Report No. 50-206/OL-86-02

Docket No. 50-206

Licensee: Southern California Edison Company
P. O. Box 800
2244 Walnut Grove Avenue
Rosemead, California 91770

Facility Name: San Onofre Nuclear Generating Station, Unit 1

Examination Conducted: November 18-20, 1986

Examiners: _____ 12-8-86
G. W. Johnston, Operator Licensing Examiner — Date Signed

_____ 12-8-86
J. P. O'Brien, Operator Licensing Examiner — Date Signed

_____ 12-8-86
T. Meadows, Operator Licensing Examiner — Date Signed

Approved: _____ 12/8/86
J. O. Ellis, Chief, Operations Section — Date Signed

Summary:

During this requalification cycle, twenty percent of the operating staff (ten licensed operators) were examined. This included eight Senior Reactor Operators, and two Reactor Operators. The NRC prepared a complete written examination and administered it to all participants. An operating examination was also administered to all ten of the participants. All of the participants passed all portions of the examination.

In accordance with NUREG-1021, ES-601 'Requalification Program Evaluations', the requalification program at San Onofre Nuclear Generating Station Unit 1 is evaluated as satisfactory.

DETAILS

1. Persons Examined

A group of ten operators, eight holding Senior Reactor Operator licenses, and two hold Reactor Operator licenses representing approximately 20 percent of the licensed staff were examined.

2. Persons Contacted

Southern California Edison:

*M. Short
*M. Kirby
*R. Mette

NRC:

*G. Johnston, Operator Licensing Examiner
*J. O'Brien, Operator Licensing Examiuer
T. Meadows, Operator Licensing Examiner
M. Royack, Operator Licensing Examiner

*Denotes those present at exit on November 20, 1986.

3. Program Evaluation

Required for Satisfactory:

The requalification program was evaluated upon the criteria of Examiner Standard ES-601 of NUREG-1021. The requirement for a satisfactory program is more than 80% of the evaluated operators must pass all operating examinations, all sections of the written examination, and the written examination overall administered by the NRC.

Performance:

The NRC administered examinations to 2 Reactor Operators and 8 Senior Reactor Operators. All ten participants passed all portions of the examination. The result is a pass rate for all the participants of 100%.

Evaluation:

Satisfactory.

4. Exit Meeting

The examiners met with the licensee representatives denoted in Paragraph 2 at the conclusion of the site visit to discuss the results of the evaluation to that point.

Resolution of Facility Comments

Facility Comment: 1.04

"Question 1.04: Question and answer OK could be clearer if question referred to '50% power, equilibrium Xenon,...'"

Resolution:

The examiner agrees.

Facility Comment: 1.06

"Question should read "..six factor formula.." instead of "..six function formula...". Answer OK."

Resolution:

The examiner pointed out the error to the candidates during the written examination.

Facility Comment: 1.10

"Reference should be SONGS Unit 1 Reactor Core Physics Book Cycle 9.

Part a: SONGS values different from answer key, see Cycle 9 book, pages 67 and 68.
Part b: SONGS peak occurs 6-7 hours following trip. See Cycle 9 book, pages 67 and 68.
Part c: OK"

Resolution:

The examiner will change the key.

Facility Comment: 2.01a

"High point value (13.3% of section). Add: Feedwater isolation MOV's (MOV's 20, 21 and 22) shut same reference. Same as question 6.01a of SRO exam".

Resolution:

Point value does not exceed the 20 percent limit imposed by the Examiner standards. The examiner will change key but not distribution of points for the indicated answers.

Facility Comment: 2.03b-3

"SIS Valves, HV851, 852, 853, and 854, are operable on loss of air since valves fail in safety related direction and air is needed to position valve only in non-safety related direction. Reference drawing is in error as it shows air is needed to position valve in safety related direction. Correct drawing is attached. Same as Question 6.05b-3 of SRO exam. NOTE: Determination of operability of components is not a RO function at SONGS but is done by the SRO on shift."

Resolution:

The examiner stated to the RO candidates that the term used was not OPERABLE/INOPERABLE as in the Tech. Specs. but only as to whether the valves actually do or do not operate. On this basis the examiner will change the question before uploading it to the exam bank but sees no reason to change the key.

Facility Comment: 2.05

"Question 2.05a-2: Answer should say 'through seal water return header' to VCT.
Question 2.05a-3: Answer should read 'approximately 2 GPH' not 2 GPM.
Question 2.05b-1: Answer should be 'Maximum flow rate of approximately 100 GPM.'"

Resolution:

The examiner will change the key.

Facility Comment: 2.06b

"Question is not clear if Administrative Maximum load or Design Maximum load is requested. Procedure S01-10-1, which is referenced, says max. load is 4725 KW with no time limit. This is Admin. limit. Answer key provides Design limit. Operationally we are limited to Admin. limit by procedure."

Resolution:

The examiner will change key to Admin. limit and time limit to indefinite.

Facility Comment: 3.02b

"Long term responses may also be included in the answer, eg. PORV may open after some time with no sprays and max. heaters, etc."

Resolution:

The examiner believes that long term responses will not be likely as the question refers only to what controls and their response.

Facility Comment: 3.03

"Question is both open ended (Number of control signals is not asked) and Double Jeopardy in that answer to Part b is dependent to answer to Part a. Answer is correct but credit need be given if only two or three inputs are provided and points should not be deducted twice."

Resolution:

The examiner does not understand the concerns here. The question would not have been considered by the reviewers to have been double jeopardy if the Part a had requested both the inputs and control of director. The examiner considers this a two part question. Secondly, the examiner considers the control of direction and input to be one answer, despite the format of the question.

Facility Comment: 3.06

"Variable Low Pressure Trip has a nominal value of 1840# with an instrumentation setpoint of 1872# to insure that trip occurs before design nominal value of 1840#. Copy of lesson plan covering this is attached. SCE accepts either answer in our exams Pzr HI Level has been changed to 50% recently. Copy of Modification is attached. SIS/LOP should be changed to sequencer since any SIS, SIS/LOP, LOP, or manual Sequencer initiation will trip reactor. See Referenced System Desc., page 6."

Resolution:

Will change key.

Facility Comment: 3.07b

"Low discharge pressure trip setpoint has been changed to 500 psi instead of 675 psi. Copy of FCN is attached. Also, note: this trip is only active in auto."

Resolution:

Agreed, will change key.

Facility Comment: 3.08

"The only Engine trip on emergency start is overspeed. Differential is a generator trip. Also, question 2.06c asked all trips active on an emergency start and is worth 1.0 points. This question asks engine trips and is worth 0.5 points. This would lead candidate to supply one answer to this question."

Resolution:

The examiner agrees that the Differential trip is inappropriate, and will drop the answer. The point value of the answer of overspeed will be increased to 0.5 points to coincide with the 2.06c answer.

Facility Comment: 4.02

Answers and reference (SO123-VIII-4.0 page 6) are for minors. But question does not ask for minors. Answers should be: a) 1.25 Rems/Qtr and b) 7.50 Rems/Qtr."

Resolution:

Will change key.

Facility Comment: 4.03

"Reference should more correctly state Technical Specifications, Section 4.0.2, page 4-1."

Resolution:

Agreed, will change key.

Facility Comment: 4.04

"Not a performance based question. SCE does not use this term in its operation of the facility."

Resolution:

The examiner points out that the NRC holds the licensed operators of a facility responsible for knowing the regulations as they pertain to his duties. This is in spite of the terminology of the facility, the question does ask for the definition of the term in the regulation.

Facility Comment: 4.05b

"Procedure does not specify that immediate action is necessary on off scale High or increasing rapidly. Also, action in fold out page. Action listed in answer is correct.

Resolution:

The examiner points out that this procedure on the foldout sheet is an always applicable requirement during the course of the Steam Generator Tube Rupture procedure. That being the case it is, arguably, an immediate instruction.

Facility Comment: 4.07a

"Same reference also says that if pump has been running for greater than 4 hours, immediate restart may be performed, i.e. no time delay."

Resolution:

Agreed, will change key.

Facility Comment: 5.02a, 5.02b

"5.02a: Question is almost the same as RO question 1.01a except SRO does not ask candidate to indicate new operating point, yet answer key is just a copy of RO answer key. Change to answer key for SRO is needed. 5.02: This is a completely different question form RO question 1.01b but answer key is for RO question! New answer key to answer SRO question is needed."

Resolution:

Agreed, will change key.

Facility Comment: 5.03c

"Answer key incorrect. Answer should be Pressure 2 not Pressure 1. DNB has not occurred at Pressure 2 at this heat flux."

Resolution:

Agreed, will change key.

Facility Comment: 5.06a

"See comments for RO question 1.10."

Resolution:

Same changes as RO 1.10.

Facility Comment: 6.01

"See comments on RO question 2.01."

Resolution:

Same changes as RO 2.01.

Facility Comment: 6.03

The reviewers noted that the Cryogenic unit has been retired in place.

Resolution:

The examiner will leave the question in the exam but delete it before uploading to the exam bank.

Facility Comment: 6.04

"See comments on RO question 2.03."

Resolution:

Same as question 2.03.

Facility Comment: 6.06

"See comments on RO question 3.03."

resolution:

Same as question 3.03

Facility Comment: 6.07

"See comments on RO Question 3.02."

Resolution:

Same as question 3.02.

Facility Comment: 7.01

"Parameters checked for Priority 4 Heat Sink are Auxiliary Feedwater and Main Feedwater. Which reflect Primary and Secondary heat removal capability."

Resolution:

Agreed will change key.

Facility Comment: 7.03

"Point value too high! 20% of the section for one question. Reference has been changed. Procedure SO1-14-17 has been superseded and replaced by SO123-0-23.1, copy attached. Answers are still OK."

Resolution:

The examiner again notes that the standards allow a 20% limit for a question.

Facility Comment: 7.04

"Max. Delta T between Pzr. and RC has changed from 200 deg. F to 190 deg. F. Also, Max. Heatup rate of Pzr. has changed from 95 deg. F/Hr to 90 deg. F/Hr. Copy of revised procedure page is attached."

Resolution:

Agreed, will change key.

Facility Comment: 7.05a

"Rod position for criticality is usually only related to Control bank 2. By Tech. specs. and admin. controls as well as this startup procedure all other groups are full withdrawn. This is usually assumed and generally not given as part of the answer for this question."

Resolution:

The examiner will eliminate the reference to the other banks.

Facility Comment: 7.06

"Reference is wrong. Question asks per procedure SO1-3-1 yet reference is to SO1-3-2. Procedure SO1-3-1 says observe bubble formation when RCS temp. 250 deg. F and PZR 440 deg. F. Procedure does not discuss answer that is in answer key although it is correct.

Resolution:

The key will stand.

Facility Comment: 8.01

This question is false if it is assumed that the rod is immovable "due to binding or friction" per Tech. specs. If immovable for other reason but untrippable then question could be answered true."

Resolution:

The key will stand.

Facility Comment: 8.02

"Question does not ask per Tech. Specs. or administratively. If Admin. is assumed then if, in Mode 3 and trip breakers are racked in, then 3 reactor coolant pumps required. Reference SO1-3-4, Precaution 4.1.6."

Resolution:

It is true the question does not specify Tech.Specs. limitations. However, the examiner feels no change is required of the key.

Facility Comment: 8.03a, 8.03b

"Question may be answered generically ie. 1.25X surveillance interval, and not just 15 hours. Typo in answer key b.1. "last done + 15 Hours = November 29 at 2000 not 2900."

Resolution:

Agreed will change key.

Facility Comment: 8.05

"Reference has been superseded by procedure SO123-0-14. Copy of Attachment 4 to the procedure is attached."

Resolution:

Will change reference.

Facility Comment: 8.06

"For answer to be correct only one of four fuel transfer pumps can be operable. If one per DG is operable then startup may proceed with no restrictions, ie. Tech. specs. require one per DG."

Resolution:

During the course of the examination the examiner informed the candidates that both fuel pumps to one DG were inoperable.

Facility Comment: 8.08

"Question is unclear, if by deviations a TCN is met then answer b and d are correct. However, SCE policy is you never deviate (meaning carry out action outside the procedure without preparing a TCN) from a procedure except in an emergency then a and d are correct. Reference should be 6.8.3 not 6.8.1 and this talks about temporary changes not deviations."

Resolution:

With two possible answers it appears that there is sufficient information for the candidates.

Facility Comment: 8.10b

"Primary responsibility for the EC is the Unit Superintendent. It is true the first recalled EC relieves the SS."

Resolution:

The examiner fails to see any concern.

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR REQUALIFICATION EXAMINATION

Facility: SONGS 1
Reactor Type: WESTINGHOUSE - 3 LOOP PWR
Date Administered: NOVEMBER 18, 1986
Examiner: G. W. JOHNSTON
Candidate: Key

INSTRUCTIONS TO CANDIDATE

Read the attached instruction page carefully. This examination replaces the current cycle facility administered requalification examination. Retraining requirements for failure of this examination are the same as for failure of a requalification examination prepared and administered by your training staff. 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 four (4) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value	Category
15.0	25.0			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
15.0	25.0			2. Plant Design Including Safety and Emergency Systems
15.0	25.0			3. Instruments and Controls
15.0	25.0			4. Procedures - Normal, Abnormal, Emergency, and Radiological Control
60.0		Final Grade		TOTALS

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

Candidate's Signature

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

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

$Pwr = W_f \dot{m}$

$P = P_o\ 10^{SUR(t)}$

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

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

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

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

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

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

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

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

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

Cycle efficiency = $\frac{\text{Net Work (out)}}{\text{Energy (in)}}$

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

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

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

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

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

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

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

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

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

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

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

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1$ seconds^{-1}

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

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

$R/hr = 6\ CE/d^2$ (feet)

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

1 ft. H_2O = 0.4335 lbf/in^2

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 Btu = 778 ft-lbf

1 inch = 2.54 cm

$^\circ F = 9/5\,^\circ C + 32$

$^\circ C = 5/9\ (^\circ F - 32)$

SECTION 1

Prinicples of Nuclear Power Plant Operation,
Thermodynamics, Heat Transfer and Fluid Flow

***QUESTION 1.01**

Refer to figure 5.1, a sketch of a typical Auxiliary Feedwater System utilizing two certrifugal pumps of similiar characteristics and capacities. The plot of volume Flow Rate versus Pressure shows the system with the "A" Auxiliary Feed Pump in operation as an initial condition.

a) Show the changes from the initial conditions to the curve(s) as the PORV or atmospheric dump opens and reduces Steam Generator pressure by 50 percent. Indicate the new system operating point. (1.0)

b) Show the changes from the initial conditions to the curves as the second pump "B" is started. (1.0)

NOTE: INDICATE DIRECTION AND MAGNITUDE OF CHANGES.

*ANSWER
Figure. (2.0)

*REFERENCE
General Physcis, Volume III, Chapter 2.

***QUESTION 1.02**

There are two ways to describe flux distribution variations that can occur in the reactor core. These are typified as flux tilts.

a) What is Axial flux tilt? (0.5)

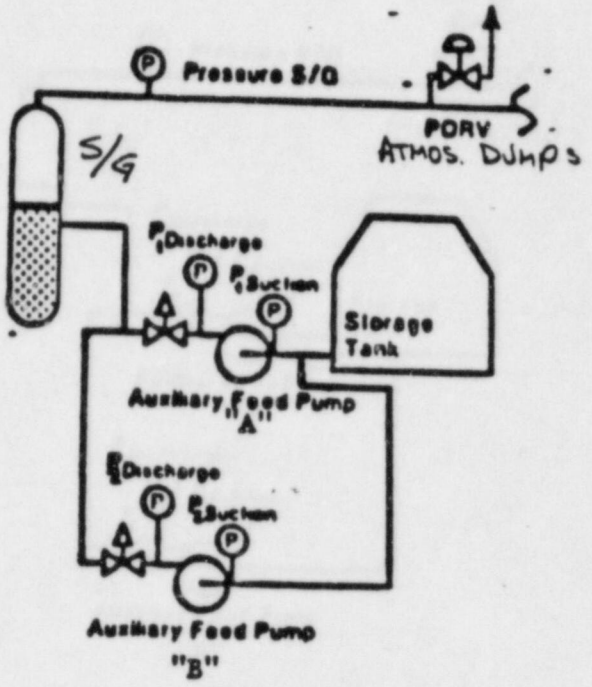
b) What is Racial flux tilt? (0.5)

*ANSWER
a) Variation of flux along the vertical axis. (0.5)
b) Uneven flux distribution across a horizontal plane view of the core. (0.5)

*REFERENCE
Westinghouse Training Notes

KEY

FIGURE 5.1



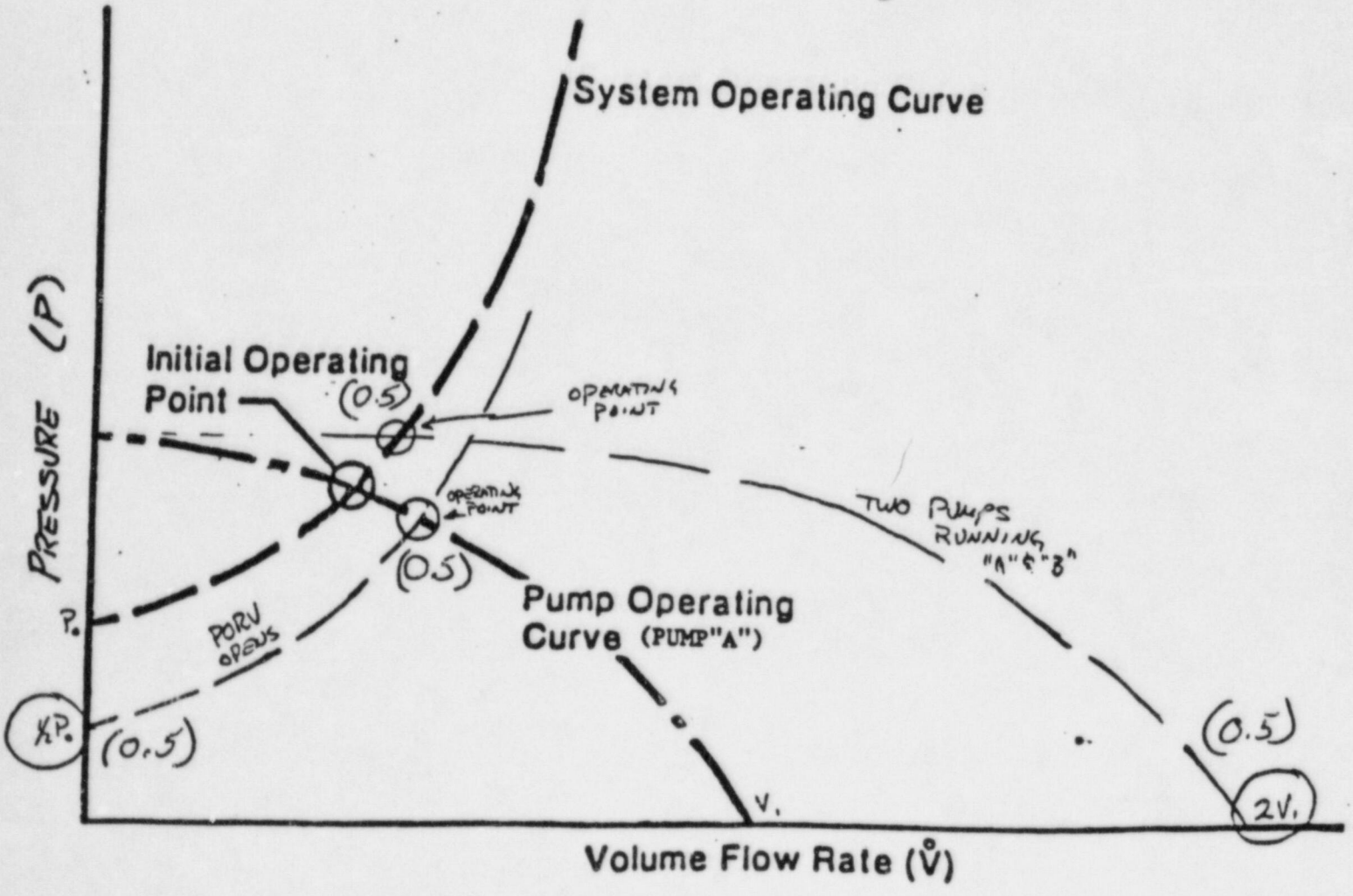
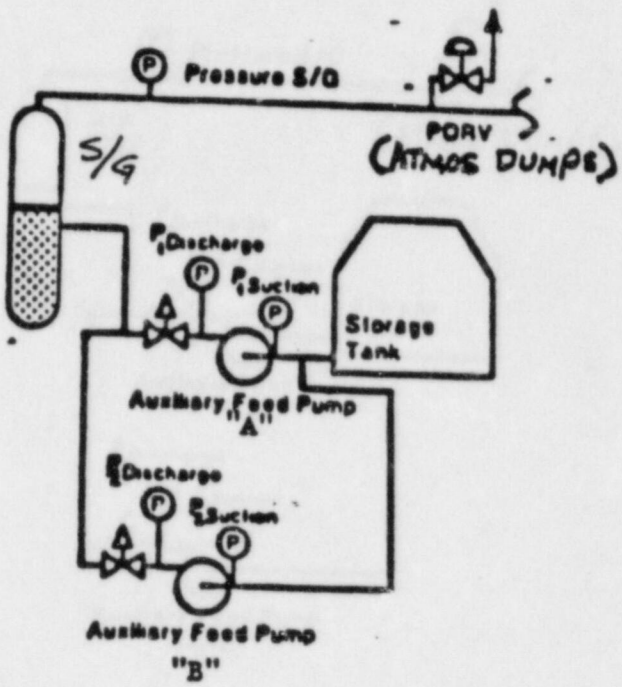
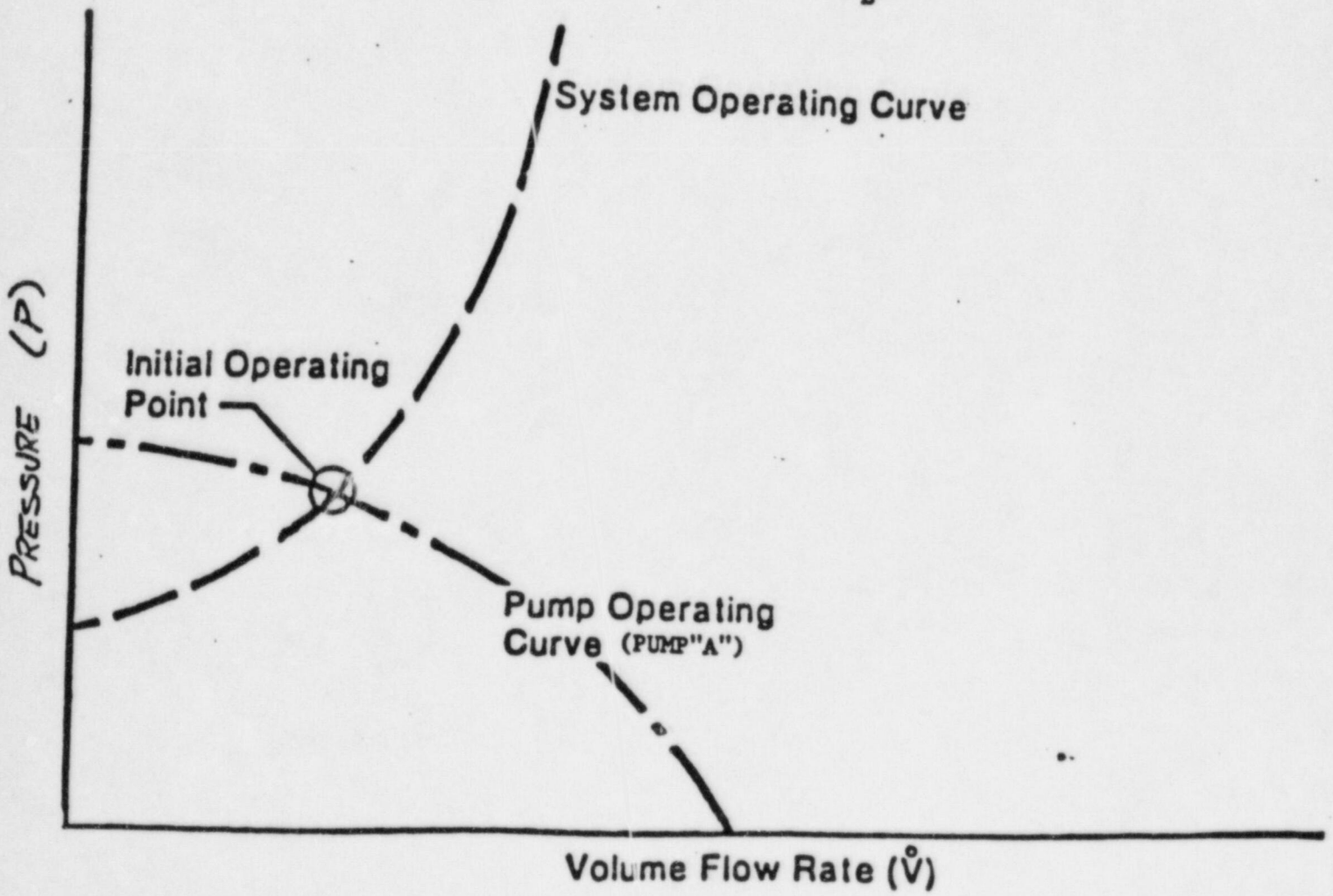


FIGURE 5.1


Pressure S/G
PORV
(ATMOS DUMPS)
S/G
Storage Tank
Auxiliary Feed Pump "A"
Auxiliary Feed Pump "B"


System Operating Curve
Initial Operating Point
Pump Operating Curve (PUMP "A")
PRESSURE (P)
Volume Flow Rate (V̊)

***QUESTION 1.03**

The flowrate through the Steam Generators on the primary side is twelve (12) times the flowrate through the secondary side. Explain what parameters cause this apparent difference. (2.0)

*ANSWER
The difference in the enthalpy on the secondary side is much higher. (1.0) This is so because the sensible heat given up by the primary in the steam generator is taken up to a great degree by the latent heat of vaporization during the change of phase. (1.0)

*REFERENCE

***QUESTION 1.04**

The Xenon peak that occurs after a reactor trip from 100 percent equilibrium xenon condition is greater than the peak for a trip from 50 percent power due to: (1.0)

a) The fission yield for Xenon is higher at 100 percent power.

b) There is more Iodine in the core at the time of a trip from 100 percent power.

c) There are more thermal neutrons in the core at 100 percent power.

d) There are more delayed neutrons in the core at 100 percent power.

*ANSWER
b) (1.0)

*REFERENCE
VIII-06 'Principles of Power Control'.

***QUESTION 1.05**

If reactor power increases from 1000 cps to 5000 cps in 30 seconds, what is the Startup Rate (SUR)? (2.0)

*ANSWER
P = Po (10E:SUR(t))
5000 = 1000 (10E:SUR(.5)) [1.0]
SUR = (LOG 5)/(0.5) = (1.699)/(.5) = 1.4 DPM [1.0]

*REFERENCE
VIII-05 'Reactor Kinetics'.

***QUESTION 1.06**

Which of the following six ~~functions~~ factor formula terms increases on a power escalation to allow reactor power to match turbine power? (1.0)

a) The fast fission factor.

b) The thermal utilization factor.

c) The reproduction factor.

d) The doppler effect.

*ANSWER
b) (1.0)

*REFERENCE
VIII-03 'Neutron Cycle'.

***QUESTION 1.07**

The most serious problem with reaching the critical heat flux (CHF) in a power reactor is caused by: (1.0)

a) the poor thermal conductivity of steam.

b) the blockage of flow through the core when steam bubble formation becomes significant.

c) the displacement of boron from the core as steam bubble formation becomes significant.

d) the high pressure surges in the reactor coolant systems caused by steam bubble formation.

*ANSWER
a) (notes talk to steam blanketing, clad burnout). (1.0)

*REFERENCE
General Physics HTFF Notes, Chapter 4, Part "B", Pages 220-230.

***QUESTION 1.08**

The Fuel Temperature Coefficient increases (becomes more negative from BOL primarily due to: (1.0)

a) the reduction of fuel to clad gap distance.

b) the reduction in the moderators boron concentration.

c) the increase in Pu-240 in the core.

d) the increase in thermal neutron flux.

*ANSWER
c) (1.0)

*REFERENCE
VIII-06 'Principles of Power Control'.

***QUESTION 1.09**

The reactivity worth of a control rod increases: (1.0)

a) as Tave increases from 150 degrees fahrenheit to 500 degrees fahrenheit.

b) as reactor power is reduced from 100 percent to 50 percent.

c) as a result of fission product buildup.

d) when the soluble boron concentration increases.

*ANSWER
a) (1.0)

*REFERENCE
VIII-06 'Principles of Power Control'.

***QUESTION 1.10**

Figure 5.5 is a sketch of Reactor Power versus Time in hours. At t=0 hours reactor startup from Xenon free conditions to 100 percent power occurs. At t=50 hours a reactor trip occurs followed by a reactor startup to 100 percent power at t=65 hours.

a) Sketch the Xenon reactivity response in the core from this power transient. (Indicate approximate magnitude and duration of each transient.) (1.0)

b) Indicate the time in hours at which the maximum negative reactivity will be inserted by Xenon. (0.5)

c) Indicate the time in hours (approximately) that the maximum rate of rod insertion will have to occur in order to overcome the slope of the Xenon transient. (Assume constant Tave and no boration or dilution). (0.5)

*ANSWER
a), b), and c) attached

*REFERENCE
General Physics Volume II, Chapter 4, Section "D".

***QUESTION 1.11**

Listed below are two factors that can affect Departure from Nucleate Boiling Ratio (DNBR). What will the effect on DNBR (increase or decrease) be, if the factors listed below are increased? (1.0)

a) Reactor Power. (0.5)

b) Reactor Coolant System flow. (0.5)

*ANSWER
a) DNBR decreases. (0.5)
b) DNBR increases. (0.5)

*REFERENCE
GP HT&FF notes.

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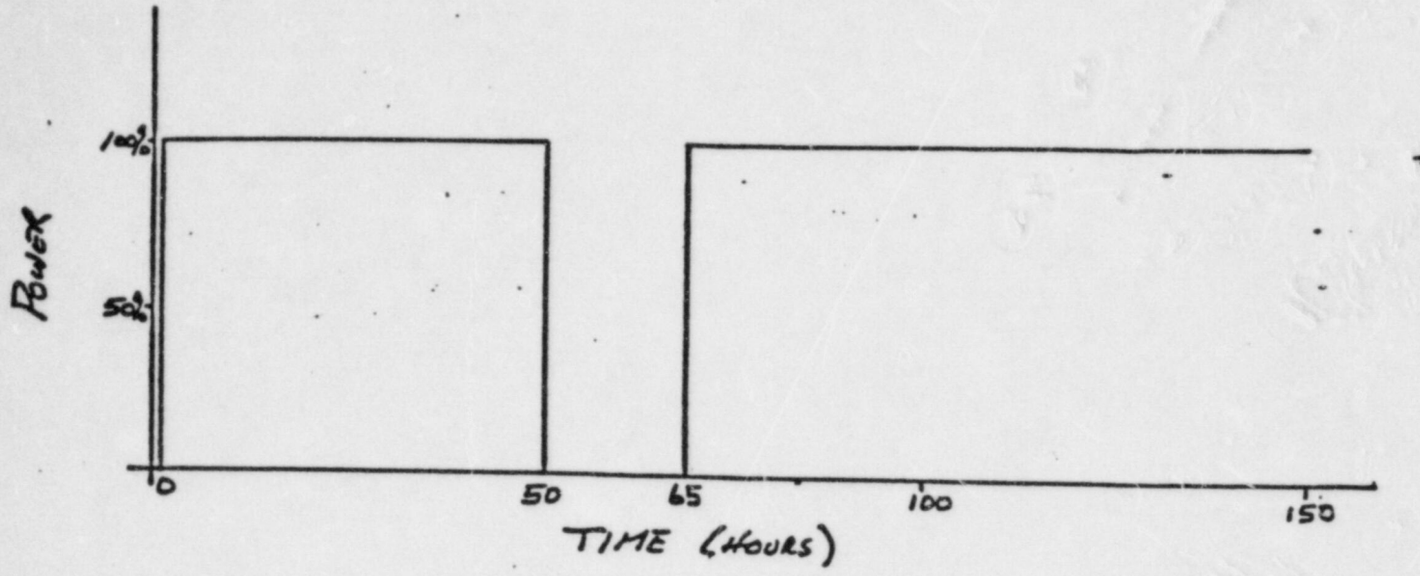
End of Section 1
Go on to Section 2

FIGURE 5.5


POWER
100%
50%
0
50
65
100
150
TIME (HOURS)

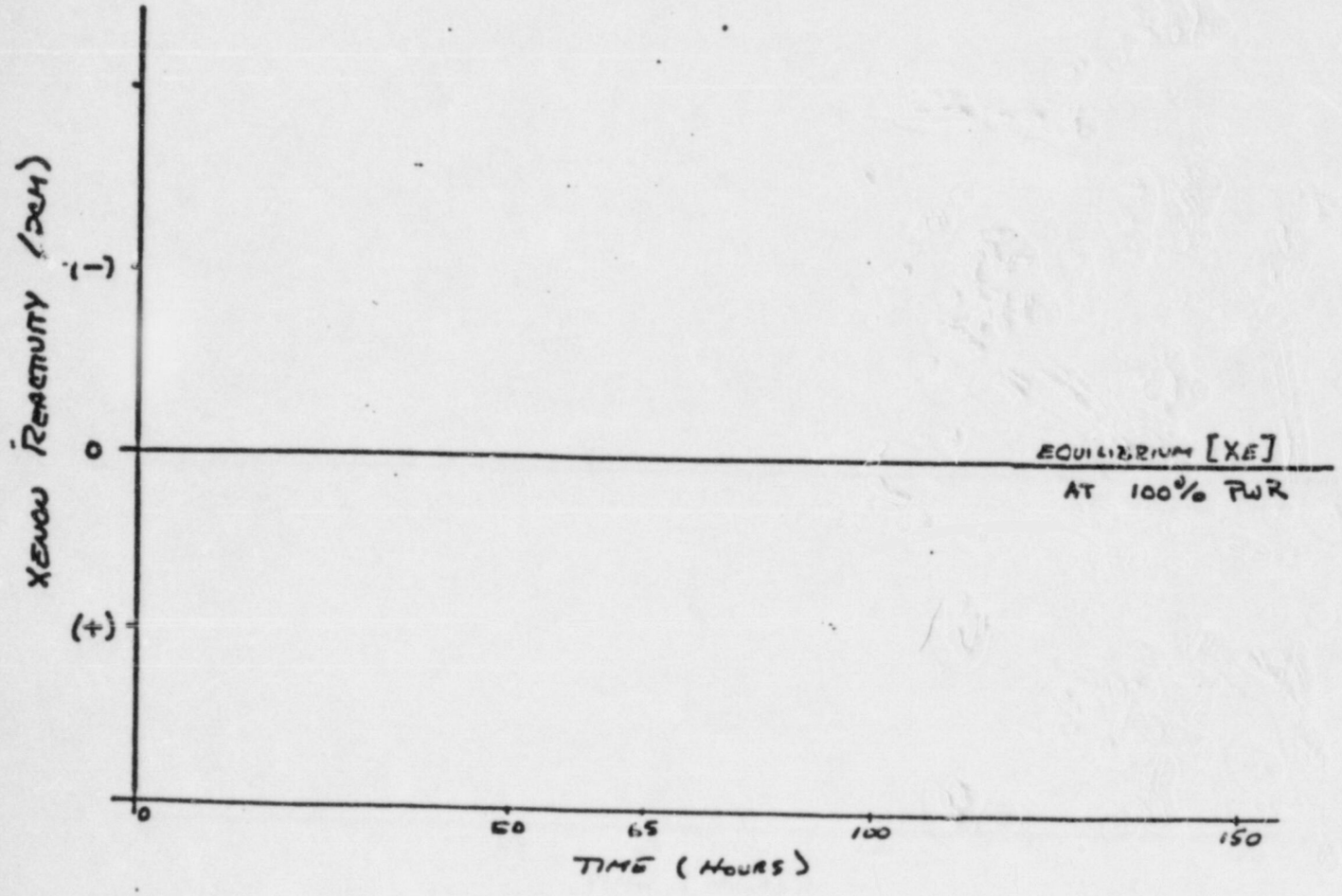
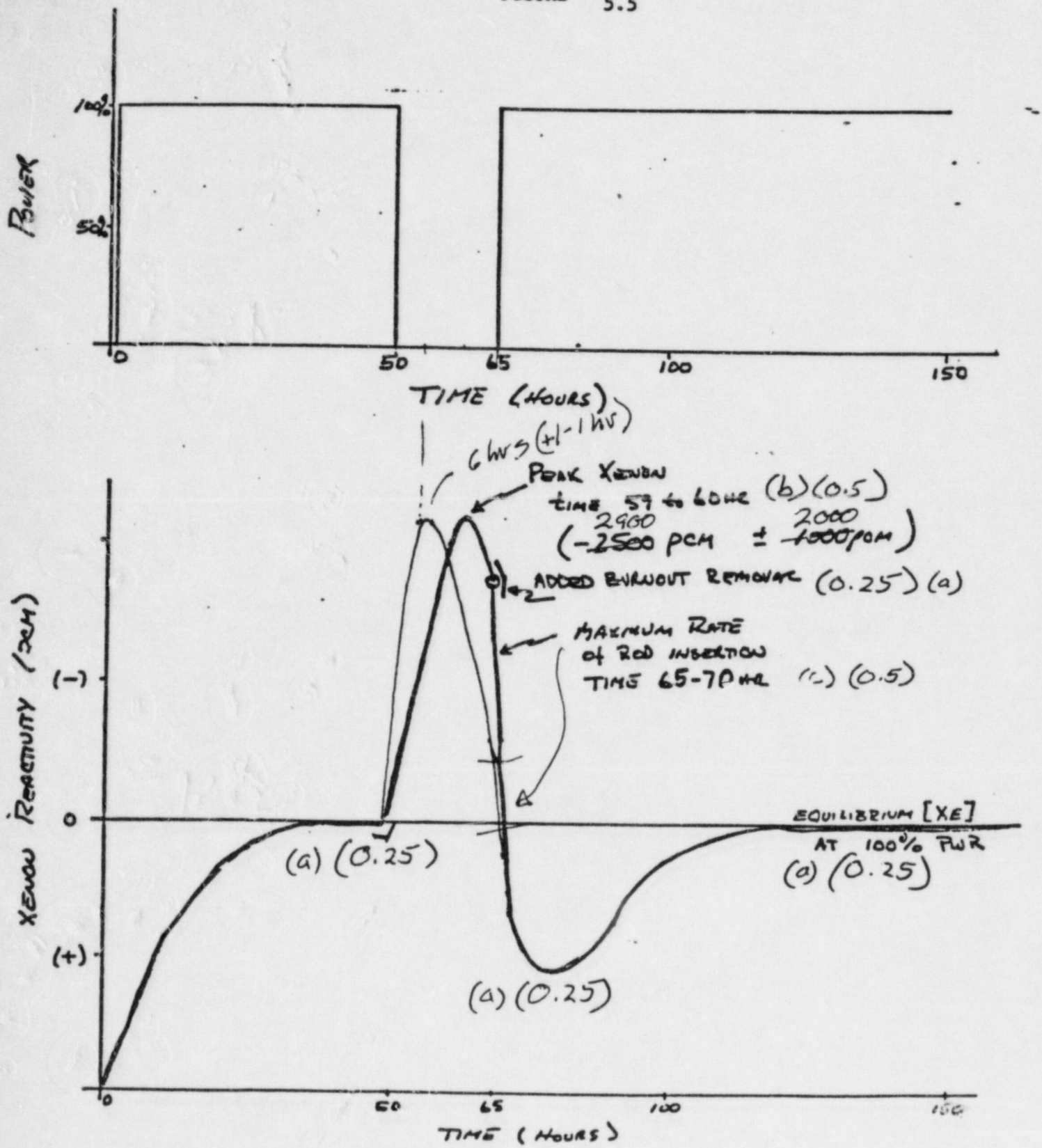

XENON REACTIVITY (ΔK/K)
(-)
0
(+)
EQUILIBRIUM [XE]
AT 100% PWR
0
50
65
100
150
TIME (HOURS)

FIGURE 5.5


POWER
100%
50%
0
50
65
100
150
TIME (HOURS)
6 HRS (+/- 1 HR)
PEAK XENON
TIME 59 to 60 HR (b)(0.5)
(-2500 PCM ± 1000 PCM)
ADDED BURNOUT REMOVAL (0.25) (a)
MAXIMUM RATE
of ROD INSERTION
TIME 65-70 HR (c) (0.5)
XENON REACTIVITY (PCM)
(-)
0
(+)
EQUILIBRIUM [XE]
AT 100% PWR
(a) (0.25)
(a) (0.25)
(a) (0.25)
0
50
65
100
150
TIME (HOURS)

SECTION 2
PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

***QUESTION 2.01**

Concerning the Feed System and Safety Injection System:

a. What pump and valve changes will automatically occur in the Feed and Condensate system when, at full unit load, a Safety Injection signal is received ? (2.0)

b. What specific valve interlock prevents the injection of Feed and Condensate water into the RCS ? (0.5)

c. If during the SIS sequence the SI pump suction isolation valve (HV-853 A&B) fails to open, will the feedwater pump run ? Explain ! (0.5)

* ANSWER

a.
- Main feed pump trips and then restarts (0.5)
- Condensate pumps trip (0.25)
- Heater Drain pumps trip (0.25)
- MFP suction valves (HV-854 A&B) shut (0.25)
- MFP discharge valves (HV-852 A&b) shut (0.25)
- Feedflow control valves (FCV-456/7/8) shut (0.25)
- Feedflow Aux control valves (CV-142/3/4) Shut (0.25)
- (SIS Suction Header Isolation Valves (HV-851 A&B) will open.) (N/A)

b. MFP Suction valves (HV-854 A&B) must be shut before the SIS header isolation valves (HV-851 A&B) will open. (0.5)

c. Yes, the FW pump will run for 30 sec. then if HV-853 does not open, the pump will trip. (0.5)

reference: SD-S01-580, pp.16, S01 exam bank.

Feedwater isolation valves MOV's 20, 21, & 22 shut.

***QUESTION 2.02**

Concerning the CVCS system:

a. Under what three conditions would the Excess Letdown System be Used ? (1.5)

b. Describe the three Excess Letdown Alignments. (1.5)

* ANSWER

a.1. inoperability of normal letdown (0.5)
2. used during PZR Steam Bubble formation (0.5)
3. To increase the cooldown rate during the initial cooldown phase of RHR operation. (0.5)

b.1. Normal - thru seal water return piping into the VCT. (0.5)
2. Thru a section of normal letdown piping and a residual heat exchanger to the VCT. (0.5)
3. Thru and then directly to the RCS drain tank. (0.5)

*REFERENCE: SD-S01-310 pp.5,fig. I-1.

***QUESTION 2.03**

a. List the sources of air that can be used to supply the Instrument Air Header? (1.5)

b. How are the following valves affected by the loss of Instrument air (operable/inoperable) ? If operable, state why? If inoperable, state status: (failed shut, failed as is, or failed open). (1.5)

1. Main feedwater flow control valves
2. Emergency Auxilary feedwater regulators
3. Safety injection valves HV851,852,853,854

*ANSWER

a.1. Three Service Air Compressors (0.5)
2. Auxilary Air Compressor (0.5)
3. Diesel Air Compressor (0.5)

b.1. Inoperable (0.25) - Failed open (0.25)
2. Operable (0.25) - Nitrogen Backup (0.25)
3. Inoperable (0.25) - alarmed in the controlroom, Operable if operator selects Nitrogen backup (0.25)

* Reference: SD-S01- 420 & 580 ; S01-2.4-2 IA Sys Malfunction

***QUESTION 2.04**

TRUE or FALSE ?

In order to move the carriage in the fuel transfer system, both upenders must be fully down. TRUE or FALSE ? (1.0)

* ANSWER

TRUE (1.0)

Reference SD-S01-350

***QUESTION 2.05**

During normal operation, about 7 gallons per minute (GPM) of seal water is supplied to each reactor coolant pump.

a. For the following, give the normal flow rate past each, and tell where the majority of that flow goes.

1. Thermal barrier (0.5)
2. #1 seal (0.5)
3. #2 seal (0.5)
4. #3 seal (0.5)

b. in the event of a # 1 seal failure:

1. What is the expected flowrate through the seal? (0.5)
2. What limits the flowrate to this value ? (0.5)

* ANSWER

a.1. approx. 5 GPM (0.25), into the RCS (0.25) Through seal water return header
2. approx. 2 GPM (0.25), to VCT (0.25)
3. approx. 2 ~~GPM~~ GPH (0.25), to Vapor Seal Head Tank/ RCS Drain Tank (0.25)
4. approx. 100 cc/Hr. (0.25), to Atmosphere (0.25)

b.1. 100 GPM (0.5) ← Maximum flow rate of approx.
2. Bushings in the Floating Ring Seals.

* Reference SD-S01-300 pp.8-11

*QUESTION 2.06

With regard to the diesel generator system:

a. How long can a diesel generator operate at full power without replenishing the Day Tank ? (0.5)

b. what is the maximum load a diesel generator can supply (0.5),
and how long can it supply this load? (0.5)

c. List the trips that remain in effect, when the diesel has started automatically due to safeguards actuation. (1.0)

* ANSWER

a. 4725 45 minutes to 1 hour. indefinitely (0.5)
b. ~~6600~~ kW (0.5) ~~for 2 hours in a 24 hr. period~~ (0.5)
c. 1. Generator differential. (0.5)
2. Overspeed. (0.5)

* Reference SD-SO1-600 pp.2,3,9; SO1-10-1

****

END OF SECTION 2
Go on to Section 3

SECTION 3
INSTRUMENTS AND CONTROLS

***QUESTION 3.01**

Which of the following is true concerning the Source Range Channel high voltage cutoff? (1.0)

a. During a reactor startup either IR channel increasing above the P-6 setpoint will turn off the high voltage.

b. If one IR channel fails low while at power, both Source Range high volatage will be reenergized.

c. If a IR channel increases above 2 X 10 E(-9), it will turn off its respective source range's high voltage.

d. During a reactor shutdown either IR channel decreasing below the P-6 setpoint will turn on the high voltage.

*ANSWER
c. (1.0)

*REFERENCE
SD-S01-380 Excore NI's

***QUESTION 3.02**

Pressurizer Pressure Transmitter PT-430 fails low during normal at power operations.

a. Name <u>one</u> alarm that you expect to annunciate in the control room as a direct result of the PT failure ? (0.5)

b. What <u>pressurizer</u> systems or controls will be affected and how will they respond? (1.0)

* ANSWER

a. Safety Injection Train A Channel 1 Alarm or VLPT Channel 1 Alarm (0.5)

b. 1. Spray Valves will close if they are open (0.5)
2. All heaters in Automatic will turn on (0.5)

* Reference Requal Exam Bank

***QUESTION 3.03**

a. What are the control signal inputs to the Automatic Rod Control System? (1.0)

b. For each of the above, State if the input controls rod speed and/or rod direction. (1.5)

*ANSWER

a.1. T avg	(0.25)	b.1. Direction and speed	(0.5)
2. T ref	(0.25)	2. Direction and speed	(0.5)
3. Nuclear Flux	(0.25)	3. Speed	(0.25)
4. P-P ref	(0.25)	4. Speed	(0.25)

REFERENCE

SD-SO1-400, and Requal Exam Material

***QUESTION 3.04**

What are the two (2) Turbine "Runbacks", and to what value will each run back the Turbine? (2.0)

* ANSWER

1. Dropped Rod (0.5) - runback to 70 % (0.5)
2. Low Frequency (0.5) - runback to 85 % (0.5)

* Reference SD-SO1-200

***QUESTION 3.05**

Concerning the Sub-Cooling Monitoring System:

a. What signals are inputs to each channel of the Sub-Cooling Monitoring System ? (1.0)

b. How is the margin to saturation computed ? (1.0)

c. At what margin does it alarm? (0.5)

* ANSWER

a.

Temperature:	4 Core Exit Thermocouples	(0.25)
	3 Hot Leg RTD's	(0.25)
Pressure:	2 Pressurizer Pressure Xmitters(ChA)	(0.25)
	1 for Ch. B	(0.25)

b. The highest of the 7 Temperature signals is compared to the saturation temperature calculated from the lowest pressure signal input for that Ch. This difference is the output or margin to Saturation. (1.0)

c. 40 degrees F(+/- 2 deg.) (0.5)

* Reference SD-SD1-390 pp. 20.

***QUESTION 3.06**

List 5 of the 11 Reactor Trips, include in your answer: name of the trip, setpoint(s), logic, and initiating devices. (3.0)

* ANSWER

Trip	Setpt.	coincidence /logic	Device
1. Hi SUR	5 DPM (accept 1840 nominal)	1/2	I.R. NIS.
2. Va.low Press.	1872 min.psi	2/3	PZR Press.
3. Turbine Trip	45 PSIG	2/3	A.S. Oil
4. *Two Loop RCS low Flow	85%or BKR position	2/3	loop Flow/BKR contacts
5. *Single Loop RCS Low Flow	85%or BKR position	1/3	Loop Flow/ contacts
6. PZR Hi Level	70 %	2/3	level channels
7. PZR Hi Press.	2200 psig	2/3	press.channels
8. Overpower	25%,85%,109%	2/4	P.R. NIS
9. SF/FF Mismatch	SF>FF by 25%	2/3	Channels
10. ~~SI3/LOP~~ Sequencer	→(SIS/LOP, LOP, SIS, Manual)	1/2	Sequencers
11. Manual		1/2	Pushbuttons

*May be listed as one trip depending on assumptions concerning P-8 Status

(any 5 lines 0.6/line or 0.15/item)

* REFERENCE SD-SO1-570 and Requal exam bank 1166

***QUESTION 3.07**

Concerning the Auxilary Feedwater System (AFW):

a. What will Auto Start the AFW pumps?
(include in your answer coincidence logic
and setpoints). (1.0)

b. What will trip the AFW pumps ? (1.0)

* ANSWER

a. Low S/G Level 5% N.R. (0.5)
2/3 S/G's per train (0.5)
(ch.A =Motor driven ch.B = Turbine driven)

b. Low Suction Pressure 0.6 psig for 10 sec. on
* auto start - both pumps
* Ch.A - Low Disch Press. - ~~675~~ 500 psi (0.25)
Ch.B - Overspeed (0.25)

* Reference SD-S01-620 and requal exam bank

*** QUESTION 3.08**

For the Diesel Generators:

a. List the Engine trips that are in service on an
Emergency start. (0.5)

* ANSWER

a. * engine overspeed ~~(0.25)~~ (0.5)
~~* Generator differential (0.25)~~

* Reference Requal Exam 1175

****

End of Section 3
Go on to Section 4

Section 4

Procedures - Normal, Abnormal,
Emergency, and Radiological Control

***QUESTION 4.01**

Regarding SO1-1.4-1, 'Response to Imminent Pressurized Thermal Shock Condition': (1.0)

a) After termination of an overcooling transient that results in a Safety Injection actuation why is it important to maintain RCS temperature and pressure stable?

*ANSWER

a) To avoid potential pressure increase that could result in potential overstress of the pressure vessel. (2.0)

*REFERENCE

SO1-1.4-1, page 19

***QUESTION 4.02**

What are the Regulatory Limits for exposure for the following areas of the body in Rems per Quarter?

a) The hands and forearms. (0.5)

b) Skin of the whole body. (0.5)

*ANSWER 18.75

a) ~~18.125~~ Rems/Qtr. (0.5)

b) ~~18~~ 7.50 Rems/Qtr. (0.5)

The above limits are all that is required for full credit.

*REFERENCE

SO123-VII-4.0 page 6

***QUESTION 4.08**

Regarding SO1-2.1-9 "Loss of Residual Heat Removal System".During refueling operations with the Reactor Coolant System drained down the operating train of RHR experiences conditions that indicate a loss of flow in the system.

a) What are two symptoms (not Alarms) that could provide that indication? (2.0)

*ANSWER

a) Any two (1.0):
   1. Increasing RHR temperature (any temp. indicator).
   2. RHR pump fluctuating current.
   3. Decreasing RHR flow or Lo flow.
   4. Lack of temperature difference across RHR heat exchangers.

*REFERENCE

SO1-2.1-9, page 2.

***QUESTION 4.09**

Why must the Volume Control Tank pressure be maintained between 14 to 20 psig when the Reactor Coolant Pumps are in operation? (1.0)

*ANSWER

To ensure the lower side of No. 2 seals are maintained wet. (1.0)

*REFERENCE

SO1-4-3, PAGE 2.

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End of Section 4
End of Examination

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1983
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1984
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F.Y. 1985

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1985
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F.Y. 1986

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FISCAL YEARS

***QUESTION 4.06**

Regarding SO1-4-11, "Letdown Demineralizer Operations":

a) The procedure includes a precaution about placing an unsaturated mixed-bed demineralizer in service. What concern can arise if an unsaturated mixed-bed demineralizer is inadvertantly placed in service? (2.0)

*ANSWER
a) Unintentional dilution of boron (Speaks to 80 ppm dilution being anticipated.) (2.0)

*REFERENCE
SO1-4-11, page 2.

***QUESTION 4.07**

Regarding Operating Instruction OI-3-4 "Reactor Coolant Normal Operation":

a) What criteria must be met before the restart of any Reactor Coolant Pump after it has tripped? (1.0)

b) How many consecutive starts are allowed for a two (2) hour interval? (1.0)

*ANSWER
a) The pump must stand idle for at least 30 minutes. (1.0)
b) Three. (1.0)

*REFERENCE
SO1-4-3, page 10.

***QUESTION 4.03**

During the performance of a monthly surveillance test you notice that the previous test was conducted on December 1, 1985. Today is January 5, 1986. This is an interval of 36 days. The test was conducted previously on November 2, 1985 and October 1, 1985 (Calender on following page.)

a) Does this situation constitute non-compliance with the Technical Specifications? EXPLAIN. (2.0)

*ANSWER

a) No (0.8). The Technical Specification allow a surveillance interval to be extended by 25 percent (0.5). With the interval for 3 consecutive surveillances not to exceed 3.25 times the interval (0.5).

*REFERENCE

Technical Specifications 4.0.2 page 4-1.

***QUESTION 4.04**

Regarding 10 CFR 55 'Operators Licenses':

a) What are 'Controls as defined by 10 CFR 55? (1.0)

*ANSWER

a) Controls are defined as aparatus and mechanisms the manipulation (0.5) of which affect the reactor or reactor power (0.5).

*REFERENCE

10 CFR 55.4 AND 55.10.

***QUESTION 4.05**

Regarding SO1-1.0-40 "Steam Generator Tube Rupture":

a) Under what condition may the RHR System alignment be made when conducting the cooldown phase of the Steam Generator Tube Rupture procedure if the RCS temperature is greater than 350 deg. F.? (1.0)

b) What action must be taken immediately if the ruptured Steam Generator level is OFF SCALE HIGH or INCREASING RAPIDLY? (1.0)

*ANSWER

a) If the RCPs are running. (1.0)

b) Depressurize the RCS to approximately the main steam pressure. (1.0)

*REFERENCE

SO1-1.0-40

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR REQUALIFICATION EXAMINATION

Facility: SONGS 1
Reactor Type: WESTINGHOUSE - 3 LOOP PWR
Date Administered: NOVEMBER 18, 1986
Examiner: J. P. O'BRIEN
Candidate: KEY

INSTRUCTIONS TO CANDIDATE

Read the attached instruction page carefully. This examination replaces the current cycle facility administered requalification examination. Retraining requirements for failure of this examination are the same as for failure of a requalification examination prepared and administered by your training staff. 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 four (4) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value	Category
15.0	25.0			5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
15.0	25.0			6. Plant Systems Design, Control and Instrumentation
15.0	25.0			7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
15.0	25.0			8. Administrative Procedures, Conditions, and Limitations
60.0				TOTALS

Final Grade

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

Candidate's Signature

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

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

$Pwr = W_f \dot{m}$

$P = P_o\ 10^{SUR(t)}$

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

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

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

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

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

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

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

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

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

Cycle efficiency = $\frac{\text{Net Work (out)}}{\text{Energy (in)}}$

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

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

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

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

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

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

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

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

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

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

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

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1$ seconds^{-1}

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

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

$R/hr = 6\ CE/d^2$(feet)

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

1 ft. H_2O = 0.4335 lbf/in^2

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 Btu = 778 ft-lbf

1 inch = 2.54 cm

$^oF = 9/5\,^oC + 32$

$^oC = 5/9\ (^oF - 32)$

Section 5

Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics

*QUESTION 5.01

The reactor is determined to be shutdown by 6 percent delta K/K with indication in the source range of 30 counts per second.

a. What is the Keff when the reactor is shutdown by 6 percent delta K/K? (0.5)

b. What would the count rate be if Keff is increased to 0.98? (0.5)

c. What would the count rate be if Keff is increased to 0.99? (0.5)

*ANSWER

delta K / K = 0.06

1 - K / K = 0.06

1 = K + 0.06(k)

1 = 1.06(K)

a. KEFF = 1/1.06 = 0.94 (0.5)

1 - K1 / 1 - K2 = CR2 / CR1 and 0.06 / 0.02 = CR2 / CR1

b. CR2 = (30)(3) = 90 cps (0.5)

Conversly CR3 / 30 = 0.06 / 0.01

c. CR3 = (30)(6) = 180 cps (0.5)

(Rule of thumb, doubles)

*REFERENCE

Glastone and Sesonske, "Nuclear Engineering".

*QUESTION 5.02
Refer to FIGURE 5.1, a sketch of a typical auxiliary feedwater system utilizing two centrifugal pumps of similar characteristics and capcities. The plot of Volume Flow Rate versus Pressure shows the system with the "A" auxiliary feed pump in operation as the initial condition.

a. Show, on the Figure provided, how the curve(s) will change as the PORV opens and reduces Steam Generator pressure by 50 percent. (1.0)

b. Show, on the Figure provided, how the curve(s) will change when the discharge valve for the "A" pump is partially shut. (1.0)

*ANSWER
a. Attached (1.0)
b. Attached (1.0)

*REFERENCE
Section III, Part B, General Physics manual on Heat Transfer and Fluid Flow.

*QUESTION 5.03
Refer to the FIGURE 5.2 that follows this page. The figure is of "Heat Flux" versus "Temperature Difference between a Wall and the Bulk Fluid" for an operating reactor.Note that there are two curves represented for two pressures(P1 less than P2).

a. What is the principle type of heat transfer that is occuring at pressure P1 and 1.0E4 BTU/Hr-ft between the wall and the bulk fluid? (0.5)

b. What is the principle type of heat transfer that is occuring at Pressure P2 and 1.0E4 BTU/Hr-ft between the wall and the bulk fluid? (0.5)

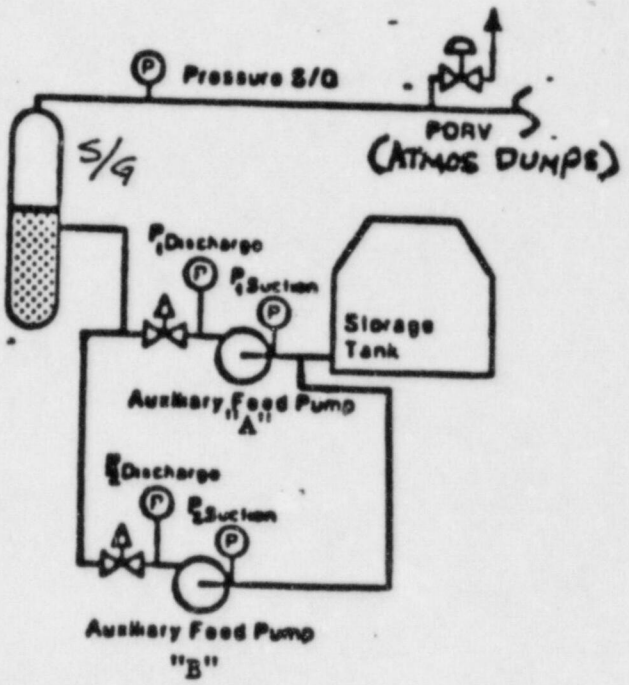
c. What pressure will yield a lower fuel centerline temperature at 3.0E5 BTU/HR-FT? (0.5)

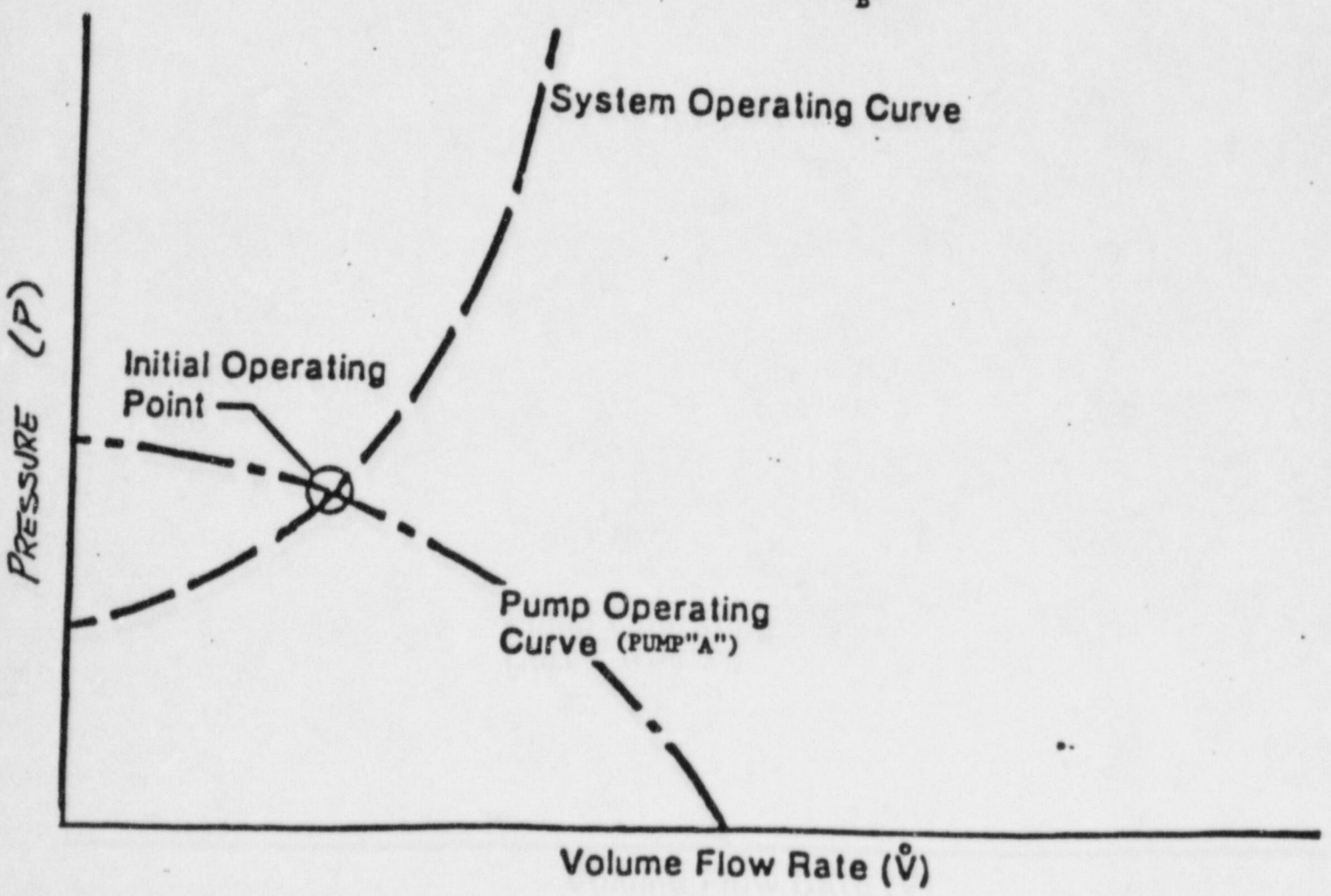
d. Assuming bulk temperature well below saturation, Will decreasing the pressure affect the bulk fluid temperature at a heat flux of 3.0E5 BTU/Hr-ft? (0.5)

*ANSWER
a. Nucleate boiling. (0.5)
b. Convection (other terms may be used). (0.5)
c. Pressure 2. (0.5)
d. No acceptable. The bulk fluid temperature remains constant (is independent of pressure below saturation.) (0.5)

*REFERENCE
General Physics Nuclear Technology, Section E, Pages 2-144, 2-151- 2-159, and 2-164.

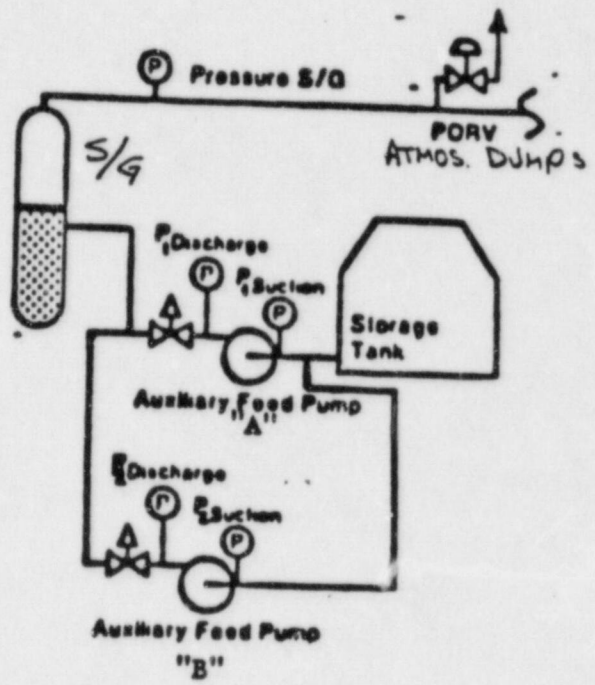
FIGURE 5.1

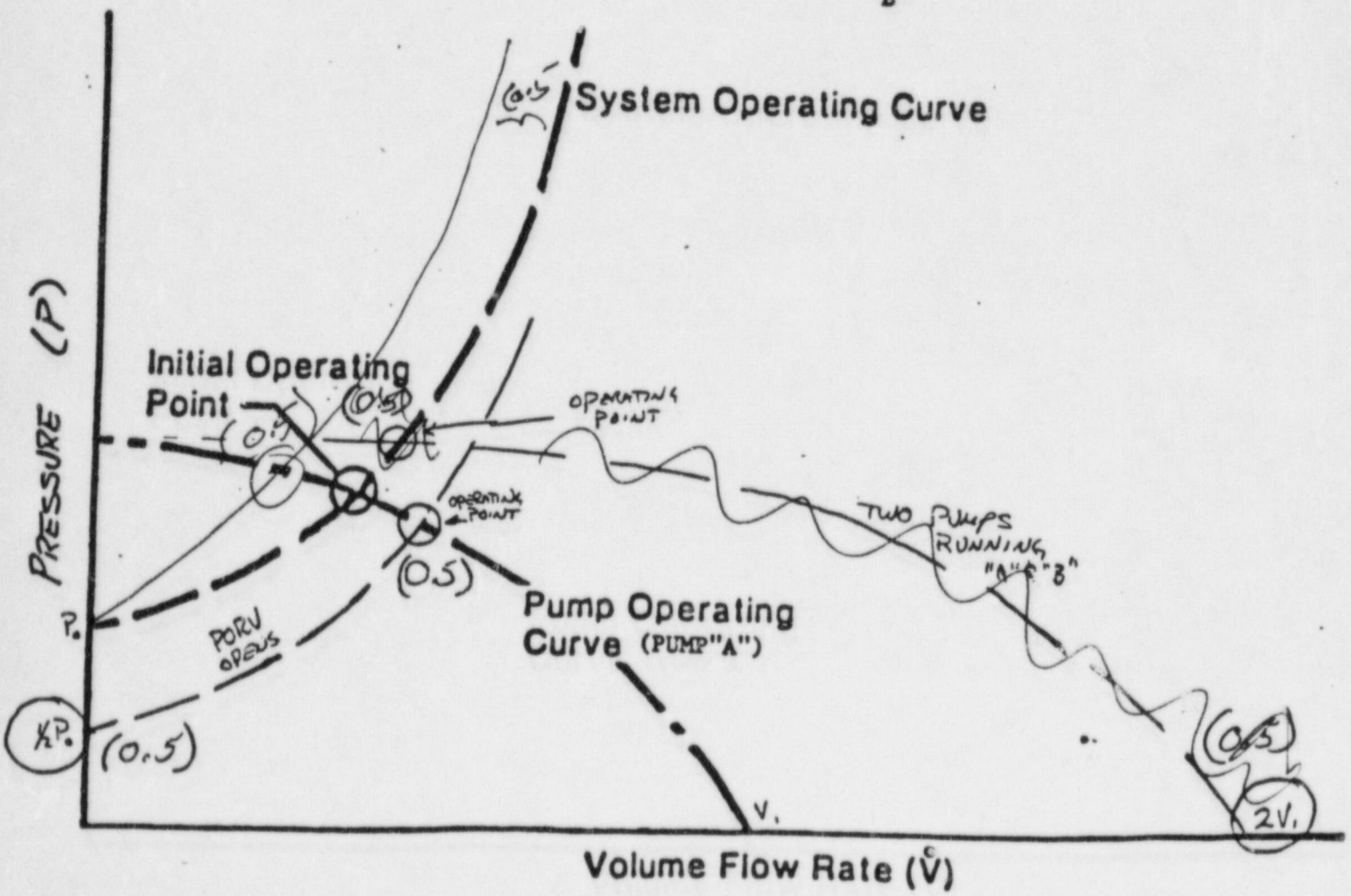

Pressure S/G
PORV
(ATMOS DUMPS)
S/G
Storage Tank
Auxiliary Feed Pump "A"
Auxiliary Feed Pump "B"


System Operating Curve
Initial Operating Point
Pump Operating Curve (PUMP "A")
PRESSURE (P)
Volume Flow Rate (V̊)

KEY

FIGURE 5.1





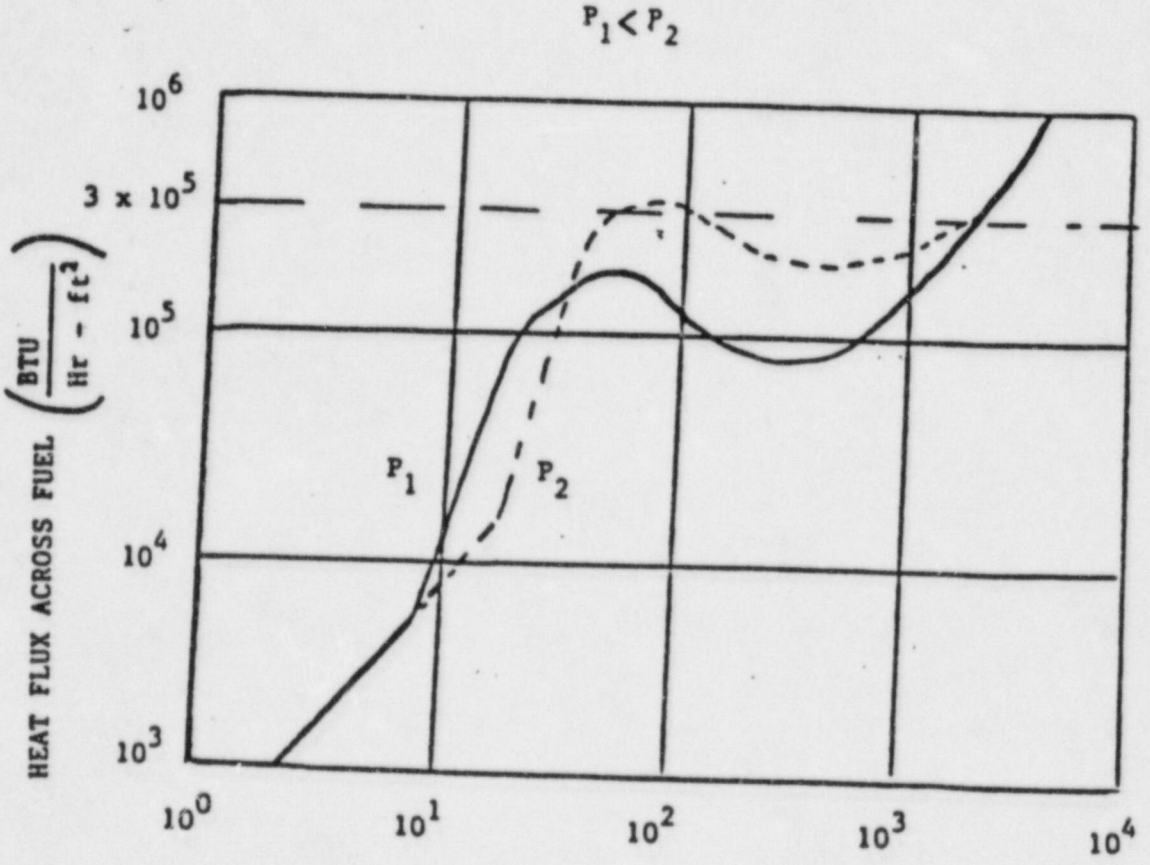

$P_1 < P_2$
10^6
3×10^5
10^5
10^4
10^3
HEAT FLUX ACROSS FUEL $\left(\frac{BTU}{Hr - ft^2}\right)$
P_1
P_2
10^0
10^1
10^2
10^3
10^4
Temperature Difference Between Wall and Bulk Fluid (F^o)

FIGURE 5 -2

*QUESTION 5.04
Regarding a Main Steam line rupture:

a. Why is a rupture of a Main Steam line at End of Life (EOL) a much more limiting accident than at the Beginning of Life (BOL)? (1.0)

*ANSWER
a. The Moderator Temperature Coefficient is less negative at BOL (~~0.~~0). Than at EOL. This difference in magnitude increases the severity of the addition of positive reactivity (~~0.~~) due to the sudden cooling of the reactor coolant from the Main Steam line rupture. (~~.33~~)

*REFERENCE
General Physics, Volume III, Chapter 3, Section 3.

*QUESTION 5.05
Periodically chemical agents are added to the Reactor Coolant System (RCS). For each of the following chemicals added to the RCS, indicate why the chemical is added and (what is it designed to control):

a. Hydrazine (0.5)

b. Hydrogen (0.5)

c. Lithium Hydroxide (0.5)

*ANSWER
a. Control of Oxygen (at low temperature). (0.5)
b. Control of oxygen (at power). (0.5)
c. Control of pH in RCS. (0.5)

*REFERENCE
System Description "Chemical and Volume Control System"

*QUESTION 5.06
Figure 5.5 is a sketch of Reactor Power versus Time in hours. At t=0 hours reactor startup from Xenon free conditions to 100 percent power occurs. At t=50 hours a reactor trip occurs followed by a reactor startup to 100 percent power at t=65 hours.

a. Sketch the Xenon reactivity response in the core from this power transient. (Indicate approximate magnitude [percent delta k/k] and duration of each part of the transient.) (1.5)

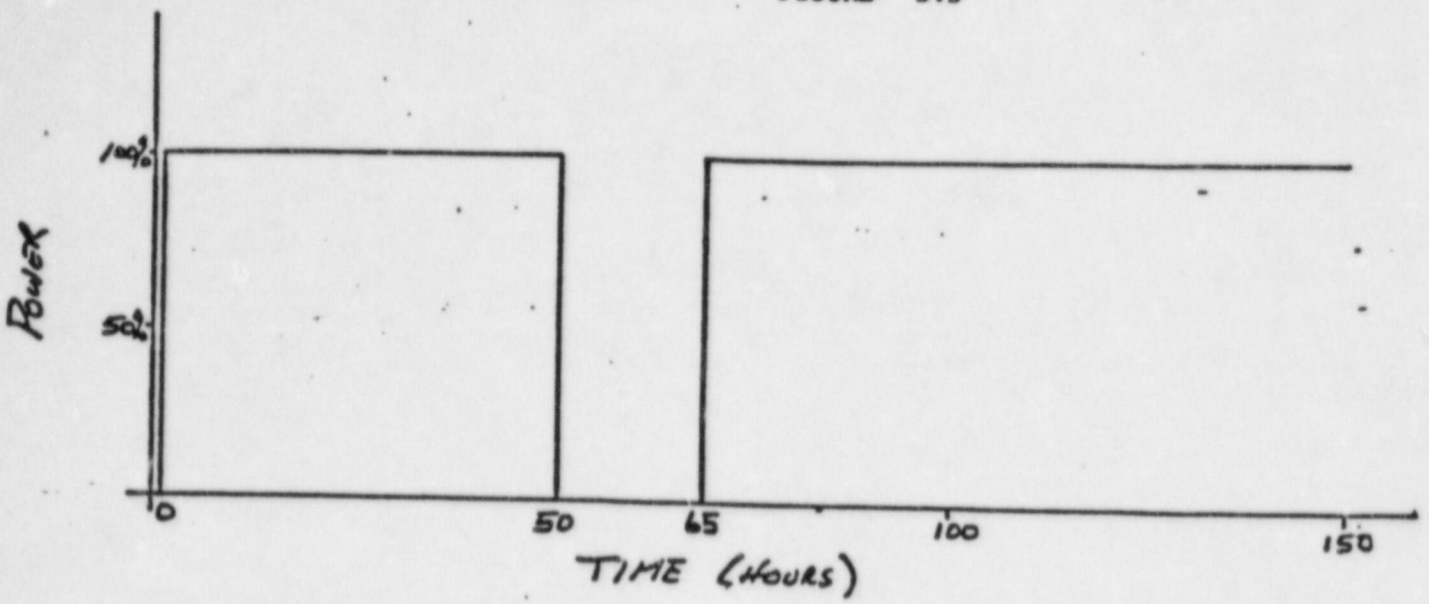
b. At what time in hours will the maximum rate of rod insertion have to occur in order to overcome the Xenon transient. (Assume constant Tavg and no boration or dilution.) (0.5)

c. What are the production and removal mechanisms for Xenon? (1.0)

*ANSWER
a. (Attached) (1.5)
b. 65 to 70 hours (0.5)
c. Xenon: Production - Iodine decay (.25)
Direct yield from fission (.25)
Decay (.25)
Burnout (.25)

*REFERENCE
"Westinghouse Training Notes", 'Introduction to PWR Control' and 'Fission Product Poisoning'.

FIGURE 5.5


POWER
100%
50%
0
50
65
100
150
TIME (HOURS)

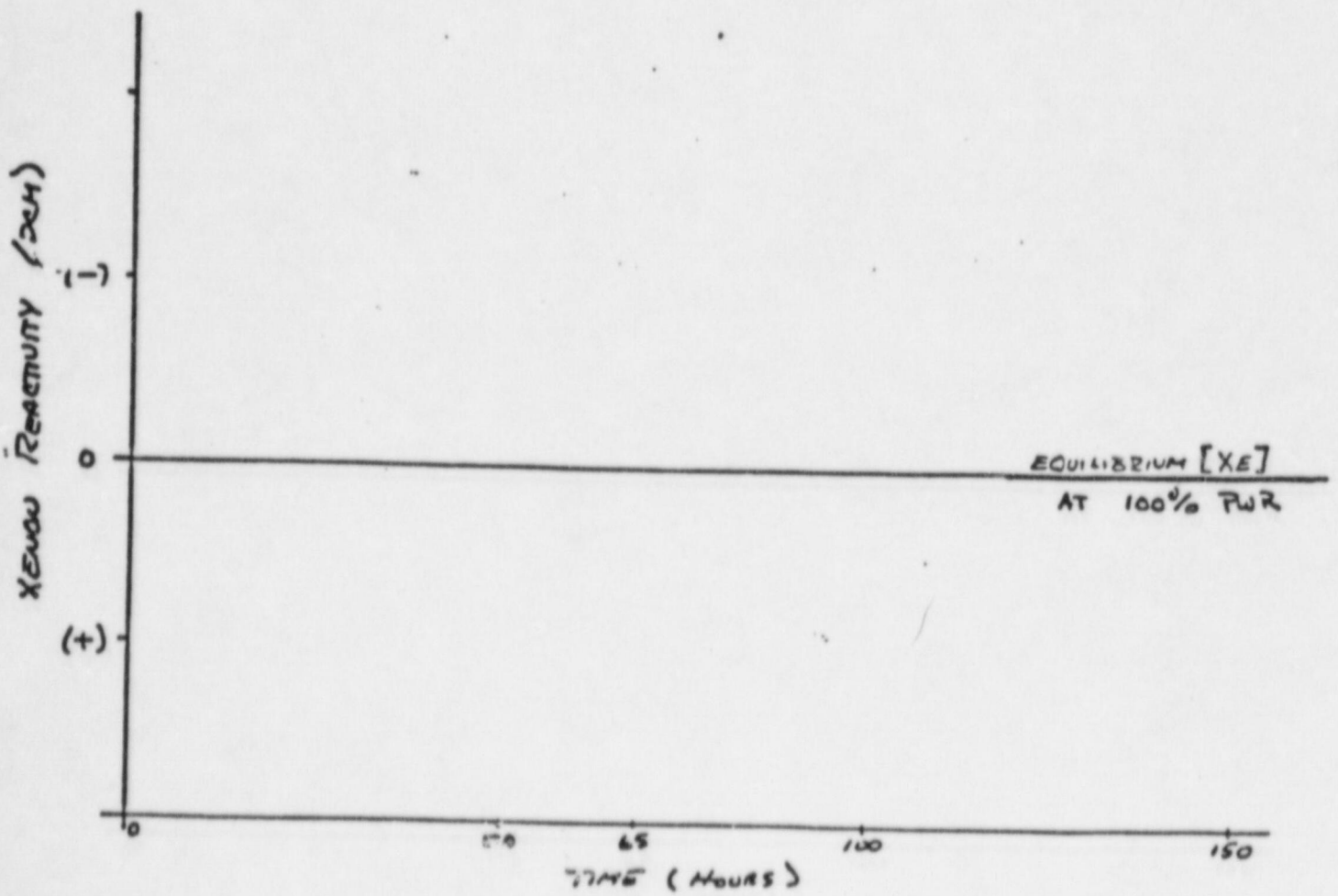
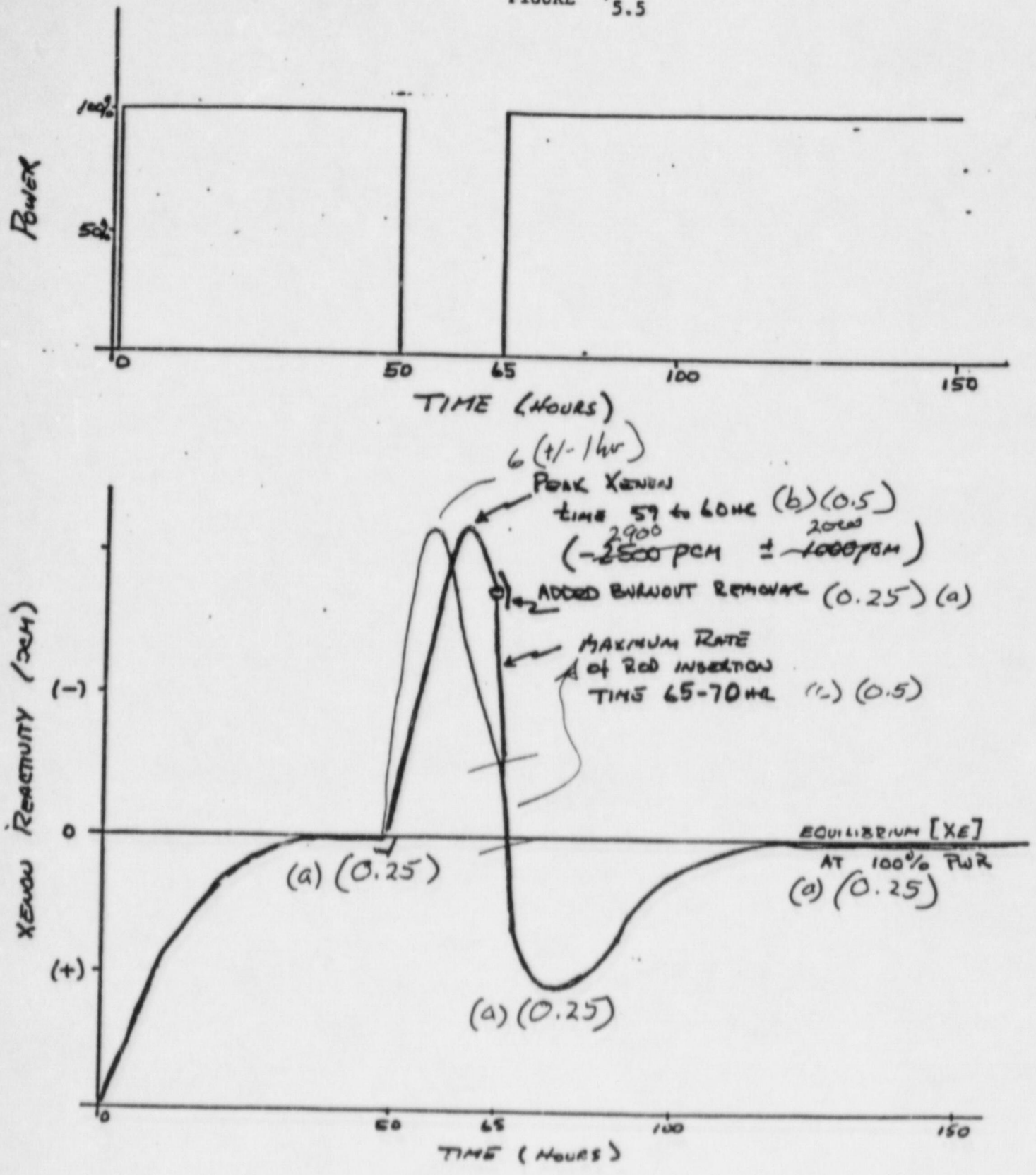

XENON REACTIVITY (ΔK/K)
(-)
0
(+)
EQUILIBRIUM [XE]
AT 100% PWR
0
50
65
100
150
TIME (HOURS)

FIGURE 5.5


POWER
100%
50%
0
50
65
100
150
TIME (HOURS)
6 (+/- 1hr)
PEAK XENON
time 59 to 60HR (b)(0.5)
(~~2500~~ 2900 PCM ± ~~1000~~ 2000PCM)
ADDED BURNOUT REMOVAL (0.25) (a)
MAXIMUM RATE
of ROD INSERTION
TIME 65-70 HR (b) (0.5)
XENON REACTIVITY (%Δk)
(-)
0
(+)
EQUILIBRIUM [XE]
AT 100% PWR
(a) (0.25)
(a) (0.25)
(a) (0.25)
0
60
65
100
150
TIME (HOURS)

*QUESTION 5.07
For the following indicate which is the most accurate statement:

a. The main condenser functions by: (0.5)

1. Removing the latent heat of vaporization at a constant temperature to allow the steam to condense.

2. Providing a low pressure volume that allows the steam to condense.

3. Cooling the steam to the point where is at satuaration temperature.

b. Condensate depression is: (0.5)

1. Maintained by adequate backpressure on the low pressure turbine exhaust.

2. Used to maintain constant temperature profile on the condenser tubesheets.

3. Used to maincain adequate Net Positive Suction Head for the condensate pumps.

4. Used to maintain adequate Net Positive Suction Head for the main feedwater pumps.

*ANSWER
a. 1. (0.5)
b. 3. (0.5)

*REFERENCE
General Physics "Heat Transfer and Fluid Flow"

*QUESTION 5.08
At Beginning of Life (BOL) the plant nuclear instrumentation count rate will increase as plant temperature increases causing a Reactor Coolant System (RCS) coolant density decrease.

a. Assuming there is no rod movement or changes in RCS boron concentration. What two factors contribute to this phenomena? (1.0)

*ANSWER
a. 1. Neutrons travel farther, more leakage. (0.5)
2. Lower boron density, less boron absorpiton. (0.3)

*REFERENCE
General Physics, Volume II, Chaper 4, Sections B & D

*QUESTION 5.09
Over the life of a core, the composition of the fuel changes.

a. How and why does this change in composition over core life affect the reactor period for equal reactivity additions? (1.0)

*ANSWER
a. The decrease in Uranium 235 and the increase in Plutonium 239 cause the Beta Bar Effective to decrease. (0.5)
This Beta Bar Effective change causes the reactor period to decrease. (0.5)

*REFERENCE
General Physics, Volume II, Chapter 5, Sect. E.

*QUESTION 5.10
Regarding the feedwater heaters:

If the level in a feedwater heater is allowed to increase, what will happen to the temperature of the feedwater exiting the heater, increase, remain the same, or decrease? Why? (1.0)

*ANSWER
Decrease (0.5). The level increase exposes less surface area in the heater tubes. There is less latent heat removed from the steam, and therefore less heat transfered to the feedwater resulting in a lower exit temperature for the feedwater. (0.5)

*REFERENCE
General Electric "Thermodynamics, Heat Transfer, and Fluid Flow", Chapter 8.

End of Section 5

CONTINUE NEXT PAGE

SECTION 6

PLANT SYSTEM DESIGN, CONTROL AND INSTRUMENTATION

* QUESTION 6.01

Concerning the Feed System and Safety Injection System:

a. What pump and valve changes will automatically occur in the Feed and Condensate system when, at full unit load, a Safety Injection signal is received ? (2.0)

b. What specific valve interlock prevents the injection of Feed and Condensate water into the RCS ? (0.5)

c. If during the SIS sequence the SI pump suction isolation valve (HV-853 A&B) fails to open, will the feedwater pump run ? Explain ! (0.5)

* ANSWER

a.
- Main feed pump trips and then restarts (0.5)
- Condensate pumps trip (0.25)
- Heater Drain pumps trip (0.25)
- MFP suction valves (HV-854 A&B) shut (0.25)
- MFP discharge valves (HV-852 A&b) shut (0.25)
- Feedflow control valves (FCV-456/7/8) shut (0.25)
- Feedflow Aux control valves (CV-142/3/4) Shut (0.25)
- (SIS Suction Header Isolation Valves (HV-851 A&B) will open.) (N/A)

b. MFP Suction valves (HV-854 A&B) must be shut before the SIS header isolation valves (HV-851 A&B) will open. (0.5)

c. Yes, the FW pump will run for 30 sec. then if HV-853 does not open, the pump will trip. (0.5)

reference: SD-SO1-580, pp.16, SO1 exam bank.

→ Feed water isolation valves MOV's 20, 21, 22 shut.

* QUESTION 6.02

Concerning the CVCS system:

a. Under what three conditions would the Excess Letdown System be Used ? (1.5)

b. Describe the three Excess Letdown Alignments. (1.5)

* ANSWER

a.1. inoperability of normal letdown (0.5)
2. used during PZR Steam Bubble formation (0.5)
3. To increase the cooldown rate during the initial cooldown phase of RHR operation. (0.5)

b.1. Normal - thru seal water return piping into the VCT. (0.5)
2. Thru a section of normal letdown piping and a residual heat exchanger to the VCT. (0.5)
3. Thru and then directly to the RCS drain tank. (0.5)

*REFERENCE: SD-S01-310 pp.5,fig. I-1.

*QUESTION 6.03

What is the purpose of the Cryogentic Waste Gas Treatment System? (1.0)

*ANSWER

To Cryogentically (low temp. & Charcoal)remove and store Xe and Kr from the Radioactive waste gas prior to release. (1.0)

*REFERENCE
SD-S01-530

* QUESTION 6.04

Which of the following is true concerning the Source Range Channel high voltage cutoff? (1.0)

a. During a reactor startup either IR channel increasing above the P-6 setpoint will turn off the high voltage.

b. If one IR channel fails low while at power, both Source Range high volatage will be reenergized.

c. If a IR channel increases above 2 X 10 E(-9), it will turn off its respective source range's high voltage.

d. During a reactor shutdown either IR channel decreasing below the P-6 setpoint will turn on the high voltage.

*ANSWER
c. (1.0)

*REFERENCE
SD-S01-380 Excore NI's

*QUESTION 6.05

a. List the sources of air that can be used to supply the Instrument Air Header? (1.5)

b. How are the following valves affected by the loss of Instrument air (operable/inoperable) ? If operable, state why? If inoperable, state status: (failed shut, failed as is, or failed open). (1.5)

1. Main feedwater flow control valves
2. Emergency Auxilary feedwater regulators
3. Safety injection valves HV851,852,853,854

*ANSWER

a.1. Three Service Air Compressors (0.5)
2. Auxilary Air Compressor (0.5)
3. Diesel Air Compressor (0.5)

b.1. Inoperable (0.25) - Failed open (0.25)
2. Operable (0.25) - Nitrogen Backup (0.25)
3. Inoperable (0.25) - alarmed in the controlroom, Operable if operator selects Nitrogen backup (0.25)

* Reference: SD-S01- 420 & 580 ; S01-2.4-2 IA Sys Malfunction

* QUESTION 6.06

a. What are the control signal inputs to the Automatic Rod Control System? (1.0)

b. For each of the above, State if the input controls rod speed and/or rod direction. (1.5)

*ANSWER

a.1. T avg	(0.25)	b.1. Direction and speed	(0.5)
2. T ref	(0.25)	2. Direction and speed	(0.5)
3. Nuclear Flux	(0.25)	3. Speed	(0.25)
4. P-P ref	(0.25)	4. Speed	(0.25)

REFERENCE
SD-SO1-400

* QUESTION 6.07

Pressurizer Pressure Transmitter PT-430 fails low during normal at power operations.

a. Name <u>one</u> alarm that you expect to annunciate in the control room as a direct result of the PT failure ? (0.5)

b. What <u>pressurizer</u> systems or controls will be affected and how will they respond? (1.0)

* ANSWER

a. Safety Injection Train A Channel 1 Alarm or VLPT Channel 1 Alarm (0.5)

b. 1. Spray Valves will close if they are open (0.5)
2. All heaters in Automatic will turn on (0.5)

* Reference Requal Exam Bank

End of Section 6

CONTINUE NEXT PAGE

SECTION 7

PROCEDURES: NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL

* QUESTION 7.01

SONGS 1 Emergency Procedures utilize a " Critical Safety Function Status Tree" (CSFST) concept.

a. List the six (6) CSFST's in order of priority and briefly describe what parameters are checked for each. (3.0)

* ANSWER

a. Priority	Type	Parameter
1	Subcriticality	Checks power levels on the PR, IR and SR
2	Core Cooling	Checks core exit TC's and Hot Leg RTD's
3	RCS Integrity	Checks cooldown rates and P/T limits
4	Heat Sink	Checks primary and secondary heat removal prameters (Aux Feed and Main Feed)
5	Containment	Checks Containment pressure, sump level, and radiation levels.
6	RCS Inventory	Checks Pressurizer level.

priority: (0.1ea.) Type: (0.3 ea.) parameters: (0.1 ea.)

* REFERENCE SO1-1.0-1

* QUESTION 7.02

TRUE or FALSE:
On your first pass through the CSFST's, you note an "orange" priority 3 condition and a "red" priority 5 condition. In Accordance with SO1-1.0-1, you would address the "orange" priority 3, first, because it is of higher priority.
TRUE or FALSE? (1.0)

* ANSWER

FALSE (1.0)

* REFERENCE SO1-1.0-1

* QUESTION 7.03

Per SO1-14-17, What are the three of four actions that should be taken when a gagging device is installed on a relief or safety valve ? (3.0)

* ANSWER

1. Document the valve gagged. (Safety Related valve or Component Control Form for Safety Related Valves, CO's Log, Procedural Steps, Etc.)

2. Initiate a maintenance order, if the gag os installed due valve failure, and hang a defiency tag on the gagged valve.

3. Evaluate the affected system or component for an operability assessment.

4. Notify engineering for a techinical evaluation to determine if continued operation is acceptable.

(3 of 4 required, 1.0 ea.)

* REFERENCE SO1-14-17

* QUESTION 7.04 (3.0)

Fill in the following blanks in accordance with SO1-3-1:

a. Maximun Delta T between the Pressurizer liquid and the Reactor Coolant is _____? (0.5)

b. Fully withdrawn position of all control rods shall be _____? (0.5)

c. RCS Pressure and Temperature should not exceed: _____psig (0.5) and _____ degrees F (0.5) when RHR system is in service.

d. Maximum heatup rates are _____ for the RCS, and (0.5) _____ for the Pressurizer. (0.5)

* ANSWER

a. 190 degrees F (0.5)
b. 318 (0.5)
c. 400 (0.5), 400 (0.5)
d. 60 deg./hr, (0.5) 90 deg./hr (0.5)

* Reference

* QUESTION 7.05

The plant is being started up from Hot Standby to Minimum Load.

a. What rod position are procedurally required for criticality unless otherwised directed by the Shift Superintendent ? (0.5)

b. When, according to the procedure, should critically be anticipated ? (0.5)

c. What is the limit on Startup Rate duringthe startup? (0.5)

d. What is the limit on turbine backpressure prior to rolling the turbine ? (0.5)

* ANSWER

a. ~~Shutdown Banks 1 and 2 and Control Bank 1 at 310 Steps~~. Control Bank 2 at 100 steps (0.5)

b. At any time when control rods are being withdrawn, or when boron dilution is in progress. (0.5)

c. less than 1.0 DPM (0.5)

d. 5.5' Hg (0.5)

* REFERENCE SO1-3-2

* QUESTION 7.06

In accordance with SO1-3-1, Plant Startup from cold S/D to Hot STBY, a steam bubble being formed in the pressurizer is indicated when ? (1.0)

* ANSWER

a. Charging flow is less than letdown flow with RCS pressure constant (1.0)

* Reference SO1-3-1

* QUESTION 7.07

Answer the following concerning Cold Leg Injection and Recirculation:

a. At what RWST level is it that the Cold Leg Injection and Recirculation Procedure is entered? (0.5)

b. How long after reaching the RWST level is it that the operator has to reset SI ? (0.5)

c. At what RWST level must the cold leg recirculation path be completed by, (0.5)
and what are the consequences if the alignment is not completed by this level ? (0.5)

* ANSWER

a. 21 %. (0.5)

b. 30 seconds (0.5)

c. 7 % (0.5)

At 7 % level adequated NPSH to the charging pumps cannot be assured (0.5) possible formation of vortices could result and air entrainment of the pumps.)

* REFERENCE SO1-1.0-23

End of Section 7

CONTINUED ON NEXT PAGE

Section 8

Administrative Procedures,Conditions, and Limitations

*QUESTION 8.01
TRUE or FALSE.

Technical Specification 3.5.3 does not require any action if one control rod is immovable, provided the immovable rod is within + or - 35 steps of it's group step counter demand position. (1.0)

*ANSWER
False. (1.0)

*REFERENCE
Technical Specification 3.5.3

*QUESTION 8.02
During MODE 3:

a. How many Reactor Coolant Loops must be in operation? (0.5)

b. How many Reactor Coolant Loops must be operable? (0.5)

*ANSWER
a. 1 (0.5)
b. 2 (0.5)

*REFERENCE
Technical Specification 3.1.2

*QUESTION 8.03
Each RPS Pressurizer Pressure channel is required by Technical Specifications to undergo a channel check on a shiftly basis (at least once every 12 hours). Some extensions of the basic interval are allowed by the Technical Specifications.

Records show that this was done on:

November 27 at 0000
November 27 at 1500
November 28 at 0000
November 28 to 1400
November 29 at 0500

a. What is the maximum allowable interval between channel check surveillances? (1.0)

b. When is the next channel check surveillance due? (1.0)

*ANSWER
a. 12 hours + 25 percent = 15 hours (1.0) ~ will accept 1.25 x interval (12hrs)
b. 1. Last done + 15 hours = Nov 29 at 2000
2. Last 3 + 3.25 x 12 hours =
Nov 28 @ 0000 + 39 hours = Nov 29 @ 1500

Therefore IT must be done no later than: Nov 29 @ 1500 (1.0)

*REFERENCE
Technical Specifications 4.02

*QUESTION 8.04
When a fire suppression Spray/Sprinkler System is declared inoperable for a portion that protects an area containing redundant safety-related equipment, the required action is to: (pick one) (1.0)

a. commence a Unit shutdown within one hour.

b. establish an hourly fire patrol for the affected area.

c. establish a continous fire watch with backup fire suppression equipment in the affected area within one hour.

d. establish a backup suppression system in one hour or be in HOT SHUTDOWN within 12 hours.

*ANSWER
c. (1.0)

*REFERENCE
Technical Specification 3.14

* QUESTION 8.05
Station Order SO123-O-14 Attachment 4 contains certain Operations Reporting Requirements:

a. State three conditions of which the Unit 1 Superintendent (or his designee) and the on-duty Shift Technical Advisor must be notified by the Shift Supervisor as soon as practical. (1.5)

*ANSWER
1. An unplanned trip or load reduction.
2. An unplanned or unexplained major system or component failure.
3. Any significant event which cannot be corrected by operators, or any abnormal operating event, which, in the judgement of the Shift supervisor, warrants notification of the Unit 1 superintendent and/or on-duty shift Technical Advisor.
4. Any event that would be reported as a Reportable Occurance.
5. Any item listed in Attachment 4.
(Any three (0.5) each.)

*REFERENCE
~~SO1-O-100 para. IV.J , and (SO123-O-14~~ SO123-O-14 Attach. 4.

*QUESTION 8.06
The plant is in MODE 4, preparing for a routine plant startup. As a result of routine surveillance, it is determined that only one Diesel Generator Fuel Transfer pump is operable. The Maintenance Supervisor has told you, that the cause of the failure is known, parts are on site, and that repairs and operability checks will be completed in less than 48 hours. Explain why the startup can or cannot proceed. (See enclosed Technical Specifications.) (1.5)

*ANSWER
The startup cannot proceed, (0.5) because "entry into an Operating Mode is not allowed if an action statement must be relied upon to do so. (1.0)
(The plant can remain in MODE 4, but cannot enter MODE 3.)

*REFERENCE
Technical Specification 3.0.4 and Technical Specification 3.7

3.7 AUXILIARY ELECTRICAL SUPPLY

APPLICABILITY: Applies to the availability of electrical power for the operation of the plant auxiliaries.

OBJECTIVE: To define those conditions of electrical power availability necessary (1) to provide for safe reactor operation, (2) to provide for the continuing availability of engineered safeguards, and (3) to ensure that the station can be maintained in the shutdown or refueling condition for extended time periods.

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SPECIFICATION:

I. In Modes 1, 2, 3 and 4 the following specifications shall apply:

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A. As a minimum the following shall be OPERABLE:

1. One Southern California Edison Company and one San Diego Gas & Electric Company high voltage transmission line to the switchyard and two transmission circuits from the switchyard, one immediate and one delayed access, to the onsite safety-related distribution system. This configuration constitutes the two required offsite circuits.

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2. Two separate and independent diesel generators each with:

   a. A separate day tank containing a minimum of 290 gallons of fuel,

   b. A separate fuel storage system containing a minimum of 37,500 gallons of fuel, and

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   c. A separate fuel transfer pump.

3. AC Distribution

   a. 4160 Volt Bus 1C and 2C,

   b. 480 Volt Bus No. 1, Bus No. 2 and Bus No. 3, and

   c. Vital Bus 1, 2, 3, 3A, 4, 5 and 6.

4. DC Bus No. 1 and DC Bus No. 2 (including at least one full capacity charger and battery supply per bus).

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5. The two Safety Injection System Load Sequencers.*

* The automatic load function may be blocked in Mode 3 at a pressure ≤ 1900 psig.

Revised: 11/28/84

B. Action

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1. With one of the required offsite circuits inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Periodic Testing Requirements A and B.1.a of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter; restore an additional offsite circuit to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

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2. If one diesel generator is declared inoperable, demonstrate the OPERABILITY of the two offsite transmission circuits and the remaining diesel generator by performing Periodic Testing Requirements A and B.1.a of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter; restore the inoperable diesel generator to service within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

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3. With one offsite circuit and one diesel generator of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Periodic Testing Requirements A and B.1.a of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in COLD SHUTDOWN within the next 36 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

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4. With two required offsite circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Periodic Testing Requirement B.1.a of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 4 hours. With only one of the required offsite circuits restored, restore the remaining offsite circuit to OPERABLE status within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

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5. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite circuits by performing Periodic Testing Requirement A of Technical Specification 4.4 within one hour and at least once per two (2) hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore both diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

6. With less than the above complement of AC buses OPERABLE, restore the inoperable bus within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

7. With one required DC bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. 84 11/14/84

8. With a required DC bus battery and both of its chargers inoperable, restore the inoperable battery and one of its chargers to operable status within 2 hours or be in cold shutdown within the next 36 hours.

9. With one Safety Injection Load Sequencer inoperable, restore the inoperable sequencer to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

II. Additionally, in Modes 1, 2 and 3 the following specifications shall apply:

A. As a minimum, the following shall be OPERABLE:

1. The MOV850C Uninterruptable Power Supply (UPS).

B. Action

1. With the MOV850C UPS inoperable, restore the UPS to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Revised: 11/28/84

*QUESTION 8.07
Provide the minimum number of individuals required by Technical Specifications for the following positions to operate the plant at full power (mode 1):

a. ------ Senior Operating Licenses (SRO)
b. ------ Operating Licenses (RO)
c. ------ Non-licensed Persons (NPEO)
d. ------ Shift Technical Advisors (STA)

*ANSWER
a. 2 (0.5)
b. 2 (0.5)
c. 2 (0.5)
d. 1 (0.5)

*REFERENCE
Table 6.2.1 and Technical Specifications 6.2.2

*QUESTION 8.08
Deviations from a procedure during normal plant operations:

a. are not allowed.

b. can be made if the original intent of the procedure is satisified.

c. can be made with verbal approval of two licensed operators. (one of which is licensed as a SRO).

d. can be made only in an emergency when immediate action is needed to protect the public health and safety.

*ANSWER
d. or b. (1.0) a. will accept -0.25
* For normal procedure B would be more correct.

*REFERENCE
Technical Specification 6.8.1

*QUESTION 8.09
TRIP/TRANSIENT REVIEWS per SO123-0-25 are performed by:

a. Any personnel licensed by the NRC.

b. Shift Supervisor and the Shift Technical Advisor.

c. Shift Technical Advisor.

d. Shift Technical Advisor, only if they hold a valid SRO license. (1.0)

*ANSWER
b. (1.0)

*REFERENCE
SO123-0-25 page 3.

*QUESTION 8.10
EPIP SO123-VIII-10 defines the Emergency Coordinator as the person who is designated to take charge of all emergency control measures, and has ultimate authority over all onsite activities and personnel.

a. Who initially fills this role in the early stages of a event? (0.5)

b. Who relieves this first person and has the primary responsibility for this role? (0.5)

*ANSWER
a. the Shift Superintendent (0.5)
b. the first Recall EC (0.5)

*REFERENCE
SO123-VIII-10

*QUESTION 8.11
Which of the following is NOT required of the A.C. and D.C. Electrical power sources by Technical Specifications in mode 5 and 6? (1.0)

a. Two circuits between the offsite transmission network and onsite ESF Electrical System.

b. One operable Diesel generator set.

c. One energized and operable 125-volt D.C. bus aligned to its associated charger.

d. One energized and operable 480-volt AC bus.

*ANSWER
a. (1.0)

*REFERENCE
Technical Specification 3.7

*QUESTION 8.12
If control power is lost to a pressurier power operated relief valve (PORV) while in Mode 1, to continue to operate:

a. no action is required by Technical Specifications provided another PORV is operable and all pressurizer code safety valves are operable.

b. Technical Specifications require the associated block valve to be verified open and then its power supply to be removed, if the PORV is not made operable within one hour.

c. Technical Specifications require the associated block valve to be shut and maintained in the closed position if the PORV is not made operable within one hour.

d. Technical Specification requires action to be initiated within one hour to place the plant in at least HOT STANDBY within the following 6 hours, if the PORV is not made operable.

*ANSWER
c. (1.0)

*REFERENCE
Technical Specification 3.1.5

END OF SECTION 8

END OF THE EXAM

**********BE SURE TO SIGN THE FRONT COVER SHEET**********