Department of
Nuclear Engineering and Science

Rensselaer Polytechnic Institute Troy, New York 12181

Class Outline 35.662

Instructors:	Dave Walthousen Critical Facility - 393-4281 Rick Kacich - 273-2450 John Guerci - 274-5158
Course Outline:	Handouts
Reference Material:	Mostly handouts and lecture material. Text that could be used is ANL 6990. Any other material will be spelled out during lecture peiod, or can be purchased at Book Store.
Course requirements:	Attendance and Laboratory reports due the week following the experiment unless two week lab session or otherwise instructed.
Reports to Include:	Abstract: Synopsis of results Objective: General test methods and applied principles. Preexperiment calculations and preparation results and discussion of such work. Presentation of Data Reduction and Analysis of Data; Sample Calculations. Discussion of Principles and Results.
Grading:	Grades will be based on performance in the laboratory and the reports. Possible quizes and final exam dependent on preceding.
Schedule:	Class meeting - Six students for lab class maximum. Tentative lab periods: Tuesday, Wednesday, and Thursday afternoons and Wednesday evenings.
Safety:	Emergency procedures Safety comments

Discussion of weeks lab: Cover reactor components and instrumentation. Calibrate startup channels

1. Generate voltage VS count rate identify plateau
2. Generate pulse height VS count rate identify plateau; identify region due to gamma and noise.
3. Select optimum operating points Assure signal to noise ratio of 10 or 100 to 1.

Preliminary Course Outline

Week 1: Introduction, orientation, and discussion of procedures, safeguards, and emergency procedures and evacuation. Demonstration of important reactor components. Discussion of course outline, schedule and required reports. The required work will be to check out the reactor system and prepare to startup the reactor. Lastly, a discussion of the following week's experiment.

Week 2: Core Loading - Part I - Including source-detecting system location and geometry, startup channels calibration and initial loading steps by subcritical multiplication.

Week 3: Core Loading - Part II - Loading to minimum critical mass. Prediction of minimum critical mass. Loading to excess reactivity core. Prediction of critical rod bank position.

Week 4: Control rod bank calibration and individual rod calibration by positive period measurements in the super-critical range.

Week 5: Control rod bank and individual rod calibrations by subcritical multiplication and/or rod drop in the subcritical range.

Week 6: Temperature coefficient of reactivity measurements.

Week 7: Poison (Boron) coefficient of reactivity measurements.

Week 8: Void coefficient of reactivity measurements.

Week ~~9~~: 11 Absorption cross-section measurements.

Week ~~10~~: 9 In core flux mapping using foil activation techniques.

Week 1~~1~~0: Power calibration.

FINAL EXAMINATION

COMMENTS ON THE USE OF THE RPI CRITICAL FACILITY

1. Manipulations of fuel shall only be accomplished under RPI supervision and only when audiable contact with the control room is established.

2. Under no circumstances shall you attempt to remove control rods from the reactor core.

   a. removal or returning fuel to the vault
   b. removal or insertion of fuel to the reactor core

3. The neutron source located within the core (or any radioactive source for that matter) shall not be handled without handling tongs and only when the source strength is known and the manipulation has been approved by RPI.

4. TLD's shall be worn at all times at the Critical Facility. Other personnel monitoring may be deemed necessary.

5. The reactor will be operated by the students, including startups, manipulation and shutdowns, but only with prior approval.

6. Operating at low powers produces some residual radiation emanating from the fuel following shutdown. Operating at out limit of 100 watts for a period of time produces proportionally more residual radiation. The radiation will decay (approximately) at a rate equivalent to $t^{-0.2}$.

7. Survey meters shall be used upon entering the reactor room. The results of the survey made shall be recorded in the log book.

8. If you have ever seen the movie on the SL-1 accident; be reminded that this is a similar type reactor.

9. For those students who wish to drive their own vehicles, be very careful entering and leaving the facility property on to Maxim Road. Traffic conditions are bad especially from 4 to 6 p.m. and visability is also poor at this intersection.

10. We encourage you to ask questions at any time. This is the only way we can alleviate any doubts or uncertainties and prevent misunderstandings. Past experience has indicated this point is most crucial to success in this course.

RENSSELAER POLYTECHNIC INSTITUTE
NUCLEAR ENGINEERING AND SCIENCE

EMERGENCY PROCEDURES
SCHENECTADY CRITICAL FACILITY

A. Power Failure to Building - A power failure will result in an automatic scram of the reactor.

1. Turn "off" reactor key on console and return the key to the Facility Supervisor.
2. Turn "off" power key on console.
3. Turn "off" all high voltage power supplied to prevent surges when power is turned on again.

B. Reactor Scram

1. Check that the neutron level falls by at least one decade from previous operating level, and below the one (1) watt power level in any case.
2. Check that the radiation levels fall below 10 mr/hr on all monitors within one (1) minute.
3. If the reactor cell radiation monitors show higher radiation levels than above, initiate the radiation emergency procedures below.

C. Radiation Emergency

1. Scram the reactor.
2. Notify Facility Supervisor or his designated representative immediately.
3. Observe control rod position indicators to insure control rods are in the shut down position.
4. Observe the recording of the air monitor located in the control room and calculate the airborne radioactivity in uc/cc. (A curve of air monitor.)

   NOTE: Under the maximum credible accident conditions, fission products are not expected to be released from the core.

5. Start the portable air sampler in the control room and the counting equipment in the counting room. Analyze the air sample for gross beta every five minutes.
6. Assemble all personnel in the shielded counting room with all available portable radiation monitors. Do not leave building unless radiation levels exceed 1 R/hr.

7. Close down all fans and air conditioners.

8. Assemble available protective equipment:
    a. Scott Air Packs
    b. Protective clothing, etc.

9. Survey shielded area from counting room to control room with portable equipment and swipe tests. Note readings on area monitors and airborne monitor frequently.

10. Observe wind direction and speed

11. Survey area outside building

12. Notify:
    a. <u>Problems involving Facility</u> - Mr. L. D. Walthousen Supervisor; phone FR7/4273; address, 715 Sanders Avenue, Scotia, New York
    b. <u>Problems involving radiation</u> - Mr. R. M. Ryan; phone 869/0147; address, 11 James Drive, Loudonville, New York

    Give available information about the emergency. Any releases to the news media are to be done through the Director of the Critical Facility.

13. Continue with surveys and clean-up

STARTUP CHANNEL CALIBRATION

Abstract

The purpose of a startup channel is to provide information on reactor behavior at very low neutron levels when other instruments are essentially insensitive. Every time the reactor is brought to criticality this channel will provide the first piece of information regarding its behavior. Consequently it is important to calibrate for optimum neutron response. Safety considerations will always dominate all others; therefore, any deviation from normal response must be thoroughly understood and corrected for before planned procedures may ensue.

Introduction and General Theory

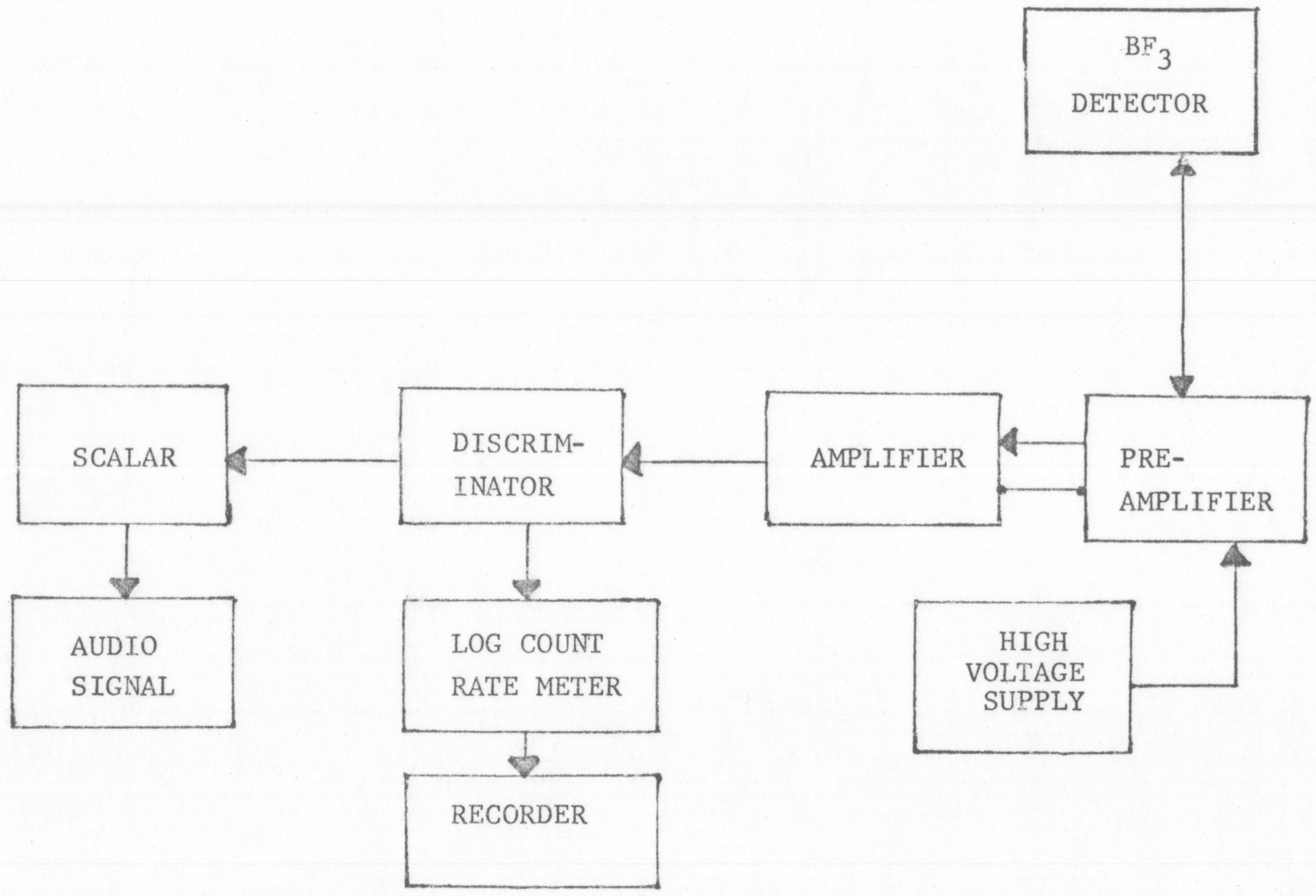
Startup channel instrumentation must be calibrated to ensure that the detecting system will provide an accurate, representative neutron count. The approximate instrument setting will be obtained from manufacturers' specifications and used as a convenient starting point. By no means, however, are these suggestions to be considered as the "correct" settings. Available adjustments usually include the following:

(1) detector high voltage
(2) amplifier gain
(3) pulse-height selector or discriminator

Increasing detector high voltage increases the pulse height in the detector itself. Increasing the amplifier gain will result in more pulses reaching the discriminator setting. The discriminator bias setting cuts off all pulses below this setting. An understanding of the functions of the varying parameters will illustrate their interdependency, i.e., increased voltage may be compensated by decreased gain or by increased pulse-height setting.

The block diagram on the following page identifies major components and explains how a neutron might be detected and counted.

Briefly a neutron is detected as follows: A neutron from a fission event enters the active volume of the detector, which con-



Block Diagram for BF_3 Neutron Detection System

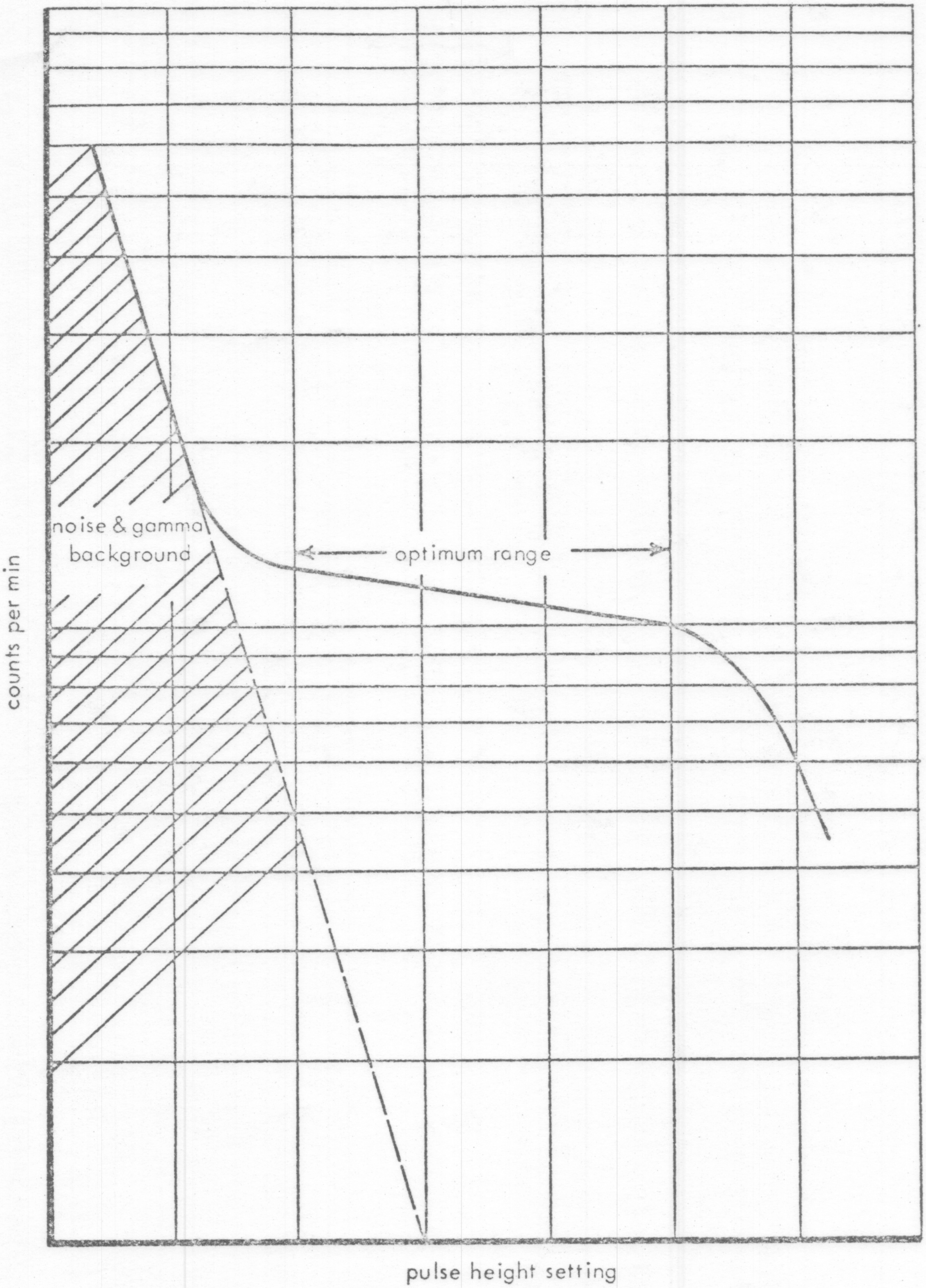
Fig. II-1

tains boron-trifluoride gas. The B^{10} in natural boron has a cross-section on the order of 3800 barns for the B^{10} (η,α) reaction over a wide energy range. The alpha particles released are of high energy and consequently cause a high degree of ionization. In general, neutron pulses are easily detectable, even in large gamma fluxes.

Pulses seen at the scalar need not originate at the BF_3 detector. Even if they originate in the detector they need not be neutrons. Ionizations due to gamma radiation through the counter give pulses much smaller than those of neutrons, on the average very small pulses in the system are caused by noise in the input resistor and input tube of the pre-amplifier. This is the primary reason for a discriminator. It screens out these smaller, unwanted pulses and allows only neutrons and perhaps a few large gamma pulses to pass.

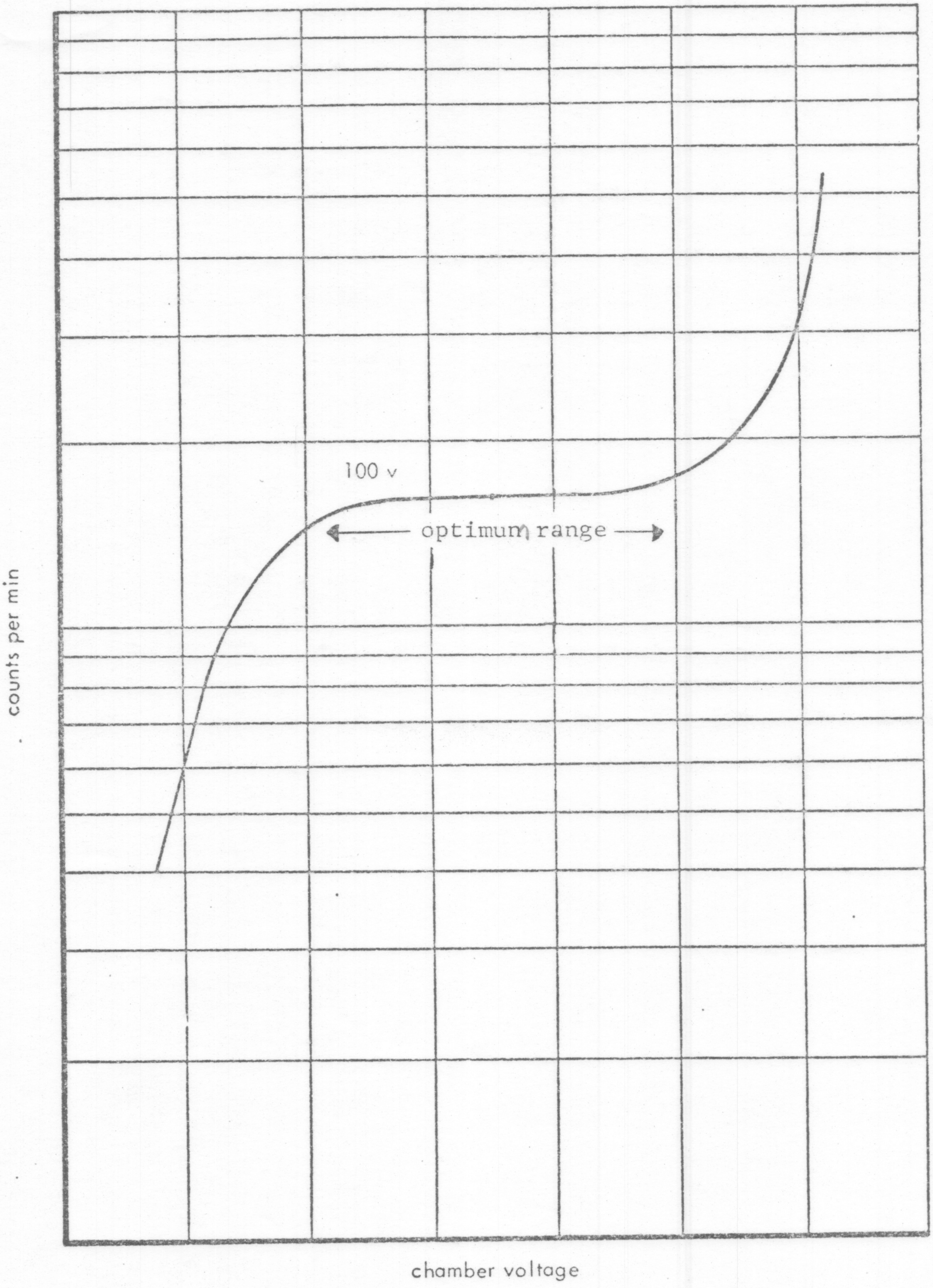
Noise, or gamma-ray, pulses are statistical in their size distribution; the number above a certain pulse-height setting is proportional to the number at that setting. Such a distribution is exponential, giving a straight line on a semi-log plot. Thus a graph of logarithmic count rate vs. pulse-height setting is a straight line and can be extrapolated to predict this "noise background" at any higher setting. Neutron pulses, however, are much larger and more uniform in size; therefore the curve will flatten out to show a plateau above the pulse-height setting. This cuts off most of the unwanted pulses. An even higher discriminator setting will eventually cut off the neutron pulses. One crucial consideration in determining the pulse-height setting is the maximization of the signal to noise ratio. This range of optimization is illustrated in Figure II-2.

A plot of count rate vs. voltage or gain (Fig. II-3) is similar, but as a mirror image. At low voltage almost all pulses are too small to pass the discriminator; at higher voltage most of the neutron pulses are counted giving a broad plateau of fairly uniform count rate. At still higher voltage the noise pulses pass the discriminator creating a steep rise in count rate. Care must



PULSE HEIGHT SETTING VS. CHAMBER RESPONSE

Fig. II-2



VOLTAGE VS. COUNT RATE

Fig. II-3

be taken so that too high a voltage is not applied. This would result in a continuous discharge in the gas-filled chamber, perhaps causing irreparable damage to the tube.

Procedure

Realizing that there are three interdependent parameters and that a gross malajustment of any of them would result in either no counts or a damaged tube, some initial manipulation without data taking is necessary. Start with the manufacturer's suggested settings of amplifier gain and compensating discriminator voltage. Cautiously increase the high voltage to the tube until a response on the scalar is noted. This condition should indicate the threshold of the plateau provided discriminator voltage is not too high. A further cautious increase in high voltage will establish the approximate range of the plateau. This manipulation will enable the student to position the high voltage near the midpoint of the plateau. Now adjustment of the compensating voltage will illustrate the interdependency of these parameters. An increase will eventually eliminate all response, and a decrease will allow an increasingly larger number of noise and gamma pulses to pass through the discriminator. After establishing the approximate range of this parameter, adjust to near the midpoint. Leave the amplifier gain at the suggested setting. Start at a voltage just below the threshold of any response and take timed count rates for various voltages. Enough data should be taken to accurately define the shape of the curve, however it is not necessary to take count rates every 10 volts along the plateau if it happens to be 400 volts long. You may want to take more frequent count rates near the curved portions of the plot as compared to the straight line portions.

After completing this plot it is necessary to select the optimum operating voltage. Your decision should be based on, among others, the following considerations:

(1) drift in gain, voltage, or discriminator setting
(2) tube-life

(3) intensity of gamma-background

(4) possible fluctuations in detector environment during the course of reactor operation.

While maintaining the same amplifier gain and the determined optimum high voltage, it is desirable to have a plot of count rate versus pulse height setting. It is acceptable to start with zero discrimination and vary in suitable increments until the shape of the curve is well defined. One time saving hint here is that it may be advantageous to vary in large increments at first. Should the student find the data insufficient it is certainly valid to go back and take count rates where desired. In this way a well-defined curve can be generated without undue delay. Now the optimum discriminator setting must be selected. Frequently it is possible to achieve a signal to noise ratio of at least 100 to 1. On the basis of this ratio and other previously mentioned considerations an optimum setting can be selected.

If time permits a plot of count rate versus amplifier gain can be generated, however the first two plots are sufficient to provide an adequate understanding of the BF_3 detector and counting system.

Analysis of Results

It is usually most convenient to plot the data on semi-log paper, but linear paper also yields easily interpretable results.

Topics for Discussion

(1) By means of graphical interpretation, consider the effect of a high gamma background on your results. Would it change your optimum settings?

(2) Consider a hypothetical plot of count rate versus high voltage. By connecting successive data points you find that a portion of the plateau region has a slightly negative slope. How would you interpret this?

(3) At zero discrimination you find that your count rates do not increase dramatically. Interpret.

(4) You are the reactor operator and before raising rods you want to insure your neutron detecting system is operating properly. Could you quickly insure this?

Chapter III

CORE LOADING BY SUBCRITICAL MULTIPLICATION

Abstract

The initial loading of a critical mass represents the only occasion when a truly unknown system must be manipulated. Certainly extensive calculations can frequently predict criticality quite accurately, but this can never be guaranteed. Potential pitfalls include erroneous computations, mistakes in fabrication, faulty detectors, mechanical failures, and a variety of human errors. Because there exists the real possibility of a serious problem, such as the inadvertent loading of a supercritical mass, extreme caution must be applied in all steps. Several rules for loading procedures have been developed, and as a further safeguard the most conservative information will always dictate the next step to be taken.

The purpose of this exercise is to _safely_ load a critical mass and, despite the magnitude of potential problems, this objective can be easily achieved.

Introduction and General Theory

Subcritical multiplication, (M), is defined as the ratio of the total thermal neutron flux due to both the source and core fissions to the flux due to the source only, or

$$M = \frac{\varphi_s + \varphi_f}{\varphi_s} = \frac{\text{count rate due to source + core fissions}}{\text{count rate due to source only}} \qquad \text{(III-1)}$$

where φ_s = neutron flux from source

φ_f = neutron flux from fissions

Subcritical multiplication can also be expressed in terms of the effective multiplication factor, K_{eff}, as follows:

$$M = \frac{\varphi_s(1+K_{eff}+K_{eff}^2 + K_{eff}^3 + \dots)}{\varphi_s} \qquad \text{(III-2)}$$

$$M = \frac{1}{1-K_{eff}} \qquad \text{(III-3)}$$

it is important to remember that (M) ≡ <u>subcritical</u> multiplication. This expression does not apply to $K_{eff} \geq 1$. Therefore for $K_{eff} = 1$ the implication is not that neutron yield or power level approaches infinity, but rather that it remains constant at some level far exceeding the original source level. Consequently, at such a level the source can be removed without noticable effect.

For K_{eff} very close to 1 M becomes very large, which is inconvient to physically handle and interpret. For graphical illustration and practical interpretation, the reciprocal of the multiplication or inverse multiplication, (1/M), is used; which is equal to $(1-K_{eff})$. Since K_{eff} is generally close to unity in practical applications, this quantity is very nearly equal to the reactivity $(\rho = \frac{K_{eff}-1}{K_{eff}})$ but opposite in sign. Essentially it is the amount of reactivity by which the reactor is subcritical. Therefore, the inverse multiplication factor is expressed in terms of reactivity and counting rates as:

$$\frac{1}{M} = 1-K_{eff} = \frac{\text{count rate due to source alone}}{\text{count rate due to source + core fissions}} \qquad \text{(III-4)}$$

in order to apply this expression with validity, two conditions must be met:

1. the detecting system must be actually counting neutrons
2. the neutron flux measured must be representative of reactor multiplication

The first condition is verified by calibrating the counting systems; this is relatively easy to accomplish with a high degree of confidence. The second condition is verified by proper location of the detecting system. It is often impossible (because of physical and other limitations) to achieve a detecting system geometry with an output that will accurately reflect reactor multiplication. Geometry considerations modify equation III-4 as follows:

$$\frac{1}{M} = 1-K_{eff} = (A) \frac{\text{count rate due to source alone}}{\text{count rate due to source + core fissions}} \qquad \text{(III-5)}$$

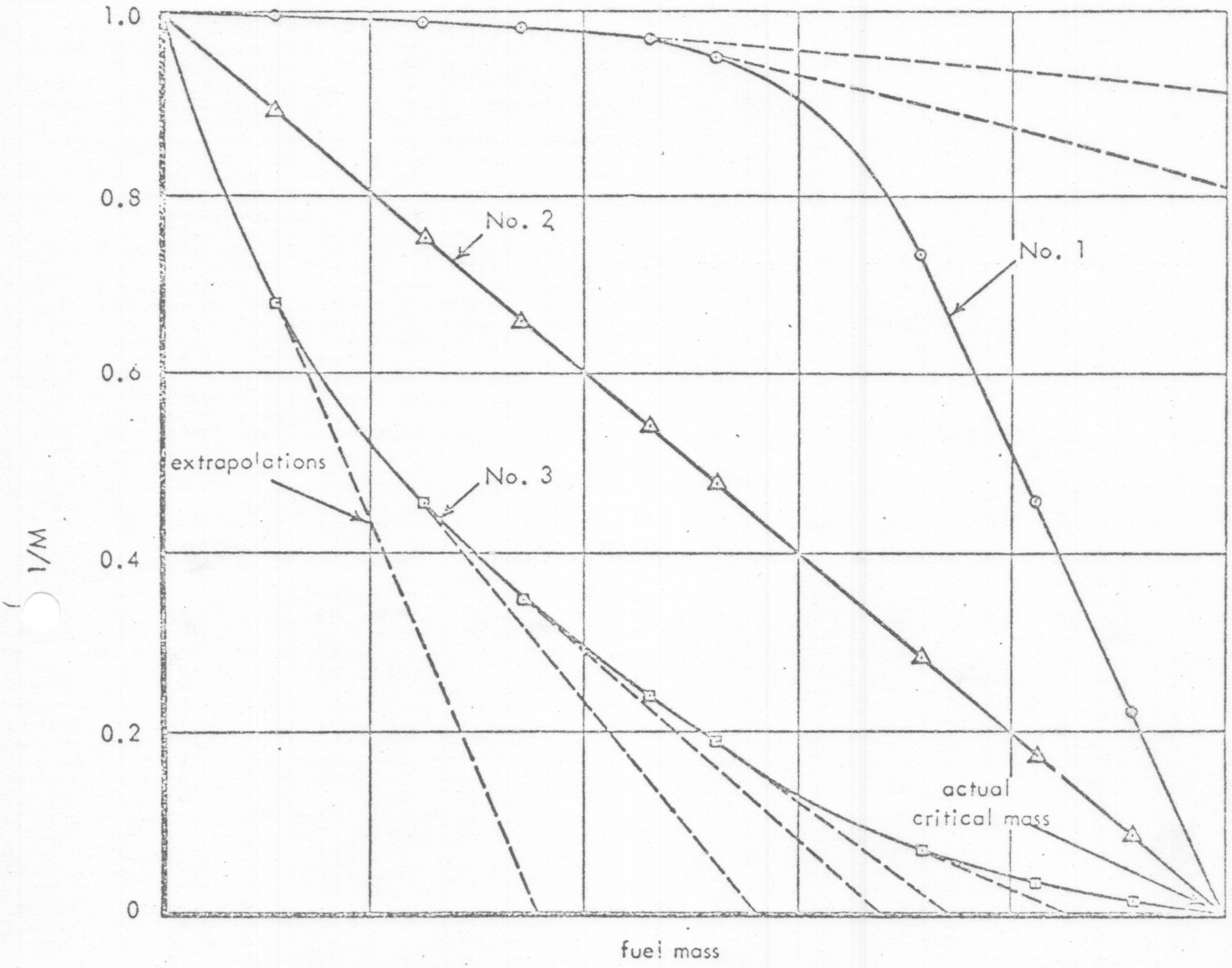
where (A) is a constant of the instrument system geometry.

Figure III-1 illustrates the effect of varied instrument system geometries on the shape of the curve of (1/M) as a function of fuel mass. When the detector system is properly located the neutron flux counted reflects actual reactor multiplication; the constant (A) becomes unity as in curve 2. A plot of (1/M) vs. fuel mass is therefore linear.

If the detector is too close to the source, the source term will tend to dominate the shape of the curve making it concave downward as in curve 1. As indicated in the illustration linear extrapolation from points on this curve overestimates the critical mass. This is the most dangerous geometrical error and must be avoided.

The condition of the detector when it is far from the source and close to the core is conservative and safe. In this case the core term will be dominant; hence the curve will reflect more core fissions than is really the case. Linear extrapolations from intermediate points on curve 3 will underestimate the critical mass.

As shown in Figure III-1, the three curves converge at the point where the actual critical mass occurs. As indicated earlier, when the reactor is critical the source term becomes insignificantly small compared to the core yield.



No. 1 - detector too close to source (overestimates critical mass)
No. 2 - proper location of detectors (expected linear relationship)
No. 3 - detector too far from source (underestimates critical mass - conservative)

NOTE: The most conservative of all 1/M plots i.e. the plot that predicts the lowest critical mass will be used during loading operations.

INVERSE MULTIPLICATION PLOTS

Fig. III-1

To determine the critical mass of an unknown assembly the inverse multiplication (1/M), usually expressed in counting rates, is plotted against the size of the assembly, usually expressed as fuel mass. With each incremental change in fuel mass the plot is <u>linearly</u> extrapolated to the zero value of the inverse multiplication to predict when criticality will be achieved.

The high degree of safety desired in this exercise demands the use of as many detectors as possible in order to determine successive loading steps. Eventually all curves should extrapolate to the same point, but during the loading the geometry effects will perturbate the counting rates. Therefore the most conservative of the inverse multiplication plots generated, the one that predicts the smallest critical mass, will be used to guide the loading process.

Procedure

The inverse multiplication method will be used for fuel loading and initial approach to criticality. The initial loading step will, of course, be the 4 control rod assemblies. Subsequent fuel additions will be determined by extrapolations of the inverse multiplication plots until the minimum critical mass is reached.

Timed count rates will be taken with the control rods fully inserted and fully withdrawn prior to fuel loading and following each subsequent loading step until the minimum critical mass is attained. The inverse multiplication factor generated from these timed counts will be plotted as a function of the total number of fuel elements or the number of grams of fissile material in the core and, where appropriate, as a function of the control rod bank position. This means that for each detector available a plot of (1/M) vs. fuel mass must be plotted for the control rods inserted; for each detector the same graph must be constructed for control rods withdrawn. The plots are then linearly extrapolated to predict the critical mass and the critical bank position.

Following attainment of the minimum critical mass the loading is continued in increments of one element. In each step the critical bank position is determined and the rod bank calibrated until the desired excess reactivity and reactor response is reached.

A fairly large neutron source is desirable and therefore utilized in the process. An examination of the expression for the multiplication (M) indicates that low levels ($K_{eff} << 1$) will be difficult to distinguish unless the φ_s term is large. Thus for any individual counting step a strong source will either reduce the counting time necessary or increase accuracy as compared to a weaker source. This general concept, when coupled with detector response, is the heart of the purpose of the neutron source.

Any potentially useful instrumentation will be located as close to the core as possible. The objective of any neutron source considerations, detector relocations and associated geometry effects, or adjustment of startup channel variables is to obtain a statistically reliable neutron count rates for all core configurations. Optimization is not achievable for <u>every</u> core configuration and any deviation is only acceptable if it is in the conservative direction. The neutron flux counted must accurately or conservatively reflect reactor multiplication. There may also be occasions when either an instrumentation unit or the neutron source must be relocated. This perturbates the data beyond a tolerable limit. In these cases count rates will be taken with the same fuel loading at both the former and latter locations to insure that subsequent readings of neutron flux levels are referenced to a common base.

At this point it is instructive to specify some of the general guidelines of the loading procedure itself. These rules are designed to facilitate the decision-making process for any given incremental loading. Occasionally some requirements conflict and in these cases as many as possible will be met. Of course safety and good judgment preclude any other considerations.

1. Along with neutron source and instrumentation, the initial loading step will be the control rod assemblies. All fuel loading will take place with control rods fully inserted.
2. All fuel handling and loading operation will be performed with the utmost care.
3. All fuel must be inserted slowly. The approximate position of each element, relative to its full-in position, must be communicated to the operator-in-charge in the control room.
4. If at any time the count rate on any channel increases by more than one-half decade during a single fuel element addition, all fueling operations will cease.
5. Fuel additions will be limited to one-half the difference between the loaded mass and the the extrapolated critical mass or to four stationary elements, whichever is lowest.
6. Throughout loading operations, each fuel element will be placed in a lattice position that will preserve, as nearly as possible, a symmetric core.
7. Each fuel element will be added in a lattice position such that the minimum surface to volume ratio is achieved.
8. The number of elements to be added at any given step can never exceed that of the previous step.
9. After the minimum critical mass is attained and until the core is loaded to achieve de-

sired response, fuel additions will be limited to one fuel element for each of these steps. Plot the critical bank position as a function of total number of elements or total fuel mass in the core. The control rods will also be calibrated by the positive period method for each new critical loading in order to provide an estimate of reactivity worth of the control rods, excess reactivity, and shutdown margin.

Because this is the first occasion the student will be dealing with a critical mass, a section is provided regarding the safety system of the RPI Critical Facility reactor. The intention here is not to provide details of the system itself, but rather to summarize the effects of the safety circuitry. Table III-1 lists the condition, and indicates if the condition results in a scram, or if an interlock is engaged, prohibiting further rod withdrawal.

Analysis of Results

As mentioned in the procedure, plots of inverse multiplication vs. fuel mass are to be generated for all conditions and all detectors. Include a calculation for the minimum critical mass. Indicate which of your count rates was used as a source level for each curve.

Topics for Discussion

1. Perhaps time limitations prevented you from monitoring a true "source level"; i.e. it was not possible to get a count rate for the system with only source, moderator, and detectors in the core. If so, discuss what effect this had on your results.
2. Why is the limit on fuel element additions set at four?

Table (III-1)

Condition	Scram	Interlock
Period less than 15 seconds		X
Period less than 5 seconds	X	
Water fill pump on		X
High level linear power 1	X	
High level linear power 3	X	
High power level	X	
Power channel switch not in operate mode		X
Less than 2 counts/second on start-up channel		X
"Reactor on" key removed	X	
Loss of 400 cycle power		X
Manual scram	X	
Loss of power	X	
Master recorder switch off		X
Individual chart drive switch on start-up channel recorder		X
Opening personnel access door	X	
Opening large truck door	X	

Note: Unless bypassed, any scram will also initiate a moderator dump through a six-inch dump line at the base of the reactor tank.

3. Considering source, detector geometry especially, are symmetric core positions truly symmetric for analytical purposes? What further perturbations were caused by the requirement of maintaining minimum surface to volume ratio.
4. Why is it possible to re-define a "source level" at any time you find it convenient?
5. Why are all extrapolations done linearly when there may exist enough data points to imply that a non-linear curve is being generated?
6. Was the control rods withdrawal plot better (i.e. more conservative) than the rods insertion plot? Why? Did one detector consistently supply more conservative data? Why? Consider the sensitivity of your analysis to the "source level" count rates.
7. What additional information about the reactor system is provided by the graph of inverse multiplication vs. control rod bank position.
8. Is it always valid to begin taking count rates immediately after a new bank position is reached? Explain the reasons for any waiting time which may be necessary.
9. What very important information is provided by the plot with control rods inserted?
10. Elaborate on the purpose of a neutron source.
11. In this procedure the student was advised to load all fuel elements with the control rods fully inserted. One argument discrediting this strategy is that in the event of a critical or supercritical mass being loaded there exists no back-up safety system. Consequently, it is advisable to load fuel with at least one rod partially or fully withdrawn. Discuss.

Chapter IV

CONTROL ROD CALIBRATION

Abstract

The safe manipulation of a critical system requires a thorough understanding and calibration of those parameters which directly govern the response of the system. A convenient terminology used to discuss these parameters is reactivity (ρ), defined as ($K_{eff}-1/K_{eff}$). Reactivity quantifies the deviation from exact criticality.

Control is most often accomplished by control rods; in the case of the R.P.I. Critical Facility these rods also have fuel followers. As poison is removed from the core region the same length of fuel is added simultaneously. One can see that it is desirable to relate reactivity to some parameter associated with control rod movement, which is the reactor period. This is accomplished through the in-hour equation, which is discussed in the theory section. The purpose of this exercise is to generate a control rod calibration curve for the entire length of travel, both as a bank and as single rods. The positive period method is the technique which relates some incremental rod withdrawal to the associated stable period. This method is the most accurate and will be used for rod calibrations wherever possible. For reactor cores with minimal excess reactivity this technique is not applicable to initial or intermediate distances of rod withdrawal. At these locations the technique of subcritical multiplication must be used.

As a further check the rod-drop method will also be employed. The rod (or bank) to be calibrated is dropped from different heights and the resulting decay of the neutron flux is observed and related to reactivity.

Introduction and General Theory

As a brief summary of the neutron cycle and the development of (K_{eff}), the schematic representation of Figure IV-1 is presented. This diagram is provided so that the student recalls the sequence of events in one neutron generation; further relationships may be derived from this development of the effective multiplication factor. A more detailed explanation of a neutron generation is provided in Glasstone and Sesonske, NUCLEAR REACTOR ENGINEERING, pp.152-170.

The (K_{eff}) of a reactor is defined as the average number of thermal neutrons remaining for each thermal neutron absorbed at the beginning of the generation. (K_{eff}) is unity for a critical reactor, less than unity for a subcritical reactor, and greater than unity for a supercritical reactor. Several factors dictate that a reactor be designed with (K_{eff}) greater than unity. For example, ($K_{eff}>1$) is the only way of increasing power level. It provides a means for over-riding the decrease in (K_{eff}) during operation due to fuel burnup, poison buildup, or any negative coefficients such as temperature, void, etc.

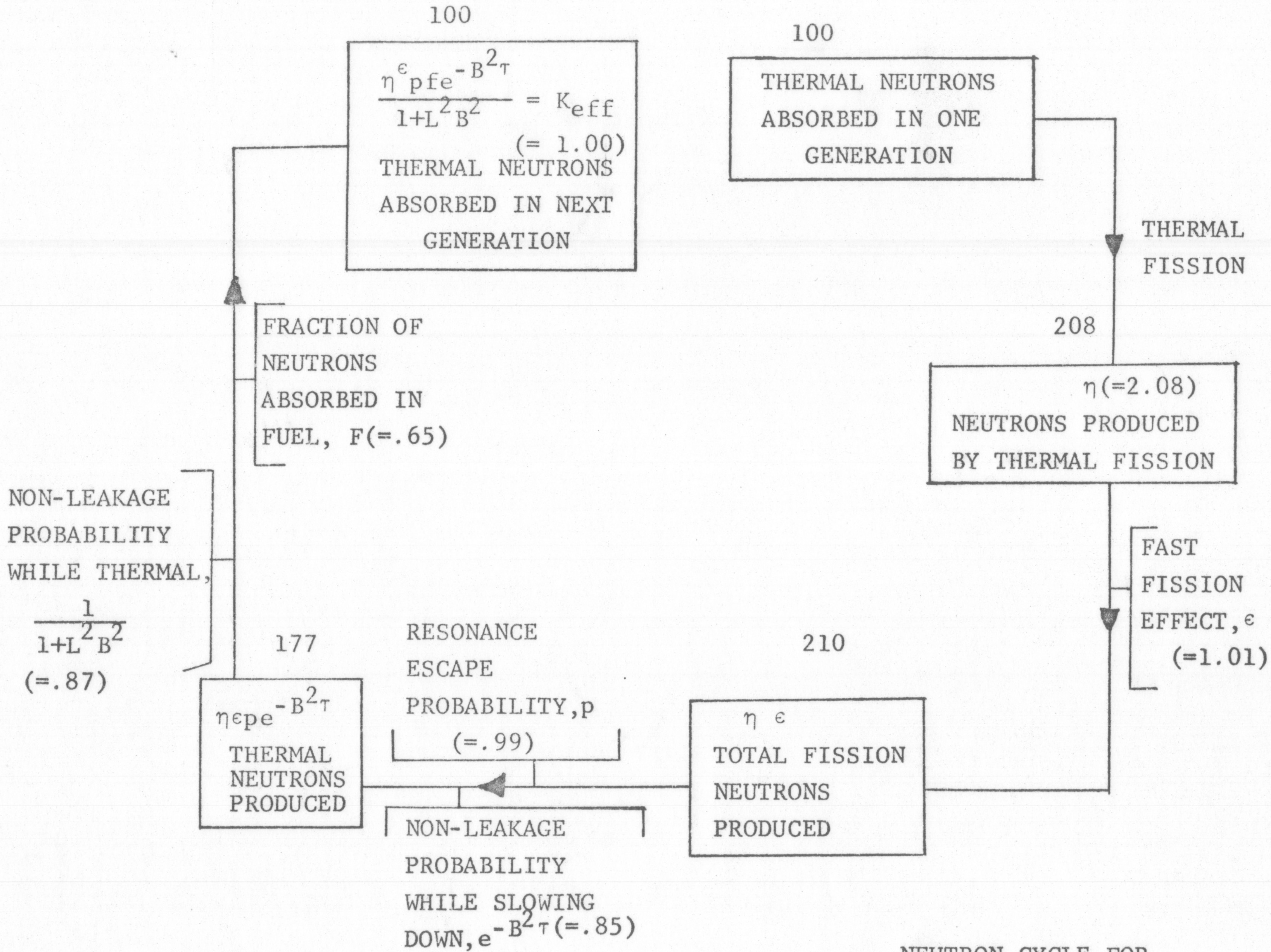
The deviation in (K_{eff}) from unity is defined as ($\Delta K = K_{eff} - 1$); the amount by which the effective multiplication factor differs from unity. A more useful quantity, mentioned in the abstract, is the reactivity (ρ), defined as:

$$\rho = \frac{K_{eff} - 1}{K_{eff}} = \frac{\Delta K}{K} \qquad \text{(IV-1)}$$

Since frequently we deal with systems with (K_{eff}) very close to unity:

$$\Delta K = K - 1 \qquad \Delta\rho = \frac{K-1}{K} \rightarrow K-1, \text{ which implies}$$

$$\Delta K \approx \Delta\rho \qquad \text{(IV-2)}$$



NEUTRON CYCLE FOR
THERMAL FISSION

Figure IV-1

This relationship is approximately correct for a reactor when (K_{eff}) is very close to unity.

The reactivity that is built into a reactor is suppressed by means of control rods. During startup control rods are withdrawn until supercriticality is achieved. The power level increases to the desired point; then the rods are re-inserted somewhat to achieve exact criticality. During the course of operation some rods may have to be repositioned to compensate for various perturbations. This re-positioning may vary from substantial changes in commercial power reactors to small or negligible changes in low or zero power test reactors.

Suppose a reactor is exactly critical and a control rod is withdrawn a small, known amount. This action will put the reactor on a sustained positive period which can be determined experimentally by utilizing the period meter, the linear power channels, or the power channels. Suppose the power increases by a factor of ten in fifty seconds. The reactor period (T) is related to power level as follows:

$$T = \frac{t}{\ln(P_2/P_1)} \qquad \text{(IV-3)}$$

given:

$$\frac{P_2}{P_1} = 10 \qquad t = 50 \text{ seconds}$$

therefore:

$$T = \frac{50 \text{ seconds}}{\ln(10)} = 21.7 \text{ seconds}$$

Thus an incremental rod withdrawal can be identified with a change in power level and eventually with reactor period. It is desirable to relate these parameters to reactivity, which is accomplished by the use of the in-hour equation.

The derivation of the in-hour equation is somewhat lengthy and consequently will not be repeated here. Again refer to Glasstone and Sesonske, NUCLEAR REACTOR ENGINEERING, pp. 246-251 for a detailed presentation. The final result is:

$$\rho = \frac{\ell*}{T} + \sum_{i=1}^{6} \frac{\beta_i}{1+\lambda_i T} \qquad \text{(IV-4)}$$

where:

$\ell* = \frac{\ell}{K_\infty}$ = prompt neutron generation time

β_i = delayed neutron fraction for i$\underline{\text{th}}$ group

λ_i = decay constant for i$\underline{\text{th}}$ group

index i = 1,2,3,4,5,6 for each of the six delayed neutron groups

It is easily seen that, given reactor period, it would be a fairly lengthy computation to obtain reactivity. The reverse procedure of obtaining period, given reactivity, would involve solving a seventh order equation. Therefore, many versions of approximating this solution have been developed; most of them use a one-group approximation for the six delayed groups. One form which has been found useful is:

$$T \approx \frac{\ell*}{\rho} + \frac{(\beta_{eff}-\rho)\tau}{\rho} \qquad \text{(IV-5)}$$

where for the RPI Critical Facility core:

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

β_{eff} = effective delayed neutron fraction ≈ .0078

τ = mean delayed neutron half-life ≈ 12.2 seconds

The expression for the period (T) is divided into two terms for convenience because one term usually dominates for a given reactivity change (ρ). For example, assume (ρ=.001) is added to a critical system. Thus:

$$T \approx \frac{.0002}{.001} + \frac{(.0078-.001)\ 12.2}{.001}$$

$$T = .2 + 82.96$$

$$T \approx 83 \text{ seconds}$$

This is reasonably close to the exact value as determined by the in-hour equation. For this case the first term is insignificant. Now assume (ρ=.0078) is added to a critical system.

Thus:

$$T \approx \frac{\ell^*}{\rho} + \frac{(.0078-.0078)\ \tau}{.0078}$$

$$T \approx \frac{.0002}{.0078} \approx .025 \text{ seconds}$$

In this case the second term is identically zero.

This rough calculation is acceptable for order of magnitude accuracy, but for control rod calibrations a much more accurate relationship between reactor period and reactivity is necessary. This relationship is plotted in Figure IV-2 for periods ranging from 20 - 80 seconds; normal operating periods. To this point reactivity, deviation from critical, has been considered in very small decimals only and more convenient units are desirable. Because of the special significance of the prompt critical condition, it has been adopted as the basis for reactivity units; namely reactivity in dollars ≡ (ρ/β_{eff}). When prompt critical a reactor has a reactivity of exactly one dollar. With (β_{eff}=.0078) a prompt critical reactor has an effective multiplication of about 1.0078. Note that one cent is equivalent to one hundreth of a dollar.

Thusfar the delayed neutron fraction has been referred to as (β_{eff}), as distinguished from (β). For U^{235} (β=.0065), yet for the RPI Critical Facility (β_{eff}=.0078). The difference between (β) and (β_{eff}) is due to the difference in energies with which the prompt and delayed neutrons are born. For applicational purposes (β_{eff}) is much more useful than (β); rather than carry the subscript the term (β) will actually mean effective delayed neutron fraction.

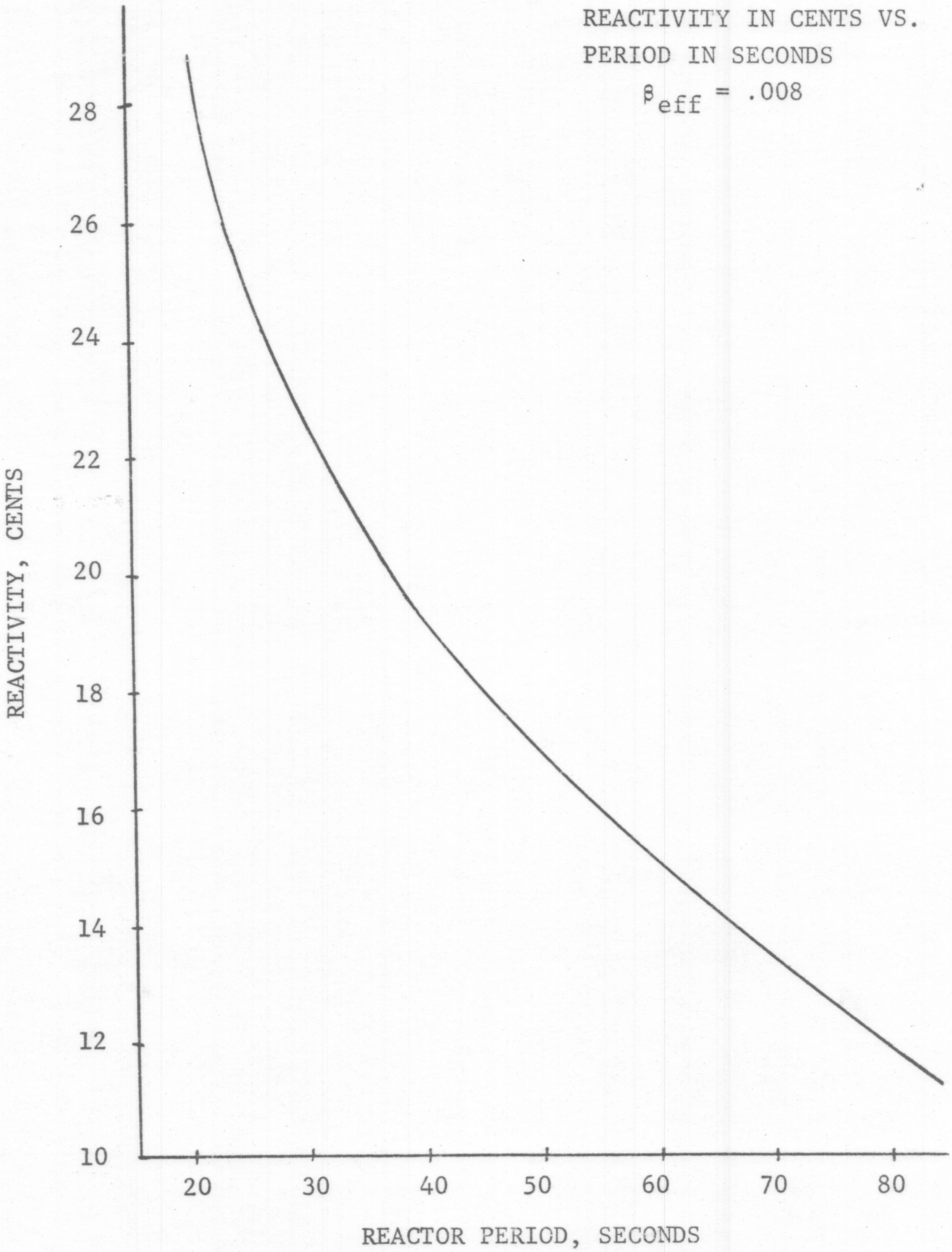


Figure IV-2. In-Hour Equation

Procedure-Positive Period Method

In the positive period method the critical position of the rods is related to some period caused by withdrawal of a rod or rods by a known amount. Suppose it is necessary to calibrate a four rod bank. First, criticality is achieved at some low power level with all rods at approximately the same height. The rods can then be withdrawn some distance such that a reasonable period, perhaps 30-50 seconds, is achieved. This period can be measured accurately with a stopwatch. By referring to the graph of the in-hour equation, Figure IV-2, the reactivity in cents is known. Divide this value by the distance from the critical position to the supercritical position; a rod worth in cents/inch is said to apply at the midpoint of the critical and supercritical positions. By repeating this procedure at various positions a curve such as Figure IV-3 can be generated for the supercritical portion of rod travel. For example:

CRITICAL BANK POSITION	19.20 INCHES
BANK WITHDRAWN TO	19.70 INCHES
PERIOD MEASURED	55 SECONDS

From Figure IV-2, reactivity = 15.5 cents, rod worth =

$$\frac{15.5 \text{ cents}}{19.70\text{-}19.20 \text{ inches}} = \frac{15.5 \text{ cents}}{.50 \text{ inches}} = 31 \frac{\text{cents}}{\text{inch}}.$$

This worth applies at:

$$\frac{19.20 + 19.70}{2} = 19.45 \text{ inches}$$

There are several purely mechanical features to be noted here. The period can be measured in several different ways. First it can be read directly from the period meters, though especially for long periods this is not particularly reliable. The linear power channels lend themselves quite nicely to timing a power increase by a factor of (e), such that the period can be read directly from the stopwatch. For

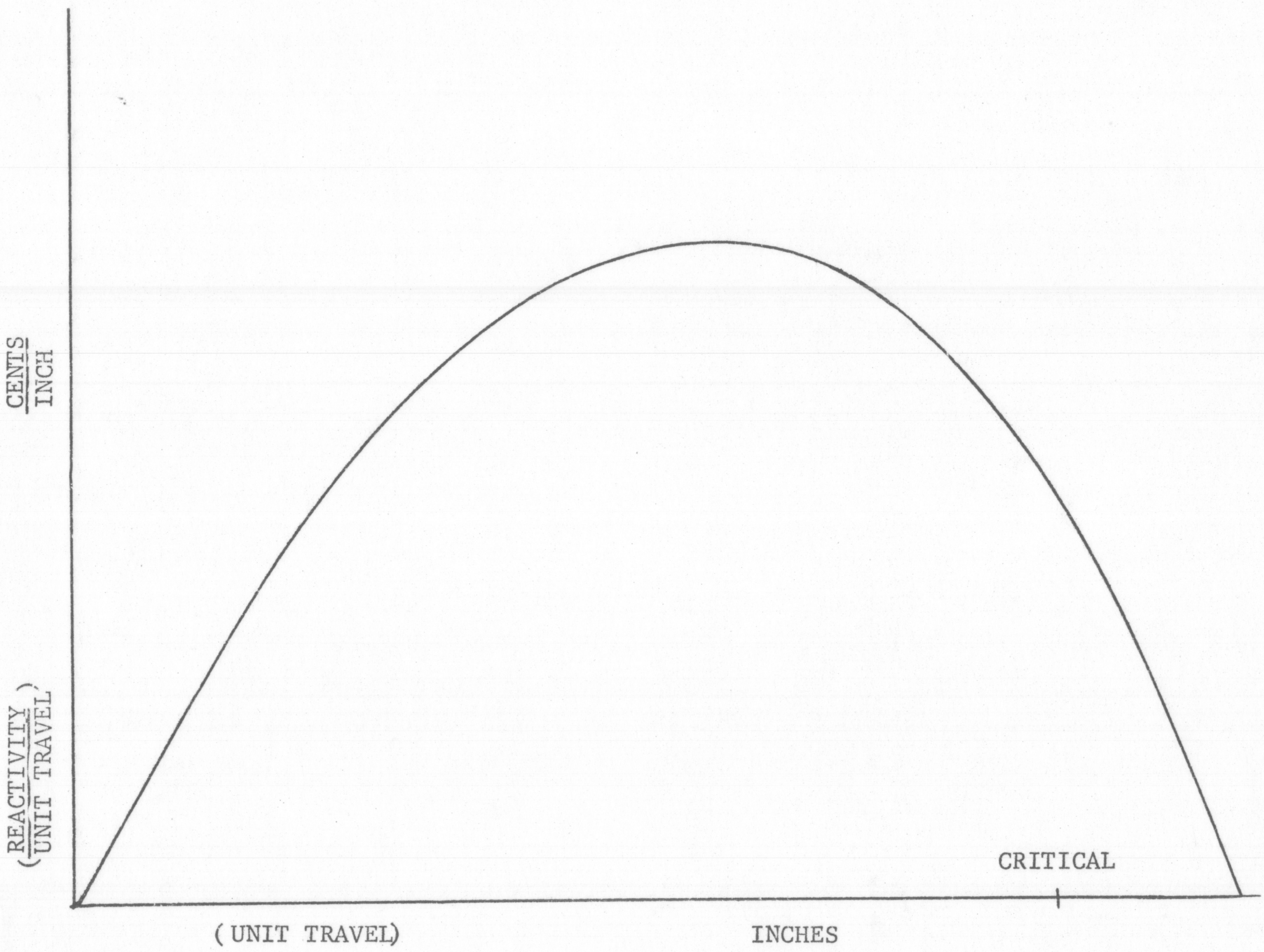


Figure IV-3. Differential Control Rod Worth Curve any given configuration this procedure can be repeated several times to obtain an average value. Still another technique is to time a one or two decade power increase on either of the power channels. Using the relation ($T = \frac{t}{\ln (P_2/P_1)}$), the period can be obtained. The reader is advised to consider the reliability or accuracy of these or other methods in a relative sense.

This procedure can be used to calibrate either the bank or a single rod. Obviously some configurations would yield a period faster than the normal operating maximum of about twenty seconds. In the case of a single rod calibration this problem is easily overcome by inserting the rod bank slightly. In this way the rod to be calibrated can be withdrawn completely without initiating excessively fast periods. It only necessitates the extra step of establishing criticality with the banked rods inserted further. In the case of a bank calibration this technique is not useful; consequently the critical bank position must be raised by the use of some poison material. It would be most accurate to borate the H_20 moderator uniformly so as to minimize the perturbation. A reasonably good approximation of uniform poisoning is the addition of boron strips in the four corner stationary fuel elements. These strips would be located far from the control rods themselves. Their positions could be adjusted slightly to yield any desired reduction in excess reactivity.

One common error in period determination is to begin timing a power level increase prior to the time the reactor system reaches equilibrium. The addition of reactivity to a critical system results in a prompt response by the system. It is necessary to wait for steady-state before timing power level increases. Another common perturbation in this exercise is the reliability or repeatability of the critical position.

One should be selective in verifying it such that a reasonable value is obtained without undue time losses. For these measurements small deviations in the critical position will not affect numerical results significantly; but in future experiments precision and refined technique become crucial. A systematic control rod operation is much more likely to yield reliable data than a haphazard one.

The last warning in this section concerns the phenomena of rod worth reversal; when rod withdrawal results in a negative reactivity addition or rod insertion results in a positive reactivity addition. Flux conditions must be somewhat unusual for this to take place. The phenomena is more characteristic of small, highly enriched cores with fuel followers near the upper limit of travel. If this is encountered it can be drawn on the differential rod worth curve.

Theory-Calibration in the Subcritical Range

The previously discussed techniques are adequate for reactivity changes above critical, however at the RPI Critical Facility a large portion of rod travel is in the subcritical range. The technique used for rod calibration here is subcritical multiplication. To be useful the rods must have been previously calibrated in the supercritical range by the positive period method. The rod worth at the critical position is used as the basis for the subcritical calibration.

The basis for rod calibration in the subcritical range is given by the following relationship:

$$\frac{\varphi_s}{\varphi_t} = \frac{C_s}{C_T} = 1-K = \Delta K \quad \text{which near critical}$$

$$\approx \frac{\Delta K}{K} = \Delta\rho \qquad \text{(IV-6)}$$

where:

φ_s = thermal flux due to source alone

φ_t = thermal flux due to source and core fissions

C_s = count rate due to source alone

C_t = count rate due to source and core fissions

K = effective multiplication factor (<1)

ΔK = deviation of K from critical

$\Delta\rho$ = amount reactor subcritical

In this relationship the ultimate aim is to determine ($\Delta\rho$). This is easily done once (ΔK) is determined, since ($\Delta\rho = \frac{\Delta K}{K}$). ($\Delta K$) is known once ($C_s$) and C_t) are known. (C_t) is quite easily found for any given configuration; it is simply the count rate for any channel. Thus the only problem is to determine (C_s). It is not practical to measure the source term directly. However (C_s) can be found with the use of the differential bank calibration curve from the positive period method. Consider Figure IV-4 on the following page. Assume a critical position of 19.5 inches with the differential rod worth curve given by the solid line in Figure IV-4. A reasonable assumption regarding the shape of the curve slightly subcritical would be a linear extrapolation, as shown by the broken line. A good approximation for (ΔK) at 19.35 inches would be the area under the curve from critical back to 19.35 inches, designated by ▨ . (C_t) can be taken; (ΔK) is known; and ($C_s = C_t \Delta K$). Should a check on (C_s) be desirable, the rods can be positioned somewhat lower, perhaps 19.20 inches. (ΔK) would now be given by the area under the curve from critical back to 19.20 inches, designated by ▨ & ▧ . A new (C_t) can be taken, and another (C_s) can be calculated. This procedure can be repeated as long as the approximation of linear extrapolation remains valid. Now that (C_s) is known, (C_t) can be taken for any configuration, and simple

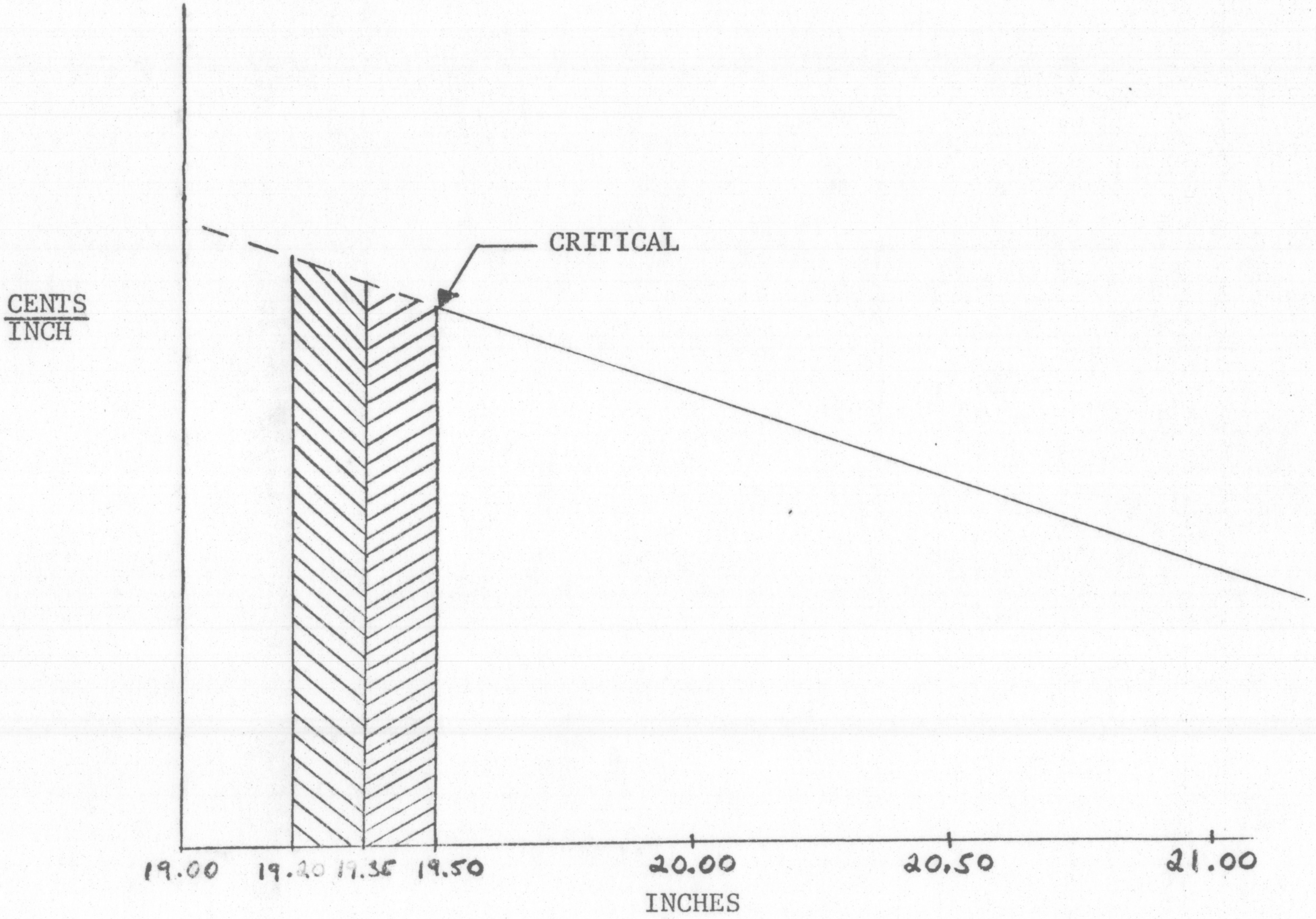


Figure IV-4. Supercritical Portion, Differential Control Rod Worth Curve division gives (ΔK). Recall that ($K = 1+\Delta K$), and ($\Delta\rho = \frac{\Delta K}{K}$), so ($\Delta\rho$) is known. Using ($\frac{\Delta\rho}{\beta} = \$$), reactivity can be converted to dollars and cents. Applying this manipulation to all positions tested will give data for an integral rod worth curve, as illustrated in Figure IV-5. Differentiation of this curve yields a differential rod worth curve for the entire length of travel.

Procedure-Subcritical Range

With the control rods fully inserted observe the neutron count rate on all channels on scale; namely the startup channels and any ionization chambers which may be on scale. The criterion for duration of counting is sufficient time for a representative count rate to be determined. Raise the rods as a bank incrementally and obtain (C_t) for each desired configuration. Be selective in choosing bank positions to obtain these counting rates. The differential rod worth curve is not a straight line and more points are needed closer to the curved portions of the plot. Remember that you want the counting rates to reflect steady-state conditions only. Some reactivity additions may require some waiting time between the end of reactivity addition and the beginning of (C_t) determination. Counting rates may vary considerably from one detection system to another; note the significance of source-core-detector geometry.

Theory-Rod-Drop Method

The rod-drop technique is rather simple to execute but complex to analyze. Since this method is somewhat cumbersome to apply at the RPI Critical Facility, it will not be discussed in any depth, but merely presented as an alternate calibration technique. A detailed analysis is presented in Glower, EXPERIMENTAL REACTOR ANALYSIS AND RADIATION MEASUREMENTS, pp. 268-272. The rod to be calibrated

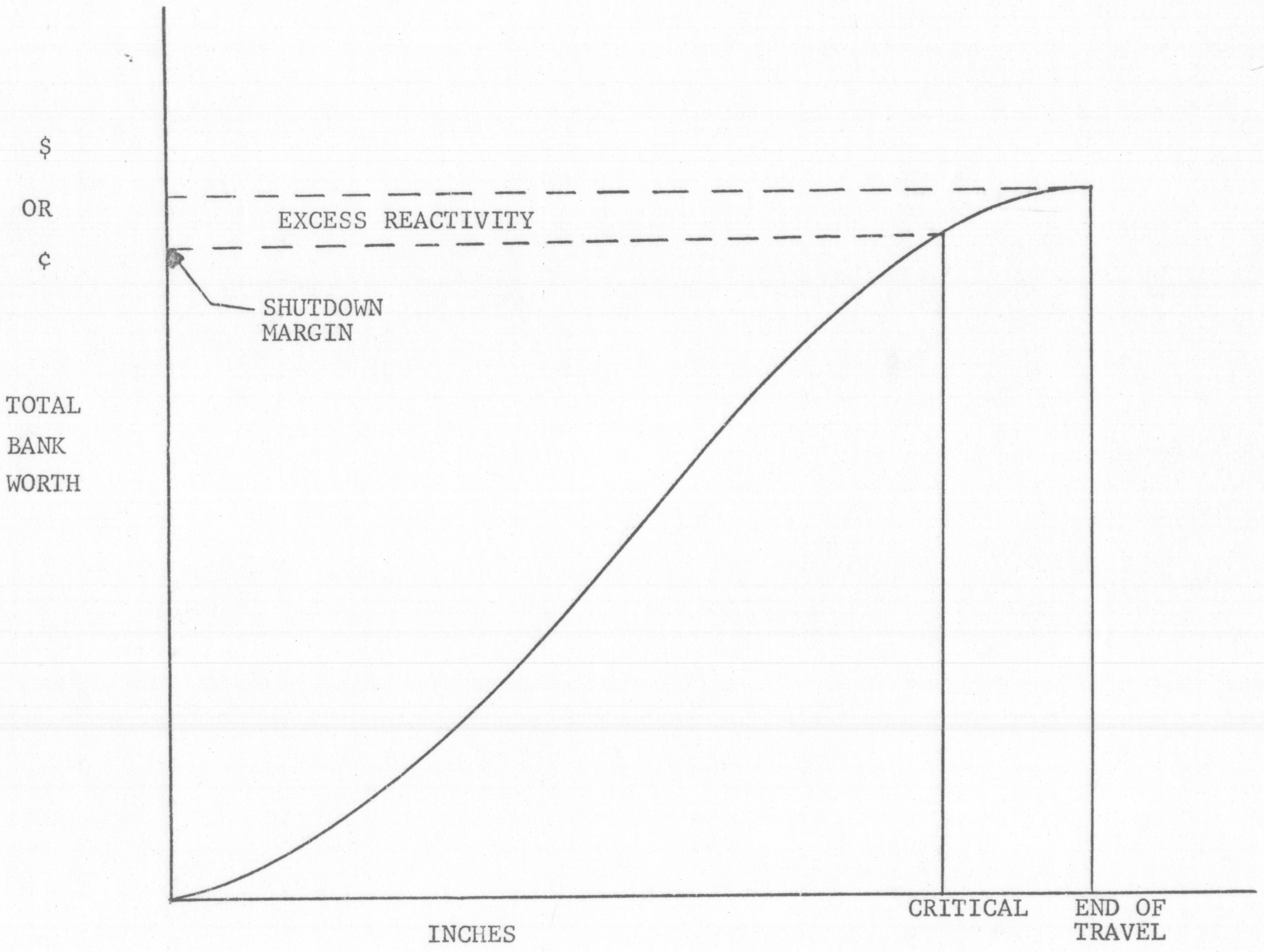


Figure IV-5. Integral Control Rod Worth Curve is dropped from different positions and the resulting decay of the neutron flux is observed and related to reactivity. For this observation it is best to have the output from the detector displayed on a fast response recorder. The equations normally used here are derived for the condition of no external neutron source; thus, it must be withdrawn before the rod is dropped. Unlike the previous method, subcritical multiplication of source neutrons must be avoided.

Establish criticality at a low power level and drop the rod to be calibrated. The decay of neutron flux can be followed by recorder or visual readings from a linear power channel. For purposes of calculation, it is desirable to have plots of normalized neutron flux as a function of reactivity. The curves differ, depending upon neutron lifetime and delayed neutron constants. An example of this is given in Figure IV-6, which is for the Argonaut reactor.

In reference to this illustration, assume that the neutron flux had decreased by a factor of two in fifty seconds. From the fifty second curve the reactivity worth of the length of the rod inserted is about thirteen cents. If this value is not reproducible for a given drop it follows that the data used in the theoretical expression is not in agreement with the physical system. In order to generate these curves for a given reactor one must solve equations for neutron density as a function of time for a step change in reactivity. This is beyond the scope of this publication, and the reader is referred to W. R. Kimel et al., NUCLEAR SCIENCE AND ENGINEERING, 6:233, 1959; for detailed analysis.

Another rod-drop analytical treatment is developed through space independent reactor kinetics equations. This method utilizes the fact that after a sudden insertion of negative reactivity (Δp), the ratio of <u>prompt</u> post-drop flux to pre-drop flux, (φ_1/φ_o), can be expressed by the following relationship:

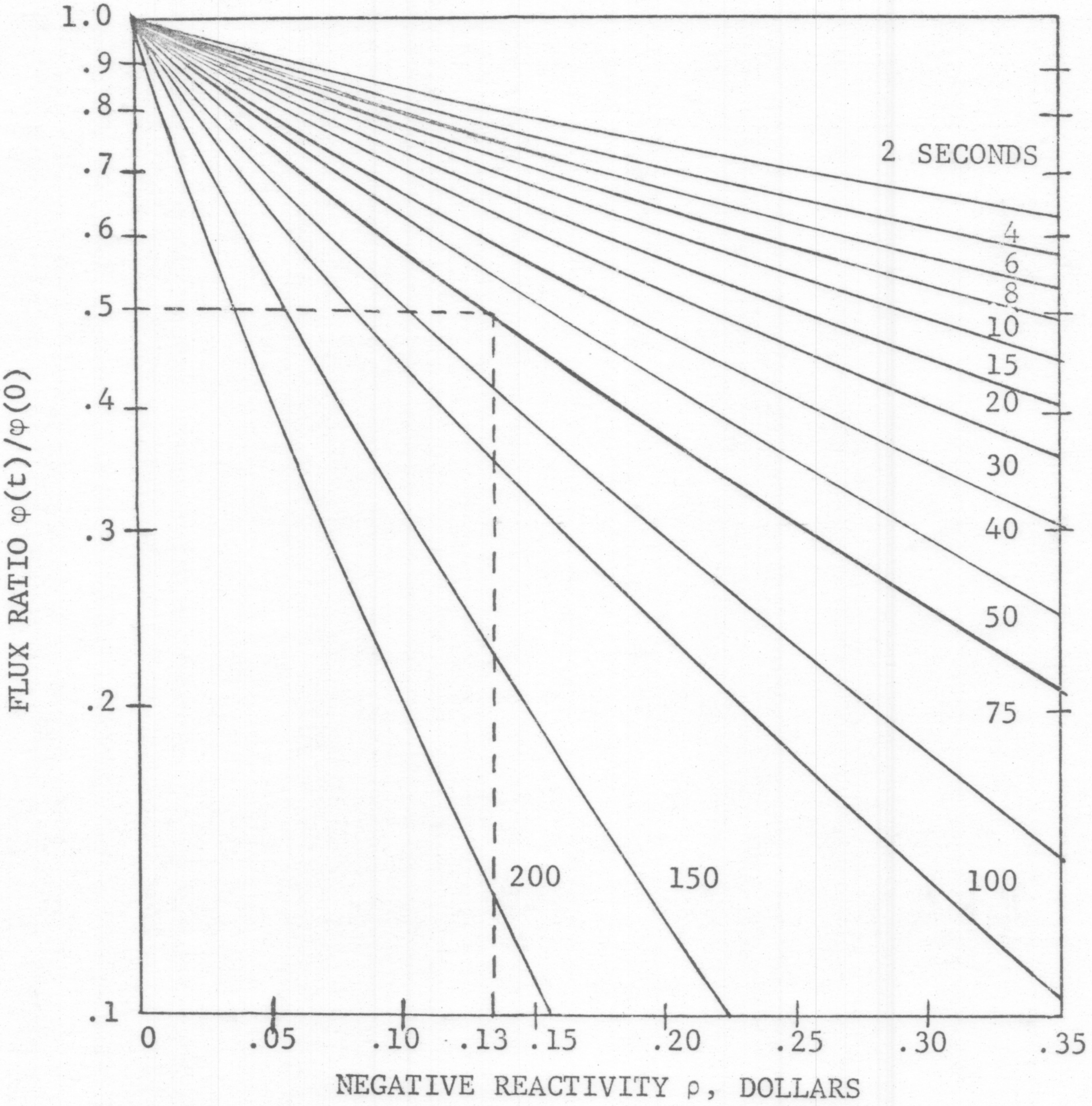


Figure IV-6. Relative Flux Ratio as a Function of Negative Reactivity for Rod-Drop Experiment

$$\frac{\varphi_1}{\varphi_o} = \frac{1}{1 + \frac{\Delta\rho}{\beta}} \qquad \text{(IV-7)}$$

rearranging,

$$\frac{\Delta\rho}{\beta} = \frac{\varphi_o - \varphi_1}{\varphi_1} \quad \text{or} \quad \frac{\varphi_o}{\varphi_1} - 1 \qquad \text{(IV-8)}$$

Note that the units of $(\Delta\rho/\beta)$ are dollars; the result is immediately available in reactivity units of dollars and cents. This relationship assumes that the reactivity insertion time is short compared to the decay time of the delayed neutron precursors; their concentration after the drop is essentially the same as the concentration at steady-state.

The success of this technique is quite dependent upon detector positioning and geometry considerations. Detectors must be so located that they measure changes in multiplication only, and not changes in that spatial distribution of the flux due to rod insertion. The experimenter is cautioned against blindly applying this simplification if the system under consideration does not lend itself to this analysis.

Analysis of Results

From the positive period method, a differential rod worth curve has been generated. Typical units are \$ or ¢ per inch versus inches. The subcritical multiplication method lends itself most conveniently to integral worth curve analysis, and this can be differentiated to yield a differential worth curve for the entire length of travel. The student is cautioned to be selective in choosing pairs of points on the integral curve so that the resulting differential curve has no inexplicable discontinuities. It is valid to use the midpoint of the two points on the integral curve as the location of the average worth on the differential curve. Of course the smaller the range, the more accurate the approximation. This data is invaluable in determination of many other core characteristics because a known control rod worth can be used as a basis of comparison for the worth of any other change in the reactor system. Some parameters of immediate interest include excess reactivity, shutdown margin, and maximum reactivity insertion rate. The results from rod drop tests can also be compiled in a similar manner and the data from the two methods can be compared.

Topics for Discussion

1. The RPI Critical Facility core is 93% enriched in the isotope U^{235}. (β) for U^{235} from textbooks is .0065; yet for this core ($\beta = .0078$). Explain this difference.
2. Why is a procedure for calculating (C_s) developed? Is it feasible to measure (C_s) directly?
3. Why is the integral worth curve shaped the way it is?
4. Where does the peak in the differential worth curve appear? Why?
5. In this experiment two or more detection systems were utilized, each had its own set of data. In all likelihood the results from these systems were not identical. Perhaps one system was far superior to the others. Discuss the reasons for this discrepancy. Include in your discussion considerations of relative reliability of data points.
6. In the procedure for subcritical multiplication you were directed to begin with the rods fully inserted and then withdraw incrementally. For analysis however, you first need data at, or slightly below, critical. Why is it preferable to begin with rods fully inserted?

Author's Note

The following four experiments: Temperature coefficient of reactivity, boron coefficient of reactivity, void coefficient of reactivity, and absorption cross-section measurements; overlap considerably with respect to both theory and experimental technique. For example, theoretical considerations of the six-factor-formula explained in detail in the temperature coefficient experiment apply to any experiment where reactor multiplication is perturbated. In any of these exercises, changes in the critical position may ultimately be explained by use of the six-factor-formula. Instructions regarding safe handling of fuel elements in the boron coefficient experiment certainly apply to the void coefficient. Because of this overlap extensive redundancy will be avoided. The reader is urged to consider the material in a cumulative sense rather than treating each experimental exercise as an individual unit.

CHAPTER V

TEMPERATURE COEFFICIENT OF REACTIVITY

Abstract

In nuclear reactors, temperature changes are inherent because the fission event releases energy, most of which quickly appears as heat in the fuel. Because of the interaction between reactor temperature and reactor multiplication, many of the physical and nuclear properties of the core are temperature dependent. The magnitude and sign of the change in multiplication caused by a unit increase in temperature is known as the temperature coefficient of reactivity. The overall temperature coefficient is the sum total of a large number of localized temperature coefficients such as: fuel, moderator, reflector, and coolant. A particular local coefficient is not necessarily of the same sign as the overall temperature coefficient.

Practically speaking, the convenient operation of a reactor requires a small temperature coefficient in order that criticality be easily maintained. This concept is especially important considering the large temperature change in the transition from a cold, clean system to full power operation. Usually in addition to its small magnitude, the overall reactor temperature coefficient is designed to be negative in order to increase stability and to enhance the safety of the reactor.

In this experiment, the overall reactor temperature coefficient of reactivity is measured. At the RPI Critical Facility this is accomplished by heating the water moderator with immersed electrical heaters and monitoring the moderator temperature. This capability is quite unique. In most power reactors the so-called temperature coefficient is actually a power coefficient where many other parameters are varying as well as those associated with the temperature

coefficient. The reactivity associated with a temperature change is determined from the measured movements of a calibrated control rod which are required to maintain criticality. The reactivity change will be measured from ambient ($\approx 68^{\circ}$F) to approximately 140°F.

Introduction and General Theory

The qualitative relationship between temperature and reactivity will be developed from a consideration of the effect of temperature on the individual parameters of the two-group criticality equation:

$$K_{eff} = \eta \epsilon p f \frac{1}{1+L^2B^2} e^{-\tau B^2} \qquad \text{(V-1)}$$

where: η = number of fast neutrons born per thermal neutron captured in fuel

$$= \nu \frac{\sigma_f}{\sigma_a} \qquad \text{(V-2)}$$

ϵ = fast fission factor; number of fast neutrons slowing down past the U^{238} fast fission threshold from thermal fission of U^{235} atoms.

p = resonance escape probability; probability of a fast neutron escaping non-fission capture while slowing down to thermal energies.

$$= \exp \left[- \frac{N_F V_F \, I}{\xi_F \Sigma_{pF} V_F + \xi_M \Sigma_{sM} V_M}\right] \qquad \text{(V-3)}$$

where: N_F = atomic density of fuel
V_F = volume of fuel
V_M = volume of moderator
I = resonance integral
$\xi_F = \overline{\Delta\mu_F}$ = average increase in lethargy for fuel

$\xi_M = \overline{\Delta\mu_M}$ = average increase in lethargy for moderator

Σ_{pF} = macroscopic (n,p) cross-section in fuel

Σ_{sM} = macroscopic scattering cross-section in moderator

f = thermal utilization; number of thermal neutrons absorbed in fuel per thermal neutron absorbed

$$= \frac{\Sigma_{aF}}{\Sigma_{aF}+\Sigma_{aM}+\Sigma_{aI}} \qquad \text{(V-4)}$$

where the subscript (I) refers to impurities.

$e^{-\tau B^2}$ = fast non-leakage probability

$$= P_F \qquad \text{(V-5)}$$

where: τ = neutron age

B^2 = geometric buckling

$\frac{1}{1+L^2B^2}$ = thermal non-leakage probability

$$= P_{TH} \qquad \text{(V-6)}$$

where (L) represents the thermal diffusion length.

The reactor is critical when ($K_{eff} = 1$) or, on the average, one of the neutrons born in a thermal fission causes another fission. In this case the neutron population is independent of time or is self-sustaining. If ($K_{eff} \neq 1$), the population will continuously change; increasing if ($K_{eff} > 1$) and decreasing if ($K_{eff} < 1$). The rate of change of neutron population is a function of the reactivity (ρ), defined as:

$$\rho = \frac{K_{eff}-1}{K_{eff}} = \frac{\Delta K}{K} \qquad \text{(V-7)}$$

The temperature coefficient is defined as:

$$\frac{d\rho}{dT} = \frac{d(\frac{\Delta K}{K})}{dT} = \frac{1}{K_{eff}} \frac{dK_{eff}}{dT} \qquad \text{(V-8)}$$

Usually for safety and stability considerations, the overall temperature coefficient is designed to be negative. Thus, as the temperature of a critical reactor increases, (K_{eff}) becomes less than one and the reactor is subcritical at the new temperature. On the other hand, if $(\frac{dK_{eff}}{dT})$ were positive, as the fission process released heat the reactor would be unstable and become increasingly more super-critical as the temperature increased.

The effects of temperature on the factors of equation (V-1) arise mainly from neutron energy dependence on cross sections, and from the variation of macroscopic cross sections due to variations in core density. The effect of volume changes on the geometric buckling can be an important effect. With increasing temperature some factors increase, some decrease, and some remain constant; the overall coefficient depends upon which factors dominate.

Differentiating equation (V-1) with respect to (T) and dividing by (K_{eff}) gives:

$$\frac{1}{K_{eff}} \frac{dK_{eff}}{dT} = \frac{1}{\eta}\frac{d\eta}{dT} + \frac{1}{\epsilon}\frac{d\epsilon}{dT} + \frac{1}{p}\frac{dp}{dT} + \frac{1}{f}\frac{df}{dT} + \frac{1}{\frac{1}{1+L^2B^2}} \frac{d(\frac{1}{1+L^2B^2})}{dT} + \frac{1}{e^{-\tau B^2}} \frac{d(e^{-\tau B^2})}{dT} \qquad \text{(V-9)}$$

Examination of this equation term by term gives a qualitative understanding of $(\frac{d\rho}{dT})$.

(a) Temperature Coefficient of η

For fuel of a single fissile isotope:

$$\eta = \nu \frac{\overline{\sigma_f}}{\overline{\sigma_a}}$$

(ν) is essentially constant at thermal energies, such that the coefficient is entirely due to variations in the ratio $\left(\frac{\overline{\sigma_f}}{\overline{\sigma_a}}\right)$ with temperature. For any given fissile isotope variations in (η) can be determined by considering energy dependence on (σ_f) and (σ_a). This calculation can be done using the Wescott formalism for computing average thermal cross sections. The $\left(\frac{d\eta}{dT}\right)$ can be either positive or negative in sign, but usually is small in magnitude relative to other factors.

(b) Temperature Coefficient of ϵ

In a heterogeneous system (ϵ) may vary slightly with temperature for two reasons. Thermal expansion of the fuel lumps which indirectly determine escape probabilities of fast neutrons from the fuel tend to increase these probabilities somewhat. Simultaneously, a temperature increase tends to flatten the thermal flux in the fuel. The resulting change in the spatial distribution of the primary fissions can be shown to decrease the escape probability of the primary fission neutrons from the fuel. This effect can be either positive or negative, but in general this effect is so small that (ϵ) can be considered to be temperature independent.

(c) Temperature Coefficient of p

Because of Doppler broadening of the (U^{238}) resonance peaks with increasing temperature, (p) decreases. The reason the resonance peaks broaden

(Doppler effect) with increasing temperature is that the microscopic cross sections actually depend upon the relative motion of the neutron and the nucleus. At room or moderate temperatures, the nucleus can be considered at rest; only neutrons in a very narrow energy range will "see" the large resonance cross section. On the other hand, at higher temperatures a non-resonant-energy neutron may strike a nucleus whose velocity is such that the interaction energy corresponds to a resonance peak. Another possibility is that resonance-energy neutrons interact with nuclei whose velocities are such that the interaction energy does not lie within a peak. The overall Doppler effect of temperature is to make the peaks broader and lower. This negative effect is quite important for natural or slightly enriched uranium fuels. However, with higher enrichments of (U^{235}) this effect decreases in importance.

(d) Temperature Coefficient of (f)

For a heterogeneous reactor the thermal utilization is given by:

$$f = \frac{\Sigma_{aF} V_F}{\Sigma_{aF} V_F + \Sigma_{aM} V_M \xi} \qquad \text{(V-10)}$$

Where (ξ) is the disadvantage factor.

It is important to know how (ξ) varies with temperature. This analysis is quite complex and cannot be presented here; the result is that the temperature coefficient of (ξ) is always negative. Details are provided in Lamarsh, Nuclear Reactor Theory, pp. 453-458. This effect implies that the temperature coefficient of (f) is positive. The reason for this may be seen most clearly in the diffusion approximation. Since increasing temperature decreases moderator density, thereby increasing absorption and scattering mean free paths, the thermal diffusion length increases with temperature. As the diffusion length increases, the flux in the unit cell tends to flatten; i.e. the flux depression in the fuel is less pronounced. This leads to a smaller value of (ξ).

A negative effect on (f) results from considering cross section dependence on temperature. The (U^{235}) does not demonstrate a (l/v) dependence, but if other materials are (l/v) this would yield a negative effect on the temperature coefficient of (f). The relative magnitudes of these effects depend upon the details of the system under consideration.

(e) Temperature Coefficient of $\frac{1}{1+L^2B^2}$

As previously explained a decrease in moderator density results in an increase in thermal diffusion length. The expansion of core volume with temperature would have a compensatory effect. The buckling decreases thereby increasing the non-leakage probability. Ususally this effect is small compared to the effect of increased diffusion length; overall, the non-leakage probability decreases.

(f) Temperature Coefficient of $e^{-\tau B^2}$

Just as the thermal diffusion length increases with temperature, so does the neutron age (τ). Even considering the effect of decreased buckling, the non-leakage probability decreases with increasing temperature.

The RPI Critical Facility core is enriched in the isotope (U^{235}) to 93%. The temperature coefficient of this reactor will involve a specialized treatment of the previous analysis. It is left as an exercise to the student to quantify the temperature coefficient of this reactor.

Procedure

The normal method of maintaining core criticality is withdrawal or insertion of a calibrated control rod as the water temperature is increased by an external, non-nuclear heat source. From the known control rod worth, the measured rod travel, and the measured change in core temperature, the temperature coefficient of reactivity may be determined as a function of moderator temperature. The H_2O moderator in the reactor tank will be heated by two electric immersion heaters and kept uniform by a motor driven mixer. Water temperatures will be monitored by thermocouples.

The integral temperature deficit is obtained by integrating the control rod worth curve within the limits of the initial and final control rod positions. The differential temperature coefficient of reactivity is obtained by integrating the control rod worth curve over a small increment of control rod travel and dividing by the corresponding temperature change to obtain the coefficient in cents per degree fahrenheit. Specifically:

$$\left[\frac{(\text{¢/IN})_{\text{initial}} + (\text{¢/IN})_{\text{final}}}{2}\right]\left[\frac{(\text{inches})_{\text{final}} - (\text{inches})_{\text{initial}}}{(\text{TEMP})_{\text{final}} - (\text{TEMP})_{\text{initial}}}\right] = \frac{\Delta\rho}{\Delta T} \qquad \text{(V-11)}$$

The temperature coefficient is considered to apply at the mid-interval temperature.

At this point the student may anticipate the fact that these reactivity changes will be relatively small and slow, thus furthering the importance of good experimental technique. The following general operating procedure is suggested: Bring the reactor to exact criticality. Note the calibrated control rod position and core temperature. At this time the agitator should be running. After syncronization of temperature recorder charts with reactor instrumentation charts, start the heaters. With the addition of heat the reactor will become either subcritical or supercritical, depending upon the sign of the temperature coefficient. After recognition of this deviation from criticality, the control rod should be withdrawn or inserted slightly to counteract this effect. This places the reactor on a very slight period. As the moderator continues to be heated the power level reaches either a maximum or a minimum. This maximum or minimum corresponds to the time (and temperature) of the new critical position. This procedure is illustrated in Figure (V-1).

There is no need to re-establish criticality with the heaters off once the first measurement has been completed. In fact, more exact measurements are possible by comparing successive power minimums or power maximums.

The last mechanical problem concerns the magnitude of rod movement necessary to yield optimum results. Again one must obtain accurate data without unnecessary expenditure of time. Initially, the heat-up rate can be calculated. This

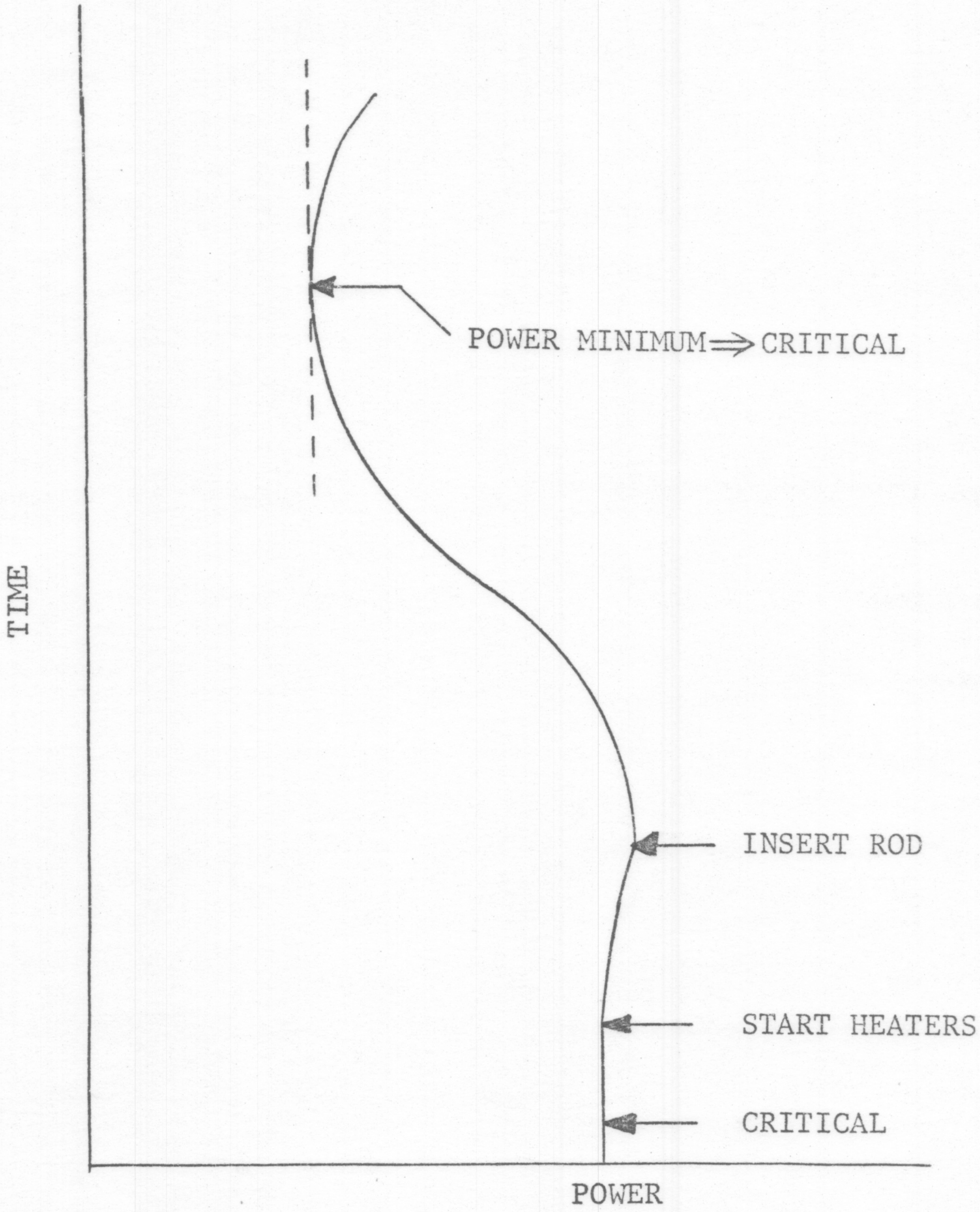


Figure V-1a. Typical Trace for Positive Temperature Coefficient

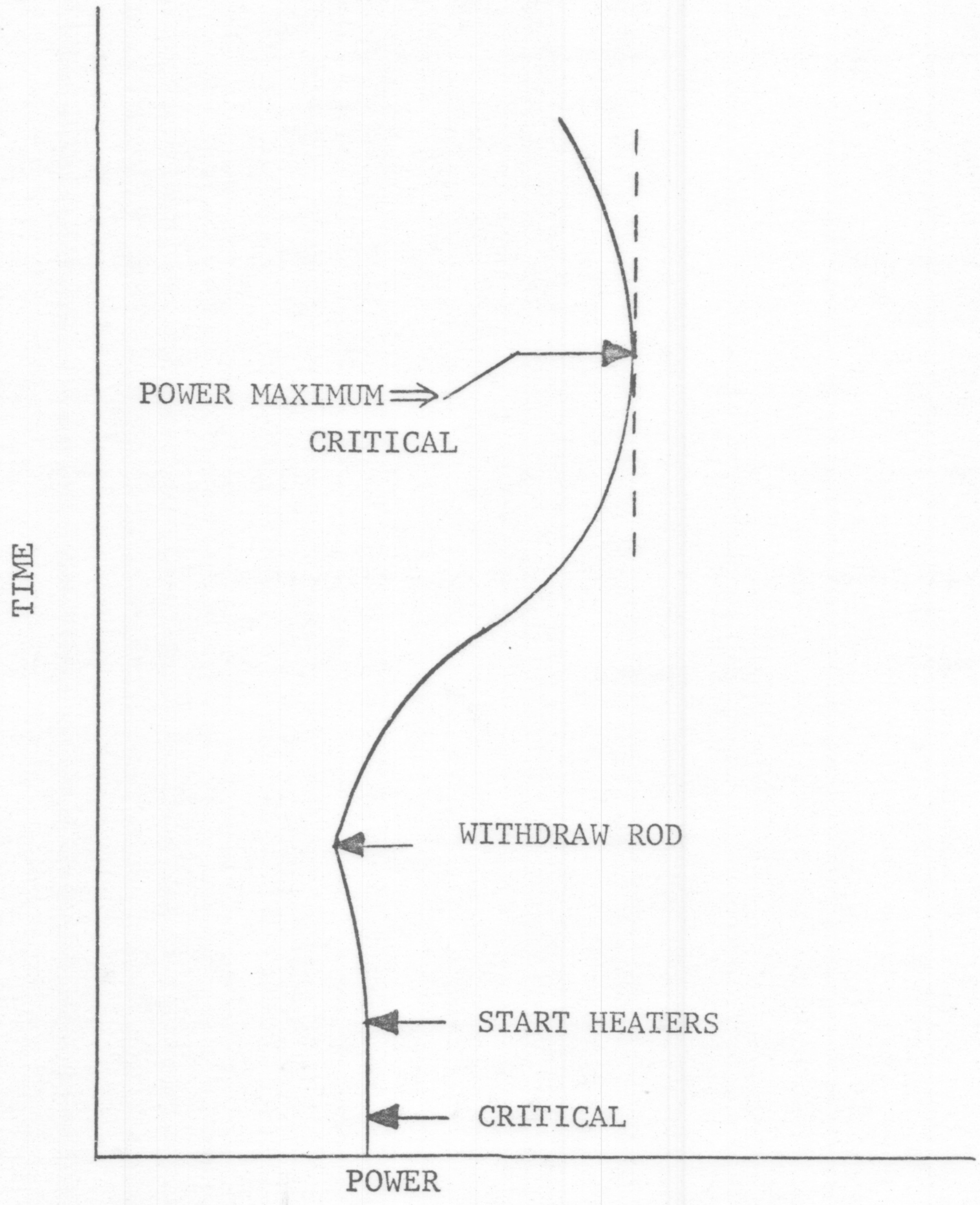


Figure V-1b. Typical Trace for Negative Temperature Coefficient information is of no use without an understanding of the magnitude of the coefficient; this can be roughly determined by observation of the power recorders after turning on the heaters. Knowing this information as well as the control rod worths enables the student to estimate a reasonable rod movement. If the distance the control rod is moved is too short, the limit of error of the data point exceeds 100%. If the distance is too long it will take hours to obtain one data point. With the use of some trivial calculations, compromise between these two extremes is possible.

Analysis of Results

A plot of the temperature coefficient versus temperature is appropriate.

The heaters at the RPI Critical Facility are rated at (18 KW) each. The reactor tank holds 2000 gallons of water. From this information the heat-up rate observed can be compared to a calculated value. What is the total reactivity worth over the temperature range measured? Calculate the maximum reactivity insertion rate.

Topics for Discussion

1. Discuss the importance of a thorough understanding of the temperature coefficient of any reactor.
2. Discuss in detail the reasons for the shape of the curve which was generated. With this data can you determine which terms of the six-factor-formula are dominant?
3. All reactors are designed to have a negative temperature coefficient. Consider the trade-offs of increased safety versus increased cost.
4. Extrapolate the curve to higher temperatures and explain why you expect the curve you indicate. Since the coefficient is negative, hypothesize what ultimately happens during an excursion.

5. In reactors where light water acts primarily as the coolant and moderation takes place in another medium, the temperature coefficient may be positive. Explain.

CHAPTER VI

BORON COEFFICIENT OF REACTIVITY

Abstract

The use of a nuclear reactor to measure absorption cross sections of certain elements was formulated early in the history of reactors. Motivation for this activity was supplied by unknown or doubtful thermal neutron cross sections and the need to measure the purity of materials which were under consideration for use as moderators, reflectors, structural components, etc.

A critical reactor is highly sensitive to the loss of neutrons when an absorbing material is placed in the core. The material is "dangerous" as it pertains to the sustention of the neutron chain reaction; hence the terminology "danger coefficient," which will be discussed later. The neutron absorbing material reduces the value of the multiplication factor (K) by decreasing thermal utilization. If the material's effect on the moderating and scattering properties of the reactor can be assumed to be small or correctable, a measurement of the change in (K) will indicate the material's ability to absorb neutrons.

Unknown cross sections are determined through the use of a known cross section; a comparison of reactivity worths can be used to obtain the unknown cross section. The purpose of this experiment then, is to calibrate the reactor with a standard material with a (1/v)cross section. Since boron has a large, well-documented thermal cross section, it will be used as the calibrating material.

Introduction and General Theory

When an absorber is placed in a reactor at a certain location, the number of neutrons absorbed depends upon the magnitude of the material's cross section over the neutron-energy range and the neutron flux-energy distribution. The effect of the loss of neutrons on the multiplication factor is dependent upon the fractional number of neutrons lost and the importance of these neutrons in keeping the reactor critical. An absorber placed at the edge of the reactor core will not have as great an effect on (K) as it would if it were placed at the core center. This is true because the absorption rate would be lower; the daughters of the neutrons on the core edge have a smaller non-leakage probability than those produced by neutrons at the core center. The maximum sensitivity of a reactor for an absorbing material can be expected to be located where the neutron flux is highest; usually near the core center.

Reactivity measurements of boron are made by observing the difference between the control rod position required to achieve criticality under a reference condition, and the position required with the boron sample at a given location. A zero or reference position will be established with a selected, calibrated rod located far from the test area and positioned where its calibration curve is essentially linear. The remaining three control rods are positioned to attain criticality. Following introduction of the test sample, the reactor will be brought critical on the calibrated rod, while the remaining three control rods are held in the same reference position. The measurement of reactivity worth is obtained from the difference in critical positions and the previously determined worth of the calibrated control rod.

Procedure

At the RPI Critical Facility boron is introduced into the reactor in the form of tape impregnated with natural boron to an area density of (1.0 mg/cm^2). This tape is convenient because self-shielding effects are negligable, even considering the large absorption cross section of boron. In addition the tape itself appears to be identical to liquid H_2O as far as neutrons are concerned. With this information the quantity of boron introduced for any test is easily determined.

Ideally, the boron coefficient of reactivity would be determined experimentally at every lattice position, but lack of time prohibits this. Having already loaded this core you can recognize some reasonably good approximations of symmetry. By measuring the boron coefficient in one quadrant and at various heights in the center port, the coefficient would be known for the entire core.

Excluding the technique of homogeneous liquid poison, there exists no flawless means of measuring the coefficient in each of the stationary elements. By measuring this parameter the same way for each lattice position however, relative magnitudes will be illustrated. One suggested procedure is to select a certain plate from the element, perhaps the plate that best approximates the average flux in that element, and tape a twenty-two inch strip of boron to it. It is also important to consider the plate's orientation in that element; i.e. does it face the center of the reactor or not. This procedure can be applied to each of the six stationary elements of the quadrant.

By considering the fuel loading of lattice position number 44, it is easily seen that the center of the reactor is essentially a thermal column. By noting the convenience of this lattice position for the purpose of future cross section measurements, one can foresee the need for more exact values here. Specifically it is desired to have the boron worth as a function of height. A smaller piece of tape can be introduced at various selected heights through the use of a center stringer supplied for this purpose. Of course a reference position will have to be established with only the center stringer in the core. When determining the size of the sample to be used, consider the sizes of unknown samples you may wish to investigate later and the error introduced by a variety of sample sizes.

A remark about safety precautions is essential at this point. Whenever materials are carefully transferred into or out of the reactor, the core instrumentation and control console must be in operation and monitored by the reactor operator. Precaution against personnel exposure to radioactive core components or test equipment must be guaranteed by strict regulation of material handling and radiation monitoring. Finally, all samples and experiments must be firmly secured in position prior to any rod movement.

Analysis of Results

Convert the information gathered into some standardized units; perhaps reactivity worth per gram of (B^{10}). If your data lends itself to graphical analysis, show the core trends.

Topics for Discussion

1. By using the six-factor-formula show quantitatively the change in (K_{eff}) due to boron poisoning.
2. List other suitable materials which may be used as a standard.
3. What error is introduced by ignoring the cross section of (B^{11}) which comprises over 80% of natural boron?
4. Discuss the core trends illustrated in measuring the boron coefficient. How does the coefficient in lattice position number 34 compare to that of number 43?
5. Discuss any errors involved in this experiment, both inherent and experimental.

CHAPTER VII

VOID COEFFICIENT OF REACTIVITY

Abstract

In the hazards evaluation of a broad class of reactors, it is important to know what is the reactivity effect introduced by a void in the core. This effect is described as being either positive or negative; negative refers to a reactivity decrease due to the void. The magnitude of the reactivity effect is usually expressed in units of reactivity worth per unit volume of void. The sign and magnitude of the reactivity due to the void is known as the void coefficient of reactivity.

Knowledge of the void coefficient is also important during the initial design and everyday operation of reactors, especially the boiling water reactor. For those reactors in which the void coefficient will be an important consideration in either the hazards or operational analysis, the coefficient is first calculated theoretically and perhaps checked experimentally in a critical facility. When the reactor is operative, void coefficient measurements are often early on the agenda.

The overall purpose of this experiment is to present some of the details and problems involved in void coefficient measurements. One is to determine how much void, and of what size and shape, should be used in this experiment. Another is to decide where and how the void should be introduced. Still another is to choose the best method of measuring the reactivity effect due to the void. The answers to the above problems will depend upon the particular reactor design and whether a certain local or a uniform void coefficient is being sought.

Introduction and General Theory

When a void is introduced into the core of a reactor, (K_{eff}) is usually changed. The magnitude and sign of the change are rather complex functions of core geometry, void size, and void location. The sign of the change is an important reactor safety consideration. If the coefficient is negative in a liquid-moderated reactor, an excursion which would result in boiling in the moderator would be self-limiting and the reactor would have an inherent safety system.

The term void coefficient of reactivity is defined as: $(\frac{\text{Reactivity Worth}}{\text{Unit Volume}})$; for example $[\frac{¢}{cm^3}]$. In stating a void coefficient, it is important to indicate whether it represents an average value for an approximately uniform void distribution throughout the core or the value for a localized void in a particular core location.

The net value of the void coefficient is actually a combination of several effects, some of which are positive and some negative. The sign of the coefficient depends upon which are dominant. For purposes of discussing the relationship between voids and reactivity, it is advisable to consider the two-group criticality equation in its usual form:

$$K_{eff} = \frac{\eta \epsilon p \; f \; e^{-B^2 \tau}}{1 + L^2 B^2} \qquad \text{(VII-1)}$$

The introduction of voids results in a decrease in the effective core density and since (τ) and (L^2) vary with density as:

$$\tau = \tau_o (\frac{d_o}{d})^2 \qquad \text{(VII-2)}$$

$$L^2 = L_o{}^2 (\frac{d_o}{d})^2 \qquad \text{(VII-3)}$$

Where (τ_o) and (L_o^2) are the age and diffusion area corresponding to the original density (d_o), and (τ) and (L^2) correspond to the new density (d); it is seen that they increase. This results in a decrease in the non-leakage probabilities and a decrease in (K_{eff}). The decrease in density might result in a small increase in core size; (B^2) would decrease thereby increasing (K_{eff}), but this effect is usually quite small.

For a heterogeneous reactor the thermal utilization can be written as:

$$f = \frac{\Sigma_{aF}}{\Sigma_{aF}+\Sigma_{aM}(\frac{V_M}{V_F})(\frac{\varphi_M}{\varphi_F})} \qquad \text{(VII-4)}$$

In the above expression (Σ_{aF}) refers to fuel and (Σ_{aM}) to the moderator, ($\frac{V_M}{V_F}$) is the volume ratio of moderator to fuel in a unit cell. The flux ratio ($\frac{\varphi_M}{\varphi_F}$) is the ratio of the average flux in the moderator to that in the fuel, the disadvantage factor (ξ). The effect of voids is to decrease both the volume ratio and (ξ), thereby increasing both (f) and (K_{eff}).

When voids are introduced the neutron moderation is decreased and the resonance flux is increased. This results in a decrease in resonance escape probability (p), therefore (K_{eff}) decreases. This decreased moderation results in an increased probability of a fast neutron with energy above the (U^{238}) fission threshold, thereby increasing the fast fission factor (ϵ) and (K_{eff}). The effect on (η) is quite small. With hardening of the neutron spectrum (ν) increases, but the ratio of (U^{238}) to (U^{235}) captures also increases.

In a reactor that is spoken of as being overmoderated, the moderator acts as a poison and the introduction of voids initially results in a net positive increase in reactivity.

If this effect dominates, the reactor has a positive void coefficient.

Another useful tool for analysis of void coefficients is perturbation theory. For a detailed presentation the reader is referred to Lamarsh, Nuclear Reactor Theory, pp530-534. The equation of interest here is:

$$\rho = \frac{\int_V [(\nu\delta\Sigma_F - \delta\Sigma_a)\varphi^2 - \delta D(\nabla\varphi)^2]dV}{\nu\int_V \Sigma_F\varphi^2 dV} \qquad \text{(VII-5)}$$

In this expression (δ) refers to the perturbation of either (D), (Σ_a), or (Σ_F), and other terms are defined in the usual manner. The results of this experiment can be explained qualitatively and semi-quantitatively by the use of this equation. This analysis is left as an exercise for the student.

Procedure

At the RPI Critical Facility polystyrene foam has been chosen to simulate voids. Given its composition, $(CH)_N$, it is possible to calculate the void fraction relative to what it is displacing, H_2O. This information must be used when calculating numerical values for the void coefficients.

With the considerations of core symmetry in mind a fairly thorough understanding of the void coefficient of this reactor can be gained by examining one quadrant and the center port. Many techniques of void introduction are possible; one technique is to use three polystyrene inserts with dimensions slightly smaller than a fuel plate for a stationary element. With these dimensions they can be easily introduced, perhaps on the reactive side of fuel plates two, five, and eight. By comparing this configuration to the clean core, reactivity worths can be determined.

This procedure is suitable for the stationary fuel elements, but the center port demands a slightly different approach. Here an understanding of the variation of the void coefficient with axial position is desirable. The center port is equipped with a stringer so that a small piece of polystyrene may be taped on at various axial positions. The determination of the size of this sample is left to the student.

One further question concerns the method of reactivity measurement. A variety of methods exist; two of these can be incorporated simultaneously. Establish criticality at some arbitrary low level, noting the critical position. This should be done first without any voids present. Now withdraw the calibrated control rod(s) to a given position, noting the stable period. It is necessary to insure that the reactivity associated with the rod motion will override the effect of the void as well as yielding a reasonable period. This assumes of course that the coefficient is negative. Return to a subcritical configuration and introduce the void in the lattice position of interest. Now return to the original power level by withdrawing rods to their original supercritical position. When this level has been reached note the new critical position. Reactivity measurements can be made by comparing the two periods or by comparing critical positions. This procedure can be repeated for a void measurement at any position in the core.

Several mechanical points should be reiterated. Fuel elements must be handled with the utmost of care at all times. By this time, safe operational techniques should be automatic. There is one last point regarding repeatability of the critical position. For various reasons the critical positions of successive runs may not be read as identical to five significant figures. Thus, while in theory it is necessary to run a reference condition (no voids) only once, you may want to spot check the critical position. A timely selection of verification of critical positions may alleviate problems in calculations.

Analysis of Results

Calculate the void coefficients for each run in cents per cubic centimeter or other convenient units. Plot the coefficient as a function of location, both axially and radially. Justify the shapes of the plots. Give the core average void coefficient; discuss its meaning.

Topics for Discussion

1. Refer to equation VII-5 and use perturbation theory to analyze the general results of the void coefficient.
2. What terms in the six-factor-formula dominate in the void coefficient of this reactor?
3. How would you expect the introduction of voids to affect control rod calibration curves?
4. Suggest possible reasons for any changes in reference critical positions you may have observed.
5. What is the similarity of this experiment to the temperature coefficient experiment with respect to moderation properties? Explain.
6. In what type of commercial reactor is the void coefficient a very crucial parameter? Why?

CHAPTER VIII

RELATIVE FLUX MAPPING AND POWER CALIBRATION

Part A: Relative Flux Mapping

Abstract

A knowledge of the flux distributions throughout the core of the reactor is basic to design of experiments and meaningful utilization of the Facility. Since neutrons of energies ranging from the maximum of the fission spectrum to thermal are present in the system, the flux distributions are functions of neutron energy as well as position. In order to interpret experimental results properly, it is desirable to map the flux as functions of both energy and position. Most important of course is the shape of the thermal flux, which is the primary objective of this segment of the experiment. These distributions are complex functions of geometry, materials composition, and other parameters which at present prohibit analytical analysis. Actual flux distributions are determined experimentally; analysis of these results leads to application of certain generalizations. Topics in this area include control rod effects, water gap effects, local core hot spots, local power levels, and the average thermal power of the core.

Introduction and General Theory

Assuming that the RPI Critical Facility core is symmetrical, only one quadrant of the core needs to be mapped to illustrate the flux distribution of the entire core. Any of the four quadrants can be chosen, but since the northwest quadrant is easy to work with and has been known to yield accurate results, it will be used in this experiment. Specifically element numbers 22,23,32,33,34,43,44 and control rod numbers 3 and 7 (lattice positions 24 and 42) will be mapped both axially (vertically) and radially (horizontally). Exact foil locations may vary somewhat, depending upon which features of the flux distribution the student would prefer to emphasize.

Symmetry indicates the following equivalences in lattice positions:

A. 22 = 26 = 62 = 66
B. 23 = 25 = 63 = 65
C. 32 = 52 = 56 = 36
D. 33 = 35 = 53 = 55
E. 34 = 54
F. 43 = 45
G. CR 3 = CR 6
H. CR 4 = CR 7
I. 44

Note that element numbers 34 and 43 are not equivalent due to the 90^o rotation of element numbers 43 and 45. This rotation also distinguishes CR 3 from CR 7 to a small extent.

Briefly the procedure consists of taping U^{235} fission foils to the fuel plates at the locations chosen by the student. Thin mylar tape is used for this purpose because of its neutron transparency. U^{235} foils are used because they approximate the fuel and perturb the system less than any potential substitute. The reader is urged to consider the reliability and accuracy of this technique compared to other flux measuring schemes. The foils are affixed to the more reactive side of a given plate, i.e. the one facing the center of the core. As illustrated in Figure (VIII-2), each element consists of nine plates, and obviously the most accurate technique would be to map each plate, both axially and radially. However time does not permit this luxury, so frequently plates two, five, and eight are selected as representative. Plates two and eight are mapped only axially, while plate five is mapped both axially and radially. Representative or suggested foil locations are illustrated in

Figure (VIII-3). The radial mapping takes place at the ten inch mark, and its distribution is assumed to apply to all plates mapped in that element. It should be noted here that foil weights differ by ± 3%. This difference should be corrected for but will be ignored in the calculation.

In order to activate the foils the reactor is operated at a known, recorded power level for thirty minutes. After a short cooling period the foils are removed and counted using the scintillation system in the counting room. With as many as 75 foils to count per run, it is obvious that some foils will have decayed in activity longer than others. This correction in decay must be applied. Knowing that U^{235} foils were irradiated, you might be inclined to pick the half-life off the chart of the nuclides. However, what is actually being counted is the radiation of the fission fragments; a rather complicated decay system to quantify analytically. Consequently Table (VIII-1) provides the decay factors for a thirty minute run, with the decay factors normalized to twenty minutes after shutdown. These numbers were generated experimentally, and apply only to a thirty minute irradiation. The times associated with each decay factor apply to the beginning of the counting interval.

A little arithmetic indicates that if the suggested foil locations were used, there would be 204 U^{235} foils to count. (This does not include the control rods which are too cumbersome to dismantle). There are too many foils to permit simultaneous activation, therefore several runs may be needed. Ideally each run should last for exactly thirty minutes at the same power level. To guarantee standardization in this regard, a gold foil is also irradiated in the same location for each run. Element number 55 plate five at the ten inch line is most frequently used. If activated at the same location and counted by the same equipment at the same time after shutdown, any deviations in exposure from one run

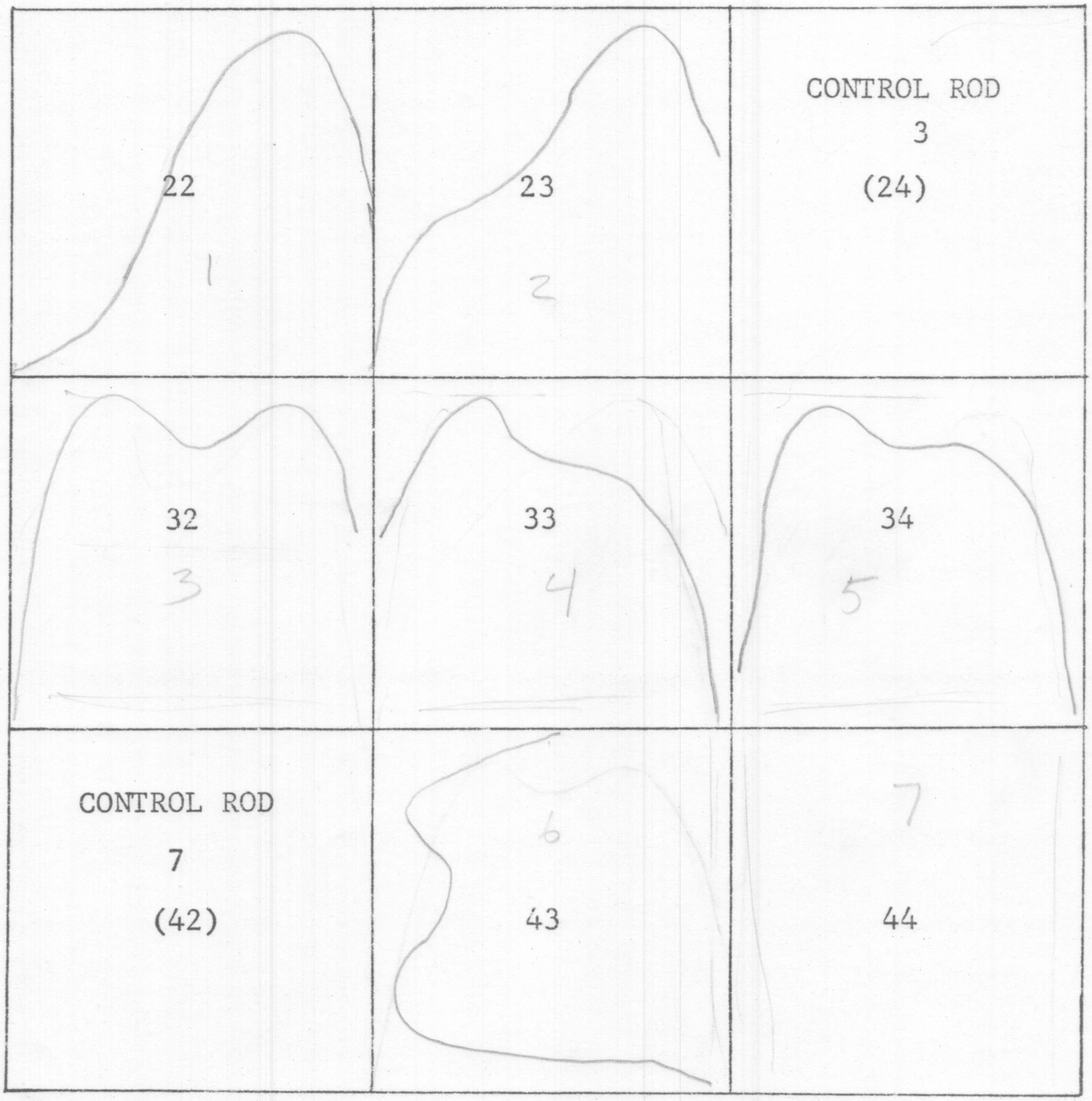


Figure VIII-1 Selected Quadrant

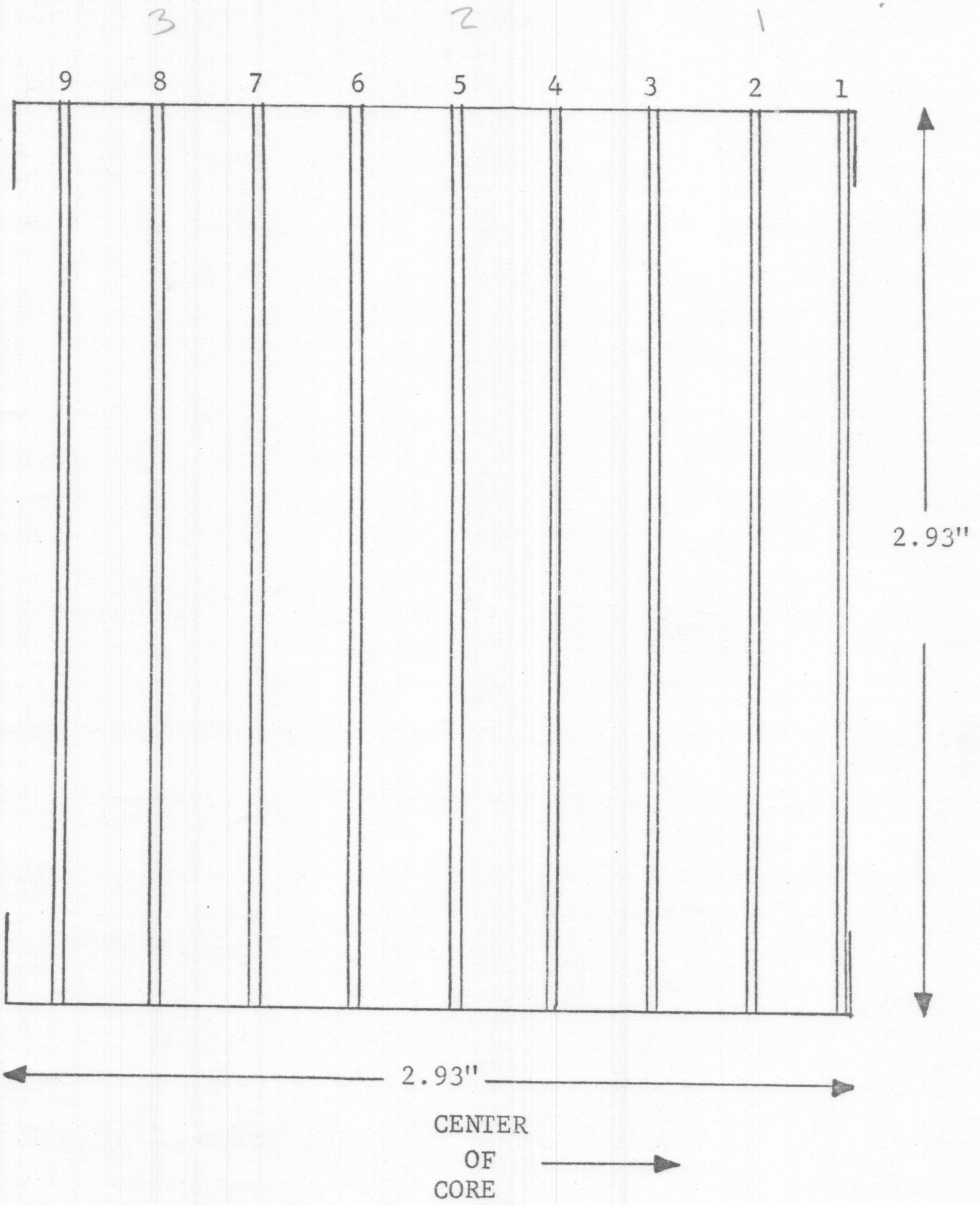


Figure VIII-2 Schematic Representation of Stationary Fuel Element

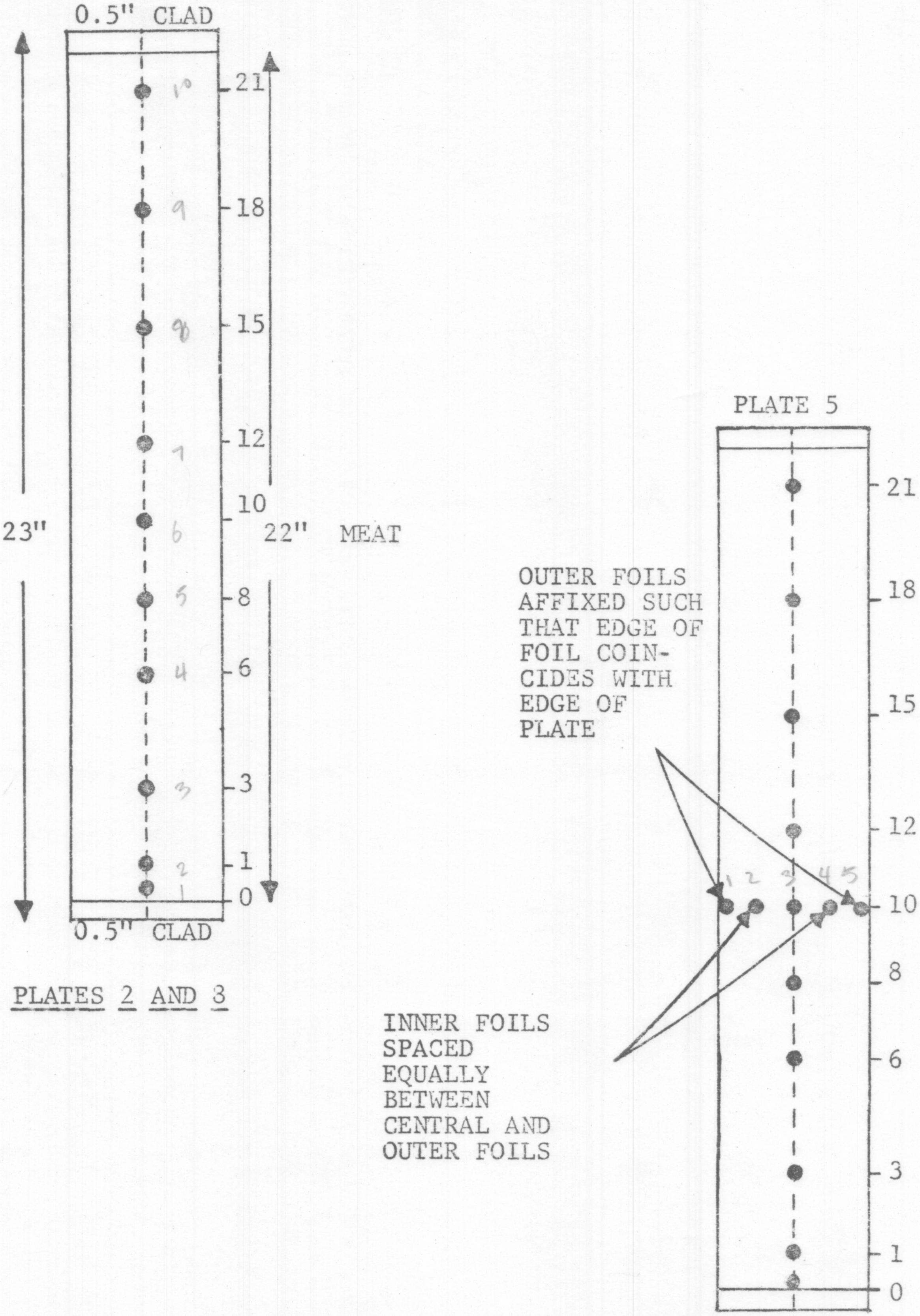


Figure VIII-3 Suggested U^{235} Foil Locations

TABLE VIII-1

FOIL NORMALIZATION FACTOR FOR DATA PROCESSING

(Uranium Foil Data)

Decay time listed is the time at beginning of count.

Decay Time (Minutes)	Decay Factor (Rel. to 20 Min. Decay Time)	Decay Time (Minutes)	Decay Factor (Rel. to 20 Min. Decay Time
10	0.678	41	1.771
11	0.709	42	1.812
12	0.742	43	1.854
13	0.775	44	1.903
14	0.805	45	1.948
15	0.836	46	1.990
16	0.867	47	2.039
17	0.902	48	2.080
18	0.935	49	2.131
19	0.967	50	2.180
20	1.000	51	2.226
21	1.035	52	2.270
22	1.069	53	2.319
23	1.102	54	2.365
24	1.137	55	2.417
25	1.174	56	2.457
26	1.208	57	2.508
27	1.245	58	2.554
28	1.282	59	2.606
29	1.318	60	2.651
30	1.357	61	2.701
31	1.392	62	2.746
32	1.426	63	2.797
33	1.466	64	2.846
34	1.500	65	2.890
35	1.539	66	2.940
36	1.574	67	2.998
37	1.612	68	3.041
38	1.648	69	3.095
39	1.689	70	3.139
40	1.732	71	3.188

TABLE VIII-1 (continued)

FOIL NORMALIZATION FACTOR FOR DATA PROCESSING

(Uranium Foil Data)

Decay time listed is the time at beginning of count.

Decay Time (Minutes)	Decay Factor (Rel. to 20 Min. Decay Time	Decay Time (Minutes)	Decay Factor (Rel. to 20 Min. Decay Time
72	3.246	104	4.947
73	3.301	105	5.006
74	3.349	106	5.089
75	3.403	107	5.134
76	3.446	108	5.195
77	3.502	109	5.250
78	3.548	110	5.304
79	3.602	111	5.375
80	3.661	112	5.412
81	3.712	113	5.488
82	3.760	114	5.545
83	3.816	115	5.605
84	3.874	116	5.665
85	3.901	117	5.745
86	3.970	118	5.833
87	4.028	119	5.883
88	4.062	120	5.926
89	4.124	121	6.003
90	4.170	122	6.074
91	4.238	123	6.143
92	4.268	124	6.213
93	4.318	125	6.278
94	4.379	126	6.354
95	4.438	127	6.431
96	4.473	128	6.501
97	4.537	129	6.577
98	4.614	130	6.650
99	4.677	131	6.724
100	4.719	132	6.801
101	4.775	133	6.878
102	4.836	134	6.953
103	4.889	135	7.027

to the next will be indicated by a variation in the activity of the gold standard. Now examination of the quadrant may take place in as many segments as necessary because any fluctuations in exposure can be corrected using this gold standard.

At this point the mechanics and general concepts of the experiment should be evident. The processing of the raw data remains to be explained. It has been previously stated that the radial mapping of the central plate in each element will be assumed to apply to <u>all</u> plates in that element. One's first impression of this distribution would be a central depression, which is indeed correct. The centerline of each plate, where the axial foils are affixed, is shielded by the U^{235} atoms on either side of it; thus, the centerline atoms are exposed to a lower flux. All of the axial foils will show a significantly lower activity than the actual plate average. A radial plot of the flux numerically integrated and averaged gives the appropriate correction factor for each element.

Having made this correction it is now possible to plot the axial activities versus position. Numerical integration again yields a plate average, and the element average is easily obtained by adding up the average plate activities and dividing by the number of plates mapped.

To summarize, the processing of the data from counting the foils is as follows:

1. Note and subtract the background of the scintillation counting system.
2. Correct all counting rates using the decay factors.
3. Normalize all runs to each other using the gold standard.
4. Correct for the flux depression using the radial distribution.

5. Numerically integrate using the axial distribution; this yields a plate average.
6. Obtain an element average.
7. Obtain the total core activity using the symmetry assumptions outlined previously.
8. Obtain the core average activity per effective fuel element.
9. Normalize all previous activities to this effective core average.

Thus far, enough information has been supplied to obtain element averages for the stationary fuel elements, namely numbers 22,23,32,33,34, and 43. Still lacking are the activities for control rods 3 and 7, and element number 44. These are calculated using ratios. For example, with respect to activities:

$\frac{43}{44} = \frac{33}{34}$; therefore an activity

for the center port can be generated. Remember that there are only two plates in element number 44 instead of nine as in the normal stationary element.

The control rod activities may be found in a similar manner using the appropriate ratios. Note that (1), there are only eight plates per fuel follower instead of nine, (2), the U^{235} loading for a fuel follower plate is only 25.07 grams instead of 28.62 grams for a stationary fuel plate, and (3), the critical bank position will be something less than fully withdrawn.

Now there is sufficient information to calculate the core average activity per <u>effective</u> fuel element. This quantity is quite important because each original data point may be normalized to this value. Using the activities, the known U^{235} loading, appropriate ratios, etc; one can determine the number of effective fuel elements, (something slightly less than the 25 lattice positions). By dividing this number into the total core activity generated by 20 stationary fuel elements, four control rods, and the center port, one obtains the core average per effective fuel element. The normalization process is quite straight forward. One arbitrarily defines the core average equal to unity, and simple division enables one to deal with much more convenient numbers. Now the data has been translated into a form which can be plotted and many core trends and boundary effects become evident.

Once the normalized value of the flux at any point in the core is known, it can be related to any other point and to the core average. If the absolute flux and power were known at any one, isolated point in the core, the absolute flux and power of all points and the core average would be known. This is the subject of part (B) of this exercise.

Procedure

Initially the experimenters must determine how much time is available to them, as this will ultimately dictate how detailed the mapping will be. On this basis the number of foils used and their exact locations are determined in any case portions of stationary element numbers 22,23,32,33,34, and 43 will be mapped. Remember to standardize the location of the gold foil. Use mylar tape to affix the U^{235} foils to the fuel plates. After verifying that the core has been reassembled completely and safely, irradiate the foils at the pre-determined power level for exactly thirty minutes. If it were not for the high radioactivity of the core after the thirty minutes, one could immediately obtain the first set of foils for counting. However a short cooling period is necessary; the exact length of this waiting time can be determined by the student.

Previously the student should have familiarized himself with the scintillation counting system, and have determined a definite repetitive procedure for counting each foil for one minute. It is advisable to retain an organization to the foils even after counting. In the event of an obvious error one may always repeat the counting rate.

Finally, a reminder on safety: the radiation levels encountered in this experiment will probably exceed those of any previous experiment. One should move quickly but carefully in handling radioactive foils from fuel elements. Caution should be taken to guard against any fuel element of fuel plate damage. Care must be taken so that loose objects or foils are not dropped into the reactor vessel. At all times monitor the reactor room, making use of the intercom. Monitor the fuel elements and foils closely for activity keeping exposure and handling to a minimum.

Part B: Reactor Power Calibration

Abstract

The object of this portion of the experiment is to obtain the absolute thermal flux at a particular point in the core, and relate this absolute thermal flux and power to the core averaged values found in Part (A). In a thermal reactor the steady state power output is directly proportional to the thermal flux only, and therefore the experimental technique employed must distinguish neutron energies.

Introduction and General Theory

Recall that a thermal neutron causing a fission event will produce 3.2×10^{-11} joules of energy. The thermal flux multiplied by the thermal neutron fission cross section represents the number of fission events taking place per unit volume. Therefore, the total power output of the reactor is:

$$p = \Sigma_f V \varphi (3.2 \times 10^{-11}) \text{ watts} \qquad \text{(VIII-1)}$$

where:

p = power

Σ_f = macroscopic fission cross section of U^{235}

V = active core volume

φ = average thermal flux

and $$\Sigma_f = N^{235}\sigma_f = \frac{MA_V}{A^{235}}\sigma_f \qquad \text{(VIII-2)}$$

where:

N^{235} = number of U^{235} atoms

σ_f = microscopic fission cross section of U^{235}

M = mass of U^{235} in core

A_V = Avogadro's number

A^{235} = atomic weight of U^{235}

Basically the problem is to determine the average thermal neutron flux. The solution is to activate a stable isotope of an element with thermal neutrons and then measure the activity of that element. If there is to be a measurable relationship between the flux and the activity of the element, the element must have a relatively strong neutron absorption cross section. Secondly it must decay with a half-life which is reasonable to yield acceptable counting statistics. An isotope meeting these requirements is Au^{197}. When activated by neutrons the following reaction occurs:

$$n_o^1 + Au^{197} \longrightarrow Au^{198} \xrightarrow{\beta-} Hg^{198} \qquad \text{(VIII-3)}$$

It is necessary to quantify a relation between the activity of the gold and the unknown flux. The familiar differential equation is:

$$\underset{\text{time rate of change}}{\frac{dN^{198}}{dt}} = \underset{\text{production}}{\varphi N^{197}\sigma_a} - \underset{\text{decay}}{\lambda N^{198}} \qquad \text{(VIII-4)}$$

initial condition $N^{198}(0) = 0$

at equilibrium $\frac{dN^{198}}{dt} = 0$, thus

$$\varphi N^{197}\sigma_a = \lambda N_\infty^{198} \qquad \text{(VIII-5)}$$

or

$$N_\infty^{198} = \frac{\varphi N^{197}\sigma_a}{\lambda} \qquad \text{(VIII-6)}$$

or

$$\varphi = \frac{N_\infty^{198}\lambda}{N^{197}\sigma_a} \qquad \text{(VIII-7)}$$

The differential equation for N^{198} is a first order differential equation which can be solved for the number of Au^{198} atoms created:

$$N^{198} = \frac{\varphi\sigma_a N^{197}}{\lambda}(1 - e^{-\lambda t_i}) \qquad \text{(VIII-8)}$$

Where t_i = irradiation time

Simplifying:

$$N^{198} = N_\infty^{198}(1 - e^{-\lambda t_i}) \qquad \text{(VIII-9)}$$

To account for the delay time from the end of the irradiation period until the foils are counted:

$$N^{198} = N_\infty^{198}(1 - e^{-\lambda t_i})e^{-\lambda t_w} \qquad \text{(VIII-10)}$$

Where t_w = waiting time before counting

The activity of Au^{198} is:

$$Au^{198} = N^{198}\lambda \qquad \text{(VIII-11)}$$

and

$$N^{198}\lambda = C^{198} = N_{\infty}^{198}\lambda(1 - e^{-\lambda t_i})(e^{-\lambda t_w}) \qquad \text{(VIII-12)}$$

Where C^{198} = measured activity of Au^{198} atoms

$$C^{198} = I_{\infty}(1 - e^{-\lambda t_i})(e^{-\lambda t_w}) \qquad \text{(VIII-13)}$$

Where I_{∞} = equilibrium activity of Au^{198} atoms

Actually this equation as written applies only to the case of 100% counter efficiency. The measured activity must incorporate the counter efficiency ε.

At equilibrium, i.e. when Au^{198} has reached saturated activity due to irradiation of Au^{197}, the flux is:

$$\varphi = \frac{N_{\infty}^{198}\lambda}{\varepsilon N^{197}\sigma_a} = \frac{I_{\infty}}{\varepsilon N^{197}\sigma_a} \qquad \text{(VIII-14)}$$

Solving equation (VIII-13) for equilibrium activity:

$$I_{\infty} = \frac{C^{198}e^{\lambda t_w}}{(1-e^{-\lambda t_i})} \qquad \text{(VIII-15)}$$

Substituting this expression for I_{∞} into equation (VIII-14):

$$\varphi = \frac{C^{198}e^{-\lambda t_w}}{(1-e^{-\lambda t_i})\varepsilon N^{197}\sigma_a} \qquad \text{(VIII-16)}$$

Knowing that:

$$N^{197} = \frac{M_{Au}A_V}{A^{197}} \qquad \text{(VIII-17)}$$

The equation for the flux is:

$$\varphi = \frac{C^{198}e^{-\lambda t_w}A^{197}}{(1-e^{-\lambda t_i})\varepsilon M_{Au}A_V\sigma_a} \qquad \text{(VIII-18)}$$

It is possible to obtain the flux from the activity of the irradiated gold foils, however the thermal flux is needed. It is known that cadmium has a very strong absorption cross section for thermal neutrons, but at neutron energies above about 1 eV the cross section is more than three orders of magnitude weaker. Therefore, a cadmium covered gold foil would be activated exclusively by fast neutrons. A bare gold foil would have an activation from both fast and thermal neutrons, so that subtracting the activity of the cadmium covered foil from the activity of the bare foil will give a reasonably accurate estimation of the thermal neutron activation:

$$\varphi_{\text{thermal}} = \frac{(C^{198}_{\text{bare}} - C^{198}_{\text{Cd}})e^{-\lambda t_w} A^{197}}{(1-e^{-\lambda t_i})\epsilon M_{\text{Au}} A_V \sigma_a} \qquad \text{(VIII-19)}$$

Recalling the expression for power, equation (VIII-1):

$$p = \frac{M\, A_V \sigma_f}{A^{235}} \; \frac{(C^{198}_{\text{bare}} - C^{198}_{\text{Cd}})e^{-\lambda t_w} A^{197}}{(1-e^{-\lambda t_i})\epsilon M_{\text{Au}} A_V \sigma_a} (3.2 \times 10^{-11}) \text{ watts}$$

(VIII-20)

Each of the quantities in equation (VIII-20) is either a constant or a measurable quantity. Thus it is possible to determine the exact power output of the core at any selected location. Knowing the relative flux distribution, the power output of the reactor can be determined by utilizing the average thermal flux. For simplicity Table (VIII-2) lists all symbols and abbreviations used in Part (B) of this experiment.

Procedure

It is necessary to find the absolute thermal flux at only one location in the core, but for more accurate coverage three locations will be used. From symmetry considerations it is known that element numbers 33 and 55 are

TABLE VIII-2

LISTING OF SYMBOLS AND ABREVIATIONS

p - Reactor Power Output

φ - Average Thermal Neutron Flux

Σ_f - Macroscopic Fission Cross Section for U^{235}

V - Active Core Volume

σ_f - Microscopic Fission Cross Section for U^{235}

M - Mass of U^{235} in Active Core

A_V - Avogadro's Number

A^{235} - Atomic Mass of U^{235}

N^{197} - Number of Au^{197} Atoms/CM^3

N^{198} - Number of Au^{198} Atoms/CM^3

σ_a - Thermal Absorption Cross Section of Au^{197}

λ - Decay Constant of Au^{198}

t_i - Irradiation Time

t_w - Waiting Time Before Counting

C^{198} - Measured Activity of Au^{198} Atoms

C^{198}_{bare} - Measured Activity of Bare Au^{198} Foil

C^{198}_{Cd} - Measured Activity of Cd covered Au^{198} Foil

I_∞ - Equilibrium Activity of Au^{198} Atoms

ϵ - Scintillation Counter Efficiency

M_{Au} - Mass of Gold Foil

A^{197} - Atomic Mass of Au^{197}

N^{235} - Number of U^{235} Atoms/CM^3

equivalent. This consideration is crucial because in order to obtain the absolute thermal flux at any one point, both bare and cadmium covered foils must be irradiated at the "same" location. Utilizing two symmetrical elements enables the student to obtain the data in one run of the reactor. Suggested foil locations are indicated in Figure (VIII-4). Note that the foils are taped with mylar to the side of plate five which faces the center of the reactor. Note also that unlike Part (A) of the experiment, foil weight is very important. Consequently, all foil weights should be determined prior to irradiation.

Activate the foils at a fixed power level for a known period of time (about thirty minutes). After a short cooling period, remove the foils and measure the activities with appropriate corrections for waiting times and other variables. Please remember to remove the Cd cover from the gold foils before counting since there is no interest in measuring activity induced into the cadmium. Relate these counting rates to the thermal flux and determine the power level at which you irradiated the gold foils.

Analysis of Results

The data gathered from Parts (A) and (B) is quite extensive and lends itself to a relatively detailed analysis. Efforts may be concentrated in certain directions if there is a particularly strong interest in one analytical aspect. Hopefully U^{235} foil locations were selected on this basis. In any case there are several basic topics to be considered.

Plot the radial and axial flux distributions for some representative plates, explaining why the curves take their characteristic shape. Make reasonable estimates where data may be insufficient. Now plot the relative flux across the core and reflector regions. Show the details of your calculation for reactor power output, and indicate how accurate

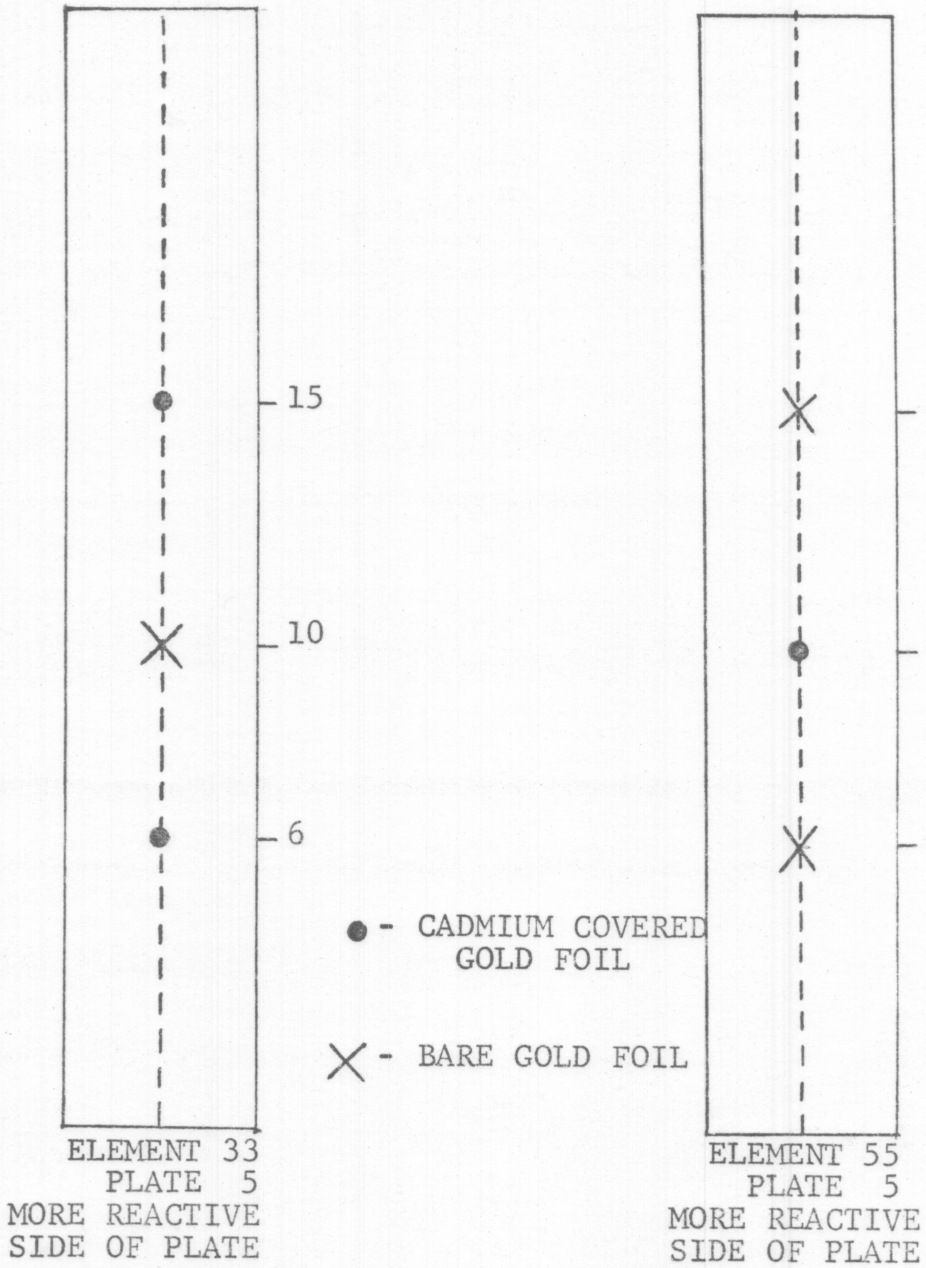


Figure VIII-4 Suggested Gold Foil Locations for Power Calibration.

you believe your number is. Indicate the primary sources of error. Determine the hot spot of the locations you measured. Show the calculation you use to determine the number of effective fuel elements.

Topics for Discussion

1. Discuss the points mentioned in the abstract: Flux shapes, control rod effects, water gap effects, core hot spots, local power levels, and average thermal core power.
2. You were told to assume that the radial correction measured for the central plate applied to all plates of that element. How valid was this assumption?
3. Hypothesize what trends you would measure if all plates of one element were mapped. Why were plates two, five, and eight suggested?
4. Discuss the relation between flux and power in the center port.
5. What was the magnitude of the flux perturbation introduced by the U^{235} and gold foils?
6. Why is there considerable interest in hot spot data, effects of heat generation and heat removal?
7. Considering the 90° rotation, would element number 43 be more or less reactive than element number 34?
8. What effect does the Cd-cover thickness used have on the power determination?
9. What other foils might be used to replace U^{235} and gold?
10. Do you think there is linearity between thermal flux and indicated reactor power?
11. What other assumptions have been made in performing this experiment? Are there any that you feel cannot be justified?

CHPATER IX

ABSORPTION CROSS-SECTION MEASUREMENTS

Abstract

The purpose of this experiment is to measure the reactivity effects associated with the insertion of various materials into the core and to determine the reactivity coefficient of each. Further, knowing these reactivity effects and knowing the boron and void effects, one is able to determine the absorption cross section of these materials. Many of the remarks prefacing the boron coefficient experiment are applicable to this exercise; thus the reader is referred to the boron coefficient section for more specific remarks.

Many factors affect the results of this experiment. For each material, every atom should be exposed to the same neutron flux. No self-shielding or flux depression should be involved. Materials whose cross sections exhibit low energy level absorption peaks in the resonance region cannot be used unless a correction factor can be developed for the material in question. The effects of scattering amd moderating on the multiplication of the system must be assumed negligible. Any change in temperature or other parameters which affect the reactivity is highly undesirable. Your material selection and experimental technique should comply with each of these guidelines as much as possible.

Introduction and General Theory

Reactivity measurements of various materials will be made by observing: the difference between the control rod position required to achieve criticality under a reference condition; and the position required under the condition being investigated. Applying the usual guidelines for materials handling and experimental technique, critical rod positions can be related to reactivity worths and ultimately absorption cross sections.

Any small change or perturbation of the given reactor system, whether it is a reactivity insertion or deletion, will be represented by a small but measurable change of the critical position of the control rods. Furthermore, if this perturbation happens to be due to insertion of material samples, the reactivity effect can be determined. If the size of the material is comparable to the size of the materials used in determining the boron and void coefficients of reactivity, one is able to calculate the thermal cross sections of these materials as follows:

$$\left(\begin{matrix}\text{Net Reactivity}\\ \text{of Sample}\end{matrix}\right) = \left(\begin{matrix}\text{Reactivity}\\ \text{Measured}\end{matrix}\right) - \left(\begin{matrix}\text{Void}\\ \text{Coefficient}\end{matrix}\right)\left(\begin{matrix}\text{Sample}\\ \text{Volume}\end{matrix}\right)$$

$$\rho_{net} = \rho_{measured} - \rho_{void} \qquad \text{(IX-1)}$$

$$\frac{\rho_{net}}{\rho_B} = \frac{N_S \sigma_S}{N_B \sigma_B} \qquad \text{(IX-2)}$$

where:

ρ_{net} = net reactivity of sample

ρ_B = equivalent reactivity of boron^{10}

σ_S = unknown sample cross section

σ_B = cross section of boron^{10}

N_S = atoms of sample

N_B = atoms of boron

To determine N_S and N_B, make use of the relations:

$$N_S = \frac{M_S A_V}{A_S} \qquad \text{and} \qquad N_B = \frac{M_B A_V}{A_B} \qquad \text{(IX-3)}$$

where:

A_V = Avogadro's number

M_S = Mass of sample

M_B = Mass of boron

A_S = Atomic weight of sample

A_B = Atomic weight of boron

Solving the resulting formula for σ_{sample} :

$$\sigma_S = \frac{(\rho_{net})(\sigma_B)(A_S)(M_B)}{(\rho_B)\qquad(A_B)(M_S)}$$

At the RPI Critical Facility the optimum location for cross section determinations is quite obviously the center port. Conveniently there exists a strong, thermalized neutron column in lattice position number 44. Using the data from previous experiments the student may determine the optimum height for the samples.

Finally the student must select the materials to investigate. It is suggested that one material which conveniently lends itself to straightforward analysis be chosen. Any elements which are of interest to the student may be selected. If experimental results severly deviate from published values it is usually possible to indicate the major cause(s).

Procedure

The operational details are very similar to previous exercises, but they will be briefly reviewed here. Position the three-rod bank at an appropriate height and go critical on the calibrated rod to establish the reference condition. Be sure this includes the center stringer. Shutdown and insert the sample; be sure you observe proper safety precautions. Bring the reactor critical at the same power level and note the difference in critical positions. In some cases this difference may be extremely small and operator skills must be refined. Repeat this procedure for each sample, and confirm the reference position when appropriate. Be sure you have obtained all necessary constants such as masses, volumes, known cross sections, etc.

Analysis of Results

Calculate the cross sections of your samples, providing the details of your calculation for at least one material.

Topics for Discussion

1. Which factor(s) in the six-factor-formula are changed when an absorber is placed in the core?
2. In addition to absorption, what other properties of the sample will change the reactivity? How can these effects be minimized?
3. Are there other materials which may serve as a substitute for boron as the standard?
4. Do you think any absorption of epithermal or fast neutrons took place?
5. Provide the details for calculating the correction which would be applied if self-shielding effects perturbed your experiment.
6. Discuss other methods of determining cross sections.