U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT
(OPERATOR LICENSING COMBINED EXAMINATION/INSPECTION REPORT)

EXAMINATION REPORT NO. 50-223/85-01 (OL)

FACILITY DOCKET NO. 50-223

FACILITY LICENSE NO. R-125

LICENSEE: University of Lowell
1 University Avenue
Lowell, Massachusetts 01854

FACILITY: Lowell Technology Institute Reactor

EXAMINATION DATES: March 11 - 12, 1985

PREPARED BY: Noel F Dudley 4-23-85
Lead Reactor Engineer (Examiner) Date

REVIEWED BY: RM Keller 4/23/85
Chief, Reactor Projects Section 1C Date

APPROVED BY: Harry B. Kister 4/24/85
Chief, Projects Branch No. 1 Date

SUMMARY: Two examinations were administered and one RO license was issued.

REPORT DETAILS

1. TYPE OF EXAMS: Replacement

2. EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	1/0	0/1
Oral Exam	1/0	1/0
Simulator Exam	/	/
Overall Results	1/0	0/1

3. CHIEF EXAMINER AT SITE: Dr. G. Robinson

EXAMINATION DETAILS

Changes Made to Written Examination:

Answer No.	Change	Reason
A.2b	Change "CAF" to "1.7%".	Incorporates value for Xenon reactivity provided by facility.
B.1	Add "may also say 1600 (nominal flow rate)"	Provides additional answer for primary flow rate.
B.4c	Change "32" to "26".	Corrects the length of rod travel.
C.1.b	Add "tau=70, delta K=.000884, Keff=1.000884".	Changes numbers to allow use of charts available in the main control room.
D.4	Add "note: above in place but not normally used Auto-louver control in secondary system or turn system on or off."	Provides information on how facility operates.
D.5b	Change "90 percent" to "1500 gpm" and "80 percent" to "1400 gpm".	Provides actual primary flow rates.
E.2 and K.3	Add "Stack Monitor".	Provides alternate means for detecting a fuel element failure.
G.3.b	Add "100 mr/wk local limit".	Provides local limit for radiation exposure.
G.5.b	Add "30,000 cpm".	Provides correct radiation instrument reading for declaration of a Limited Radiation Emergency Alarm.
H.5.a	Add "rho=.0014136".	Allows value to be read from table.
H.5.b	Add "tau=13.24 sec".	Allows value to be calculated from rho which was read from table.

Answer No.	Change	Reason
H.7	Add "or, agree, because number of source nuetrons available, poisons, temp. etc. would not change significantly since operations at 100% power over a weeks time is limited".	Incorporates facility contention that their reactor always reaches criticality at the same count rate.

EXIT INTERVIEW DETAILS

1. DATE: March 13, 1985 LOCATION University of Lowell

2. PERSONNEL PRESENT AT EXIT INTERVIEW:

   NRC CONTRACTOR PERSONNEL

   Dr. G. Robinson (NRC Consultant)

   FACILITY PERSONNEL & TITLE

   Mr. T. Wallace

3. SUMMARY OF COMMENTS MADE AT EXIT INTERVIEW:

   Both candidates were evaluated as clear passes on the oral portion of the examination. Both candidates were very familiar with the facility and facility operations as demonstrated by their performance during the operational examinations.

Attachments:

1. Written Examination(s) and Answer Key(s) (SRO/RO)

MASTER

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility: University of Lowell

Reactor Type: 1 MW Research Reactor

Date Administered: 3/ /85

Examiner: G. E. Robinson

Candidate: ______

INSTRUCTIONS TO CANDIDATE

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

Category Value	% of Total	Candidate's Score	% of Cat. Value	
19	19	___	___	A. Principles of Reactor Operation
13.5	13.5	___	___	B. Features of Facility Design
12.5	12.5	___	___	C. General Operating Characteristics
17	17	___	___	D. Instruments and Controls
12	12	___	___	E. Safety and Emergency Systems
11.5	11.5	___	___	F. Standard and Emergency Operating Procedures
14.5	14.5	___	___	G. Radiation Control and Safety
100		___		

Final Grade ______%

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

______
Candidate's Signature

A. PRINCIPLES OF REACTOR OPERATION (19)

A.1 There are 1000 neutrons in a particular generation, of which 993 are "prompt" neutrons and 7 are "delayed" neutrons. There were 995 neutrons (total) in the preceding generation.

a. Define "prompt" and "delayed" neutrons. (1.0)

b. Is the reactor subcritical, critical, or supercritical? (0.5)

c. Explain how only 0.7% of the neutrons in a generation can have a significant effect on the control of the reactor. Use the above case as an example. (2.0)

A.2 Consider a situation where the reactor is scrammed from full power and repairs undertaken which require 4 days to complete.

a. Sketch a curve of Xenon concentration versus time for the period of the shutdown. (2.0)

b. How much reactivity is taken up at peak Xenon? (0.5)

c. Describe the effect Xenon will have on critical rod position when compared to the critical rod position before scram. (1.0)

d. Describe the effect (for 3 hours) that Xenon will have on the full power rod positions when compared to full power rod position before scram. (1.0)

A.3 a. Define "Temperature Coefficient of Reactivity," as it pertains to your reactor. (1.0)

b. What reactivity change would result from a 5 degree centigrade <u>decrease</u> in the moderator and reflector temperature? (1.0)

c. Explain why moderator temperature changes affect reactivity. (2.0)

A.4 If the neutron flux diminishes during each succeeding generation when k-effective is less than 1.0, why is it that the count rate increases in your reactor after each increment of control rod withdrawal, while subcritical? (2.0)

A.5 Assume a reactor is shutdown with a shutdown margin of 1.8% $\Delta k/k$. If control rods are withdrawn until the count rate increases by a factor of 20 and the reactor is still subcritical, what is the new K-effective? Show all work. (3.0)

A.6 Briefly describe the problems and hazards involved in starting up a reactor without an adequate neutron source. (2.0)

B. FEATURES OF FACILITY DESIGN (13.5)

B.1 Draw a simple schematic of the Primary Coolant System. Include the following:

- i) all major components (1.0)
- ii) location of inlets and outlets to the reactor pool (0.5)
- iii) name and locate the detectors which actuate reactor scrams (1.0)
- iv) indicate normal flow rate (0.5)

B.2 Consider the Reactor Cleanup System

a. List two reasons why it is important to control the conductivity of the primary coolant (2.0)

b. Indicate normal flow rate for this system (0.5)

B.3 a. The Ventilation System goes from the "Normal" mode to the "Emergency Mode." Describe the path that the air follows from the containment to the stack. Include all major equipment and valves and flow rates. (2.0)

b. Why will the Emergency Exhaust operation stop if differential pressure is improper? (2.0)

B.4 a. Why was the core initially loaded with 26 fuel elements? (1.0)

b. Why are additional fuel elements added to the core over time? (0.5)

c. How does the active fuel bearing length of the fuel compare to the length of travel of the control rod? (0.5)

B.5 Briefly explain the effect on core reactivity if H_2O were to leak into several graphite reflector elements. (2.0)

C. GENERAL OPERATING CHARACTERISTICS (12.5)

C.1 a. The reactor power is increasing on a stable 70 second period. How long will it take to increase the power from 100 watts to 1 KW? (Show your work) (1.0)

b. What is K-effective while the reactor is on this period? Show all work. (2.0)

C.2 Sketch a trace of power versus time as seen on the Log N for the following:

a. Step insertion of positive reactivity at 1 KW putting the reactor on a 60 sec period (2.0)

b. Reactor scram from full power (2.0)

NOTE: Label axis and indicate why changes in trace occur. Indicate any expected periods.

C.3 During steady-state operation, would you expect that there would be any significant difference between the critical rod positions? (In each case assume negligible Xenon concentration)

a. at 100 watts and 1000 watts? (0.5)

b. at 1 KW and 100 KW? (0.5)

c. Briefly explain your reasoning in each of the above cases. (1.0)

C.4 Consider your regulating rod

a. Is the regulating rod worth more for an inch of travel, in the middle of the core, or at the bottom of the core? Briefly explain. (1.5)

b. What is the total worth of the Regulating Rod? (0.5)

C.5 Indicate what effect (increase, decrease, or none) each of the following has upon the reactivity of your reactor.

a. A Beamport flooding with water (0.5)

b. An increase in the conductivity of the primary cooling water (0.5)

c. Replacing a center fuel element with graphite (0.5)

D. INSTRUMENTS AND CONTROL (17)

D.1 Consider the Log N (Intermediate) Channel

a. Indicate the detector used and its principle of operation. (2.5)

b. Briefly describe how a period signal is generated from the signal from the detector in part a. (1.0)

D.2 a. Briefly describe how a Resistance Temperature Detector (RTD) measures temperature. (2.0)

b. What will happen to the RTD readout if the detector develops an open circuit? (1.0)

D.3 a. List the two means available to take the reactor out of automatic power level control. (2.0)

b. Describe the reactor condition (interlock) which prevents the operator from putting the reactor into automatic control. (0.5)

D.4 Briefly explain how the primary coolant system temperature is controlled during normal high power operation. (2.0)

D.5 a. Briefly explain how primary coolant flow is measured. (2.0)

b. What two alarms and automatic corrective actions are associated with primary coolant flow; include setpoints. (1.0)

D.6 Describe what would happen (and why) to the indicated power level on the linear flux recorder if the compensating voltage on the CIC is suddenly lost when:

a. The reactor is operating at 900 KW (1.5)

b. The reactor is operating at 1 KW (1.5)

E. SAFETY AND EMERGENCY SYSTEMS (12)

E.1 Briefly describe the design features which protect the core from being uncovered should a rupture occur in a primary system pipe. Include method of operation. (2.0)

E.2 List the two most likely means by which a small leak in the fuel element cladding would be discovered. (2.0)

E.3 Consider the Emergency Generator

a. Briefly describe what will cause the generator to automatically start. (0.5)

b. How is power provided during an automatic start? (0.5)

c. In addition to the Nuclear Center and Accelerator Emergency Requirements, Emergency Reactor Lighting, Emergency Horns and Flashing Lights, Fire Alarm, Intercom and Public Address Systems and various compressor motors, list five systems powered by the Emergency Generator. (2.0)

E.4 Briefly describe the difference between an electronic scram and a relay scram. (2.0)

E.5 Immediately following a scram, how does an operator ensure that the control blades have dropped fully into the core? Give two possible means. (1.0)

E.6 The picoammeters in the Flux Level Safety Channels are equipped with four trip outputs. Two are used for the scram logic unit and the scram safety chain. Briefly describe how the other two trip outputs are used? (2.0)

F. STANDARD AND EMERGENCY OPERATING PROCEDURES (11.5)

F.1 In accordance with R.O. 8, "Moving and Positioning of Core," describe the three reactor control and instrument conditions required prior to moving the core. (2.0)

F.2 In accordance with the Reactor Building Evacuation during regular working hours found in E.O.1 "Radiation Emergency," list the actions required of the console operator prior to leaving the control room. (2.2)

F.3 List the three conditions that must be satisfied for the reactor to be considered secured. (2.0)

F.4 Consider the situation described below (assume you are operating at full power) and indicate which of the following actions are required for each situation

i) Scram the Reactor
ii) Initiate Building Evacuation
iii) Insert all rods
iv) Maintain power

a. Loss of power supply to Log N channel (1.1)

b. Fire in the Nuclear Center (1.1)

c. Rupture of a beam tube (1.1)

F.5 In accordance with E.O.7 "Stuck Rod or Safety Blade," what are the two main objectives in handling this emergency? (2.0)

G. RADIATION CONTROL AND SAFETY (14.5)

G.1 Define "Relative Biological Effectiveness" or "Quality Factor" and explain how this is used in determining exposures to radiation (2.0)

G.2 a. Briefly explain the major differences in characteristics between a portable monitoring instrument using a GM tube and an Ion Chamber (2.0)

b. Under what circumstances would it be better to use a GM instrument rather than the Ion Chamber? (1.0)

G.3 After working in an area for 2 hours, a man discovers his pocket dosimeter is off-scale. He immediately leaves the area. A survey indicates that a radioactive source left out of its shield is reading 2400 mr/hr gamma at a distance of 2 feet. The man had been working about 4 feet from the source

a. Estimate the dose which the worker received. Show all calculations (2.0)

b. What exposure limits, if any, were exceeded? Briefly explain (Assume no previous exposure during the quarter) (1.0)

G.4 a. Give the following information concerning Nitrogen-16

i) Source (0.5)

ii) Primary hazard (0.5)

b. Is Nitrogen-16 a radiation problem when the reactor is shutdown? Briefly explain (1.0)

G.5 Consider the stack gas monitor

a. Indicate the type and number of detectors and type or radiation detected for

i) airborne radiactive particulates (1.0)

ii) radioactive gases (1.0)

b. What reading on the instruments indicated in part (a) would require a Limited Radiation Emergency Alarm to be declared? (0.5)

G.6 Briefly explain the difference between radiation and contamination (2.0)

10 CFR 20 APPENDIX B

ISOTOPE	$T_{1/2}$	MEV	TABLE I COL I AIR uc/ml	TABLE I COL II WATER uc/ml	TABLE II COL I AIR uc/ml	TABLE II COL II WATER uc/ml
A-41	1.83 h	1.3	$2x10^{-6}$		$4x10^{-8}$	
Co-60	5.3 y	2.5	$3x10^{-7}$	$1x10^{-3}$	$1x10^{-8}$	$5x10^{-5}$
I-131	8.1 d	0.36	$9x10^{-9}$	$6x10^{-5}$	$1x10^{-10}$	$3x10^{-7}$
Kr-85	10.8 y	0.04	$1x10^{-5}$		$3x10^{-7}$	
Ni-65	2.5 h	0.59	$9x10^{-7}$	$4x10^{-3}$	$3x10^{-8}$	$1x10^{-4}$
Sr-90	28 y		$1x10^{-9}$	$1x10^{-5}$	$3x10^{-11}$	$3x10^{-7}$
Xe-135	9.2 h	0.25	$4x10^{-6}$		$1x10^{-7}$	

LINEAR ABSORPTION COEFFICIENTS (cm^{-1})

Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49

$P = P_0 e^{t/\tau}$ $\quad P = P_0 10^{t \cdot SUR}$ $\quad SUR = \frac{26}{\tau}$ $\quad P = \frac{\Sigma_f \phi V}{3 \times 10^{10}}$

$\tau = \frac{\beta - \rho}{\lambda \rho}$ $\quad \tau_p = \frac{\ell^*}{\rho - \beta}$ $\quad T_{1/2} = \frac{.693}{\lambda}$ $\quad \rho = \frac{\ell^*}{\tau} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i \tau}$

$I = I_0 e^{-\mu x}$ $\quad N = N_0 e^{-\lambda t}$ $\quad R/hr = \frac{6CE}{(ft)^2}$ $\quad C = \frac{C_0}{1-K}$

$q = hA\Delta T$ $\quad q = \frac{KA}{L}\Delta T$ $\quad q = \sigma A T^4$

$V_{cyl} = \pi R^2 H$ $\quad R_1 D_1^2 = R_2 D_2^2$

$1\ ft^3 = 7.48$ gal.

$$\Delta K_x = \Delta K_0 + P(\Delta K_1 - \Delta K_0)$$

where $P = \dfrac{T_x - T_0}{T_1 - T_0}$

TABLE I

% $\Delta k/k$ VERSUS $+\tau$

$\ell^* = 7.2 \times 10^{-5}$ sec; $\bar{\beta} = 0.007$

	0.	1.	2.	3.	4.	5.	6.	7.	8.	9
0.	100.00000	0.55477	0.47841	0.42843	0.39156	0.36265	0.33907	0.31931	0.30241	0.287
10.	0.27474	0.26320	0.25282	0.24343	0.23486	0.22700	0.21976	0.21306	0.20683	0.201
20.	0.19558	0.19047	0.18567	0.18114	0.17686	0.17281	0.16896	0.16530	0.16182	0.158
30.	0.15532	0.15229	0.14938	0.14660	0.14393	0.14136	0.13889	0.13652	0.13423	0.132
40.	0.12990	0.12784	0.12586	0.12394	0.12208	0.12028	0.11854	0.11685	0.11521	0.113
50.	0.11207	0.11057	0.10911	0.10770	0.10632	0.10498	0.10367	0.10240	0.10116	0.099
60.	0.09877	0.09762	0.09649	0.09540	0.09433	0.09328	0.09226	0.09127	0.09029	0.089
70.	0.08840	0.08749	0.08660	0.08572	0.08486	0.08403	0.08320	0.08240	0.08161	0.080
80.	0.08008	0.07933	0.07860	0.07789	0.07718	0.07649	0.07582	0.07515	0.07450	0.07[illegible]
90.	0.07323	0.07261	0.07200	0.07140	0.07081	0.07023	0.06967	0.06911	0.06856	0.068
100.	0.06748	0.06696	0.06644	0.06594	0.06544	0.06495	0.06446	0.06398	0.06351	0.06[illegible]
110.	0.06260	0.06215	0.06170	0.06127	0.06084	0.06041	0.05999	0.05958	0.05918	0.05[illegible]
120.	0.05838	0.05799	0.05761	0.05723	0.05685	0.05648	0.05612	0.05576	0.05540	0.05[illegible]
130.	0.05471	0.05436	0.05403	0.05369	0.05337	0.05304	0.05272	0.05240	0.05209	[illegible]
140.	0.05147	0.05117	0.05087	0.05058	0.05029	0.05000	0.04972	0.04943	0.04916	[illegible]
150.	0.04861	0.04834	0.04807	0.04781	0.04755	0.04729	0.04704	0.04679	0.04654	[illegible]
160.	0.04605	0.04581	0.04557	0.04533	0.04510	0.04487	0.04464	0.04441	0.04419	[illegible]
170.	0.04375	0.04353	0.04332	0.04310	0.04289	0.04268	0.04248	0.04227	0.04207	[illegible]
180.	0.04167	0.04147	0.04128	0.04108	0.04089	0.04070	0.04052	0.04033	0.04014	[illegible]
190.	0.03978	0.03960	0.03942	[illegible]	0.03907	0.03890	0.03873	0.03856	0.03839	[illegible]
200.	0.03806									

G. E. Robinson
3/85
MASTER

ANSWER SHEET

University of Lowell

R.O. Exam

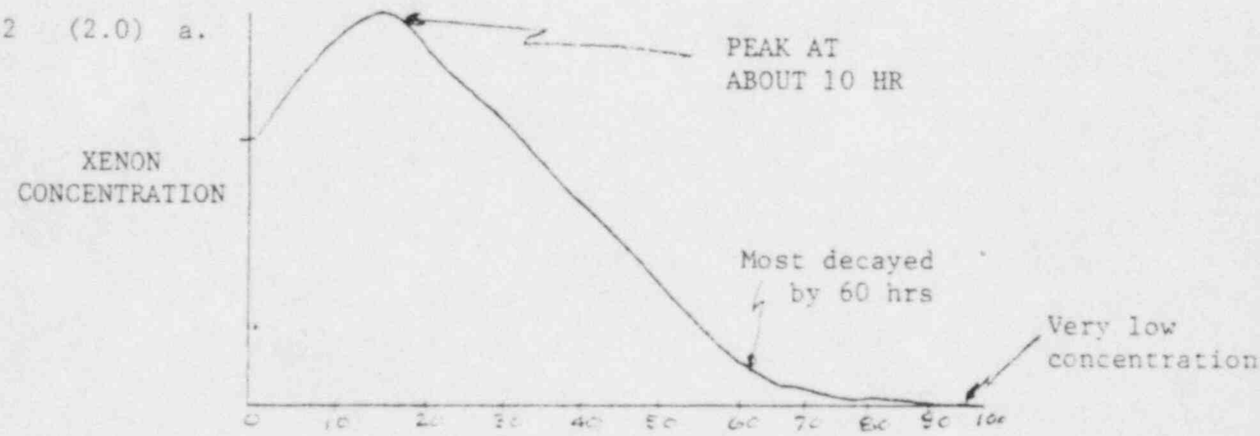
A. PRINCIPLES OF REACTOR OPERATION (19)

A.1 (1.0) a. Prompt neutrons are given off at the time of the fission event. Delayed neutrons result from radioactive decay of fission fragments and appear some time after the fission event.

(0.5) b. Supercritical

(2.0) c. In the given example, a generation of 995 neutrons causes 993 prompt neutrons (appear at time of fission). The reactor would be subcritical were there no delayed neutrons. The delayed neutrons begin appearing slowly (compared to prompt neutrons) with the total generation of 1000 neutrons indicating that the reactor is supercritical; but the slow rate at which delayed neutrons are added to the chain reaction, allows for easy control of the reactor.

A.2 (2.0) a.



(0.5) b. ~~(CAF)~~ 1.7 %

(1.0) c. After 4 days the core will be almost Xenon free. The critical position of the rods will be lower, therefore, than for a startup in which the core is poisoned by Xenon. The position of the rods at full power will initially be lower than they were at the time of shutdown for the same reason.

(1.0) d. As Xenon builds back into the core, rods will have to be withdrawn to compensate for the negative reactivity. After about 3 hrs of full power operation, the rods would still be in the process of being withdrawn.

A.3 (1.0) a. **Reactivity change (positive or negative) per degree change in moderator temperature ~~([illegible])~~.**

(1.0) b. Ref. FSAR, pg. 4-67,
$-5°C \times (-0.88 \times 10^{-4}\ \Delta k/k/°C) = +4.4 \times 10^{-4}\ \Delta k/k$.

(2.0) c. Moderator density changes with temperature. The moderator becomes less dense with increasing temperature. This reduces neutron moderation and allows more neutrons to leak into control rods or out of the core.

A.4 (2.0) The source neutrons are being multiplied by the increased k-effective (without the source the subcritical neutron count rate would fall to a very low value).

A.5 (3.0) Keff = 1 - .018 = .982

$$\frac{CR_2}{CR_1} = \frac{1-K_1}{1-K_2} = 20 = \frac{1 - .982}{1 - K_2}$$

$$1-K_2 = \frac{.018}{20} = .0009$$

$$K_2 = 1.0009 = .9991$$

A.6 (2.0) Without a neutron source it is possible to obtain a supercritical mass without neutron multiplication and without indication on the nuclear instrumentation.

When the chain reaction is initiated, the reactor would be supercritical and on a short period.

A serious power excursion could result with little warning.

B. FEATURES OF FACILITY DESIGN (13.5)

B.1 (3.0) See enclosed drawing 74-233
Flow rate 1400 gpm ~~(GAF~~) (Ref. FSAR, page 4-20)
may also say 1000 (normal flow rate)

B.2 (2.0) a. Keeps corrosion to a minimum
Maintains low levels of radioactivity in the water

(0.5) b. 40 gpm (Ref, FSAR, pg. 4-22)

B.3 (Ref. FSAR, pg. 3-17 and Fig. 3-5)

(2.0) a. The emergency system draws air through charcoal filters via a 320 cfm blower (#E-F-14) into a separate duct which passes through the containment where the integral blast valve is located and then connects to the main exhaust downstream from all other valves in order to allow air passage up the stack.

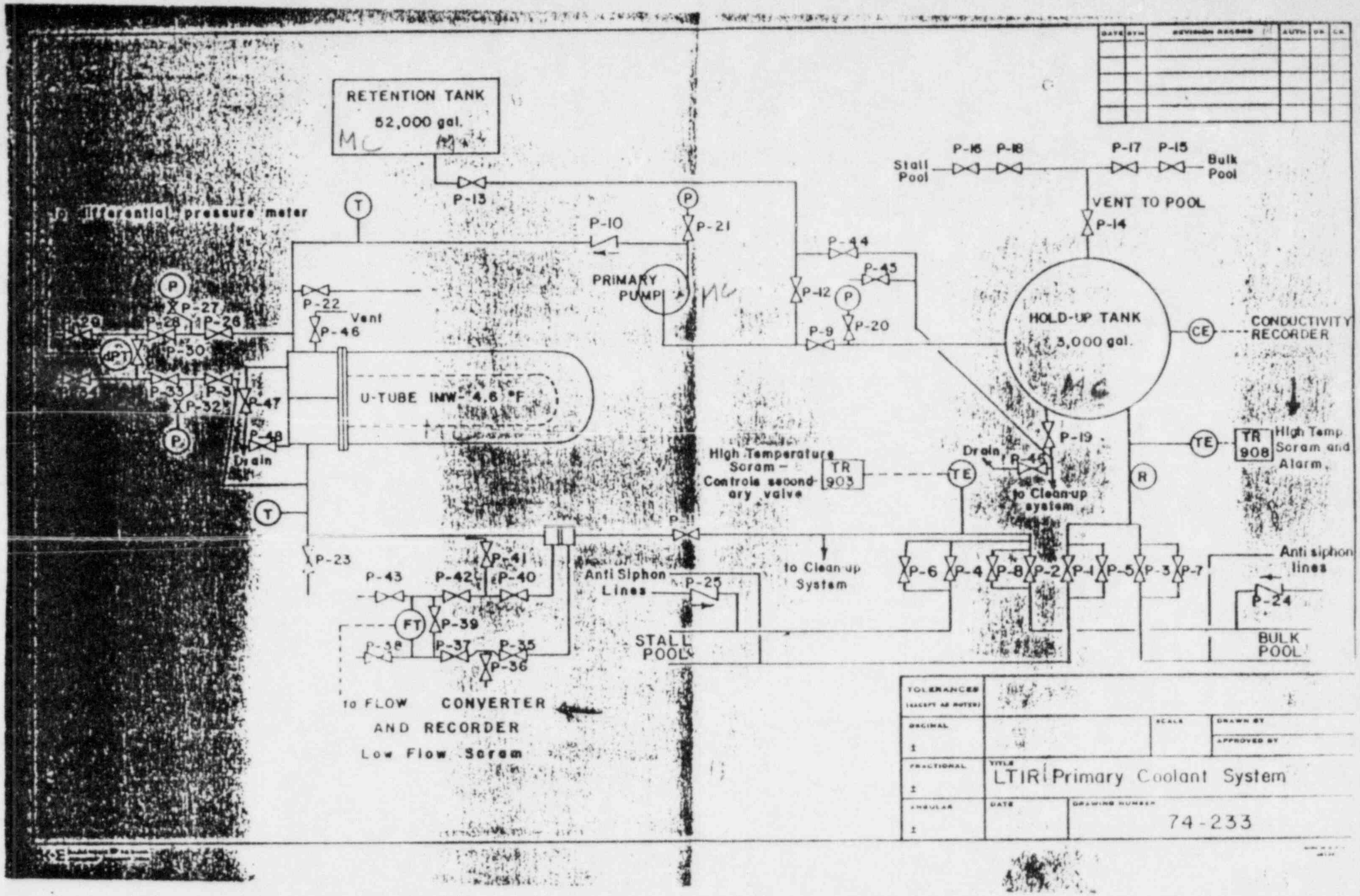
(2.0) b. to protect the integrity of the containment by holding the ΔP within limits that will not significantly increase leakage (positive ΔP) and will not allow too large a vacuum (negative ΔP) ~~(GAF)~~

B.4 (1.0) a. To provide adequate reactivity to make up for poison, temperature, fuel burnup and to provide a margin for control

(0.5) b. To make up for burnup

(0.5) c. Active fuel is 24 inches; rod travel is ~~32~~ 26" inches

B.5 (2.0) Since H_2O has a larger absorption cross section than graphite, the reactivity would decrease.



* MC Major Component

C. GENERAL OPERATING CHARACTERISTICS (12.5)

C.1 (1.0) a. $\frac{P}{P_o} = e^{t/\tau} \qquad t = \tau \ln P/P_o$

$t = 70 \text{ sec} \ln 10 = 70 \times 2.3 = 161 \text{ sec}$

(2.0) b.

They use Charts in the Control room so I obtained a copy and included it with the exam

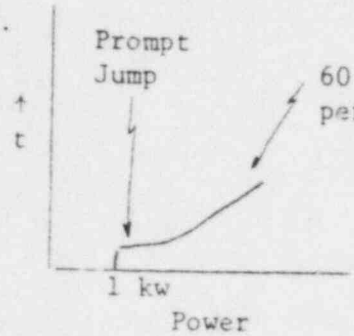
$\tau = \frac{\beta-\rho}{\lambda \rho}$ (CAF) $\beta = .007 \qquad \lambda = .08 \text{ sec}^{-1}$

From U of L Charts For a $\tau = 70$ $\Delta K = .00088$

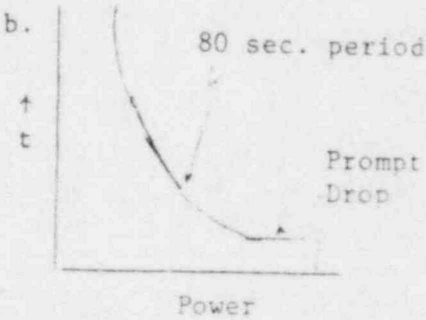
$\rho = \frac{\beta}{1+\lambda\tau} = \frac{.007}{1 + .08 \times 70} = \frac{.007}{6.6} = 1.1 \times 10^{-3} \Delta k/k$

K-effective = 1.0011 ∴ Keff = 1.00088

C.2 (1.0) a.



(1.0) b.



(1.0) a. There is a "prompt jump" increase in neutron flux due to immediate effects of prompt neutrons. Power then increases on a stable period.

(1.0) b. A prompt drop in neutron flux occurs as prompt neutron generation is stopped. Flux then gradually decreases on an 80 sec. negative period as delayed neutrons continue to be generated from the decay of precursor groups.

C.3 (0.5) a. very little if any effect

(0.5) b. at 100 KW rod position higher

(1.0) c. Heating effect significant at 100 KW, essentially no effect below 1 KW. (Heating decreases moderator density, less moderation adds negative reactivity. Therefore rods must be pulled out).

C.4 (1.5) a. regulating rod worth more in the center than at the top because it sees a higher thermal neutron flux at the center.

(0.5) b. 0.34% $\Delta k/k$

C.5 (0.5) a. increase (reference FSAR, pg. 4-67)

(0.5) b. no effect

(0.5) c. decrease (reference FSAR, pg. 4-67)

D. INSTRUMENTS AND CONTROL (17)

D.1 (Ref. FSAR, pg. 4-52)

(2.5) a. Compensated Ion Chamber. The CIC is a double chamber detector. With one chamber lined with boron thus detecting neutrons. Both chambers detect gammas. The signal from the unlined chamber is subtracted from the lined chambers signal, thus giving a signal proportional to the neutron population. Ionization of gas in each chamber produces a signal due to a voltage differential across each chamber.

(1.0) b. After amplification the log. of the signal is obtained and then the signal is differentiated (by an R-C circuit) to produce a signal proportional to the inverse of the period.

D.2 (Ref. FSAR, pg. 4-59)

(2.0) a. Resistance varies with temperature. Thus as temperature goes up, resistance increases and a bridge arrangement allows readout.

(1.0) b. Reading will be high.

D.3 (Ref. FSAR, pg. 4-43)

(2.0) a. mode transfer (MAN) button is pressed
reactor scram relays are deenergized

(0.5) b. reactor period less than 30 seconds

D.4 (2.0) Ref. FSAR, pg. 4-21, Drawings 74-233 and 74-226

A temperature control valve in the secondary cooling system controls amount of secondary water by passing the heat exchanger. The control of this valve is obtained from a temperature sensor measuring primary coolant temperature near pool entrance.

Not above in place but not normally - [illegible] control in secondary [illegible] system or [illegible] system [illegible]

D.5 Ref. FSAR, pg. 4-57 and 4-58

(2.0) a. Detects changes in differential pressure between the upstream and downstream side of the calibrated flow orifice, and converts it to a proportional d-c signal which is transmitted to the alarm switch and flow recorder.

(1.0) b. Actuates annunciator relay if flow drops to ~~90 percent~~ 1500 gpm of normal. Actuates scram relay if flow drops to ~~80 percent~~ 1400 gpm of normal (~~CAF for normal flow rate~~).

D.6 (1.5) a. very slight increase in indicated power since most of the signal is due to neutrons.

(1.5) b. significant increase in indicated power because signal produced by neutrons are about same magnitude as that produced by gamma's.

E. SAFETY AND EMERGENCY SYSTEMS (12)

E.1 (2.0) (Ref. FSAR, pg. 4-19)

Two automatic antisiphon lines are provided. One in the discharge line of the primary coolant loop and one in return primary coolant line. In the event of a primary piping failure, the pool will start to siphon. After the pool level has dropped to the level of the primary pipes, air admitted to the antisiphon risers will prevent a continuing siphon effect.

E.2 (2.0) Any two of the following (CAF)

Activity in primary water (weekly sample)
Fission Product Monitor
Continuous Air Monitor
Stack monitor

E.3 (Ref. FSAR, pg. 6-10 and Table 6.2)

(0.5) a. Loss of house power

(0.5) b. Startup powered by 24 VDC wet cells

(2.0) c. Any five of the following:

Ventillation Supply Fan AC-2
Radiation Monitor System
Emergency Exhaust System
Console Power
Nuclear Instrumentation
Process Control Cabinet
Air Lock Doors

E.4 (Ref. FSAR, pg. 4-49)

(2.0) The electronic trip (fast) deenergizes the scram magnets via the logic unit and trip actuator amplifier while the relay scram (slow) shuts off input power to the trip amplifier which deenergizes the scram magnets.

E.5 (1.0) Decreasing power and any one of the following

Visual inspection or
Proper combination of blade position lights (CAF)

E.6 (Ref. FSAR, pg. 4-44)

(1.0) i. upscale trip opens a relay in the rod withdrawal prohibit circuit

(1.0) ii. downscale trip used in the rod withdrawal circuit to prevent rod withdrawal if the picoammeter reads downscale when the reactor is operating in the rated power region.

F. STANDARD AND EMERGENCY OPERATING PROCEDURES (11.5)

F.1 (2.0) (Reference R.O. 8)

The regulating rod and safety blades fully inserted into the core. Master switch in "Test position". The control and safety system instrumentation is on.

F.2 (2.2) (Reference E.O.1, pg. 2)

a. Initiate LREA or GREA as indicated

b. Assure the reactor has scrammed by visual inspection of the neutron channel information

c. Assure the containment system is closed by reading the status of the indicator panel

d. Lock the master switch and remove key

e. Take a portable radiation monitor and the reactor operations log book

F.3 (Reference Tech. Spec., pg. 3)

(0.5) a. The reactor is shutdown

(0.5) b. electrical power to the control rod circuits is switched off and switch key is in proper custody

(1.0) c. no work is in progress involving fuel or incore exmperiments or maintenance of the core structure, control rods or control drives

F.4 (1.1) a. Insert all rods (CAF)

(1.1) b. Scram Reactor and evacuate containment (Ref. E.O.2)

(1.1) c. Scram Reactor (Reference E.06)

F.5 (Reference E.O.7)

(1.0) First get the reactor far subcritical (Assured by running the other blades in)

(1.0) Second determine the cause of the stuck rod

G. RADIATION CONTROL AND SAFETY (14.5)

G.1 (1.0) Def. A quantity used to relate the biological effectiveness of any specific type of radiation to that of X or gamma radiation

(1.0) Exposures (RAD) to different types of radiation are multiplied by the appropriate QF or RBE to determine the comparable dose (REM) of X or gamma radiation.

G.2 (2.0) a. GM instrument is more sensitive due to amplification of the ionization by the GM probe and the ion chamber gives an indication of the energy of the incident radiation because the greater the energy the more ionization there will be in the chamber.

(1.0) b. As a survey instrument looking for and measuring low level radiation

G.3 (2.0) a.

$$\frac{D_1}{R_2^{\;2}} = \frac{D_2}{R_1^{\;2}} \qquad \frac{2400}{4^2} = \frac{X}{2^2} \qquad X = \frac{1200 \times 4}{16} = 600 \text{ mr/hr}$$

$\therefore$ for 2 hrs he received 2 hr x 600 mr/hr = 1200 mr

(1.0) b. Since the man received 1.2R he has not exceeded the 1.25 REM/quarter. (~~GAF to see if any facility limits have been~~ exceeded) however 100 mr/wk local limit

G.4 (1.0) a. i) N-16 comes from 0-16 via an (n,p) reaction

ii) Its primary hazard is the very high energy gamma ray given off

(1.0) b. No, beaause of its short half life (7 sec) it decays away to an insignificant level quickly.

G.5 (Reference FSAR, pg. 7-25 & Emergency Prep. Plan, Table 1

(2.0) a. i) 2 Beta-scintillation detectors

ii) one G-M tube - primarily gamma rays

(0.5) b. Reference E.O.1, pg. 1 (~~[illegible]~~) 30 cc [illegible]

G.6 (2.0) Contamination refers to foreign materials which are radioactive and cover the outer surfaces of an object. It presents the danger of spreading radioactive material. Radiation refers to particles or rays given off by the radioactive decay of material. Contamination is the presence of these radioactive atoms in an unwanted location.

G. E. Robinson
3/85

ANSWER SHEET

University of Lowell Reactor
SRO Exam

H. REACTOR THEORY (21)

H.1 (1.5) a. A blade's worth is proportional (approximately) to the square of the flux it sees. Since the flux drops off as you move away from the core centerline, so does the blade worth.

(1.5) b. placing a second blade adjacent to a first will depress the flux in the area and decrease the worth of the first blade. (This is known as "shadowing")

(1.5) c. The blade worth will decrease as temperature decreases. This occurs because the neutron diffusion length decreases as temperature decreases, thus the blades see less neutrons.

H.2 (2.0) The initial change in neutron flux is determined as if all neutrons were prompt. This behavior is caused by the fact that when the amount of poison in the reactor is increased, the vast majority of neutrons absorbed are prompt. However, after this initial period, the delayed neutrons are released in direct proportion to the neutron population at an earlier time. Thus, the neutron population is being determined by the delayed neutrons alone.

Physically, this can be explained in the following manner. All precursors are formed at a high rate during the time before the negative step change in reactivity. After the step change in reactivity the neutron population is greatly decreased. Thus, the delayed neutron precursors are now being formed at an extremely slower rate. The decay of the large concentration of precursors formed before the step change in reactivity is, of course, determined by their half-lives. The half-lives of precursor groups 1 through 5 are substantially shorter than that for precursor group 6, the longest-lived precursor. Relative to the concentration of precursor group 6, the concentrations of the other five precursors quickly become very small. Because the majority of the longest-lived precursor, the neutron population can decrease only as rapidly as the concentration of this precursor decreases. The period of neutron-population decrease equals the lifetime of the longest-lived precursor and corresponds to -80 sec.

H.3 (2.0) $(1-K)/K = .05;\ K = .952$

$$\frac{CPS_2}{CPS_1} = \frac{1-0.952}{1-0.99};\quad CPS_2 = 20\left(\frac{(.048)}{.01}\right) = 96\ CPS$$

H.4 (3.0) NO, as critical is approached the time to level out increases. The steady state multiplication of the neutron in the system can be represented by

$$\frac{S(1 + k_{eff} + k_{eff}^{2} + \ldots)}{S}$$

Thus, as k_{eff} approaches 1.0 many more of the succeeding generations make a significant contribution to the multiplication factor.

H.5 (2.0) a. *U of Lowell uses ρ tables in control room. Copy in back of exam and was given to trainee for test*

$$\tau = \frac{D.T.}{.693} = \frac{\beta-\rho}{\lambda\rho}$$

$\beta = .007$ (CAF)

$\lambda = .0.08\ sec^{-1}$ (CAF)

$$\frac{35\ sec}{.693} = \frac{.007-\rho}{.08\rho}$$

$\rho = .00139$

From Table ρ = .0014136

(1.5) b. $\rho = .00139 + .001 = .00239$

.001416 + .001 = .002416

$$\tau = \frac{.007 - .00239}{.08 \times .00239} = 24.1\ sec$$

τ = 13.24 sec From Tables

H.6 (1.0) a. Both poisons are fission products of uranium fuel. Additionally, Xenon is the decay product of radioactive iodine, I-135, which for all practical purposes is a fission product.

(1.0) b. Both poisons are removed from the core by absorption of a neutron. Xenon undergoes radioactive decay and is also removed by this process. Samarium is essentially a stable isotope.

(2.0) c. When the reactor is shut down, both poisons increase in concentration (thus adding negative reactivity). Samarium increase is relatively small and reaches a maximum and remains there until the reactor is operated again. This increase is due to the fact that no Samarium is being removed by neutron absorption while the reactor is shut down.

Xenon reaches a peak (thus maximum negative reactivity) because no Xenon is being removed by neutron absorption while I-135 is still decaying into Xenon and the half-life of Iodine is shorter than the half-life of Xenon. However, since Xenon also decays, Xenon reaches a peak and then the Xenon concentration slowly is reduced because more Xenon is decaying than is being produced by iodine decay.

H.7 (2.0) Disagree. It would be exceedingly unlikely that this would happen since conditions are not the same. The time to take the reactor critical, no. of source neutrons available, changing poisons, temp. etc. all would cause criticality to occur at a different count rate.

or [agree, because no of source neutron available, temp etc would not change significantly since operation at 100% power over a weeks time is limited.]

I. RADIOACTIVE MATERIALS HANDLING AND HAZARDS (19)

I.1 (2.0) a. Total Energy per disintegration = 1.17 + 1.33 = 2.50 MeV

$$D.R. = \frac{6\ CE}{D^2} = \frac{6 \times 20 \times 2.5}{3^2} = 33.3\ R/HR$$ ✓

(1.0) b. 100 REM dose

Probably nausea, fatigue, blood changes (i.e., low white blood cells)

Very good chance of recovery. Almost 100%

I.2 (2.0) 4×10^{-6} μc/ml is twice 10 CFR 20 limits for restricted area (Standard Exam Table provided with exam). This limit is for 40 hour occupancy. Thus occupancy would be restricted to 20 hrs/week in the reactor room.

I.3 (2.0) About the same, GM's are not energy dependent.

I.4 (2.0) Assume β will not travel 10 ft in air therefore the 0.5 mr is all gamma's

$$\text{dose at 6 inches (0.5 ft)} = \frac{R_1 D_1^2}{D_2^2} = \frac{0.5\ (10)^2}{(0.5)^2} = \frac{50}{.25} = 200\ mr/hr$$

at 6 inches γ = 200 mr/hr β = 2000-200 = 1800 mr/hr

$$\therefore\ \beta/\gamma \text{ ratio} = \frac{1800}{200} = 9$$

I.5 (Reference R.0.4)

(0.5) a. licensed NRC operators and Senior Reactor Operators

(0.5) b. Reactor Supervisor (with H.P. Coverage)

(1.0) c. If sample expected to read less than 1R/hr on contact

I.6 (Reference E.0.1)

(1.0) a. A potential radiation emergency exists whenever the radiation monitoring system signals an alarm from preset levels.

(1.0) b. A radiation level in excess of 1R/HR or an airborne level in excess of 100 times MPC in personnel accessible areas.

I.7 (2.0) a. Assume one isotope present

$$A = A_o e^{-\lambda t} \qquad 294 = 900\ e^{-\lambda 180} \qquad \frac{294}{900} = e^{-180\lambda}$$

$$0.327 = e^{-180\lambda} \qquad -180\lambda = -1.12$$

$$\lambda = \frac{-1.12}{-180} = 0.00623\ \text{min}^{-1}$$

$$t_{1/2} = \frac{0.693}{\lambda} = \frac{0.693}{0.00623}$$

$$= \underline{\underline{111\ \text{minutes}}}$$

(1.0) b. Ar-41 has a 1.83 (110 minutes) hour half-life in accordance with data sheet provided with exam and it is the expected gas.

I.8 (Reference Tech Spec. 3.4, pg. 17

(1.0) a. Gaseous and particulate sampling of the stack effluent by the stack monitor (readout in the control room)

(1.0) b. at least one CAM unit located in the containment building on the reactor pool level (readout in the control room)

(1.0) c. minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool (capable of warning personnel of high radiation levels).

J. SPECIFIC OPERATING CHARACTERISTICS (21)

J.1 (3.0) (Reference FSAR, pg. 4-67)

Temperature Coefficient = -0.88×10^{-4} $\Delta k/k/°C$

Thus $\Delta k/k = 10 \times -0.88 \times 10^{-2}\% = \sim 0.88\%$ $\Delta k/k$

From Integrated rod worth curve - new position is 16.8 inches

J.2 (3.0) (Reference S.P. 15)

a. Establish criticality at low power but several decades above initial criticality use a pattern prescribed by SRO (i.e., all blades banked; Regulating rod full in)

b. withdraw Regulating Rod until reactor is supercritical using a stable period between 200 and 30 seconds. Measure period using the stop watch

c. After taking period data, insert balance blade to reduce power level

d. change the base pattern as required by balancing at critical with a new position of the balancing to compensate for the withdrawn position of the blade being calibrated

e. repeat above steps until entire length of regulating rod has been utilized

f. Reactivity is then determined using charts, graphs, and equations and the data plotted first as ρ/in versus inches. This curve is then integrated to obtain integral rod worth (ρ vs. rod position).

J.3 (1.0) a. (Reference Tech. Specs, pg. 2)

A moveable experiment is one where it is intended that the entire experiment be moved in or near the core or into or out of the reactor while the reactor is operating.

(0.5) b. (Reference Tech. Spec., pg. 12)

0.1% $\Delta k/k$

(1.5) c. ([redacted]) Take reactor critical without experiment; note regulating rod position, insert regulating rod, place experiment in core, take reactor critical with regulating rod, and note rod position. Calculate reactivity using regulating rod worth curces.

J.4 Reference Tech. Specs., pg. 12

(0.5) a. 4.7% $\Delta k/k$

(2.0) b. provides sufficient reactivity to accommodate fuel burnup, Xenon and samarium poisoning buildup, experiments and control requirements, but gives a sufficient shutdown margin even with the highest rod fully withdrawn.

J.5 (1.5) a. Since boron has a large absorption cross-section when compared to water, negative reactivity would be added and thus the rods will have to be withdrawn.

(1.5) b. Variations on conductivity within the allowable operating limits will have no effect on reactivity and thus rod movement.

(1.5) c. Since water has a larger absorption cross-section than graphite, the reactivity would decrease and the rod would have to be withdrawn.

J.6 (3.0) Previous blade positions and control rod position
Xenon shutdown reactivity curves (or calculate it)
Worth of experiment added
Reactivity decrease due to burnup
Negative temperature coefficient

J.7 (2.0) Negative temperature coefficient (CAF)
Negative void coefficient if boiling occurs
Maximum core reactivity as limited by Tech. Specs.

K. FUEL HANDLING AND CORE PARAMETERS (21)

K.1 (3.0) a. See attached sketch - 11 fuel elements

(2.0) b. Good geometry for B & C but not A - probably A is too close to source

(1.0) c. Count Rate for all rods in the core

K.2 (1.0) (Reference Tech. Specs, pg. 28)

All reactor fuel element storage facilities shall be designed in a geometrical configuration where k-eff is less than 0.8 under quiescent flooding with water.

Any Two of the following:

K.3 (2.0) a. ~~(CAF)~~ Analyses of samples of primary coolant

Continuous air monitor

Stack Gas Monitor, Bridge Monitor

(2.0) b. Any four of the following

i) danger of leak becoming worse

ii) increased radioactive hazard to operating personnel

iii) increased release of radioactive gaseous products to the environment

iv) increased pool, piping, tank contamination

v) increased waste disposal problem

K.4 (Ref. R.O. 1)

(1.0) a. Reactor Supervisor or other competent SRO appointed by the Reactor Supervisor

At least one licensed operator

At least one other technical qualified scientific person (to act as an independent observer)

(2.0) b. no more than four elements are to be out of mechanically enforced planar geometry at any one time

no more than one array of nine elements (placed in a single pool storage rack) may be out of the storage vault or proper in-pool location at any one time

K.5 a. (Ref. Tech. Spec., pg. 3)

(1.0) i) the reactor is shutdown

(1.0) ii) electrical power to the control rod circuits is switched off and switch key is in proper custody

(1.0) iii) no work is in progress involving fuel or incore experiments, or maintenance of the core structure, control rods or control rod drives

(1.0) b. (Ref. Tech. Specs., pg. 1)

When all isolation system equipment is operable or secured in an isolating position

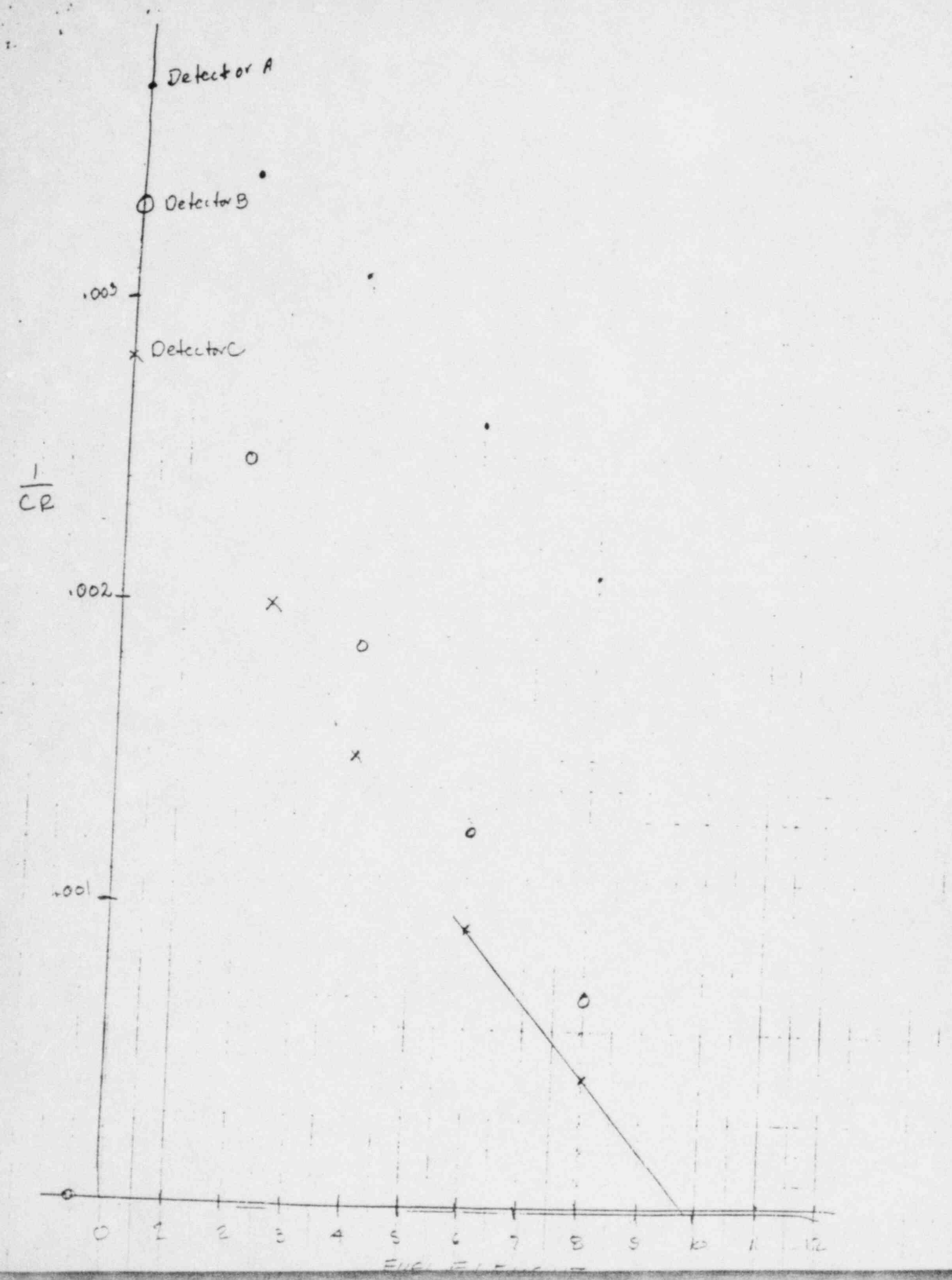
K.6 (Reference R.O. 11)

(1.0) a. any fuel element or partial element that emits sufficient radiation capable of delivering a whole body dose equivalent rate greater than or equal to 100 mrem/hr at 1 foot.

(1.0) b. because both the aspects of radiation and of criticality must be considered and care must be taken to prevent damage to the fuel element surfaces

(0.5) c. SRO (and requires notification of H.P.)

(0.5) d RO



Detector A
Detector B
Detector C
.003
.002
.001
1/CR
0
1
2
3
4
5
6
7
8
9
10
11
12

L. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (18)

L.1 (Reference Tech. Specs. pg. 27)

(1.0) i. Safety Limits insure that the integrity of the fuel cladding is maintained.

(1.0) ii. Limiting Conditions for Operation assure that the reactor can be shut down at all times and that the Safety Limits will not be exceeded.

(1.0) iii. Limiting Safety System settings assure that the plant is operated within assumed conditions of the safety analyses.

L.2 (1.5) a. (Reference Tech. Spec. 3.3, pg. 16 also T.S. 3.5, pg. 18)
Violation - Reactor should have scrammed

(1.5) b. (Reference Tech. Spec. 5.1, pg. 27)
Violation of Tech. Spec. - must be clad with aluminum

(1.5) c. (Reference Tech. Spec. 3.1)
No Violation - shutdown margin is greater than 3% $\Delta k/k$

(1.5) d. (Reference Tech. Spec. 3.2, pg. 14)
Violation - Coolant Flow Rate Channel must be operable

(1.5) e. (Reference Tech. Spec. 3.3, pg. 15)
No Violation - required level is 24.25 ft. above core center line

(1.5) f. (Reference S.P. 10)
No Violation - procedure requires pH to be between 4.5 and 8.5

L.3 (Reference E.O. 4)

(1.5) a. Leave ventilation system on
assure reactor building is cleared of personnel
assure truck door is padlocked
chain and padlock the outer doors of both air locks

(0.5) USAEC Directorate of Reg. Operations Region 1.

L.4 (Reference, 10 CFR, Part 55.4f and 55.9b)

(1.0) a. Control means apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor

(2.0) b. An individual may manipulate the controls of a facility as a part of his training to qualify for an operator license under the direction and in the presence of a licensed operator or senior operator, or as part of his training as a student in a nuclear engineering course under the direction and in the presence of a licensed operator or senior operator.

L.5 (Reference Tech. Spec. 6.3, pg. 33)

(0.5) a. Reactor Supervisor

(0.5) b. Reactor Safety Subcommittee

MASTER

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

Facility: University of Lowell

Reactor Type: 1 MW Research Reactor

Date Administered: 3/ /85

Examiner: G. E. Robinson

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 parenthesis after the question. The passing grade requires at least 70% in each category and a final grade of at least 70%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Applicatn's Score	% of Cat. Value	Category
21	21	____	____	H. Reactor Theory
19	19	____	____	I. Radioactive Materials Handling Disposal and Hazards
21	21	____	____	J. Specific Operating Characteristics
21	21	____	____	K. Fuel Handling and Core Parameters
18	18	____	____	L. Administrative Procedures, Conditions and Limitations
100	100.0	____	____	Totals

Final Grade: ______ %

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

____
Candidate's Signature

H. REACTOR THEORY (21)

H.1 Briefly explain how and why the control blade worth varies:

a. with its radial position in a reactor (1.5)

b. when it is moved close to another rod (1.5)

c. when the moderator temperature decreases (1.5)

H.2 If 1% excess reactivity is added to a reactor that is critical, the resulting period is very short. Explain in detail why the addition of 1% negative reactivity to a critical reactor results in a long stable period. (2.0)

H.3 A reactor is shutdown by 5% Δk/k and has a source level of 20 CPS. If k_{eff} of the reactor is increased to 0.99, what should be the approximate count rate? Show all work. (2.0)

H.4 While withdrawing control rods to take the reactor critical, does the start up instrumentation require the same time to level out at each subcritical level? Briefly explain. (3.0)

H.5 A critical reactor is placed on a stable reactor period at low power. When measured with a stop watch, the doubling time was 35 seconds.

a. What is the approximate reactivity which has been inserted? (Show all work) (2.0)

b. If an additional .001 Δk/k was added to the supercritical reactor, what would be the resulting period? (1.5)

H.6 Xenon and Samarium are two poisons which have a significant effect on reactor operation. Discuss and compare these two poisons for the following:

a. Sources of the poisons in the core (1.0)

b. Means of removal from the core (1.0)

c. Effect on reactor reactivity after shutdown (2.0)

H.7 A trainee for a reactor operator license took your reactor critical on Monday morning and noted that the reactor was critical at a given count rate on the Startup Channel. The following Monday morning he was to take the reactor critical again. He tells you that he expects to go critical at the same count rate that he observed the previous Monday. Do you agree or disagree? Briefly explain. (2.0)

I. RADIOACTIVE MATERIALS HANDLING AND HAZARDS (19)

I.1 An operator in his haste to run from a suddenly exposed 20 curies Cobalt-60 source, trips and strikes his head. Assuming that he falls unconscious three feet from the source, answer the following:

a. What is the average whole body dose received in one hour? Cobalt-60 emits a 1.17 and a 1.33 MeV gamma ray at every disintegration. Show your work. (2.0)

b. If the man receives 100 REM before he is removed, discuss probable physical symptoms due to radiation exposure. Also indicate recovery chances (Neglect head injury). (1.0)

I.2 During an extended reactor run, it is determined that in the reactor room the air will contain 4×10^{-6} μc/ml of Ar-41 activity. In accordance with 10 CFR 20, are there any restrictions that should be placed on the reactor room occupancy? Briefly explain. (2.0)

I.3 Consider two point sources each having the same curie strength (for example, 1 curie each). Source A gamma's have an energy of 1 MeV while source B gamma's have an energy of 2 MeV. You obtain a reading from the same Geiger counter 10 feet from each source. Would the reading from source A be about twice, one-half, or about the same as that from source B? Briefly explain. (2.0)

I.4 The following measurements are made from a beta-gamma point source

2r/hr at six inches
0.5 mr/hr at ten feet

What are the relative fractions of betas and gamma emitted? (State assumptions and show calculations) (2.0)

I.5 Consider R.O. 4 "Adding or Removal of Samples to the Core"

a. Who are allowed to add or remove a sample from the core if it is expected to have significant reactivity worth? (0.5)

b. Whose approval must be obtained to irradiate samples via the pneumatic system when it has been shown that this type of sample has no reactivity worth? (0.5)

c. Under what condition(s) may an operator or senior operator remove a sample from the pool without Health Physics coverage? (1.0)

SECTION I CONTINUED ON NEXT PAGE

I.6 Consider E.O.1 "Radiation Emergency"

a. When does a potential radiation emergency exist? (1.0)

b. List the radiation level and airborne level above which remedial action or restricted access is required. (1.0)

I.7 A radioactive sample was taken from the air going to the stack. (Assume only one isotope) The sample was counted at the following times:

Time, Minute	Counts per Minute
Initial Count	900
30	740
60	615
90	512
180	294

a. What is the half-life of the sample? (Show all work) (2.0)

b. Does this half-life agree with the most likely radioactive gas to be going up your stack? Briefly explain. (1.0)

I.8 What are the three Tech. Specs, requirements for operable radiation monitors during reactor operation to assure evaluation of radiation conditions in restricted and unrestricted areas? (3.0)

J. SPECIFIC OPERATING CHARACTERISTICS (21)

J.1 During a long run of your reactor, the temperature of the water moderator is increased by 10°C. If the regulating rod position originally was 50% withdrawn and if the attached regulating rod calibration curve is applicable, what would be the position of the regulating rod at the end of the run? Show all work. Assume no other reactivity changes occur. (3.0)

J.2 In accordance with S.P. 15 "Rod Reactivity Worth Calibration," briefly indicate how the regulating rod integral reactivity worth curve is obtained. Include the reactor condition prior to starting the test. (3.0)

J.3 You are in charge of loading a moveable experiment into the reactor

a. What is the definition of a moveable experiment? (1.0)

b. What is the maximum allowable worth of this experiment? (0.5)

c. How could you insure that the experiment is within the limits of part (b) if the experiment hadn't been run before? (1.5)

J.4 In accordance with the Technical Specification requirements on maximum allowed excess reactivity

a. What is the maximum allowed excess reactivity? (0.5)

b. What is the basis for the value given in part (a)? (2.0)

J.5 During operation at your reactor, what direction of movement of the control rod, if any, will be required to maintain a constant power level by the following changes. Briefly explain each answer. (Assume each change occurs independently)

a. a moveable experiment containing boron is put into the core via the rabbit system (1.5)

b. conductivity of the primary coolant decreases (1.5)

c. several of your graphite reflectors leak and become saturated with water (1.5)

J.6 Assume the reactor has been at rated power for one week. The reactor is shutdown for a period of 1 hour. During this time an experiment is inserted which adds positive reactivity. Also during the one hour shutdown, the primary coolant temperature decreases 5°F. Briefly describe what information would be necessary to calculate the critical blade positions. (3.0)

J.7 What limits a power excursion of your reactor other than instrumentation initiated scrams? (Your answer should include administrative or Tech. Spec. limits as well as physical considerations) (2.0)

K. FUEL HANDLING AND CORE PARAMETERS (21)

K.1 The following data was taken during a core loading

Fuel Elements	Detector A	Detector B (CR with All Rods Out)	Detector C
0	270	303	250
2	290	400	476
4	323	526	666
6	385	800	1075
8	472	1250	1818

a. Estimate the number of fuel elements needed to go critical. (3.0)

b. Was the source/detector geometry satisfactory for each detector? Briefly explain. (2.0)

c. What additional information could have been obtained so that the total worth of the control rods could have been estimated? (1.0)

K.2 What is the Technical Specification requirement that assures safe storage of your fuel when it is not in the core? (1.0)

K.3 a. Give the two (most likely) means of detecting a defective (leaking) fuel element in the core. (2.0)

b. What are four possible hazards involved with operation with a defective (leaking) fuel element? (2.0)

K.4 According to R.O. 1, "Critical Experiments"

a. What personnel must be available during a critical experiment? (1.0)

b. Briefly describe the two limits on the number of fuel elements that can be transported from the vault to the reactor. (2.0)

K.5 According to your Technical Specifications

a. What three conditions must be met for your reactor to be considered secured? (3.0)

b. When is containment building integrity considered maintained? (1.0)

K.6 Consider R.O. 11, "Handling of Irradiated Fuel."

a. How is irradiated fuel defined? (1.0)

b. Why does irradiated fuel require special treatment? (1.0)

c. Who may authorize movement of irradiated fuel? (0.5)

d. Who must supervise movement of irradiated fuel in or out of the core? (0.5)

L. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (18)

L.1 Explain the relationship between Safety Limits, Limited Safety System Settings, and Limiting Conditions for Operation. (3.0)

L.2 Indicate whether or not the following are violations of procedures and/or Technical Specifications. Briefly explain why it is or is not a violation.

a. Operating at full power with the truck door open. (1.5)

b. Operating with an experimental fuel element made of 93% uranium-aluminum alloy fuel and clad with stainless steel. (1.5)

c. The reactor is subcritical by 3.2% $\Delta k/k$ when the blade of maximum reactivity worth is fully withdrawn. All other rods are fully inserted. The reactor has been shutdown for six days. (1.5)

d. Operating at 500 KW with the primary coolant flow rate meter inoperative. (1.5)

e. Operating at full power with the pool water level 24.8 feet above the core centerline. (1.5)

f. Operating with the primary coolant pH at 5.0. (1.5)

L.3 Consider a situation where you have learned that students are marching toward the Nuclear Center to protest the presence of a nuclear reactor on campus. In accordance with E.O. 4 "Attack Warning and Civil Disorder":

a. How should the operator prepare the reactor building for such a situation? (1.5)

b. Who must be notified? (0.5)

L.4 Title 10, Part 55, Operators Licenses, defines an operator as any individual who manipulates a control of a reactor.

a. Define "control" (1.0)

b. When can an unlicensed operator manipulate a control? (2.0)

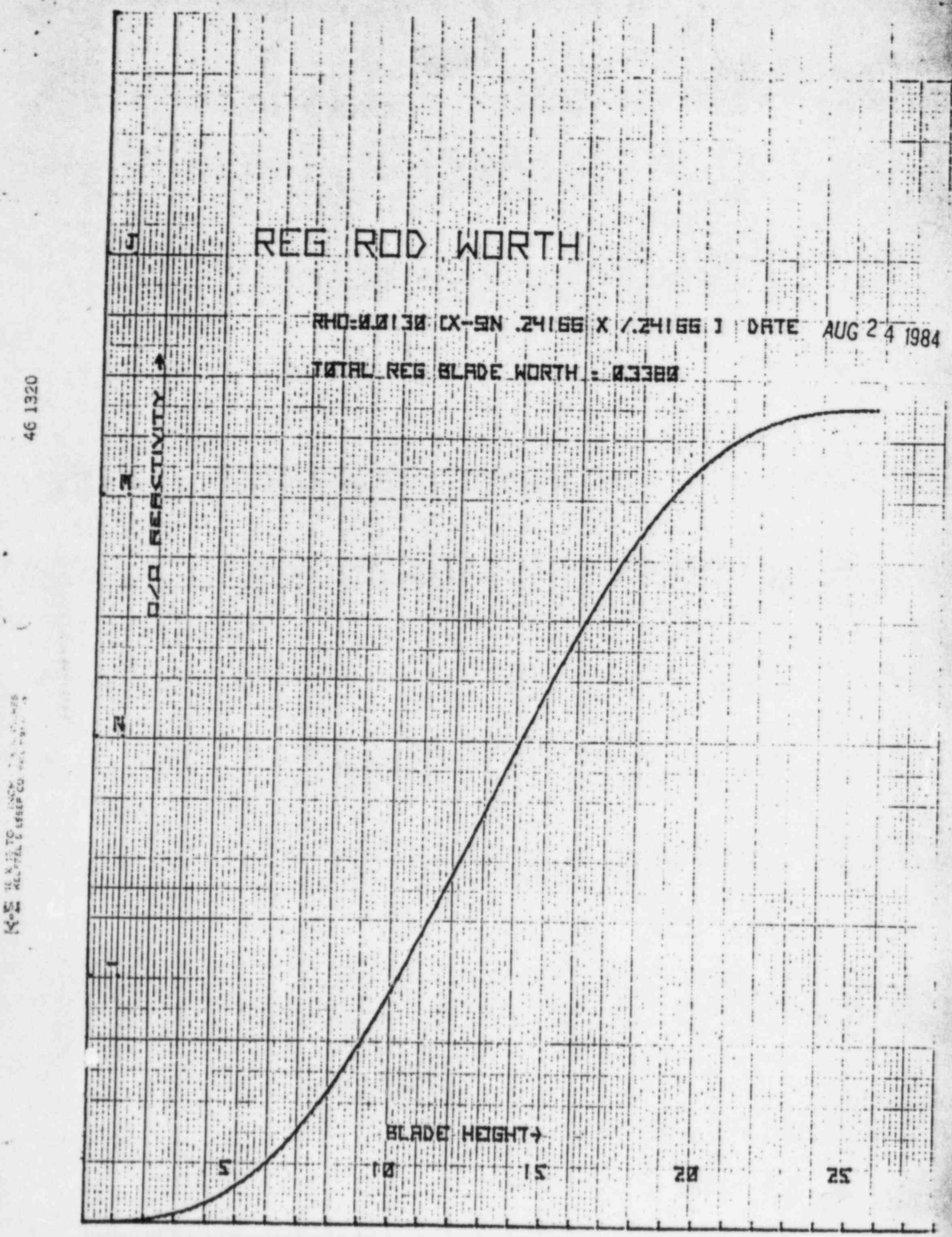
L.5 Consider the operating procedures required by your Technical Specifications

a. Who may make temporary changes to the procedures providing these changes do not change their original intent? (0.5)

b. Who must eventually review this temporary change? (0.5)

K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM 46 1510 MADE IN U.S.A.

KEUFFEL & ESSER CO.


REG ROD WORTH
RHO:0.0130 [X-SN .24166 X /.24166] DATE AUG 24 1984
TOTAL REG BLADE WORTH : 0.3380
D/D REACTIVITY →
BLADE HEIGHT→
5
10
15
20
25
46 1320

10 CFR 20 APPENDIX B

ISOTOPE	$T_{1/2}$	MEV	TABLE I COL I AIR uc/ml	TABLE I COL II WATER uc/ml	TABLE II COL I AIR uc/ml	TABLE II COL II WATER uc/ml
A-41	1.83 h	1.3	$2x10^{-6}$		$4x10^{-8}$	
Co-60	5.3 y	2.5	$3x10^{-7}$	$1x10^{-3}$	$1x10^{-8}$	$5x10^{-5}$
I-131	8.1 d	0.36	$9x10^{-9}$	$6x10^{-5}$	$1x10^{-10}$	$3x10^{-7}$
Kr-85	10.8 y	0.04	$1x10^{-5}$		$3x10^{-7}$	
Ni-65	2.5 h	0.59	$9x10^{-7}$	$4x10^{-3}$	$3x10^{-8}$	$1x10^{-4}$
Sr-90	28 y		$1x10^{-9}$	$1x10^{-5}$	$3x10^{-11}$	$3x10^{-7}$
Xe-135	9.2 h	0.25	$4x10^{-6}$		$1x10^{-7}$	

LINEAR ABSORPTION COEFFICIENTS (cm^{-1})

Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49

$P = P_0 e^{t/\tau}$ $\quad P = P_0 10^{t \cdot SUR}$ $\quad SUR = \frac{26}{\tau}$ $\quad P = \frac{\Sigma_F \phi V}{3 \times 10^{10}}$

$\tau = \frac{\beta - \rho}{\lambda \rho}$ $\quad \tau_p = \frac{\ell^*}{\rho - \beta}$ $\quad T_{1/2} = \frac{.693}{\lambda}$ $\quad \rho = \frac{\ell^*}{T} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T}$

$I = I_0 e^{-\mu x}$ $\quad N = N_0 e^{-\lambda t}$ $\quad R/hr = \frac{6CE}{(ft)^2}$ $\quad C = \frac{C_0}{1-K}$

$q = hA\Delta T$ $\quad q = \frac{KA}{L}\Delta T$ $\quad q = \sigma A T^4$

$V_{cyl} = \pi R^2 H$ $\quad R_1 D_1^2 = R_2 D_2^2$

$1\ ft^3 = 7.48$ gal.

$$\Delta K_x = \Delta K_o + P(\Delta K_1 - \Delta K_o)$$

where $P = \dfrac{T_x - T_o}{T_1 - T_o}$

TABLE I

% Δk/k VERSUS +τ

$\ell^* = 7.2 \times 10^{-5}$ sec; $\bar{\beta} = 0.007$

	0.	1.	2.	3.	4.	5.	6.	7.	8.	9
0.	100.00000	0.55477	0.47841	0.42845	0.39156	0.36265	0.33907	0.31931	0.30241	0.287
10.	0.27474	0.26320	0.25282	0.24343	0.23486	0.22700	0.21976	0.21306	0.20683	0.201
20.	0.19558	0.19047	0.18567	0.18114	0.17686	0.17281	0.16896	0.16530	0.16182	0.158
30.	0.15532	0.15229	0.14938	0.14660	0.14393	0.14136	0.13889	0.13652	0.13423	0.132
40.	0.12990	0.12784	0.12586	0.12394	0.12208	0.12028	0.11854	0.11685	0.11521	0.113
50.	0.11207	0.11057	0.10911	0.10770	0.10632	0.10498	0.10367	0.10240	0.10116	0.099
60.	0.09877	0.09762	0.09649	0.09540	0.09433	0.09328	0.09226	0.09127	0.09029	0.089
70.	0.08840	0.08749	0.08660	0.08572	0.08486	0.08403	0.08320	0.08240	0.08161	0.080
80.	0.08008	0.07933	0.07860	0.07789	0.07718	0.07649	0.07582	0.07515	0.07450	0.073
90.	0.07323	0.07261	0.07200	0.07140	0.07081	0.07023	0.06967	0.06911	0.06856	0.068
100.	0.06748	0.06696	0.06644	0.06594	0.06544	0.06495	0.06446	0.06398	0.06351	0.063
110.	0.06260	0.06215	0.06170	0.06127	0.06084	0.06041	0.05999	0.05958	0.05918	0.059
120.	0.05838	0.05799	0.05761	0.05723	0.05685	0.05648	0.05612	0.05576	0.05540	0.055
130.	0.05471	0.05436	0.05403	0.05369	0.05337	0.05304	0.05272	0.05240	0.05209	[illegible]
140.	0.05147	0.05117	0.05087	0.05058	0.05029	0.05000	0.04972	0.04943	0.04916	[illegible]
150.	0.04861	0.04834	0.04807	0.04781	0.04755	0.04729	0.04704	0.04679	0.04654	[illegible]
160.	0.04605	0.04581	0.04557	0.04533	0.04510	0.04487	0.04464	0.04441	0.04419	[illegible]
170.	0.04375	0.04353	0.04332	0.04310	0.04289	0.04268	0.04248	0.04227	0.04207	[illegible]
180.	0.04167	0.04147	0.04128	0.04106	0.04089	0.04070	0.04052	0.04033	0.04014	[illegible]
190.	0.03978	0.03960	0.03942	0.03925	0.03907	0.03890	0.03873	0.03856	0.03839	[illegible]
200.	0.03876									

MAY 6 1985

Docket No: 50-223

University of Lowell
ATTN: Dr. Leon E. Beghian
Vice President for Academic Services
and Technical Research
1 University Avenue
Lowell, Massachusetts 01854

Gentlemen:

SUBJECT: EXAMINATION REPORT NO. 50-223/85-01 (OL)

This transmits the Examination Report of Operator Licensing Examinations conducted by USNRC Region I at the University of Lowell Facility the week of March 11, 1985. At the exit interview held on March 13, 1985 the preliminary results of these examinations were discussed.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Original Signed By:

Edward C. Wenzinger, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosure: Examination Report No. 50-223/85-01, (OL) w/attachments

cc: w/enclosure and attachments
T. Wallace, Reactor Supervisor
Dr. J. Phelps, Program Director for Nuclear Engineering
Dr. G. H. R. Kegel, Administrative Assistant for Reactor and Accelerator
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
State of Massachusetts (2)

bcc: w/o attachment to enclosure
DRP Section Chief
Chief Examiner
Chief, OLB/DHFS, NRR
OL File 12.0
Region I Docket Room (w/concurrences)
Master Exam File
D. Weiss, LFMB

8

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