THE DESIGN, CONSTRUCTION, AND TESTING OF THE UNIVERSITY OF ARIZONA SUBCRITICAL REACTOR

by

Charles E. Backus

A Thesis Submitted to the Faculty of the

DEPARTMENT OF NUCLEAR ENGINEERING

In Partial Fulfillment of the Requirements
For the Degree of

MASTER OF SCIENCE

In the Graduate College

THE UNIVERSITY OF ARIZONA

1 9 6 1

STATEMENT BY AUTHOR

This thesis has been submitted in partial fulfillment of requirements for an advanced degree at The University of Arizona and is deposited in the University Library to be made available to borrowers under the rules of the Library.

Brief quotations from this thesis are allowable without special permission, provided that accurate acknowledgement of source is made. Requests for permission for extended quotation from or reproduction of this manuscript in whole or in part may be granted by the head of the major department or the Dean of the Graduate College when in their judgment the proposed use of the material is in the interests of scholarship. In all other instances, however, permission must be obtained from the author.

SIGNED: Charles E. Backus

APPROVAL BY THESIS DIRECTOR

This thesis has been approved on the date shown below:

Ray L. Chapman — 5/11/61

Ray L. Chapman — Date

Associate Professor of Nuclear Engineering

ABSTRACT

A natural uranium-light water, subcritical reactor was designed and built for the Nuclear Engineering Department, University of Arizona. The reactor was designed as a teaching tool and thus easily facilitates experimentation and instruction. The entire assembly is safe, portable, and highly versatile. The subcritical reactor utilizes 5500 pounds of natural uranium in the form of Savannah River fuel slugs. The core lattice is 42 inches high and is about 30 inches in diameter. The lattice is completely immersed under water in a stainless steel tank that is 50 inches high and 60 inches in diameter. The water to uranium volume ratio in the core is about 1.5. Preliminary investigations indicate the maximum neutron flux using five, one curie PuBe sources, is about 2×10^4 neutrons/cm^2-sec.

ACKNOWLEDGEMENTS

The author would like to acknowledge the Department of Nuclear Engineering, whose financial support made this project possible.

The author would also like to express deep gratitude to his thesis advisor, Ray L. Chapman, who offered many valuable criticisms and suggestions.

CONTENTS

CHAPTER 1

INTRODUCTION

The purpose of this thesis is to provide information on all aspects of the design and construction of the University of Arizona Subcritical Reactor. The reactor was built to supplement the facilities for instruction in the Department of Nuclear Engineering.

A subcritical reactor, or exponential pile, is an excellent laboratory tool.[1*] It is relatively inexpensive in both initial and operating costs. It can never become critical and thus is inherently safe for student use. It is usually simple in construction which allows students to work more directly with the nuclear properties being investigated. Even with these many simplicities almost all measurements made on a power reactor can be made equally well on a subcritical assembly.

Exponential piles are used quite extensively in practice. When building large power reactors, it is extremely expensive and time consuming to obtain experimental verification of critical size estimates of a complex heterogeneous reactor. These experiments are customarily done on a subcritical assembly of the actual system. The

---

* See numbered references in the bibliography.

assembly is a crude mock-up which serves primarily to simulate the essential nuclear features and geometry of the actual reactor.[2]

In recent years, there have been two general types of educational subcritical assemblies built. One is a natural uranium-graphite pile and the other is a natural uranium-light water assembly. (Natural uranium is usually used because the AEC loans this type of fuel to educational institutions.) Of the two types of systems, the uranium-water assembly has many advantages. It is cheaper, more flexible, usually smaller, impossible to make critical, and more accessible for flux measurements. One can also measure the temperature coefficient. Its main disadvantage is that it may introduce errors in exact positioning of foils because of the small diffusion length of neutrons in water. On these arguments, it was decided that the Arizona assembly was to be a light water-natural uranium system.

There are a number of factors to be taken into account in the basic design of a university assembly. The foremost factor is that the assembly is to be used as a teaching tool. As an educational device, the system must be safe, versatile, and as simple as possible. It requires that the internal assembly be easily accessible to facilitate experimentation and instruction.

Along with the teachability requirement, the system should optimize the nuclear properties involved. This essentially means maximizing the material buckling, B_m^2, of the system. The buckling depends, among other things, on the water to uranium volume ratio. Therefore, there is an optimum ratio. Also, the core should be surrounded by a reflector to conserve neutrons.

The first thing done toward the construction of the subcritical was to conduct a literature search. This revealed the types of assemblies in existance and the components on the industrial market necessary for construction. Chapter two discusses the nuclear design considerations. It includes an elementary mathematical analysis of the reactor. The computations give an order of magnitude of the expected neutron flux. From this rough approximation of the flux, the physical design, such as the shielding requirements, can be determined. Chapter three describes the physical components of the system. Chapter four gives the results of experiments with the completed subcritical assembly. This chapter also includes a correlation of calculations with experimental results and a discussion of the subcritical reactor in general.

CHAPTER 2

COMPUTATIONS AND ESTIMATIONS

2.1 Background

One of the first systems considered for nuclear reactors was that of natural uranium in light water. However, research along these lines was not carried very far. Early calculations led to the belief that the relatively large thermal neutron absorption cross section of hydrogen would not permit its use in obtaining a chain-reacting system with natural uranium.

By early 1944, the experimental values of cross sections had changed enough to cause the use of water to be reconsidered. A re-examination of the problem by Weinberg and others showed that water-metal lattices might have much higher reproduction factors than was first thought. At least, the uncertainties in the analysis were large enough to allow the possibility that k could be greater than one. An experimental program to settle the question was therefore carried out in 1944-45, at Oak Ridge.[3]

The Oak Ridge measurements were made in exponential experiments, with natural uranium rods of diameter 1.18 inches, 1.10 inches, and 0.787 inches. Several volume ratios of water to uranium were assembled with each rod size. The experiments were primarily meant to provide values of the material buckling, B_m^2.

The preliminary report on the problem, October 3, 1944, states: "The problem of building a water uranium lattice with a k greater than one still remains. For 25°C and no gap, the best k obtained with the materials used (1.18 inch rods), was for a volume ratio of between 1.4 and 1.5."[4] The results of these experiments appear in the form of a graph in Figure 2.1.

This graph is of interest since the AEC loaned Savannah River fuel slugs to the University of Arizona for its subcritical. The diameter of these slugs is about 1.20 inches. (For other specifications of the Savannah River slugs, see Figure 2.3). According to Figure 2.1, the maximum value of B^2 obtainable with 1.18 inch rods is about -3×10^{-4} at a water-to-uranium ratio of about 1.5. Of course, the Oak Ridge experiments used solid rods and the Savannah River slugs are hollow, so their results can only be used for a first approximation.

On the basis of the previous information, the Arizona subcritical was designed to have a water to fuel ratio of between 1.4 and 1.5. This was done with the knowledge that this ratio is probably too low for an optimum system using hollow slugs. However, literature was not found available on Savannah River slugs in light water.

Again referring to Figure 2.1, it appears that there will not be a great difference in systems with ratios between about one and two.

After the water to fuel ratio was decided upon, the specifications for the grid plates had to be determined. A triangular spacing is preferred since each fuel element is the same distance from each adjacent fuel element. The spacing dimensions must be standard distances so the industrial firm can make them economically. The distance between centers of the fuel rods in a row is 1 9/16 inches. The distance between rows is 1 3/8 inches. The fuel spacing is shown in Figure 2.2.

All the calculations that follow in this chapter are based on the unit cell shown in Figure 2.2. According to notation used in Reactor Theory, this is a triangular spacing with a hexagonal unit cell. The specifications for the cell are given in Table I.

TABLE I

THE UNIT CELL

Material	Area (in^2)	% of Volume	V_{H_2O}/V_U	V_{AL}/V_U
Water	1.192	55.5		
Uranium	0.820	38.1	1.46	
Aluminum	0.138	6.4		0.116
Total Cell	2.150	100.0		

Of course, the numbers in Table I take into consideration the aluminum cladding around the uranium and the aluminum tubes through the center of the slugs. All end effects have been neglected.

2.2 Assumptions and Limitations

The United States Atomic Energy Commission has an educational assistance program to help academic institutions develop their facilities for instruction. The AEC loans all types of nuclear materials including natural uranium. They will loan up to 5500 pounds of uranium metal in the form of aluminum cladded slugs. These slugs have been fabricated for either the Hanford Reactor or the Savannah River Reactor, but have been rejected for not meeting certain ridged specifications.

In order for an educational institution to obtain this fuel, they must submit two proposals. A request for the material must be granted and also a license for the fuel. No license is required for a subcritical assembly as such. However, material licenses are required for the natural uranium slugs and the plutonium-beryllium neutron sources used in such subcritical assemblies. Natural uranium is a source material and its receipt, possession and use is licensed in accordance with Title 10, Code of Federal Regulations, Part 40.

The applications for materials and license must include rather detailed information as to the utilization of the uranium. The AEC requires such information as: type of material desired; description of the assembly including the type of moderator, reflector, and shield; and estimates as to the maximum neutron and gamma fluxes or dose rates at the boundaries of the assembly.

The remainder of this chapter gives the basis for the estimates included in the Arizona proposal for Savannah River fuel slugs. Since an order of magnitude is all that is required in the estimations, the calculations are very crude. However, very conservative estimates are made from these calculations to find the maximum dose rates to expect. The calculations are not intended to be rigorous and should not be interpreted as a mathematical analysis of the reactor.

Since the purpose here is only to get an estimation of the fluxes, an elementary mathematical model can be assumed to approximate the system. The calculations are greatly simplified by the assumptions and limitations of the following model:

1. The reactor core is treated as a homogeneous mixture that has the same volume ratios as the unit cell.

2. The core is considered to be a rectangular parallelepiped that has the same height and same cross sectional area as the hexagon.

3. The reflector savings due to the water are added to the core dimensions to find an equivalent bare reactor.

4. The Fermi Age is used for the fast leakage factor.

5. The Reactor Handbook was used to obtain values for as many parameters as possible.[5]

The first assumption is by far the most limiting. This assumption will make the resonance escape probability appear lower than reality. Also the peaking of the flux between fuel elements is neglected.

In the handbook, under their description of natural uranium lattices in water, they give the constants of a system very similar to ours. A description of this system is in Table II.

TABLE II

DESCRIPTION OF NATURAL URANIUM LATTICE FROM REACTOR HANDBOOK

Lattice number	11
Description	Slugs in tubes, no gap
Cell type	Hexagon
V_{H_2O}/V_u	1.42
V_{al}/V_u	0.140
Diameter of rods	1.1 inches
L^2 at 25°C	2.6 cm^2
τ at 25°C	51.0 cm^2
k_∞	0.990

Since the system described in Table II is very similar to ours, the values for the thermal diffusion length L^2, the Fermi age τ, and the infinite multiplication factor k_∞, will be assumed in our calculations.

2.3 Neutron Flux Distribution[6]

From conventional reactor theory, the general diffusion equation is

$$\frac{1}{v}\frac{\partial \phi(\bar{r},t)}{\partial t} = D\nabla^2 \phi(\bar{r},t) - \Sigma_a \phi(\bar{r},t) + \int_{vol.}\left[\frac{k}{p}\Sigma_a \phi(\bar{r}_o) + S(\bar{r}_o)\right] P(\bar{r},\bar{r}_o E_{th})\, d\bar{r}_o$$

where $\phi(\bar{r},t)$ = Neutron flux as a function of space vector $\bar{r}$ and time t (neutrons per cm^2 per sec)

D = Diffusion coefficient

Σ_a = Macroscopic absorption cross section

k = Infinite multiplication factor

p = Resonance escape probability

$S(\bar{r}_o)$ = Extraneous source of fast neutrons at $\bar{r}_o$ (assumed the same energy as fission neutrons)

$P(\bar{r},\bar{r}_o E_{th})$ = Slowing down density at thermal energy at $\bar{r}$ due to a point source at $\bar{r}_o$

v = Velocity of the neutrons

The solution of the above differential equation can be obtained for the simple model described in section 2.2. The expression for the steady state flux is

$$\phi(\bar{r}) = \sum_n \frac{S_n Z_n(\bar{r}) p e^{-B_n^2 \tau}}{\Sigma_a (1+L^2 B_n^2)(1-k_{n\ eff})}$$

$$k_{n\ eff} = \frac{k\ e^{-B_n^2 \tau}}{(1+L^2 B_n^2)}$$

The B_n^2 and Z_n come from the condition that

$$\nabla^2 Z(\bar{r}) + B^2\ Z\ (\bar{r}) = 0$$

This has assumed that the space part of the flux is equivalent to the space form of the wave equation. The condition that $Z = 0$ on the boundary results in a set of eigenvalues B_n^2. The corresponding eigenfunctions, of which Z_n is the nth member, form a complete orthogonal set of functions. The S_n is the coefficient of a Fourier series representing the extraneous source.

$$S(\bar{r}) = \sum_n S_n\ Z_n\ (\bar{r})$$

The Macroscopic Cross Sections

The macroscopic absorption cross section Σ_a, is defined by the expression,

$$\Sigma_a = N\ \sigma_a = N^u\ \sigma_a^{\ u} + N^{al}\ \sigma_a^{\ al} + N^w\ \sigma_a^{\ w}$$

where N^u = Nuclear density of the uranium

N^{al} = Nuclear density of the aluminum

N^w = Nuclear density of the water

σ_a = Microscopic absorption cross section for respective components

The nuclear densities can be found from the volume ratios in Table I.

$$N = \frac{\text{normal density x Avogadro's no. x fraction of volume}}{\text{atomic weight}}$$

For example,

$$N^u = \frac{18.68 \times 6.025 \times 10^{23} \times 0.381}{238.07} = 1.80 \times 10^{22} \frac{\text{Nuclei}}{\text{cm}^3}$$

Putting the thermal cross sections and densities into the above expression for Σ_a,

$$\begin{aligned}\Sigma_a &= (1.80 \times 10^{22})(7.42 \times 10^{-24}) \\ &+ (3.51 \times 10^{21})(0.215 \times 10^{-24}) \\ &+ (1.86 \times 10^{22})(0.657 \times 10^{-24}) = 0.147 \text{ cm}^{-1}\end{aligned}$$

The macroscopic scattering cross section is defined similar to Σ_a, except that the microscopic scattering cross section, σ_s, is used in place of σ_a.

$$\Sigma_s = N^u \sigma_s^u + N^{al} \sigma_s^{al} + N^w \sigma_s^w$$

$$\Sigma_s = 0.148 + 0.911 + 0.00492 = 1.064 \text{ cm}^{-1}$$

The total cross section is simply the sum of the absorption and scattering.

$$\Sigma_t = \Sigma_a + \Sigma_s = 1.211 \text{ cm}^{-1}$$

The Resonance Escape Probability

The resonance escape probability is the fraction of fast neutrons that are not absorbed by the fuel while slowing down to thermal energy. This probability, as derived from slowing down theory, is represented approximately by

$$P(E) = \exp\left[-\int_{E}^{E_o} \frac{\Sigma_a(E)}{\bar{\xi}\,(\Sigma_s(E) + \Sigma_a(E))} \frac{dE'}{E'}\right]$$

Where E_o is the neutron fission energy, Σ_s (E) and Σ_a (E) are the scattering and absorption cross sections, respectively, the $\bar{\xi}$ is the average of the logarithmic energy decrement per collision.

$$\bar{\xi} = \frac{\sum_{i=1}^{N} \Sigma_s^i \xi_i}{\Sigma_s} = \frac{\Sigma_s^u \xi_u + \Sigma_s^{al} \xi_{al} + \Sigma_s^w \xi_w}{\Sigma_s}$$

$$\bar{\xi} = \frac{(0.148)(0.00837) + (0.911)(0.93) + (0.005)(0.073)}{1.064}$$

$$\bar{\xi} = 0.798$$

For mixtures of natural uranium with several different moderators it has been found that the resonance escape probability can be expressed empirically to a good approximation by[7]:

$$P(E) = \exp\left[-\frac{3.9}{\bar{\xi}}\left(\frac{N_u \times 10^{-24}}{\Sigma_s}\right)^{0.585}\right]$$

$$P(E) = \exp\left[-\frac{3.9}{.798}\left(\frac{.018}{1.064}\right)^{.585}\right]$$

$$P(E) = 0.640$$

Dimensions of Equivalent Bare Reactor

The actual core of the subcritical is in a hexagonal shape. The height is 42.5 inches and the side of the hexagon is 14.7 inches. (For all calculations it is assumed that the slugs are five high.) An equivalent parallelepiped of the same height will be 23.7 inches on each side.

The reflector savings must be added to these dimensions to find an equivalent bare reactor. It is assumed that the core is completely surrounded by a thick reflector of water. The savings for a thick reflector are[2]

$$\text{Savings} \simeq \frac{D_c}{D_r} \quad L_r$$

The subscripts, r and c, refer to the reflector and the core respectfully. The diffusion coefficient for the core is found from the expression,

$$D_c = L_c^2 \, \Sigma_a^c = 2.6(0.147) = 0.382 \text{ cm}$$

$$\text{Savings} \simeq \left(\frac{0.382}{0.18}\right) 2.88 = 6.11$$

Therefore, the dimensions of an equivalent bare parallelepiped are:

h = height = 120.2 cm

a = side = 72.4 cm

b = side = 72.4 cm

The Eigenfunctions, Z_n

Assume that there is a point source of fast neutrons at the center of the parallelepiped. The system is symmetrical with respect to the Cartesian coordinate axes, x, y, and z, which are parallel to the edges of the parallelepiped, and selected with their origin at the center. The eigenfunctions are therefore even functions, or

$$Z_n = \cos \frac{n\pi x}{a} \quad \cos \frac{n\pi y}{b} \quad \cos \frac{n\pi z}{h}$$

where n is a positive odd interger.

The Eigenvalues, B_n

The eigenvalues that corresponds with the above eigenfunctions are

$$B_n^2 = \pi^2 \left[\left(\frac{n}{a}\right)^2 + \left(\frac{n}{b}\right)^2 + \left(\frac{n}{h}\right)^2 \right]$$

$$B_1^2 = \pi^2 \left[\left(\frac{1}{120.2}\right)^2 + \left(\frac{1}{72.4}\right)^2 + \left(\frac{1}{72.4}\right)^2 \right] = 0.00451 \text{ cm}^{-2}$$

$$B_3^2 = 0.0401 \text{ cm}^{-2}$$

$$B_5^2 = 0.1118 \text{ cm}^{-2}$$

The Fourier Coefficient S_n

S_n is defined in the expression of the Fourier series,

$$S(\bar{r}) = \sum_n S_n Z_n(\bar{r}).$$

It can be solved for when $Z_n(\bar{r})$ is known, as above.

$$S_n = \frac{8 \; S(\bar{r})}{abh}$$

In our case we are using 4 Pu-Be sources which collectively emit about 8×10^6, 5 mev, neutrons/sec. Therefore

$$S_n = \frac{8 \times 8 \times 10^6}{(72.4)^2(120.2)} = 102 \ \frac{\text{neutrons}}{\text{cm}^3\text{-sec}}$$

The Effective Multiplication Factor, $k_{n\ eff}$

The effective k is defined as

$$k_{n\ eff} = \frac{k\ e^{-B_n^2 \tau}}{1 + L^2 B_n^2}$$

$$k_{1\ eff} = \frac{0.99\ e^{-.230}}{1 + (2.6)(0.00451)} = 0.774$$

$$k_{3\ eff} = 0.120$$

$$k_{5\ eff} = 0.00284$$

The Maximum Neutron Flux

Of course, the maximum flux will occur at the center of the core where $x = y = z = 0$. Then $Z_n = 1$.

$$\phi_{max} = \phi(0) = \sum_n \frac{102\ (0.640)\ e^{-50B_n^2 \tau}}{0.147\ (1 + 2.6B_n^2)(1 - k_{n\ eff})}$$

$$= 1730 + 61.2 + 1.3$$

$$\phi_{max} = 1793 \simeq 2 \times 10^3 \text{ neutrons/cm}^2\text{-sec}$$

The Maximum Flux at the Sides

To find the estimated flux at the edge of the core it was assumed that $z = x = 0$, and $y = 30.1$cm.

Then, ϕ(edge of core) $= 1730(.257) - 61.2(.710) + 1.3(.966)$
$= 401$ neutrons/cm^2-sec

This is a very conservative estimate since the y value assumed is the distance to the nearest edge of the hypothetical parallelepiped. Also the minimum water shield distance will be assumed. This is the distance from the corner of the hexagon to the side of the tank, 37 cm.

The dose from a cylinder of radius r, at a distance from its axis, $\bar{r}_o$, is[7]

$$D\ (\text{cylinder}) \approx \sqrt{\frac{r}{\bar{r}_o}}\ \frac{S_a}{2}\ \exp\left[\frac{-(r_o - r)}{\lambda}\right]$$

Where S_a = The strength of the source at the surface of the cylinder

λ = The relaxation length of the shield material for a given radiation.

Then the

$$\text{Dose at side} \approx \sqrt{\frac{38.8}{76.2}}\ \frac{401}{2}\ \exp\left[\frac{-37.3}{10}\right]$$

$$\approx 3.4 \text{ neutrons/cm}^2\text{-sec}$$

2.4 Gamma Ray Flux Estimations[7]

Again being extremely conservative, assume that the average flux in the reactor is equal to the maximum flux, 1.8×10^3. Then the strength of the volume source is

$$S_v = N^u \sigma_f^u\ \bar{\phi}(\bar{r}) \text{ Fissions/cm}^3\text{-sec} \times 20 \text{ mev/fission}$$

Where σ_f^u is the fission cross section for the fuel, 20 mev is the total gamma radiation energy produced per fission, and $\overline{\phi}(\bar{r})$ is the average neutron flux.

$$S_v = (1.80 \times 10^{22})(3.92 \times 10^{-24})(1.8 \times 10^3)\ 20$$

$$S_v = 2.54 \times 10^3 \text{ mev/cm}^3\text{-sec}$$

The next problem is to find the relaxation length, which is equal to $1/\mu$, where μ cm^{-1} is the linear absorption coefficient of the gamma rays produced in the reactor core. This is found by obtaining an average density for the core and assuming that the mass absorption coefficient, μ/ρ, is about 0.032 cm^2/g for fission gamma rays.

$$\mu = 0.032\,\rho = 0.032\ (7.84) = 0.251 \text{ cm}^{-1}$$

Therefore, the relaxation length λ, of the core is

$$\lambda = \frac{1}{\mu} = 3.98 \text{ cm.}$$

The strength, S_a, of an isotropic surface source which is equivalent to the volume source, is given by

$$S_a = S_v \lambda = 2.54 \times 10^3 \times 3.98 = 10^4 \frac{\text{mev}}{\text{cm}^2\text{-sec}}.$$

Using the expression in section 2.3 for the dose from a cylinder,

$$\text{Dose at side} = \sqrt{\frac{38.8}{76.2}} \times \frac{10^4}{2} \quad \exp\left[-\frac{37.3}{30}\right]$$

$$= 1{,}700 \frac{\text{mev}}{\text{cm}^2\text{-sec}}$$

Even this extremely conservative estimate is well within the AEC minimum requirements.

2.5 Summary

A rough summary of estimations for the system are given in Table III.

TABLE III

Uranium, natural (Savannah River Slugs)	5500 lbs
Unit cell	hexagon
Center spacing	1 9/16 in.
V_{H_2O}/V_u	1.46
Fuel rod length (five slugs high)	42.5 in.
Ratio of lattice length to diameter	1.3
Reflector	15 in.
k_{eff}	$\sim .92$
Neutron amplification	~ 12
Maximum flux	$\sim 2 \times 10^3$ neutrons/cm^2-sec
Neutron flux at side of tank	< 4 $\frac{\text{neutrons}}{\text{cm}^3\text{-sec}}$
Gamma flux at side of tank	$\ll 1{,}700$ $\frac{\text{mev}}{\text{cm}^2\text{-sec}}$

BUCKLING VS VOLUME RATIO

(Oak Ridge Experiments)

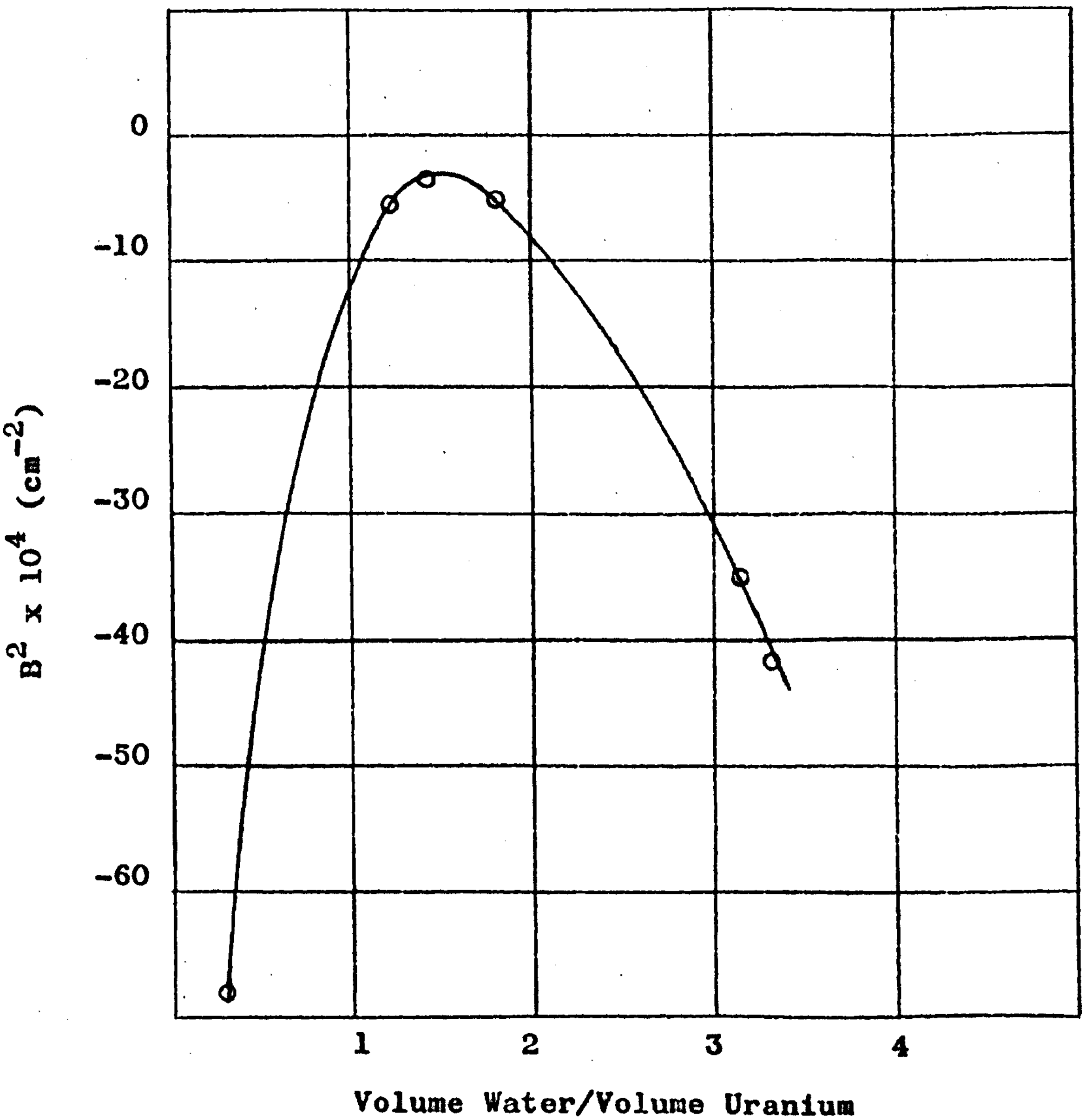


Figure 2.1

FUEL SPACING

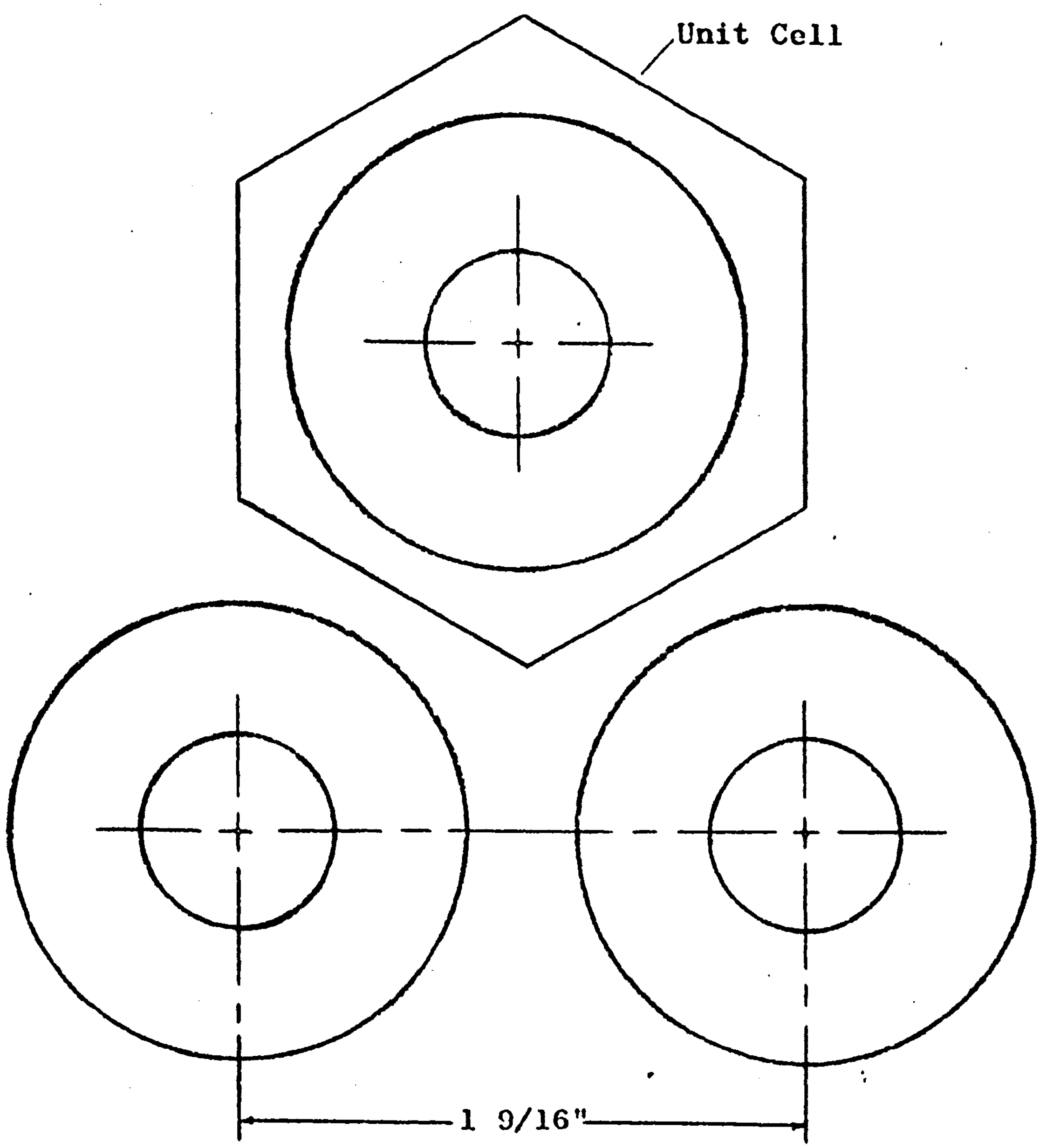


Figure 2.2

SAVANNAH RIVER FUEL SLUG

(Cross-Section - Not to Scale)

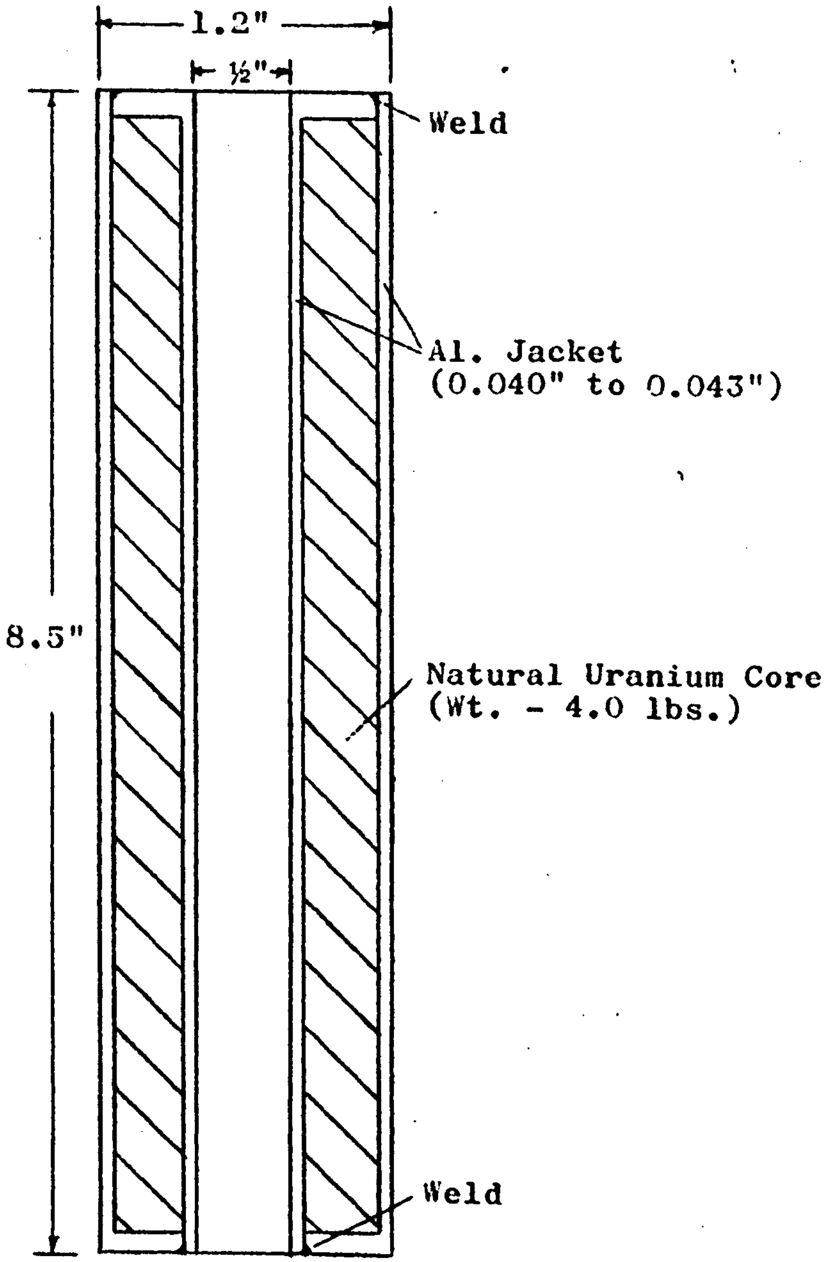


Figure 2.3

CHAPTER 3

DESIGN AND DESCRIPTION

3.1 General

The calculations in the previous chapter indicate that many of the considerations necessary for a large reactor can be eliminated for our assembly. For example, the maximum heat generation in the subcritical is not enough to warrant a cooling system. Also, no extra shielding is necessary other than the moderating water. Since the assembly can never reach criticality, an elaborate control system is not required. The instrumentation necessary is determined only by the type of experiments that are performed with the reactor.

However, many things, such as corrosion, need consideration in any water system. The possibility of the aluminum fuel cladding corroding away and allowing the uranium to enter the water is a serious problem. The water must therefore be very near pure to minimize corrosion. This requires a circulating system with a rather high flow rate, containing a filter and a deionizer.

All materials used in the reactor must be as corrosion resistant as possible. The only two metals used inside the reactor tank are stainless steel and 6061-T6 aluminum. All metal fabrication for the assembly was done by T.A. Caid and Sons of Tucson, Arizona.

From the requirements of the computations, the grid plates were made with fuel spacings of 1 9/16 inches. The source holder and poison rod were made with 1 1/4 inch aluminum tubing so they would be inter-changeable with any of the fuel rods. This allows adequate versatility in the arrangements of the source, poison, and fuel.

The complete subcritical assembly fits comfortably in a 15 by 20 foot laboratory. The platform around the reactor tank occupies 81 square feet of floor space. The circulating system and other supplementary equipment takes additional floor space. The laboratory is required to have a nine foot ceiling to allow ease of removing the fuel elements and source holders.

Both the reactor tank and the platform were made in two detachable parts. The assembly is, therefore, mobile and can be moved into any room having a standard doorway. The platform is on casters, thus making the equipment underneath easily accessible.

3.2 Reactor Tank

The reactor tank is made of 0.104 inch stainless steel, type 304. It is 60 inches in diameter and 50 inches high. The tank is made in two sections. The lower section is 32 inches high with one inch angle around the top and has a welded in bottom. The upper section is 18 inches high with one inch angle on both ends. The angles that fit together at the midsection of the tank are welded all around to their respective section to insure water tightness. The two sections are joined with a rubber gasket and drawn together tightly. (See Figures 3.1 and 3.2)

3.3 Core Structure

The fuel rods rest on the bottom of the tank but are held in place by the grid plates. There are two, 1/8 inch, circular, aluminum grid plates. They are 36 inches in diameter and are spaced 28 vertical inches apart. The lower plate is about 1 1/2 inches above the bottom of the tank. The two plates are held in position by six, 1 1/2 by 1/8 inch angle braces. Six adjustable locating arms extend from the top of these braces. These are to center the core and to keep it from leaning. There are three locating arms on the tank floor positioning the braces from the side. There are three hundred and seven 1 1/4 inch holes in the grid plates in the triangular pattern suggested by chapter two. (See Figures 3.1 and 3.2)

3.4 Dust Cover

When the reactor is not in use it is covered to prevent dust and other contamination from entering the water. The cover is made of 1/16 inch aluminum to make it light-weight. It is 62 inches in diameter with one inch angle around the perimeter. This angle fits down over the angle of the tank. Support to the cover is given by two, one inch channel crossmembers. Holes in the angle allow it to be locked to the tank.

3.5 Fuel Rod

The fuel rods are simple in construction and allow maximum versatility. The desired number of fuel slugs are slid over a 48 inch long, aluminum tube. The slugs can be placed four, five, or six high. The 7/16 inch tubing has a 0.035 inch wall so that a 5/16 inch neutron detector can enter the tube for flux measurements. A stainless steel cotter pin extends through the tube, one inch up from the bottom. Over the pin is an aluminum washer that the slugs rest on. The one inch of moderator at the bottom allows one diffusion length for reflector savings. (See Figure 3.3)

3.6 Source Holder

There are four source holders, since four Pu-Be neutron sources are available for use. Each holder is a 1 1/4 inch aluminum tube, 48 inches long and welded at the bottom. The four sources can be put in individual tubes or all in the same tube. The tubes have 0.035 inch walls and are completely filled with polyethylene except for the space taken by the sources. (See Figure 3.3)

3.7 Poison Rod

The poison rod is a 1 1/4 inch aluminum tube, 41 inches long, and welded closed at both ends. The inside contains a 40 x 3 1/2 x 0.020 inch cadmium sheet rolled into the form of a cylinder. This rod is not for control, but rather for experiments measuring flux depressions. (See Figure 3.3)

3.8 Platform

The wooden platform is nine foot square, and one foot high. It is supported by 2 x 6 inch members. There is a 42 inch high railing around the platform except at the entrance. The platform is in two sections, held together by two hooks on either side. The whole platform is mounted on casters which lock into position. The top is covered with linoleum.

3.9 Circulation System

The circulation system was supplied by the American Demineralizer Company. The system consists of a pump, filter, deionizer, and a two probe purity meter. The inlet pump is connected to a perforated, one inch plastic pipe resting on the bottom of the tank. The water is recirculated from the pump through the deionizer and filter and hence into the tank at the rate of approximately 10 gallons per minute. For the original filling a temporary deionizer was used to fill and recirculate during the first week of operation. The components of the system are as follows:

1 MB 10 x 54 Deionizer, epoxy lined containing two cubic feet of resins

A bronze recirculating pump with an operating head of 55 feet and a flow rate of 15 gallons per minute.

Purity Meter - two cells, one before and one after the deionizer, battery operated.

1 AD #20 Kleer Flo cartridge filter.

A plastic pipe assembly for the inlet at the bottom of the tank.

Necessary hoses and fittings.

(See Figure 3.4)

3.10 Extra Equipment

Scaler - Baird-Atomic, Model 131A

Power Supply - Baird-Atomic, Model 312A

Amplifier - Baird-Atomic, Model 212

Neutron Scintillation Detector - Nuclear-Chicago, Model DS8-10 (with 5/16 inch outside diameter by 42 inch long probe)

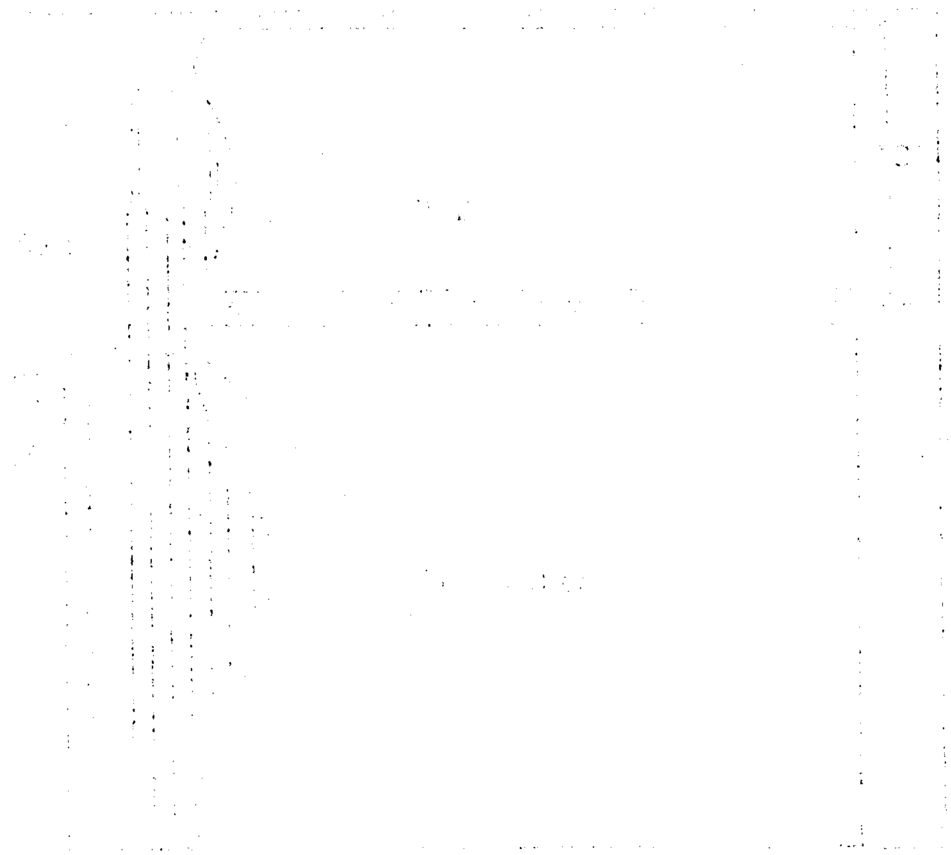
Boron Trifluoride Detector - Nuclear-Chicago, Model NC-202 (5/16 inch by 42 inch probe)

Survey Meter - Nuclear-Chicago "Cutie Pie" Model 2586

<u>Slow Neutron Meter</u> - EKCO, Model SN87

<u>Fast Neutron Meter</u> - National Radiac, Model EP 1

<u>Geiger Counters</u> - Universal Atomic, Model 4



REACTOR

(side view)

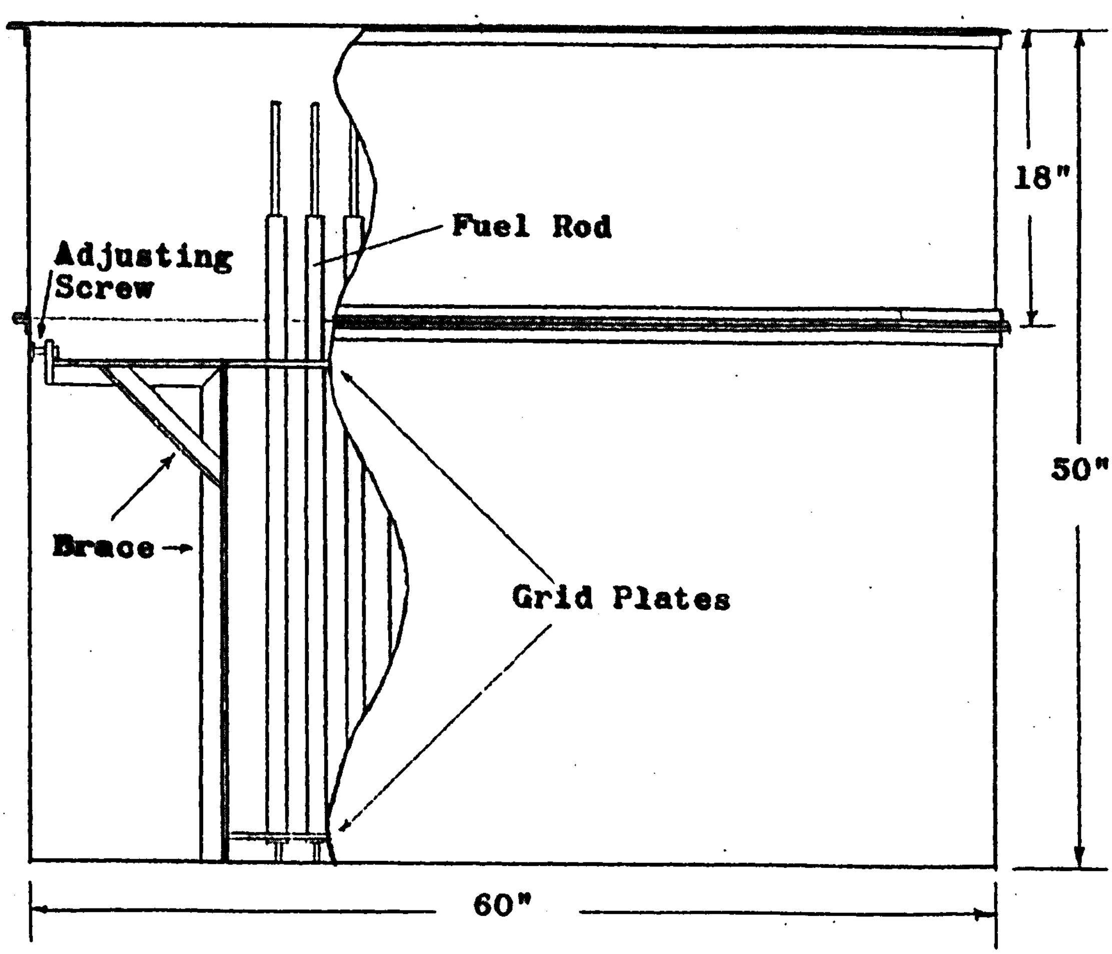


Figure 3.1

REACTOR

(top view)

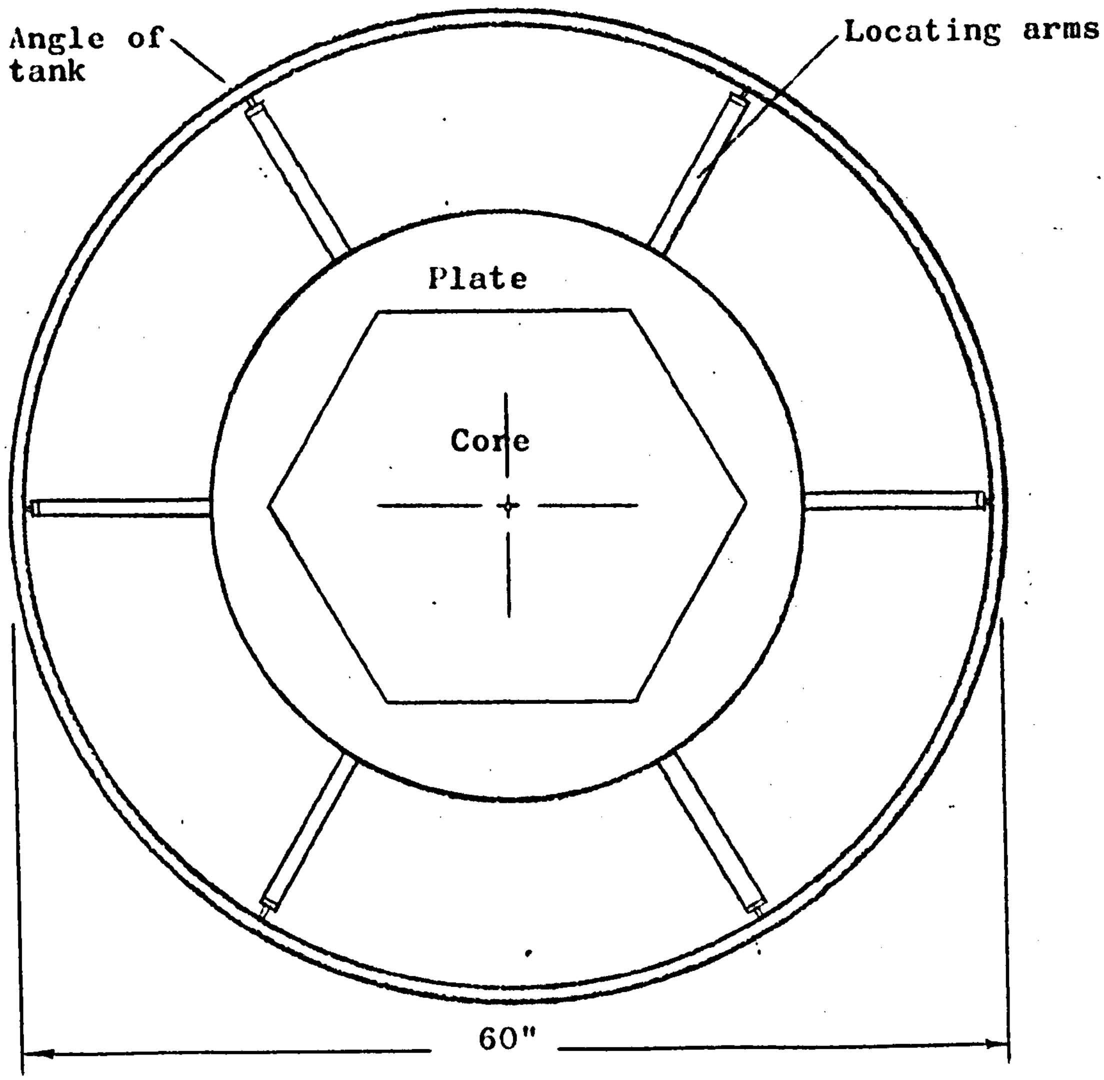


Figure 3.2

CORE ELEMENTS

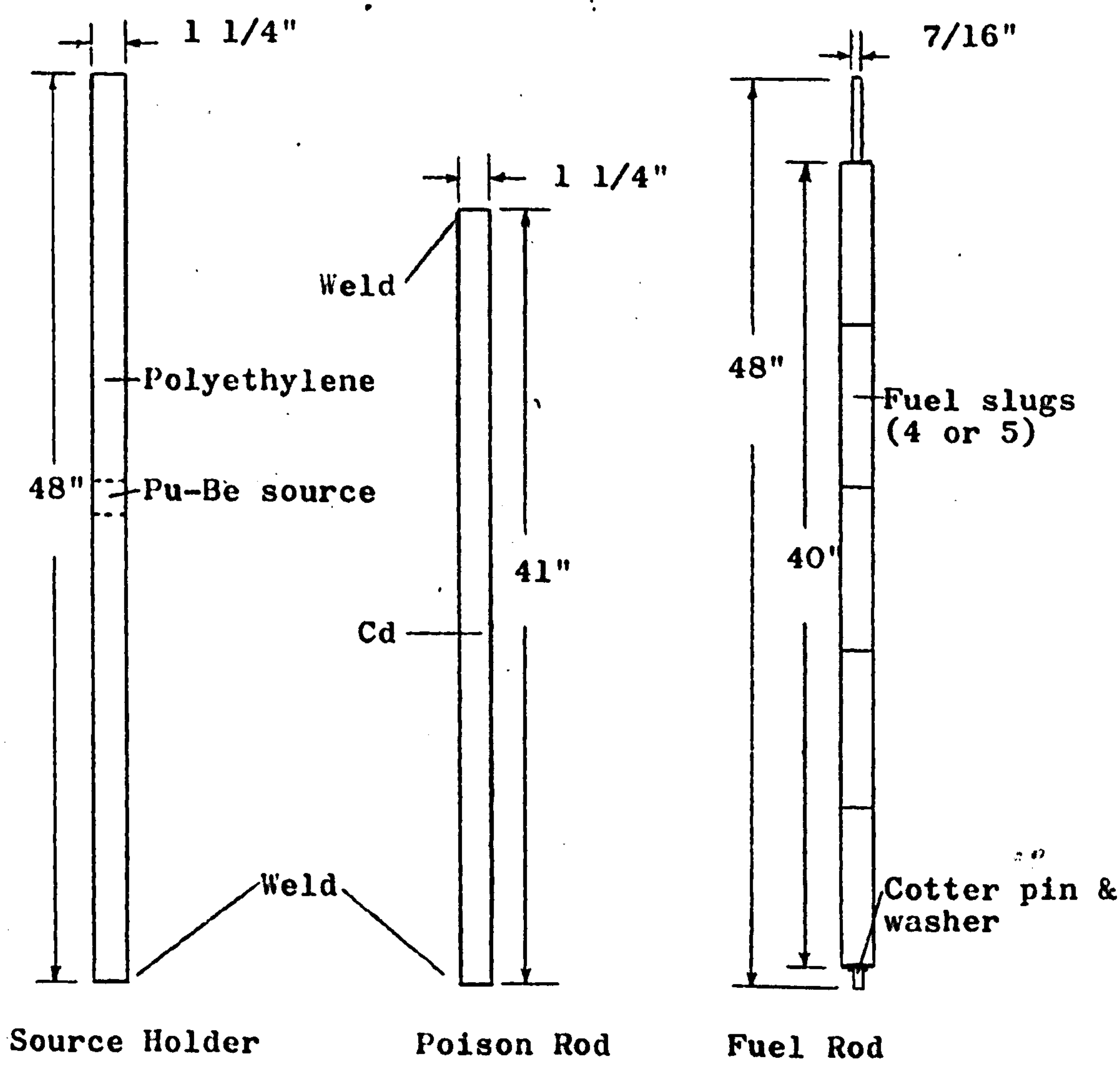


Figure 3.3

CIRCULATING SYSTEM

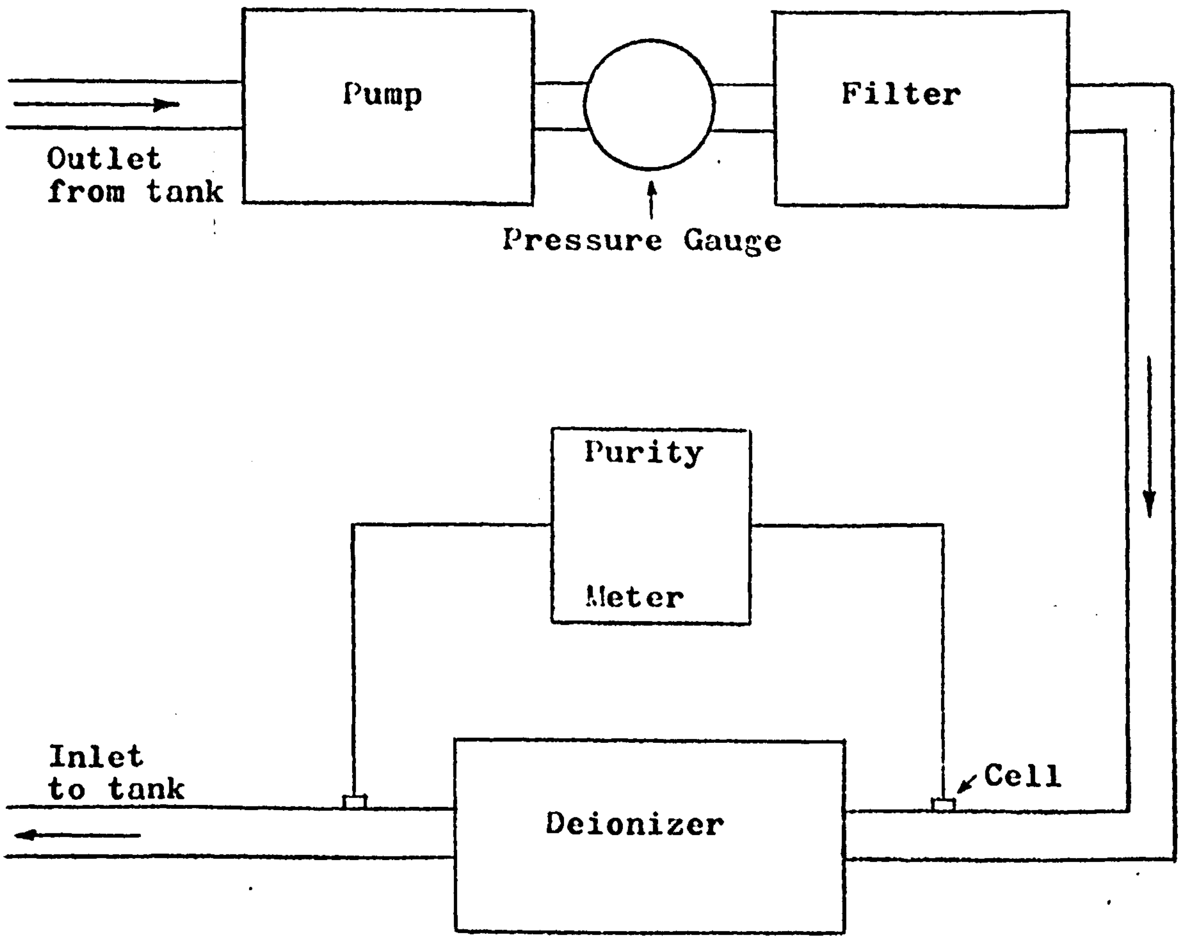


Figure 3.4

CHAPTER 4

TESTING AND DISCUSSION

4.1 Flux Distributions

Upon completion of the construction of the various components, the tank was filled with water. The circulation system was operated for two days to filter and deionize the water. When the resistivity of the water exceeded about 10 meg ohms, the fuel was inserted. The initial loading of the lattice was treated as an approach to criticality, and was accompanied by the standard safety practices established for critical approaches.

The neutron detector used in all the experiments was a Nuclear Chicago, boron trifluoride counter. The small detector is enclosed in a 42 inch probe that has a 5/16 inch diameter. The active volume is 1/2 of an inch long and is centered 1 1/2 inches from the tip of the probe. The pressure of the $B^{10} F_3$ (96 % enrichment) gas is 70 cm of mercury. The operating voltage range for the detector is 1200-1450 volts.

The amplifier, scaler, and power supply used were the Baird-Atomic models specified in section 3.10. The signal that orginated in the detector was first sent through a pre-amplifier and then into the 212 Amplifier.

The low energy noise was discriminated out by the pulse height selector in the amplifier. There is also a pulse height selector in the scaler, but it was not used to discriminate pulses. The multiplier setting on the amplifier was always on 10. From the amplifier, the signal entered the 131A, glow tube scaler. Here the pulses were counted for the preset time period.

The high voltage supply has an 85 to 1450 volt range. This range is in 17 steps of 85 volts each. The steps are denoted by the letters of the alphabet. For all experiments a positive high voltage of 1360 volts was applied to the detector. The 1360 volts corresponds to the step setting of P, and is in the middle of the operating voltage range of the detector.

Traverses were made to find the variation of the neutron flux with distance from the central axis of the assembly (the height being fixed). The spacial distribution sought in these instances was the cell-to-cell variation, with the intracell variation having been factored out. Values on this macroscopic curve are found by sampling the neutron flux at those points along a radius which are equivalent from the standpoint of local flux depression by the fuel. For convenience, these equivalent points were choosen in the center of the fuel rods. The center of the active volume of the detector was the same height as that of the single neutron source in the center position of the reactor. The points on all of the curves are plotted with a plus and minus one standard deviation (normalized, of course).

Figure 4.1 shows the horizonal flux distribution four rows over from the source. This traverse was parallel to the point-to-point diameter of the hexagon. For this curve the count rate ranged from 500 to 1710 counts/min, and the percentage error ranged from $\pm$ 1.2 % to 2.0 %. The percentage error is defined as,

$$\text{Percentage error} = \pm \frac{100}{(\text{Total Counts})^{1/2}} \ \%.$$

Figure 4.2 shows the radial flux distribution. This traverse was made from the central source to the point of the hexagon. This curve is an exponential of the form,

$$\phi(\bar{r}) = \text{const.} \times \exp\left[-1.08\ \bar{r}\right].$$

The percentage error for Figure 4.2 ranged from 1.8 % to 4.4 %.

By comparing the two horizonal traverses one can see the effect of the source on the flux distribution. In an attempt to minimize the effect of the source, a radial flux traverse was made with the source on the outside of the core. This curve appears in Figure 4.3. Due to the small count rates in this experiment, the percentage error range was 6.4 % to 14 %.

A curve of the axial flux appears in Figure 4.4. The measurements were made inside a fuel rod that was in the fourth row from the source. The percentage error range for this curve is 2.5 % to 14 %.

In general, the flux distributions were about as expected.[8]

4.2 Absolute Neutron Flux

The estimations in chapter two were made to predict the safety of the dose rates around the reactor tank. To check the dose rates at the boundaries, various survey meters were used. The meters used were as follows: National Radiac Fast Neutron Meter, minimum reading possible—0.4 mrem/hr; EKCO Slow Neutron Meter, minimum reading—1.0 mrem/hr; and Universal Atomic Geiger Counter, minimum reading—0.1 mr/hr. With one source in the center of the core, no reading could be obtained from any of the survey meters. This means that the dose rates were below the thresholds of the survey meters and therefore far below the AEC maximum permissible dose rates.

In order to determine the neutron fluxes at the center and edge of the core, the BF_3 counter was again used. However, the sensitivity of the counter had to be determined. Since the industrial supplier only suggested that the sensitivity was about 0.02 count/sec per unit neutron flux, a calculation was made to check this accuracy. For a Maxwell-Boltzmann energy distribution, the sensitivity of a BF_3 counter is given by,[9]

$$\text{Sensitivity} = 3.56 \times 10^{-21}\ NV$$

where N is the number of B^{10} atoms per cubic centimeter and V is the sensitive volume of the counter. For our detector, with the specifications given in section 4.1, this becomes

Sensitivity = 0.0192 count/sec per unit neutron flux.

Since this agrees well with the value given by the manufacturer, the value of 0.02 was assumed for the calculation of fluxes.

A comparison of the experimental flux with the estimated flux is given in Table IV. These fluxes are obtained with four of the Pu-Be sources in the center of the core.

TABLE IV

REACTOR FLUX
(n/cm^2-sec)

	Experimental	Estimated
Flux at flat edge of lattice	610	400
Maximum flux in fuel rod	5,400	1,800
Maximum flux in water	10.000	
Maximum flux obtainable (Measuring in removed fuel rod space)	16,000	

The experimental fluxes, as expected, are higher than the estimations in chapter two predict. However, considering the crude calculations of chapter two, the correlation between the estimated flux and experimental flux is fairly good. The largest assumption in chapter two is that the system is homogeneous. Not only does this make a big difference in the resonance escape probability, but the microscopic flux is radically different. In a homogeneous system, of course, there is a uniform thermal neutron sink throughout the system, lowering the flux.

Even with the neutron flux higher than the estimated value, the dose rate estimations are still conservative. This is because of the extremely conservative assumptions in the dose rate estimations.

4.3 Discussion

All of the design objectives for the subcritical were met. The total cost of the entire assembly was considerably less than half the price of a commercial subcritical. The internal assembly is simple and accessible for experimentation. It is quite versatile as to core dimensions, source distributions, and etc. The maximum flux is comparable to that in other university subcriticals.

The water-to-metal ratio is fixed for the original grid plate. The optimum ratio is not accurately known for Savannah River slugs in light water. Recent investigations at Hanford seem to indicate that the optimum ratio might be between 1.6 and 1.8.[10] Therefore, a slightly better system might be obtained if the ratio is raised from the 1.46 used in the orginal grid plates.

The water in the tank is kept highly pure (resistivity is greater than 100 meg ohms). However, there is evidence that the aluminum under water is corroding. Fortunately, the core structure is corroding in preference to the fuel cladding.

There are a number of avenues open for further investigation. New grid plates could be made to determine the optimum arrangement of fuel and moderator. Perhaps even a variable lattice could be developed. Calibrated foils should be used to accurately determine the macroscopic flux variations. The intracell flux distribution could also be determined with the use of foils. All the determinations of reactor parameters have been left for future experimentation.

Of the many experiments that are normally done on a subcritical, the only one that cannot be performed on this one is the determination of the temperature coefficient. To do this experiment it is necessary to incorporate a heating unit into the circulation system. With that limited exception, the subcritical reactor appears to be a safe, versatile, and useful addition to the facilities for instruction at the university.

HORIZONAL NEUTRON FLUX
(Four rows from center)

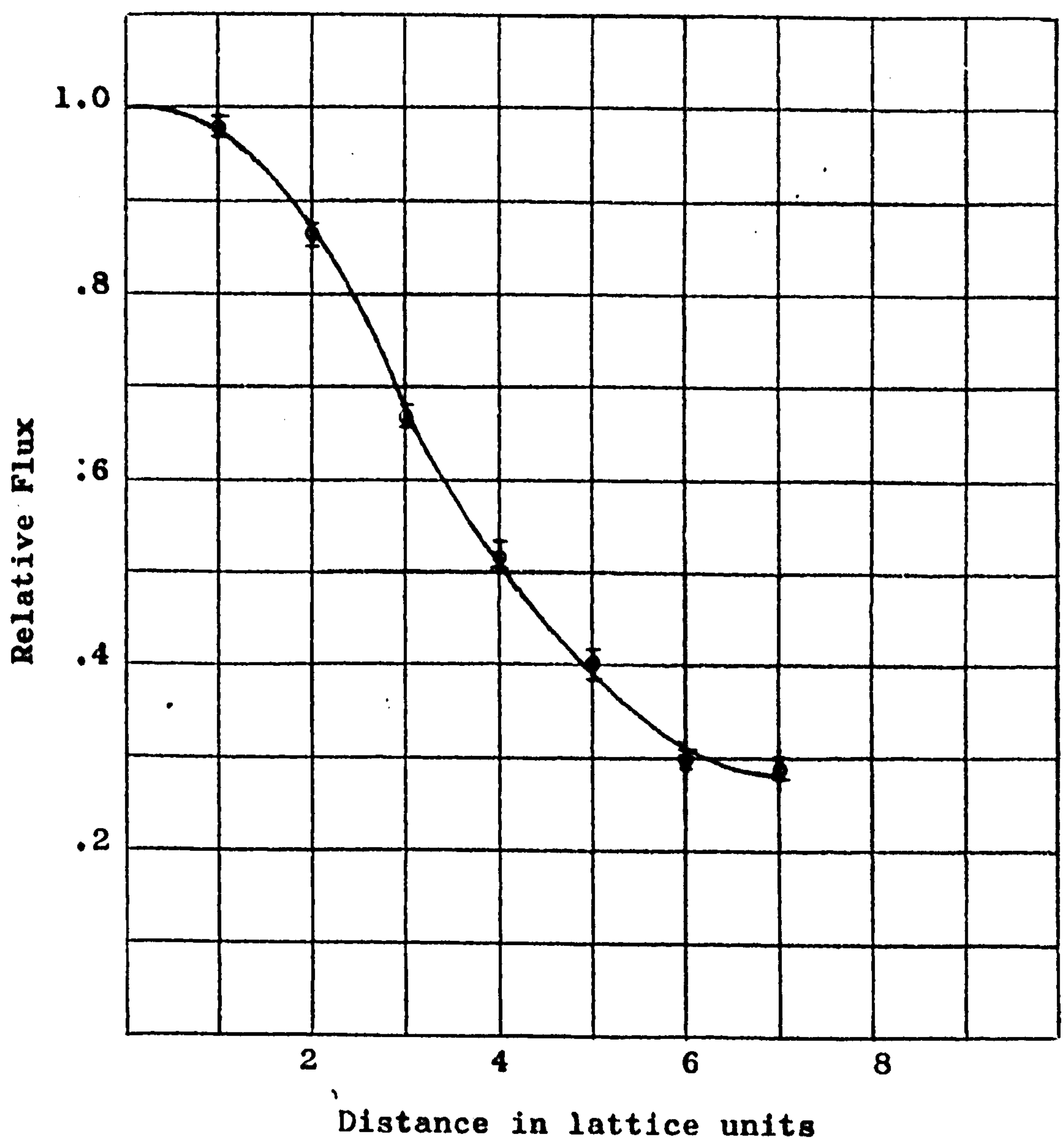


Figure 4.1

RADIAL NEUTRON FLUX
(From center to point)

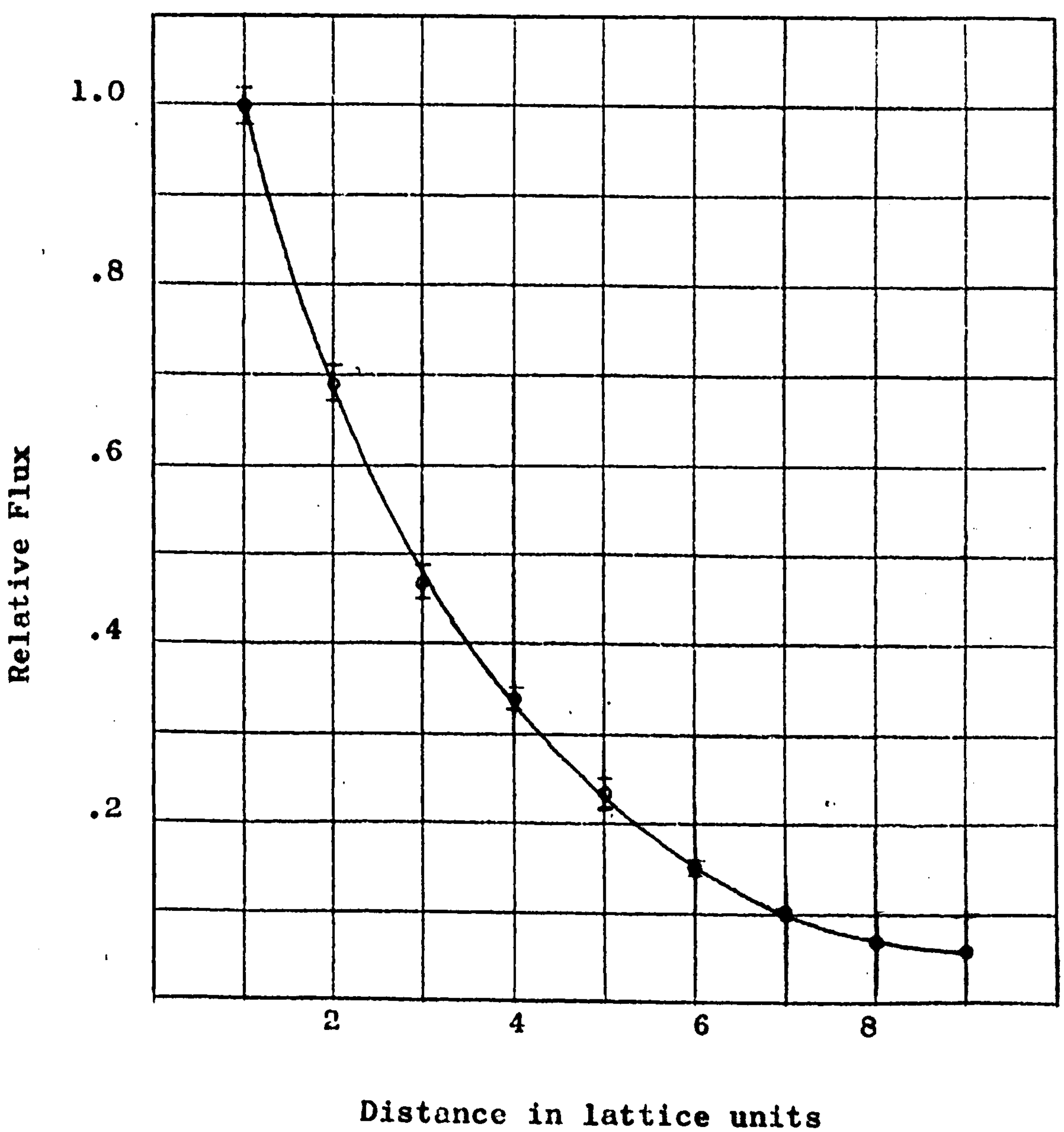


Figure 4.2

HORIZONAL NEUTRON FLUX
(Source on outside of core)

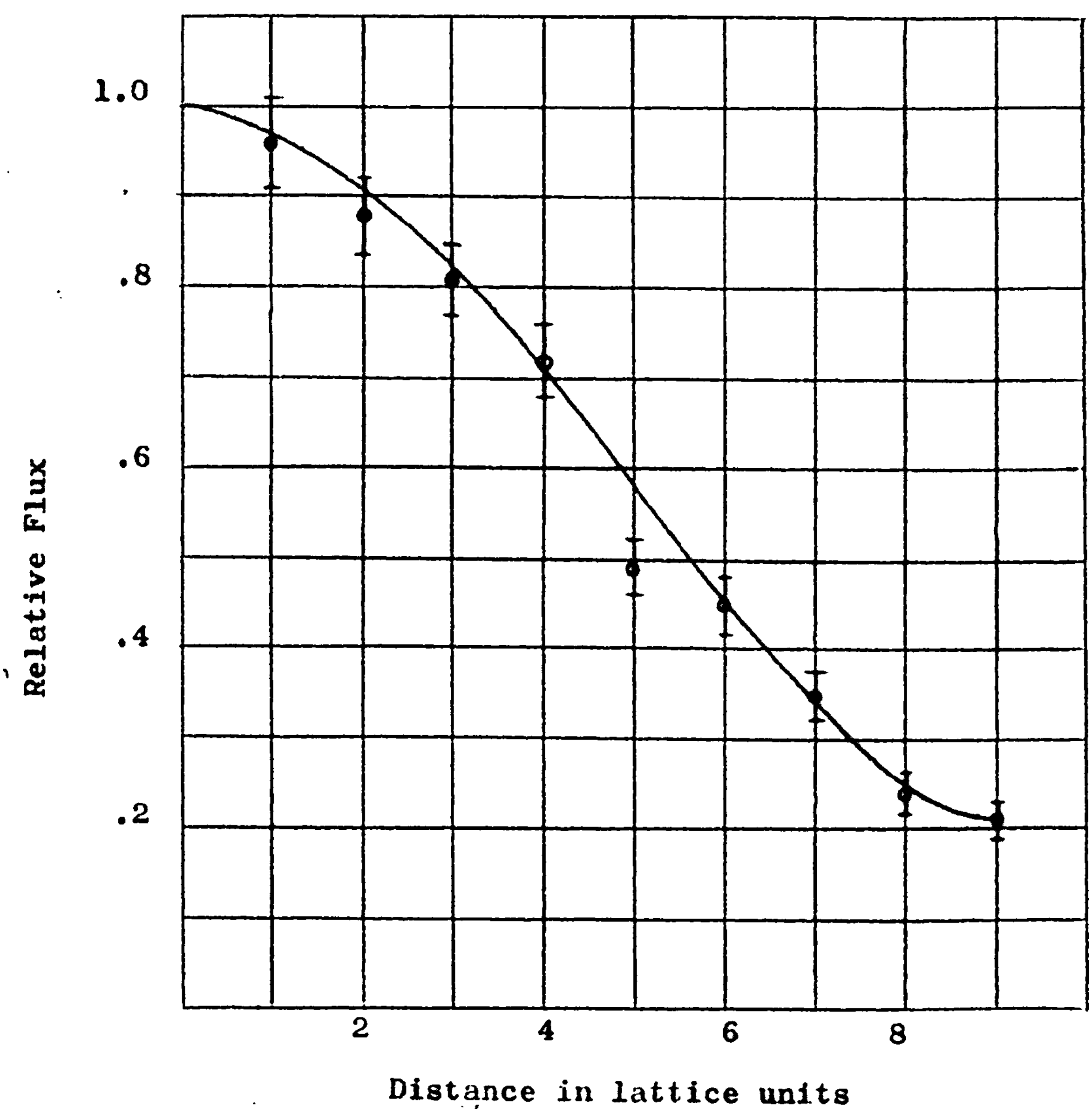


Figure 4.3

VERTICAL NEUTRON FLUX
(Four rows from the source)

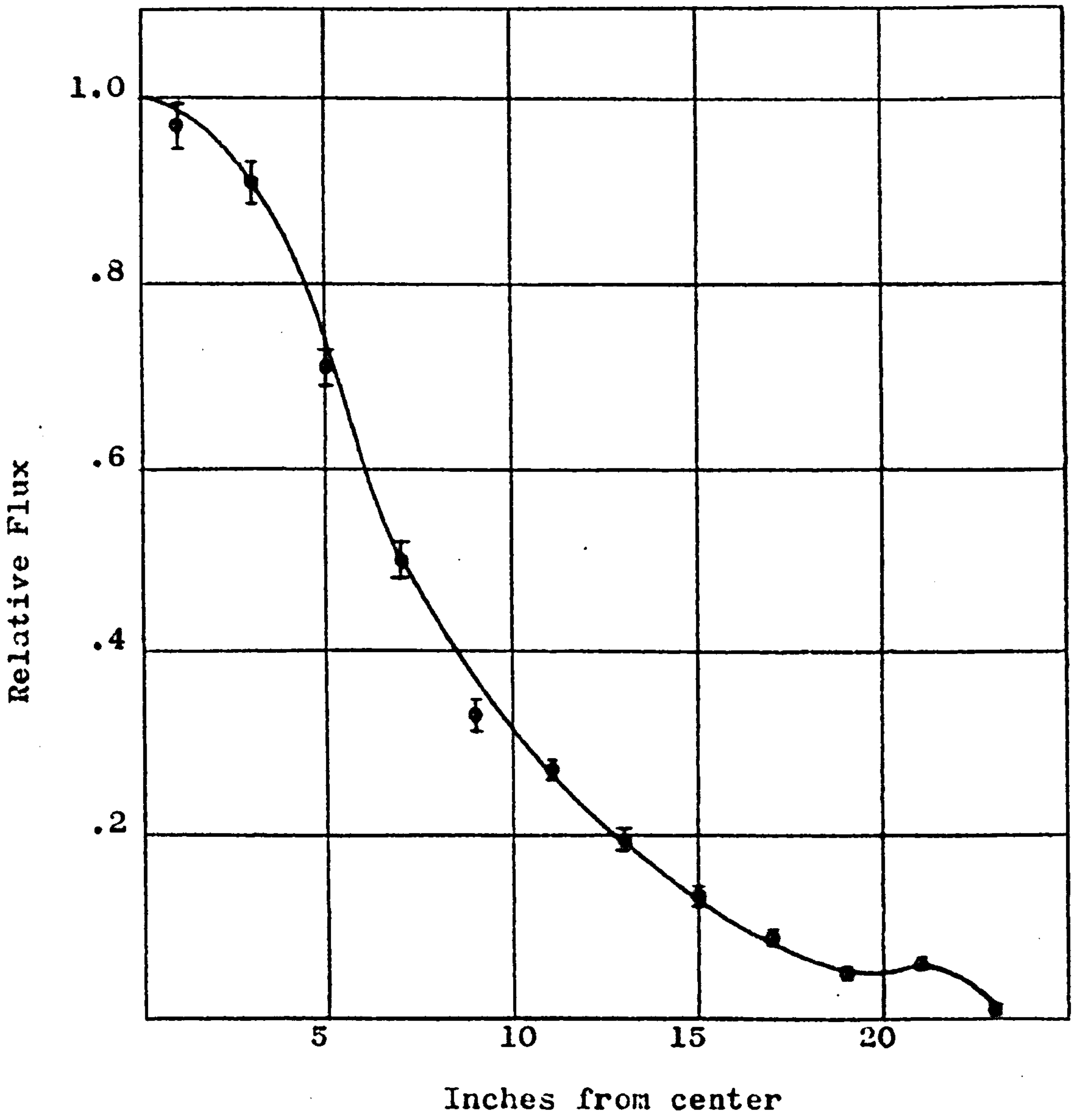


Figure 4.4

BIBLIOGRAPHY

1. Richard Stephenson, "Educational Uses of the Pickle-Barrel," presented to the Second Nuclear Engineering and Science Congress, 1957.

2. Robert V. Meghreblian and David K. Holmes, Reactor Analysis, (New York: McGraw-Hill, 1960).

3. H. Kouts et al. "Exponential Experiments with Slightly Enriched Uranium Rods in Ordinary Water," A/Conf. 8/5, P/600, 1956.

4. R. Arnette et al. "Exponential Experiments with Water-Metal Rod Lattices," CP-2048, October 3, 1944, P. 28.

5. U.S., Selected Reference Material on Atomic Energy, Vol. II: Reactor Handbook: Physics, August 1955.

6. Samuel Glasstone and Milton C. Edlund, The Elements of Nuclear Reactor Theory, (New York: D. Van Nostrand, 1952), Chapter 12.

7. Samuel Glasstone, Principles of Nuclear Reactor Engineering, (New York: D. Van Nostrand, 1955).

8. Herbert Kouts and Rudolph Sher, "Experimental Studies of Slightly Enriched Uranium, Water Moderated Lattices," BNL 486, September 1957.

9. William J. Price, Nuclear Radiation Detection, (New York: McGraw-Hill, 1958), P. 262.

10. Nuclear Physics Research Quarterly Report, January - March 1959, HW-60220, (Washington: U.S. Atomic Energy Commission), P. 57.