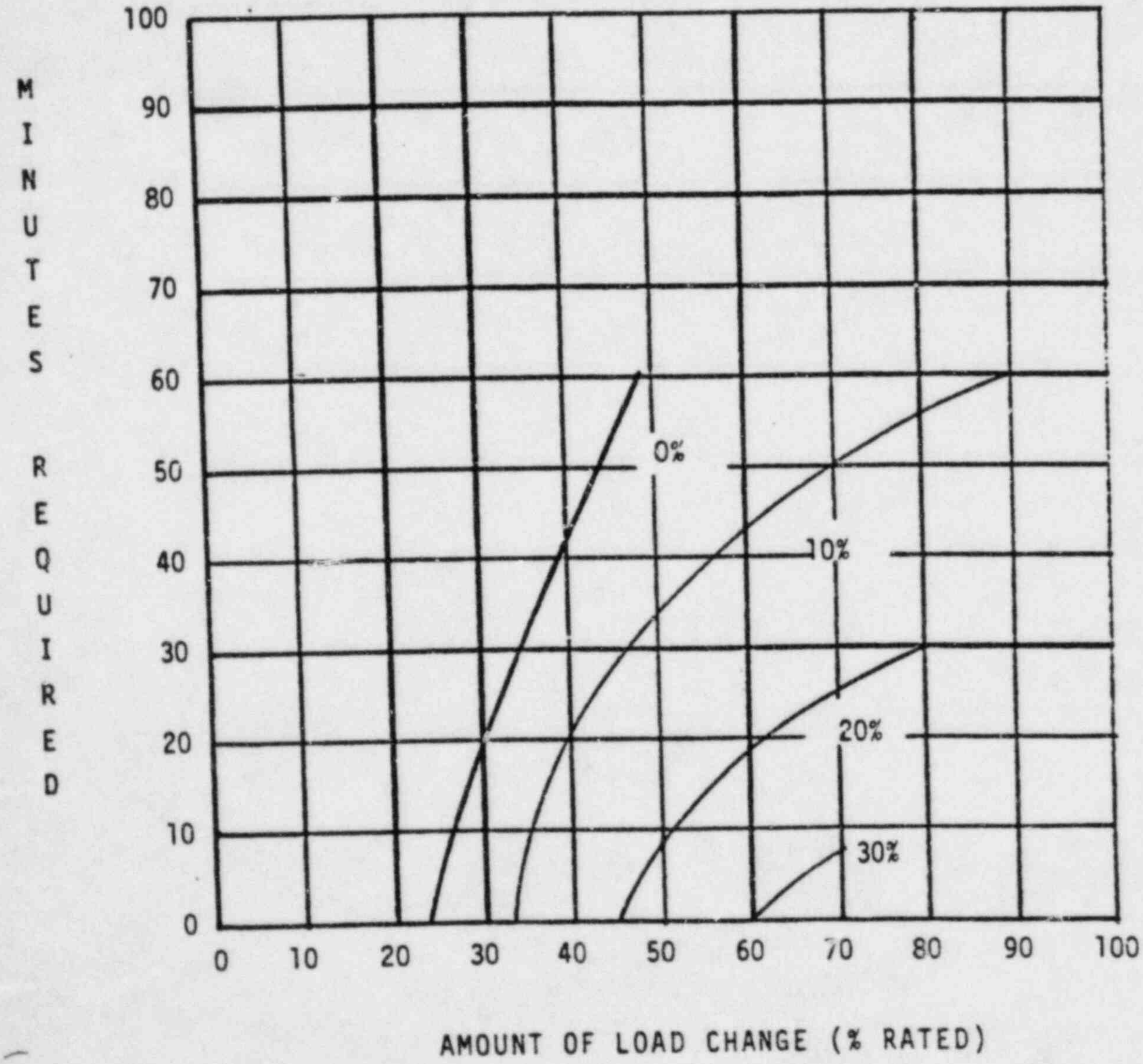


FIGURE B.6.1

QUESTION 7.02 (2.00)

a. What is the minimum time restriction placed on an increase in power from 10% to 60%? Figure B.6.1 is provided. (0.6)

b. What TWO Turbine Generator control circuits can be used to increase load and how do they differ? (1.4)

QUESTION 7.03 (2.00)

a. Why must the Steam Dump and Bypass Control System controllers be kept in AUTO during normal plant operations?

b. Why must the turbine bypass valves be verified shut if vacuum is lost to the condenser?

QUESTION 7.04 (2.50)

What actions should be taken by the control room operator if during refueling the Main Ventilation Stack radiation monitor alarms soon after an approved waste gas discharge is started?

QUESTION 7.05 (3.00)

What corrective actions should be taken in each of the following situations in accordance with Reactor Trip procedure EOP-1. Consider each situation seperately.

a. An anticipated transient without a scram (ATWS) has occurred. (2.0)

b. RCS pressure drops to 1500 psia following a reactor trip. (1.0)

QUESTION 7.06 (2.00)

Where should the following personnel report when a Site Area Emergency is declared?

a. Operations shift personnel
b. Shift Technical Advisors
c. Duty Officer
d. Director of Station Emergency Operations
e. Duty Health Physics Technicians

QUESTION 7.07 (3.00)

An entry into the containment is required while at 100% power and will result in an estimated whole body dose of 120 mrem. The following four candidates are equally qualified to perform the task. Which candidate may be allowed to perform the task in accordance with administrative procedures.

Explain your reasons for accepting of rejecting each candidate. No waivers can be obtained.

CANDIDATE	1	2	3	4
SEX	male	male	female	male
AGE	27	38	24	20
WK/EXPOSURE	200mrem	100mrem	0mrem	30mrem
QT/EXPOSURE	?	900mrem	20mrem	800mrem
ACCUM LIFE EXPOSURE	?	54000mrem	2200mrem	4000mrem
REMARKS	History unavailable	None	3 months pregnant	None

QUESTION 7.08 (2.50)

What are FIVE (of the seven) conditions which must be verified prior to selecting the Automatic Sequential Mode on the CEA Control System?

QUESTION 7.09 (3.00)

In accordance with EOP 7~~2~~A, Reactivity Control, what are the THREE reactivity control success paths and WHEN would each be used?

QUESTION 7.10 (3.00)

A Recirculation Actuation Signal (RAS) has occured following a large break LOCA.

a. What THREE automatic actions should have occurred? (1.8)

b. What manual actions should be taken to direct cooled water from the SDC Heat Exchangers to the HPSI pumps? (1.2)

QUESTION 8.01 (1.00)

What THREE items must the Senior Control Room Operator document in his log upon termination of a liquid or gaseous radioactive waste release?

QUESTION 8.02 (2.00)

What THREE requirements must be met to allow a Shift Supervisor to make a temporary change to a procedure?

QUESTION 8.03 (2.00)

What actions, if any, should the Operating Shift Supervisor take if the plant is in Mode 6, fuel loading is in progress, and the Refueling Senior Reactor Operator reports, "We have just lost the audio count rate out here in the containment, but I'm in position to load this fuel bundle, so I'll lower it down to prevent leaving a suspended load over the core." Justify your answer.

QUESTION 8.04 (2.50)

Indicate by position and quantity the minimum number of persons required by Technical Specifications during a plant heatup?

QUESTION 8.05 (2.00)

During the mid shift the four Pressure Channels read 2240 psia 2250 psia, 2350 psia, and 2260 psia respectively and all track with pressure changes. What actions, if any, should the Shift Supervisor take if no instrument technician is available on shift.

QUESTION 8.06 (3.00)

What actions, if any, should be taken if the plant is at 95% power with all rods withdrawn and the computer printout indicates:

Axial Shape Index	~~+1.3~~ +.13
Radial Peaking Factor (Fr)	1.5
Azimuthal Power Tilt (Tq)	0.04
Planar Radial Peaking Factor (Fxy)	1.65
Maximum allowable fraction of rated thermal power (N) presently in use	1.0

Operation at this power level is desirable if at all possible. JUSTIFY your decision. Technical Specifications 3.2.1 through 3.2.4 are provided.

QUESTION 8.07 (2.50)

If while in mode 6 with the reactor vessel head removed and core alterations in progress, the reactor operator discovers that the boron concentration of the primary system is 1600 ppm.

a. As the shift supervisor what are your actions as outlined in the Tech. Specs.? (2.0)

b. How soon do you have to perform these actions? (0.5)

QUESTION 8.08 (2.50)

a. If a Safety Limit has been violated, what THREE actions must be performed? (1.5)

b. What are the TWO Safety Limits? (1.0)

QUESTION 8.09 (3.00)

For each of the following events explain briefly why the NRC SHOULD or SHOULD NOT be notified within 1 hr.

a. During instrument testing while at power, three pressurizer pressure safety channels are momentarily place in bypass.

b. While at power, Tave momentarily dips to 510 F and then returns to normal.

c. Refueling water tank level falls below 400,000 gallons and cannot be restored.

d. During surveillance testing an expected actuation of LPIS train A occurs.

QUESTION 8.10 (4.50)

For each of the following situations indicate what REQUIREMENT, if any, applies and what ACTION, if any, should be taken. Consider each situation seperately.

a. Diesel generator A's operability load test, which is required every 31 days, is scheduled for today. The last three tests were completed 36, 68, and 102 days ago respectively. The plant is at 100% power. (1.5)

b. The plant is at 295 F and heating up at 1 F per minute, when an HPSI pump is found inoperable. (1.5)

c. The plant is at 100% power when it is determined that the discharge valve for one auxillary feedwater pump is failed shut and cannot be repaired for 4 days? (1.5)

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
05.01	2.00	DUD0000468
05.02	1.50	DUD0000390
05.03	2.00	DUD0000471
05.04	1.50	DUD0000478
05.05	1.50	DUD0000474
05.06	1.50	DUD0000473
05.07	2.00	DUD0000008
05.08	2.00	DUD0000476
05.09	2.00	DUD0000469
05.10	3.00	DUD0000472
05.11	3.00	DUD0000475
05.12	3.00	DUD0000477

	25.00	
06.01	2.50	DUD0000348
06.02	2.50	DUD0000479
06.03	2.00	DUD0000480
06.04	2.00	DUD0000481
06.05	2.50	DUD0000482
06.06	2.50	DUD0000483
06.07	3.00	DUD0000486
06.08	2.00	DUD0000503
06.09	3.00	DUD0000484
06.10	3.00	DUD0000485

	25.00	
07.01	2.00	DUD0000487
07.02	2.00	DUD0000488
07.03	2.00	DUD0000489
07.04	2.50	DUD0000490
07.05	3.00	DUD0000491
07.06	2.00	DUD0000492
07.07	3.00	DUD0000493
07.08	2.50	DUD0000495
07.09	3.00	DUD0000496
07.10	3.00	DUD0000494

	25.00	
08.01	1.00	DUD0000497
08.02	2.00	DUD0000498
08.03	2.00	DUD0000499
08.04	2.50	DUD0000500
08.05	2.00	DUD0000502
08.06	3.00	DUD0000501
08.07	2.50	DUD0000424
08.08	2.50	DUD0000428
08.09	3.00	DUD0000427
08.10	4.50	DUD0000378
	25.00	

ANSWERS -- C-E SIMULATOR -84/06/18-DUDLEY,N.

ANSWER 5.01 (2.00)

1. D [.5]
2. C [.5]
3. B [.5]
4. D [.5]

REFERENCE
General Reactor Operating Characteristics p 9,26; fig. 15
Nuclear Physics, Reactor Theory p 150

ANSWER 5.02 (1.50)

When rods were inserted only prompt neutrons were affected and reactor power was stablized.[0.75] During the time that the operator was taking critical data, the delayed neutrons contributed to the overall neutron population [0.75] thus power increases. (1.5)

REFERENCE
C-E Reactor Theory pgs. 74-76

ANSWER 5.03 (2.00)

a. Decrease (due to MTC)
b. ~~Increase~~ Remains the same [illegible] (due to PLCS program)
c. Increase (due to Tavg program)
d. Increase (due to removing more heat) [0.5 each]

REFERENCE
SD: General Reactor Operating Characteristics p 14
Pressurizer Level and Pressure Control Systems p 23
Steam, Feed, and Condensate Systems p 12-14

ANSWER 5.04 (1.50)

See attached figure [0.5 for each point].
Removal of boron from RCS. [0.5] (1.5)

REFERENCE
General Reactor Operating Characteristics p 14-20

K_{EFF} VS WATER TO FUEL RATIO

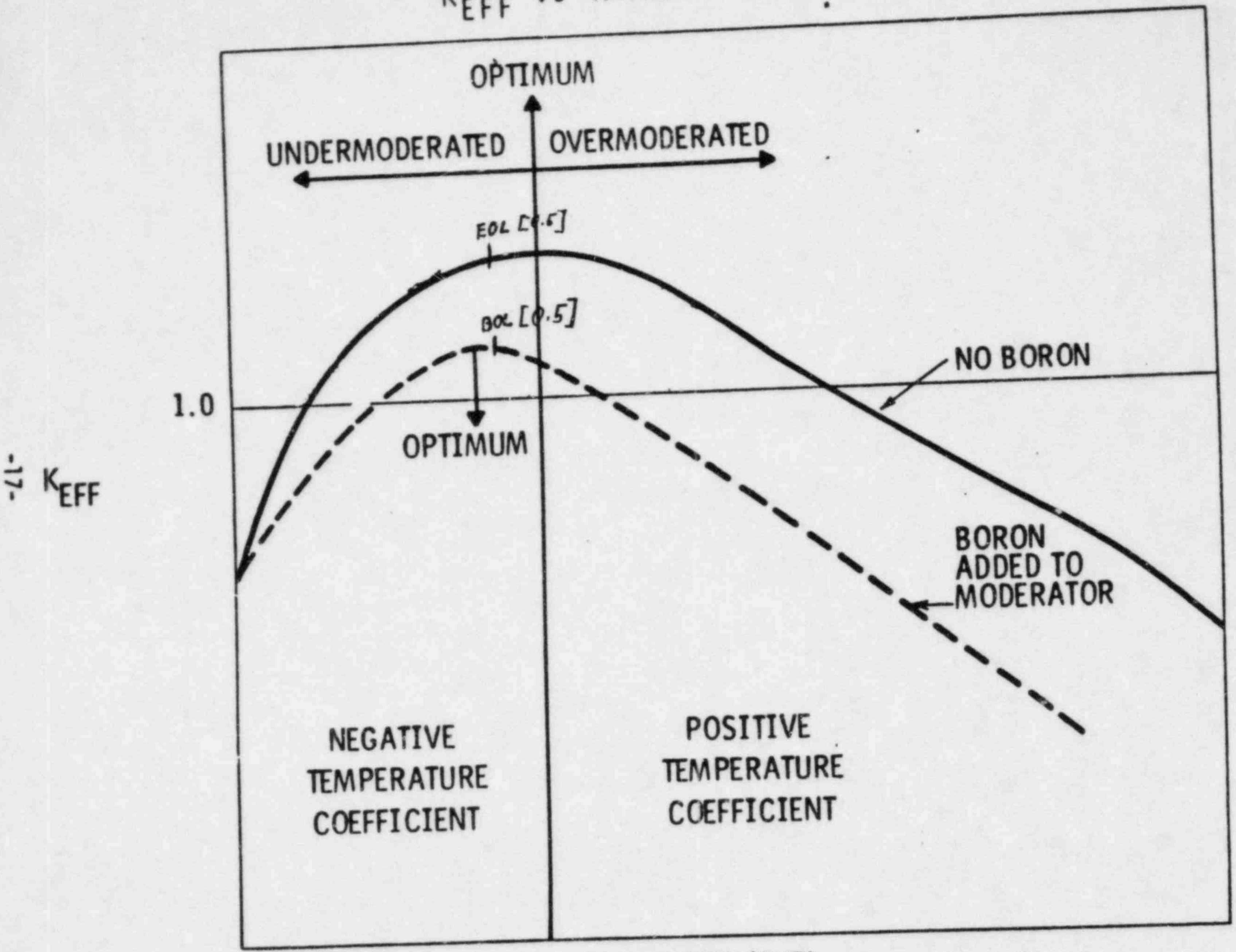


Figure 14

ANSWER 5.05 (1.50)

a. Measures the difference in vector angle between current and and voltage. [0.25] Provides indication of circulating currents which affects heating of windings. [0.75] (1.0)

b. Using generator voltage adjust (0.5)

REFERENCE
Basic Theory Review p 3-12,3-13

ANSWER 5.06 (1.50)

Steam is directed through a nozzle (into a mixing chamber), which lowers pressure. [0.6] The lower pressure causes gases ~~and ac~~ to flow into nozzle. [0.6] The mixture is then carried out the diffuser. [0.3] (1.5)

REFERENCE
Basic Theory Review p 3-4-3

ANSWER 5.07 (2.00)

a. Cooldown is controlled by controlling the steam and feed flow from the S/G.[0.5] If steam and feed flow goes up, the cooldown rate is increased. [0.5] (1.0)

b. The higher the amount of decay heat which must be removed, the higher the delta T and flow rate will be. [0.5] Ten minutes after the trip decay heat will be about 3% reactor power and 1 hour later decay heat will be about 1.5%. [0.5] (1.0)

to setup.

REFERENCE
SD: Natural Circulation - loss of forced coolant flow p. 7

ANSWERS -- C-E SIMULATOR -84/06/18-DUDLEY,N.

ANSWER 5.08 (2.00)

Yes. [0.4]
HPIS shut off head is 1270 psi [0.4] which corresponds to a saturation temperature of 574 F [0.4]
Tavg is maintained at 532 F by SBCS [0.4] therefore the plant would be 42 F subcooled. [0.4] (2.0)

REFERENCE
SD: Engineered Safety Features, p 8
SD: Steaam, Feed, and Condensate System, p 14
Steam Tables

ANSWER 5.09 (2.00)

See Figure 1-1. With one pump stopped,the flow rate at a given head is halved.[0.5] Sketch a one pump head capacity curve and pick off off the flow rate where the one pump curve intersects the system curve;(100gpm)[0;5]. Draw a straight line to the left until it intersects with the HX. #4 curve(40 gpm).[0.5] 100gpm-40 gpm leaves 60 gpm for the remaining HX's; or 20 gpm each. [0.5] (2.0)

REFERENCE

CCNPP module 14.0 requal man.

ANSWER 5.10 (3.00)

a. Decrease[0.25], smaller delta T required to transfer more energy from RCS. [0.5]
b. Increase[0.25], center line temperature responds to RCS temperature inorder to maintain constant delta t across cladding. [0.5]
c. Decrease[0.25], fuel swelling and clad creep reduce clad gap which reduces delta T across gap and lowers center line temp. [0.5]
d. No change[0.25], pressure has little effect on heat transfer in subcooled fluids. [0.5]

REFERENCE
SD: Thermal Hydraulics p 18-27

ANSWER 5.11 (3.00)

a. Reactor would be supercritical [0.75].
By doubling count rate 1-keff is halved. [0.5]
By adding same amount of reactivity keff would be greater than 1. [0.25] (Numerical calculations may be used) (1.5)

b. About 30% power [0.75]
Total power defect changes linearly [0.75] (1.5)

REFERENCE
CE Nuclear Physics Reactor Theory and Core Operating Characteristics, p 139-147
Technical Data Book, p22

ANSWER 5.12 (3.00)

a. Decreases (~15 F) [0.3] due to buildup of Xe [0.3]
b. Held constant [0.3] by PPCS spray and heaters [0.3]
c. Decreases [0.3] due to the decrease of Tavg [0.3]
d. Increases [0.3] due to lower S/G pressure [0.3]
e.* Constant [0.3] increases slightly due to increased steam flow but decreases slightly due to more BTU/lbm transfered at lower pressure. [0.3] (3.0)

REFERENCE
SD: General Reactor Characteristics p 23, 39, 45
Pressurizer Level and Pressure Control Systems p 2
Turbine Generator p 15
Steam Tables

ANSWER 6.01 (2.50)

a. Valves on both HX open
b. No effect
c. Two of three pumps start
d. Two of three pumps start
e. Valves open [0.5 each] (2.5)

REFERENCE
System Descriptions: Support Systems pp 3,6,7
ESF pp 22-23

ANSWER 6.02 (2.50)

a. RCS is protected by the primary safety valves [0.5] since Tave is greater than NDT. [0.5] OR OVERPRESSURIZATION OF RCS BY DESIGN (1.0)
b. Yes [0.5] SDC connection to the CVCS is downstream of the letdown heat exchangers, which cools the RCS water to below 140 F [0.5] and downstream of the temperature sensor used for shifting the demineralizer bypass valves [0.5] (1.5)

REFERENCE
SD: CVCS p 12

ANSWER 6.03 (2.00)

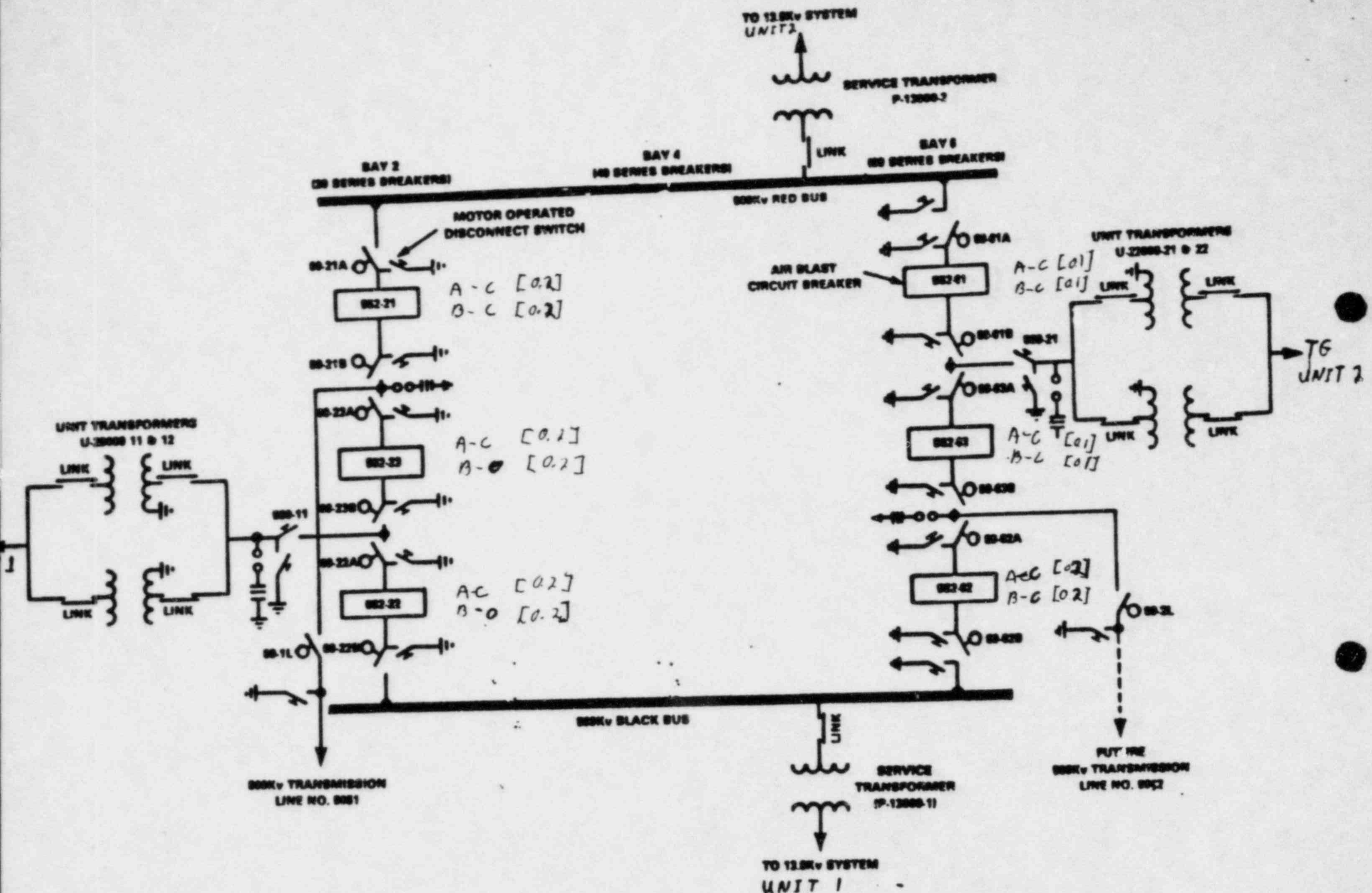
a. Drain SIT to RCDT (0.7)
b. Fill using RWT and containment spray pump [0.7] Drain to RCDT. [0.6] (1.3)

REFERENCE
SD: Engineered Safety Feature p 3,4
OP C.13 p 4,5

ANSWER 6.04 (2.00)

a. see attached sheet [0.2 each]

b. see attached sheet [0.2 each]



TO 13.8Kv SYSTEM
UNIT 2
SERVICE TRANSFORMER
P-13000-2
LINK
BAY 2
(20 SERIES BREAKERS)
BAY 4
(40 SERIES BREAKERS)
BAY 6
(60 SERIES BREAKERS)
500Kv RED BUS
MOTOR OPERATED
DISCONNECT SWITCH
A - C [0.2]
B - C [0.2]
A-C [0.2]
B-0 [0.2]
A-C [0.2]
B-0 [0.2]
AIR BLAST
CIRCUIT BREAKER
A-C [01]
B-C [01]
A-C [01]
B-C [01]
A-C [02]
B-C [02]
UNIT TRANSFORMERS
U-22000-21 & 22
TG
UNIT 2
UNIT TRANSFORMERS
U-22000 11 & 12
LINK
500Kv BLACK BUS
500Kv TRANSMISSION
LINE NO. 5051
SERVICE
TRANSFORMER
(P-13000-1)
TO 13.8Kv SYSTEM
UNIT 1
FUT. 1RE
500Kv TRANSMISSION
LINE NO. 5052

REFERENCE
SD: Electrical Power Distribution
EOP -7E p 4

ANSWER 6.05 (2.50)

a. 1. Delta T Power [0.5]
   2. Feed flow, feed temp., SG press. (calorimetric) [0.5] (1.0)
   3. In-core
b. PZR pressure, Tavg [0.5] (0.5)

c. 1. Th, Tc
   2. SG pressure
   3. Incore TC's [0.5 each for any two] (1.0)

ANSWER 6.06 (2.50)

a. APD
   Rod Drop alarm
   Automatic Withdrawal Prohibit (AWP) alarm
   Level 1 bistable energized ~~light~~
   Subchannel comparator deviation lights
   (lower power on that channel) [0.3 each]

b. High power level channel trip
   Axial flux Offset channel trip
   TM/LP channel trip
   Subchannel comparator deviation light [0.3 each]
   Tave - Tref mismatch

REFERENCE
SD: Nuclear Instrumentation p 28-31

ANSWER 6.07 (3.00)

a. Yes [0.35] deenergizing channels provide a trip signal. [0.4]
b. Yes [0.35] TM/LP channel A and B trip. [0.4]
c. No [0.35] flow, power, TM/LP, and APD values are all less than the reduced setpoints. [0.4]
d. Yes [0.35] each SG pressure channel auctioneers lowest pressure from the SG's. [0.4]

REFERENCE
SD: RPS p 5,13,14-16,36-37

ANSWER 6.08 (2.00)

Cool the reactor core
Limit containment pressure (magnitude and duration)
Provide long term post incident cooling
Reduce airborne radioactivity in containment [0.5 each] (2.0)

REFERENCE
SD: Engineered Safety Features p 1

ANSWER 6.09 (3.00)

Reactor trips on low SG level (-50") [0.6]
AFW starts on low SG level (-40") and maintains level [0.6]
TG trips on Reactor trip [0.6]
SBCS bleeds steam to maintain SG pressure (900 psia) [0.6]
PZR level controls Pzr level (120" to 160") [0.3]
PZR pressure control controls PZR pressure (2225-2300) [0.3] (3.0)

REFERENCE
EOP 6: Loss of Feedwater
SD: Steam Feed and Condensate System p 14,44
PZR Level and Pressure Control System p 7,18

ANSWER 6.10 (3.00)

a. LPSI needed to circulate water through the core [0.75]
b. CCW needed to provide cooling to SDHX [0.75]
c. SW needed to provide cooling ~~to CCW HXe~~[0.75] To EQUIPMENT (ANY BUILDING OR CTMT AIR COOLERS)
d. Seperate buses needed to provide reliability [0.75]

REFERENCE
O.P. . C.12.1
SD: ESF p 11
Electrical Distribution p 8
Support Systems p 1
T.S. p B 3/4 7-3

ANSWERS -- C-E SIMULATOR -84/06/18-DUDLEY, N.

ANSWER 7.01 (2.00)

1. D
2. B
3. A
4. ~~C~~ D

REFERENCE
EOP 7B fig 5.1
EOP 7C p 19
EOP 7B p 12
EOP ~~8E p 11~~ 7D FIG 5.21

ANSWER 7.02 (2.00)

a. 35 min. (0.6)
b. Load set [0.3] changes the setpoint at which the TG attempts to control load [0.2] at selected rate [0.2]
Load limit [0.3] requires that the load set be above the load limit and load is controlled on the limiter [0.2] at maximum rate [0.2] (1.4)

REFERENCE
OP B.6 p 2-3, 7

ANSWER 7.03 (2.00)

a. Provides heat removal capacity following a reactor trip inorder to remove decay heat.

b. Prevents over pressurizing the condenser following the reactor trip.

REFERENCE
OP F.3 p 11

ANSWER 7.04 (2.50)

Initiate a containment evacuation signal [0.6]
Initiate a containment radiation signal [0.6]
Shift the four containment cooling fans to high speed/high SW flow [0.6]
Secure waste gas discharge [0.7]

REFERENCE
EOP 8 p 2

ANSWER 7.05 (3.00)

a. Depress both Rx trip pushbuttons [0.5]
De-energize CEDM MG sets [0.5]
Manually initiate auxillary feedwater [0.5]
Manually initiate SIAS [0.5] (2.0)

b. Verify SIAS or manually initiate [0.5]
Stop all RCP's [0.5] (1.0)

--- REFERENCE --- (DUD0000491) ---

EOP 1 pp 2-5

ANSWER 7.06 (2.00)

a. Control room
b. Control room
c. *Control room ~~(Technical Support Center)~~
d. Emergency Operating Facility
e. Control room - [0.3 each]

REFERENCE
EPIP 4103 p bh3-bh4, bh6

ANSWER 7.07 (3.00)

1 No [0.35] because he would exceed 300 mrem/qt [0.4]
2 No [0.35] because he would exceed admin. limit of 1000 mrem/qt [0.4]
3 No [0.35] because she would exceed limit of 125 mrem/qt [0.4]
4 Yes [0.35] exposure would be less than 1000 mrem/qt [0.4]

REFERENCE
Radiation Science fig. 1-1

ANSWERS -- C-E SIMULATOR -84/06/18-DUDLEY, N.

ANSWER 7.08 (2.50)

Plant computer in service
Reactor power greater than 15%
Tavg and Tref matched (2 F)
No CWP or AWP alarms
Reactor power and Turbine load stable
RRS 1 and 2 stable with no CEA motion demand
PZR level controlling

[0.5 each for any 5]

REFERENCE
OP F.3 p 4

ANSWER 7.09 (3.00)

CEA insertion [~~0.4~~ 0.7] used when greater than 2 rod bottom lights not lited[~~0.3~~ 0.2] and rods indicate out on metrascope [~~0.3~~ 0.1]

Borate via CVCS [~~0.4~~ 0.7] when greater than 2 CEA not inserted ~~[0.2]~~ reactor power > 5% or increasing [~~0.2~~ 0.1] RCS pressure > 1200psia [0.2]

Borate via HPSI [~~0.4~~ 0.7] when greater than 2 CEA not inserted ~~[0.2]~~ reactor power >5% or decreasing [~~0.2~~ 0.1] and RCS pressure < 1200 psia [0.2]

REFERENCE
EOP 7A p 9

ANSWER 7.10 (3.00)

a. LPSI pump stops [0.6]
Recirculation valves from containment sump will open [0.6]
Recirculation miniflow valves close [0.6]

b. Open HPSI pump suction crossover valves [0.6]
Adjust containment spray valves to adjust flow [0.6]

REFERENCE
OP C.16 p 3-5

ANSWER 8.01 (1.00)

Time [0.3]
Release number [0.3]
Maximun flow rate [0.4] (1.0)

REFERENCE
AP -11 p 7

ANSWER 8.02 (2.00)

Intent not altered [0.6]
Approval by two members of plant staff [0.4] one of whom holds an SRO license [0.4]
Change documented, reviewed by POSRC, and approved by Plant Superintendant [0.6] (does not address Unreviewed Safety Question) (2.0)

REFERENCE
T.S. p 6-10
10CFR 50.59 = AP-10 p5

ANSWER 8.03 (2.00)

Order Refueling SRO to move fuel to a safe and conservative position ~~and then lower the fuel.~~ [0.75]
Suspension of fuel loading is required by loss of audio count rate. [0.5]
SRO is in charge of overall plant operations. [0.75]
(2.0)

REFERENCE
AP -12 p 3

ANSWER 8.04 (2.50)

2 SOL [0.5]
2 OL [0.4]
3 Non-licensed [0.4]
1 STA [0.4]
1 HP [0.4]
6 Fire Brigade [0.4] (2.5)

REFERENCE
T.S. p 6-1, 6-3

ANSWER 8.05 (2.00)

*Declare channel C inoperable ~~[0.75]~~ [1.00]
Bypass channel C (ans operate in 2/3 coincidence ~~[0.75]~~ which is allowed by T.S.) ~~[0.5]~~ [1.00] (2.0)

~~Continue to operate with channnel C at pre-trip alarm set points depending on 1/3 coincidence for trip. [0.75]~~ c

REFERENCE
T.S. 3-1

ANSWER 8.06 (3.00)

Within 2 hours and once every 8 hours verify Ftr and Ftxy are within limits. ~~[1.5]~~ [1.0]
Replace N=1.0 with N=0.~~95~~60 in the M x N calculation and verify operations in acceptable region. [0.5]
Ftr is within limits (1.56) [0.5]
Ftxy ~~is~~ NOT within limits (~~1.584~~ 1.716) ~~[0.5]~~ REDUCE POWER TO RETURN F_{xy}^{T} TO WITHIN LIMITS. [1.0] (3.0)

REFERENCE
T.S. p 3/4 2-1 to 2-12

ANSWER 8.07 (2.50)

a. Suspend all operations involving core alterations or positive reactivity changes [1.0]. Initiate and continue boration [EMERGENCY BORATION] at > 40 gpm of 1750 ppm boric acid until Keff is < 0.95 (or boron concentration is >1750 ppm) [1.0] (2.0)

b. Immediately (0.5)

REFERENCE
Calvert Cliffs Tech. Specs. pg. 3/4 9-1

ANSWER 8.08 (2.50)

a. 1. Facility placed in at least hot standby within one hour.

2. NRC operations center notified by phone as soon as possible and in all cases within one hour.

3. Safety limit violation report shall be prepared and submitted
ACCEPT APPLICABLE ACTION STATEMENTS [3 @ 0.5 each] (1.5)

b. 1. The combination of thermal power, pressurizer pressure and highest Tc shall not exceed the limits of the T. S. curve (2.1-1) [0.5]

2. RCS pressure shall not exceed 2750 psia [0.5] (1.0)

REFERENCE
T.S. p 6-9 and 2-1

ANSWER 8.09 (3.00)

a. Should report [0.35] plant is operated outside design basis [0.4]

b. No report [0.35] needed when an action statement for LCO is entered [0.4]

c. Should report [0.35] shutdown due to inability to meet LCO action statement requirement[0.4]

d. No report [0.35] for ESF actuation during surveillance testing [0.4] (3.0)

REFERENCE
AP -5 p 4

ANSWERS -- C-E SIMULATO… -84/06/18-DUDLEY,N.

ANSWER 8.10 (4.50)

a. Each test is within 25% of required time [0.35] and each three conserutive tests within 3.25 of reqired time [0.4].

Declare DG A inoperable. [0.25] Prove operability of DG B within 1 hr. [0.3] Conduct load test on DG A. [0.2] (1.5)

b. ~~Cannot enter a higher mode with reliance on action statement. [0.75]~~
2 HPI PUMPS REQUIRED OPERABLE [0.75] INSURE OPERABLE PUMPS OFF SEPERATE POWER SUPPLIES [0.75]
~~Stop heatup. [0.45] Restore HPSI pump operability prior to entering mode 3. [0.3]~~ (1.5)

c. Unable to comply with LCO or Action Statement. [0.75]
(general statement T.S. 3.03)
Start shutdown within one hour. [0.25] Hot standby within next 6 hours. [0.25] Hot Shutdown within following 6 hours. [0.25] (1.5)

REFERENCE
T.S., pp. 3/4 0-1, 3/4 0-2, 3/4 5-3, 3/4 7-4, 3/4 8-1 378