

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION



ORNL - TM - 277

Ref
206

**FUEL CYCLES AND LOADING PROGRAMMING
FOR WATER-COOLED RESEARCH REACTORS**

A. L. Colomb
D. Cavin

NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report. The information is not to be abstracted, reprinted or otherwise given public dissemination without the approval of the ORNL patent branch, Legal and Information Control Department.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ORNL-TM-277

Contract No. W-7405-eng-26

FUEL CYCLES AND LOADING PROGRAMMING
FOR WATER-COOLED RESEARCH REACTORS

A. L. Colomb
D. Cavin

Paper presented at Conference on
Light-Water-Moderated Research Reactors
June 11-14, 1962
Gatlinburg, Tennessee

Date Issued

JUL 31 1962

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. Atomic Energy Commission



FUEL CYCLES AND LOADING PROGRAMMING
FOR WATER-COOLED RESEARCH REACTORS

A. L. Colomb, D. Cavin

ABSTRACT

A general discussion of the problems encountered while developing loading methods for the ORNL water-cooled research reactors is presented in this paper.

The methods used to program the fuel loading for the LITR and ORR are described as well as the experimental methods that produced the data necessary for the solution of these problems.

FUEL CYCLES AND LOADING PROGRAMMING
FOR WATER-COOLED RESEARCH REACTORS

Burning a fissionable material in a nuclear reactor is a process that is, in general, well understood; and good analytical models are available for the solution of this problem.¹ Nevertheless, when these models have to be applied to an actual reactor, some serious difficulties are encountered.

Some of the basic data needed to carry out the computations are not sufficiently accurate; fission product cross section is a good example of this difficulty. If the spacial distribution of the flux and the fuel has to be taken into consideration, the calculations become very intricate. Added to these difficulties is the problem of costs which is still not well defined. Information on reprocessing costs and on the refund value of reprocessed fissionable material is difficult to obtain.

Because of the lack of adequate answers to some of these questions, this work will only describe a particular case of the more general problem.

The methods used at ORNL to make efficient use of the nuclear fuel in the water-cooled research reactors will be commented on; and, in the light of this experience, some general conclusions concerning fuel-loading and fuel-inventory problems for reactors of a similar type will be drawn.

Fuel Procurement

Before burning fuel, it has to be purchased; the next paragraph will describe some important points of this operation.

¹M. Benedict and T. H. Pigford, Nuclear Chemical Engineering, McGraw-Hill, New York, 1957.

The ORR fuel element consists of 19 composite plates containing U²³⁵ fuel in the form of aluminum-clad uranium-aluminum alloy. It is required that the cladding be metallurgically bonded to the fuel cores and that the cores be hermetically sealed.

Prior to 1960 these fuel elements were produced at ORNL by a furnace brazing technique; however, since that time, they have been purchased from a commercial manufacturer using a mechanical-assembly procedure. While it is not the purpose here to discuss in detail the specifications of these fuel elements, there are several areas in which experience indicates the necessity for care in specifying the details of manufacture; and the requirements placed on ORR elements will be mentioned briefly.

Blisters

The presence of nonbonded regions or blisters between the fuel core and the cladding can result in local increases in heat flux. In the case of mechanically assembled elements, each plate is subjected to a heat treatment at a minimum temperature of 500°C for a period of one hour prior to the final 15 to 25% cold-roll reduction. Finished plates showing a single blister of more than 0.016 in. in diameter are rejected. In the case of furnace-brazed elements, the heat-treatment temperature is increased to 607°C and the final cold roll is omitted.

Surface Defects

Prior to assembly the surfaces of the aluminum cladding on the fuel-bearing section are examined for pits, scratches, and dents. Any pits or scratches greater than 0.003 in. deep are cause for rejection, and dents greater than 0.012 in. in diameter and greater than 0.012 in. deep are not acceptable. In the case of brazed assembly, the braze material

shall not extend over the fuel-bearing portion of the plate.

Contamination

The fuel surfaces shall contain less than 5 μg of uranium per sq ft. The surface layer is considered to extend into the plate to a depth of 0.003 in. The presence of small amounts of fission products in the cooling system is usually of greater concern in the case of open pool reactors than is true for tank-type reactors; therefore, in the former case, it is recommended that contamination be held to 1 μg of uranium per sq ft if possible.

Experience at ORNL with commercial fuel elements has, so far, been very satisfactory; however, much of the credit for this must be attributed to the existence of reasonable, yet adequate, specifications and the wholehearted cooperation of the manufacturer.

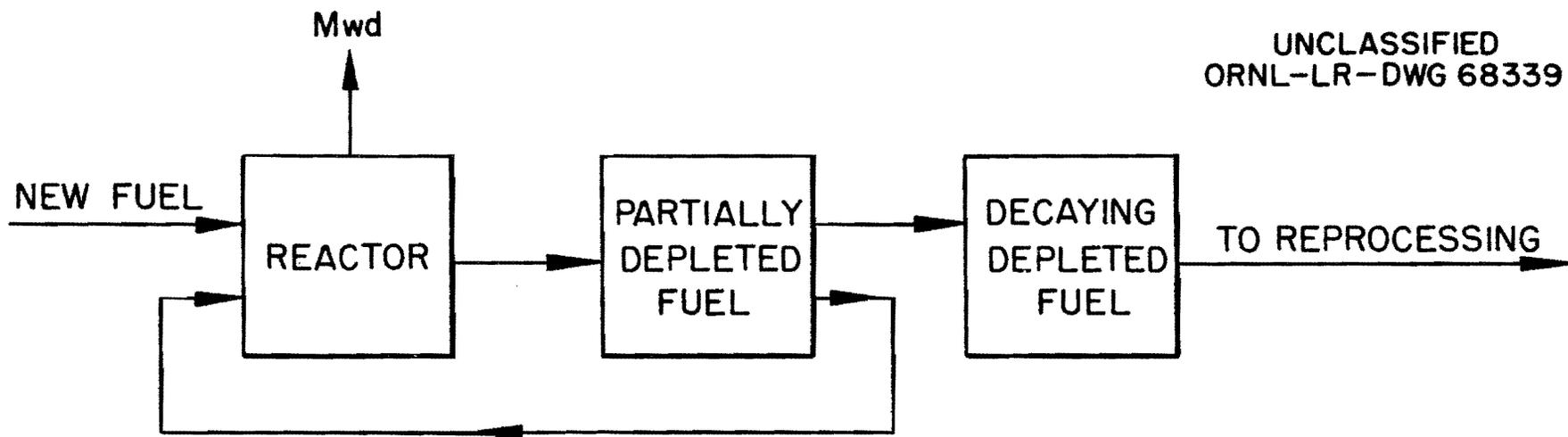
Fuel Cycle

The efficient burning of nuclear fuel in an MIR-type reactor can be condensed in the following statement:

What is the highest burnup achievable with a realistic cycle duration, with as small a fuel inventory as possible, and without exceeding some excess reactivity limitation.

A relatively simple flow diagram, Figure 1, can describe the system and will help in the discussion of the parameters affecting the achievement of a high burnup.

This general system can be simplified for reactors operating at a power level lower than 6 to 8 Mw. With this type of reactor, the xenon peak can always be overridden if the allowable amount of excess reactivity is sufficient; therefore, there is no need for a partially-depleted fuel



Fuel Flow Diagram.

Fig. 1.

inventory. New fuel can be added at the beginning of each fuel cycle to compensate for the amount of fuel converted into energy and for the amount of unburned fuel removed. The size of the depleted fuel inventory will be a function of the operating power level, of the achieved burnup, and of the cooling time necessary to minimize shield weight during the transfer to reprocessing. The burnup will be determined by the fuel content of the new elements and by the excess reactivity limitation.

In this type of reactor only a small number of new fuel elements have to be replaced every cycle. The remaining elements are moved in the core to equalize the burnup variations produced by the flux distribution. These moves can be made by displacing every element along a predetermined path² or, as in the LITR, by balancing the total fuel weight contained in every row and column.

When the operating-power level is higher than 6 to 8 Mw, the xenon peak reaches so high a value that most of the fuel contained in the core at the end of a cycle has to be removed and replaced partly by new elements and partly by partially depleted elements. The reactor could, of course, be loaded only with new elements at the beginning of every cycle; but this scheme results in a low burnup--approximately 10 to 15%. This burnup can easily be increased by a factor of two to three by recycling partially depleted fuel elements.

This fuel recycling introduces some difficulties. The balance of the system has to be kept carefully in equilibrium; that is, the addition of new fuel has to be equal to the removal of fuel. The selection

²E. E. Hill, F. J. Shon, "Fuel Programming in Pool-Type Research Reactors of Intermediate Power Level", Nuclear Science and Eng.: II, 105-110 (1961)

of the partially depleted fuel elements has to be made so as to keep the weight distribution of the inventory as flat as possible.

Here again, the burnup achievable is a function of the fuel content of the new elements and of the excess reactivity limitation. The size of the partially depleted fuel-element inventory is now much more difficult to determine. This inventory will have to contain at least the number of elements required for the next loading. In practice, the number of elements contained in this inventory will have to be much larger. Irregularities in cycle duration raise difficulties if the required weight distribution in an inventory is too limited. Two to three core loadings will have to be stored as partially depleted elements to make the selection of a new loading possible.

It is evident that an increase of the fuel contents of the new elements will increase the burnup without changing the cycle duration appreciably. In first approximation and for a given amount of excess reactivity, the average fuel weight of the critical core will always be the same whatever the amount of fuel contained in the new element. The larger amount of fuel introduced in the core by new elements can be compensated by using lower weight, partially depleted elements. Therefore, a 10% increase in initial fuel weight will increase the life span of the usable, partially depleted element inventory by 20%; and the burnup will increase by an appreciable amount.

The effects of this amelioration should be carefully estimated. Increasing the initial fuel weight will increase the amount of fission products in the core and, therefore, increase the absorption cross section of the core resulting in a hardening of the neutron spectrum. It

could also create some loading difficulties because of the flux peaks that may result from the proximity of a light element and a heavy element. It will also be more difficult to equalize individual burnup of the elements because the burnup at constant flux is proportional to the fuel weight at the beginning of the irradiation.

The last remark before illustrating the preceding discussion with some practical examples is concerned with the lack of exact knowledge of spatial distribution of the flux in a reactor of the type studied. For operational reasons, the flux is usually measured by irradiating one cobalt wire per element; and this time-consuming measurement is done as seldom as possible. Figure 2 shows the results of a measurement made at the ORR with 16 cobalt wires per element. The three elements were intentionally selected with dissimilar weights because such an arrangement is very likely to occur in an actual reactor core. It can be seen from this flux map that if the one-wire technique yields the actual average flux it is purely accidental. This state of affairs has a profound effect on any attempt to compute fuel cycles and loadings for this type of reactor. It actually means that the only time the U^{235} weight of an element is known with high accuracy is when the element arrives from the factory.

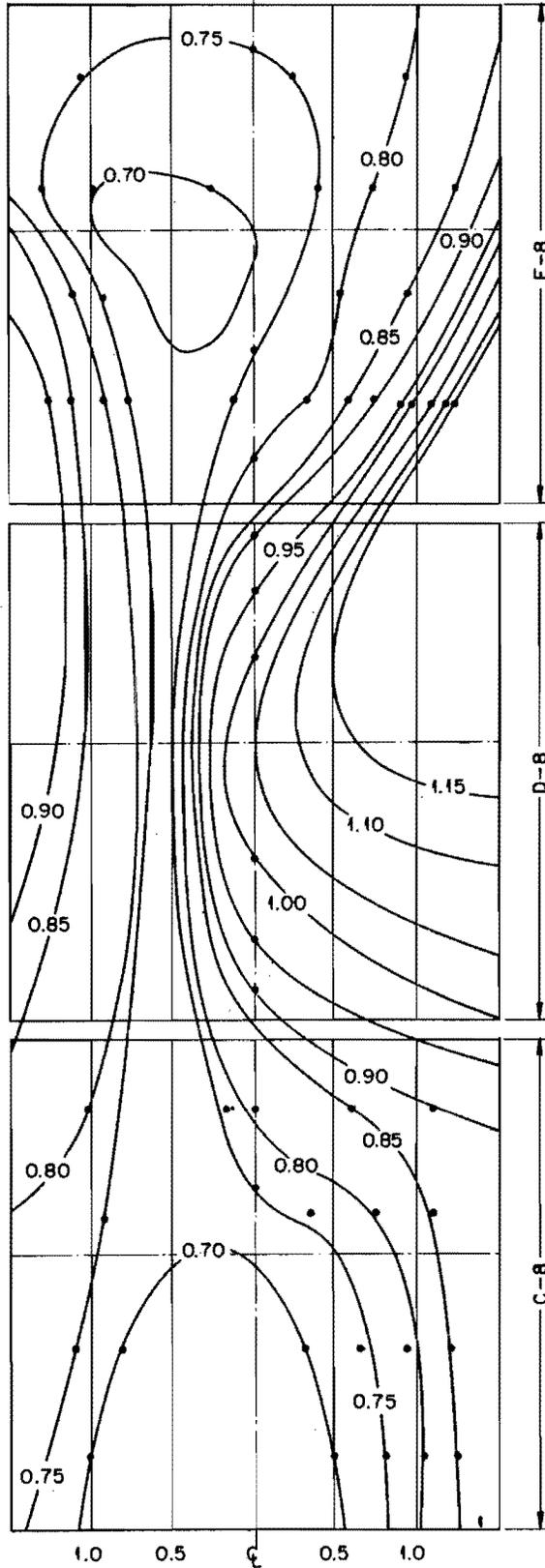
With this uncertainty in depleted-element weights in mind, the fuel cycle and loading computation methods for the LITR and the ORR will be examined.

LITR*

The Low-Intensity Testing Reactor operates at a power level of 3 Mw and has a utilization factor of 87%. This represents a power production

*The loading method described in this paragraph was developed by Mr. H. V. Klaus, LITR Supervisor.

UNCLASSIFIED
ORNL-LR-DWG 54036



DISTANCE FROM FUEL ELEMENT CENTER LINE (in.)

Fig. 2 Iso-Flux for Core Positions C-8, D-8, E-8 Normalized to Highest Measured Value in E-8.

of approximately 950 Mwd per year or an annual consumption of 1200 grams of U^{235} .

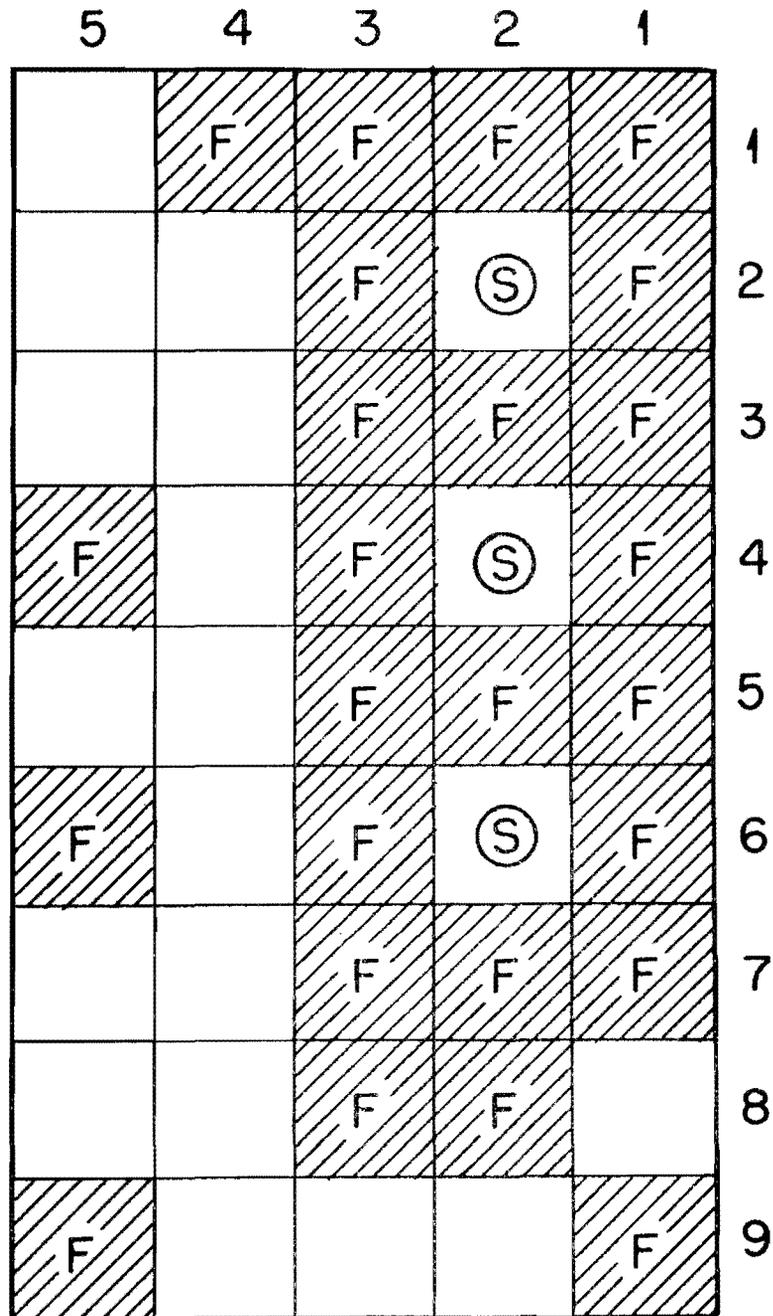
To avoid fission-product leakage in the cooling system due to cladding corrosion, the elements are not kept in the core more than two years.

The core configuration is fixed, as shown in Figure 3, by experimental requirements. In addition to these conditions the reactivity limitation (excess reactivity shall never be larger than half the total control reactivity) and, for convenience, a not-too-short cycle duration almost imposes a fuel-cycle pattern. It was experimentally found that with a cycle duration of 1 month or 100 grams burnup xenon can at all times be overridden without exceeding the reactivity limitation. Under these conditions the average burnup achieved is 33%, indicating that 18 new fuel elements each containing 200 grams of U^{235} will have to be loaded into the core every year. This is done by alternately loading one element then two and so on. The new element or two new elements are always loaded in column 2, between the shim rods; that is, in positions 2-3 and 2-5.

The elements loaded outside of row 7 and column 3 are not taken into consideration when arranging a new loading, and they are always the lowest weight elements available. The elements removed from the reactor will always come from one of those outside locations.

Once the new fuel is introduced and the depleted fuel is removed, the fuel contained in the 3 by 7 array (this fuel produces approximately 88% of the total power) is arranged so that the total weight contained in the nine elements surrounding each shim rod is the same. This insures an even reactivity balance between the shim rods. Then the fuel weight con-

UNCLASSIFIED
ORNL-LR-DWG 68340



LITR Fuel Loading Configuration.

Fig. 3.

tained in column 1 is made equal to that of column 3. The weight of fuel contained in Column 2 is fixed by the condition that the heaviest fuel elements available have to be loaded in this column.

This simple method has proved to work very well. It takes approximately two hours of a supervisor's time to compute the new fuel arrangement, and 10 to 15 elements have to be moved in the core. The amount of time required for this operation is short compared with the time required to open and close the reactor tank. The most satisfying results of this method is that the local fuel-weight variation between cycles is never larger than ± 5 grams, thus making the flux measurements and burnup computations more reliable.

The other advantages resulting from this situation are constant flux in the experiment positions and constant flux at the location of the control-system sensors. This last advantage is quite important because it minimizes the need for calibration of control-system channels.

ORR

The Oak Ridge Research Reactor operates at a power of 30 Mw; and, therefore, the xenon effect will have a strong influence on the fuel-utilization scheme.

The utilization factor of this reactor is about 80% or 8800 Mwd per year representing an annual U^{235} consumption of 11,000 grams. As usual with research reactors, the fuel arrangement is determined by the experiments installed and assumes odd shapes (Figure 4). The main effect of such a core configuration is to make an analytical approach to fuel-cycle computation almost impossible. For safety reasons, the total excess

	1	2	3	4	5	6	7	8	9
A				F	F	F			
B			F	(S)	F	(S)	F		
C			F	F	F	F	F	F	
D		F	F	(S)	F	(S)	F	F	
E		F	F	F	F	F	F	F	
F							F		
G									

ORR Fuel Loading Configuration.

Fig. 4.

reactivity loaded into the core is not allowed to exceed half the total control reactivity.

To further complicate the problem, the reactor has to be shut down, on a fixed schedule, once every four weeks to extract irradiated isotopes and to introduce new target materials. This requirement fixes the fuel-cycle duration to four weeks, with some exceptions due to the necessity of having some long shutdowns for maintenance and in-pile experiment installation. However, since the allowable excess reactivity is not sufficient to operate the reactor longer than 18 days, the fuel is re-loaded approximately every two weeks.

Because of xenon buildup after shutdown, an almost complete new loading has to be installed in the core every time the reactor is shut down. The elements composing the new loading must have had sufficient time out of the core to allow for the xenon to decay. This fact requires that a relatively large partially depleted fuel inventory be maintained. Having to handle a large inventory presents some problems. If the inventory is not watched carefully, its weight distribution will become unbalanced; and, therefore, it will become difficult to select a new loading. These conditions--a large amount of repetitive calculations and certain arithmetical difficulties--advocate strongly the use of a computer to solve this problem.

Besides accuracy and saving time, the computer offers another advantage; it impersonalizes the solution of the problem. A human being will always, consciously or not, introduce some of his past experience in the process of computing a new loading. This certainly helps to obtain a

new loading rapidly, but it has a detrimental influence on the fuel weight distribution of the inventory.

To develop a loading program, one must first assume a certain model describing the reactor. As was said before, this is very difficult to do with the existing analytical method mainly because of the complicated core shape and the numerous material compositions (fuel elements with various fuel concentrations, experiments, reflector, etc.) contained in it.

It is, therefore, much simpler to approach the problem in a very elementary manner. Assume the reactor can be described by the very general formula

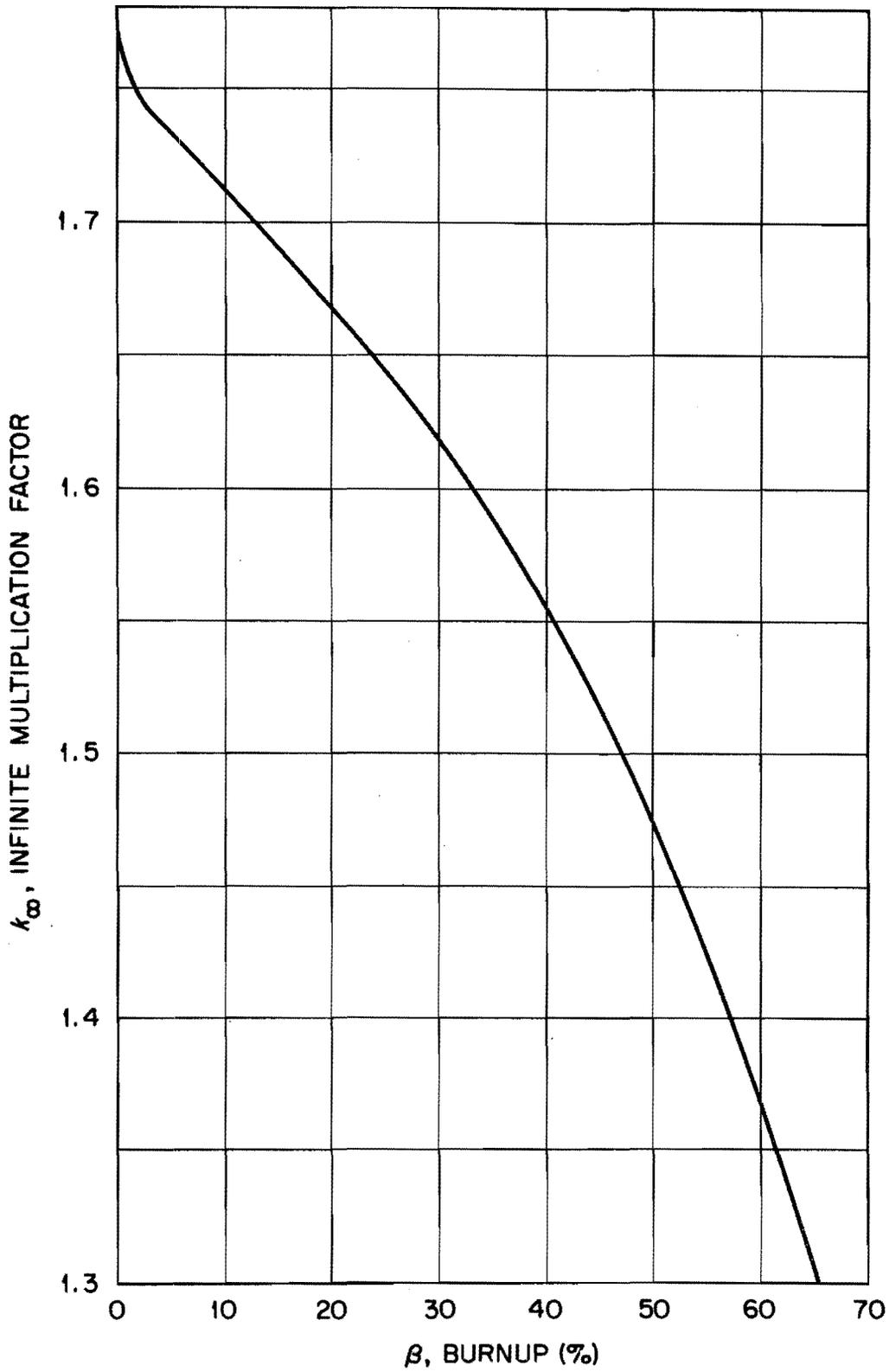
$$k_{\text{eff}} = P \bar{k}_{\infty} \quad (1)$$

where P is a nonleakage probability for the complete neutron spectrum and \bar{k}_{∞} the average k_{∞} 's of all the elements loaded into the core. This equation is correct as long as no assumptions are made concerning P .

In order not to make any assumptions concerning P , this factor was determined experimentally by using 18 operational loading critical experiments. From the known fuel elements loaded in the core, \bar{k}_{∞} could be computed. Figure 5 shows the variation of k_{∞} with burnup for ORR fuel elements containing 200 U²³⁵ when new. k_{eff} could be computed from the measured position of the control rods at critical, from the fuel contents of the rod followers, and from the rod-position factors.

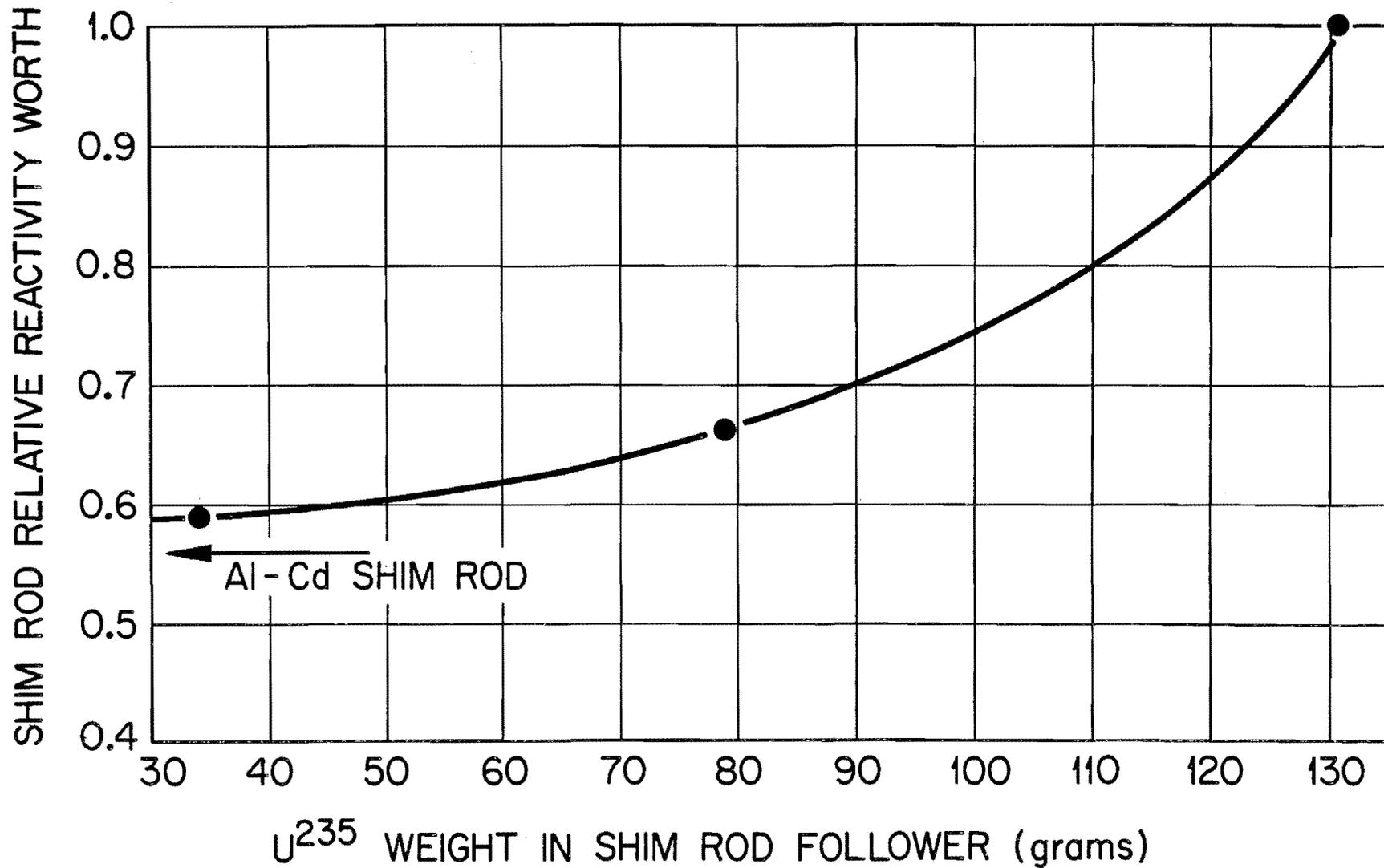
The effect of the fuel contained in the rod followers was measured by calibrating rods containing various fuel weight in their followers. These measurements were made in the same core location and with the same fuel distribution in the core. The results of these measurements are presented in Figure 6. Rod-position factors were obtained by a comparison

UNCLASSIFIED
ORNL-LR-DWG 68532



Infinite Multiplication Factor k_{∞} as a Function of Burnup.

Fig. 5.



Shim Rod Reactivity Worth as a Function of Follower U²³⁵ Contents.

Fig. 6.

method consisting of keeping the reactor critical and compensating the reactivity controlled by one rod with that of another.

Knowing k_{eff} and \bar{k}_{∞} and using equation 1, one can obtain an experimental value of P for each loading.

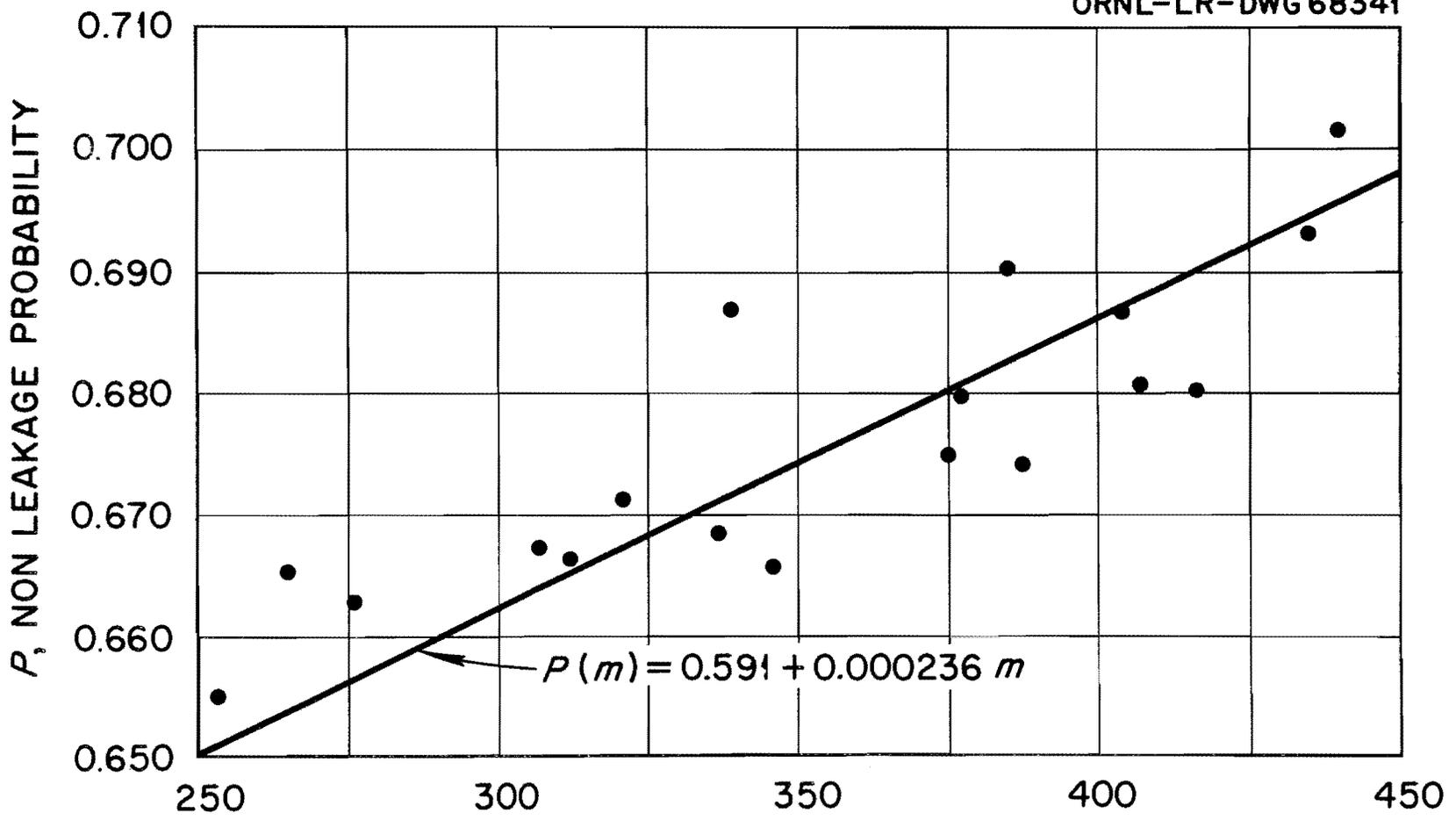
An analysis of P as a function of the total mass M of fuel in the core indicated that the function P(M) was randomly distributed around its average value. The same analysis made as a function of the mass m of fuel contained in the control-rod followers showed that P(m) was not randomly distributed around its average value but varied coherently with m. This is shown in Figure 7. A linear approximation seems satisfactory and was found by least square fit to be

$$P(m) = 0.519 + 0.000236m \quad (2)$$

with a standard deviation of 0.005 or approximately 0.7%.

Two interpretations of this effect are plausible. The first is to assume that the leakage of the core is increased when more fuel is loaded in the shim rods; the second is to assume that the rod interaction increases with the fuel weight contained in the followers or, in other words, with the amount of reactivity controlled by each individual rod.

The lack of experimental data makes it impossible to decide which of these two interpretations is the right one; and, therefore, the decision to use one or the other is arbitrary. For the sake of simplicity, the second alternative was chosen. In this way, all the variations due to the fuel-weight contents of the control-rod followers can be put on the left-hand side of equation 1 and all the variations resulting from the fuel weight in the core on the right-hand side.



The Effect of Control Rod Fuel Content on the Non Leakage Probability.

Fig. 7.

All the information collected up to now allows for the selection of the elements to be loaded into the new core, but the location of these elements is still not specified. Since the amount of time required to obtain a loading is, in some way, proportional to the degree of freedom left to the loader, a compromise should be found leaving enough freedom to the loader to take care of unexpected changes but not burdening him with an unnecessary amount of work.

To accomplish this, the core was divided in four regions and the following equation

$$K_{\text{eff}} = P \sum_{j=1}^4 \bar{k}_{\infty j} I_j \quad (3)$$

$\bar{k}_{\infty j}$ = average k_{∞} for region j

I_j = importance factor of region j

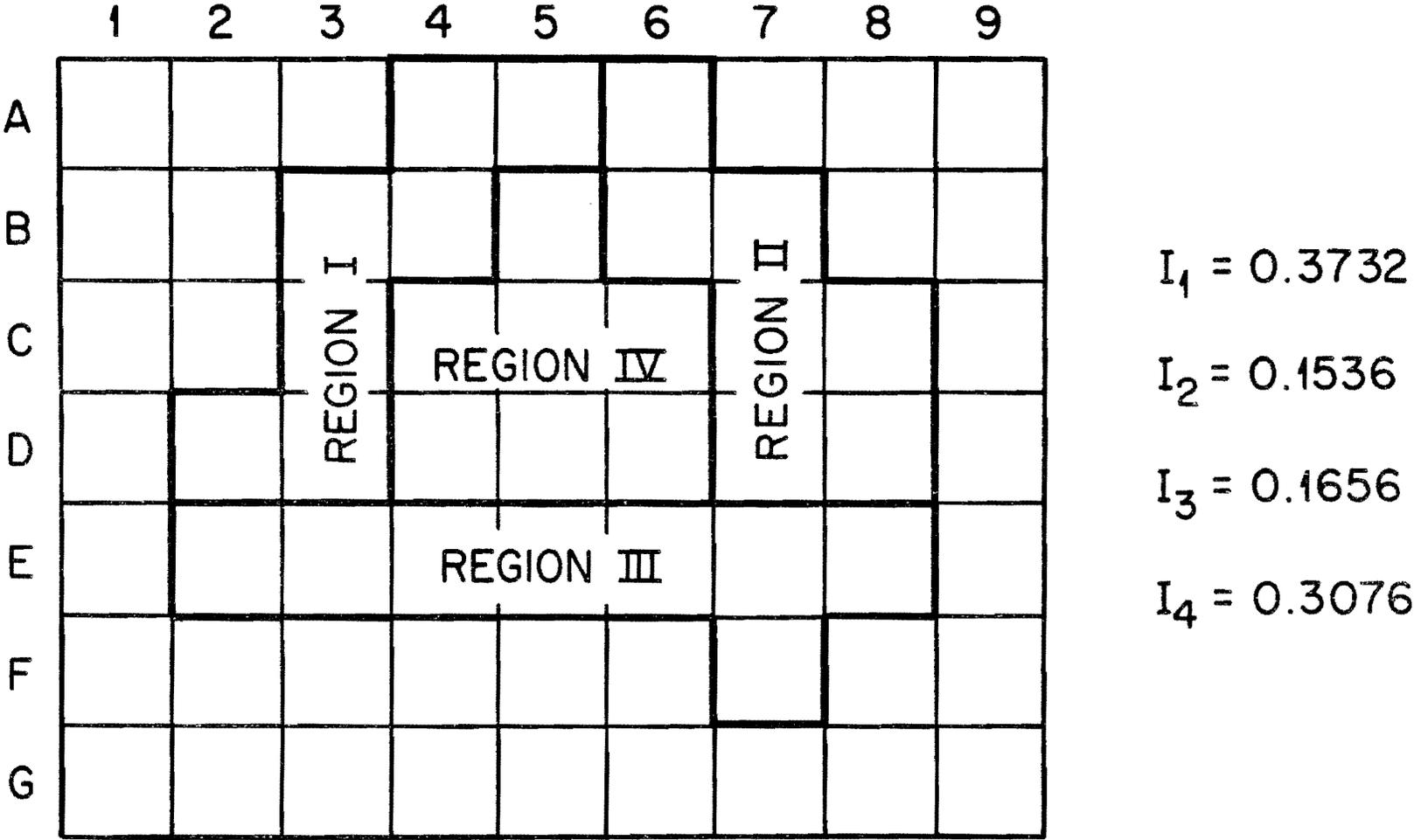
was fitted by least square to the 18 experimental loadings. Figure 8 shows the four regions and their importance factors I_j . These factors are certainly not correct according to reactor theory. They are, however, the only ones obtainable with the limited amount of experimental data available; and, as can be seen in Figure 9, the results obtained with them are quite satisfactory.

Sixty-seven percent of the 18 loadings fall between $\pm 0.5\% \Delta k/k$ of the desired value and 95% between $\pm 1.0\% \Delta k/k$, and the maximum deviation is $1.5\% \Delta k/k$.

Using the results obtained above, a loading method can now be developed.

One first selects the control rods to be loaded into the core, then computes their reactivity contents

$$\frac{\Delta k}{k} = \frac{d}{S} \sum_{i=1}^4 W_i i_i \quad (4)$$



ORR Core Importance Regions.

Fig. 8.

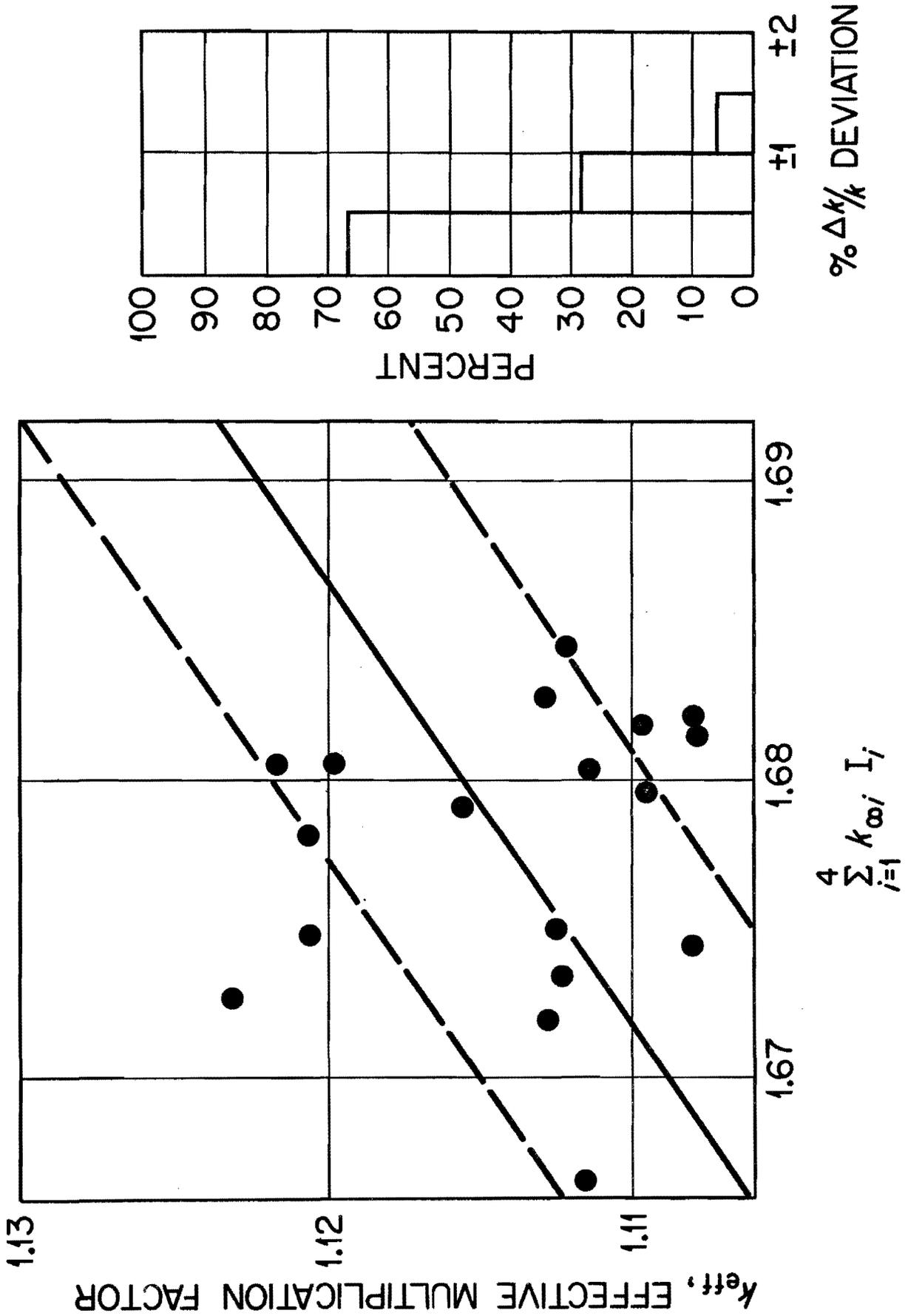


Fig. 9.

where i_i is the position factor, W_i the weight factor of the rod i . The withdrawal factor d specifies at what rod elevation the reactor should reach criticality. With the excess reactivity limitation in use for the ORR, d is equal to one-half. The interaction factor S was obtained from equation 3 and is shown in Figure 10.

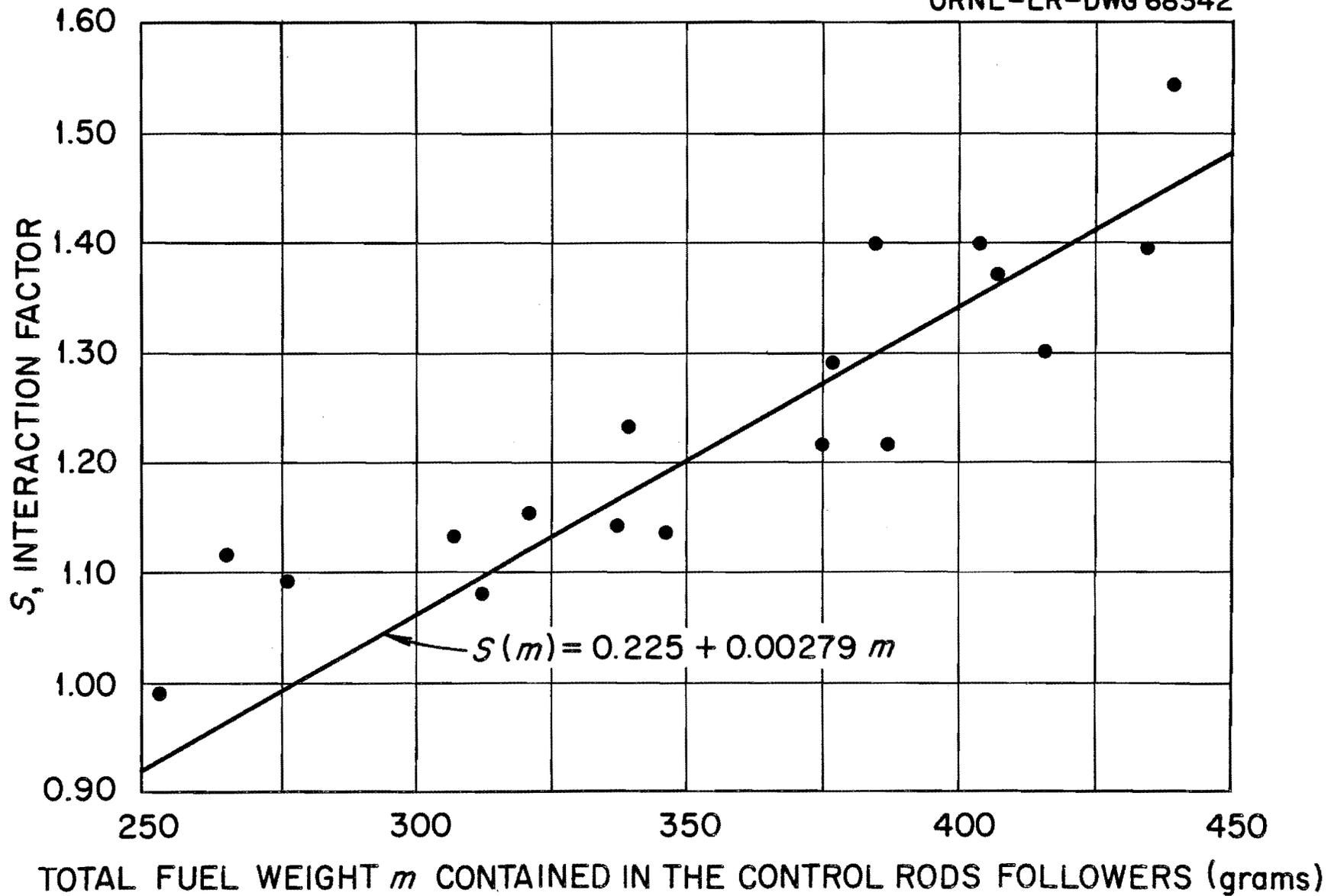
Having obtained $\Delta k/k$, this value is converted to k_{eff} and then, using equation 1, the average k_{∞} required for the core is obtained.

At this time the shape of the inventory spectrum must be considered. Twenty-five elements have to be selected. Five to six new elements, and sometimes one or two special experimental elements, have to be selected from the partially depleted fuel inventory. Some of these may be reused from the last loading, but ~ 20 elements must have been out of the reactor long enough for xenon to decay. Figure 11 shows a typical inventory spectrum.

If the number of elements to be selected is odd, the first element is chosen as close as possible to the desired \bar{k}_{∞} value. Then the distance between \bar{k}_{∞} and the highest k_{∞} available in the inventory is measured and reflected from \bar{k}_{∞} toward the lower values of k_{∞} . In this manner a portion of the spectrum having \bar{k}_{∞} as median is selected.

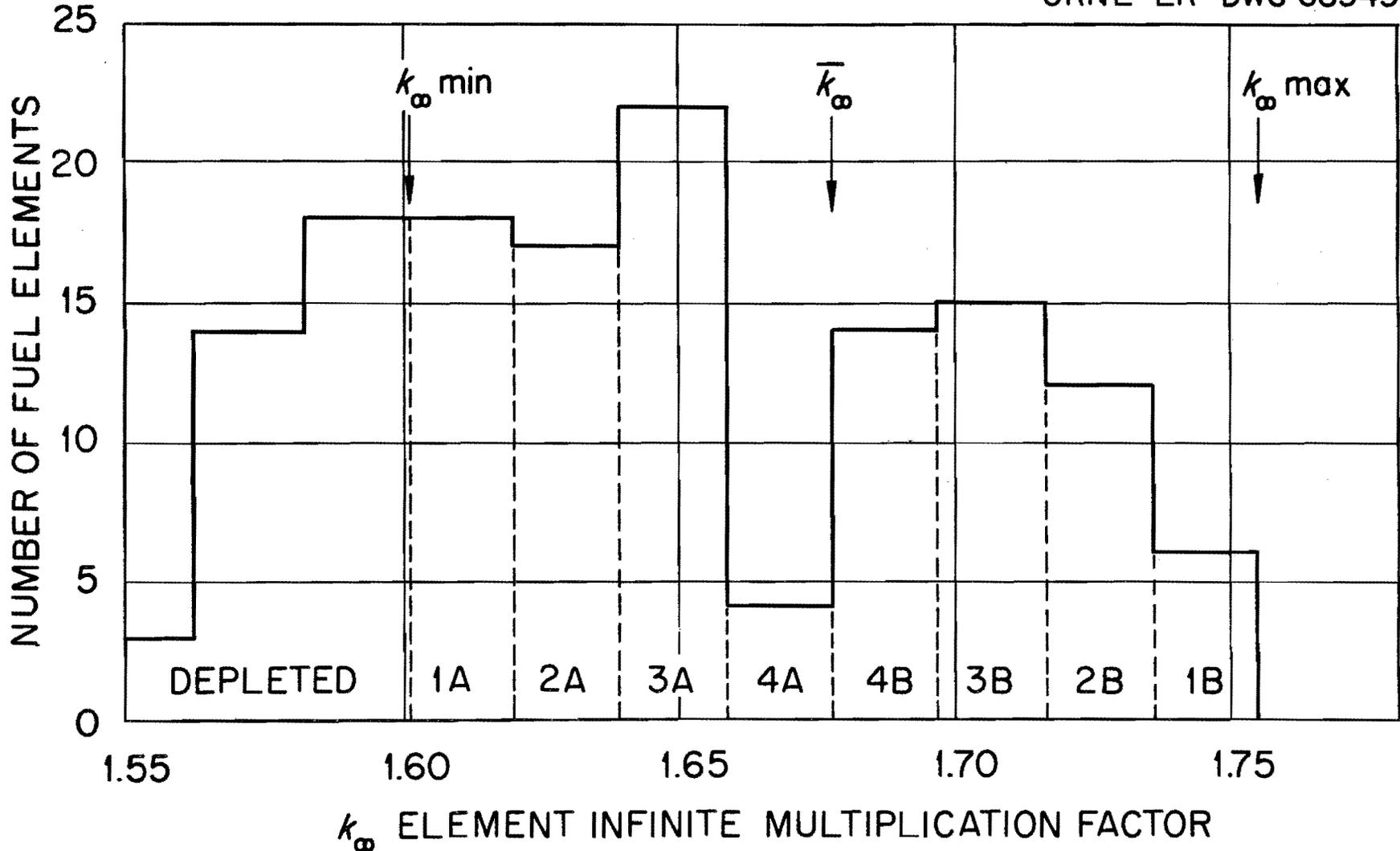
This range is then divided into 8 subranges, and these subranges are grouped in reflective (i.e., symmetric about \bar{k}_{∞}) pairs 1A - 1B, etc., as shown in Figure 11.

The selection of the elements is now made by pairing an element of one subrange with an element of the reflective subrange so that the average k_{∞} of the pair falls as close to \bar{k}_{∞} as possible.



Control Rod Interaction Factor.

Fig. 10.



k_{∞} ELEMENT INFINITE MULTIPLICATION FACTOR
Partially Depleted Fuel Elements Distribution.

Fig. 11.

This selection starts at the lower groups 1A to 4A and always chooses the oldest element in the group. This avoids having some elements remain in the inventory for a long time and keeps the oldest elements moving into the depleted inventory for eventual shipment.

Once 8, or a multiple of 8, elements have been selected, two, four, or six elements can remain. These elements will be selected from the reflective pairs that contain the greatest number of elements so as to introduce a spectrum-flattening action.

The limited number of elements available will usually cause the average of each selected pair to deviate slightly from \bar{k}_{∞} . To improve this situation, the deviation from the first selection is carried over in the second selection and so on. In this manner the accumulated error of all selected elements is only the error of the last pair. If this error is larger than a preset value, one element exchange will correct it.

Once this selection is accomplished, the elements have to be distributed in the four importance regions. The preselected elements are usually also prelocated; therefore, they are placed first at their requested locations.

The remaining elements are then distributed randomly between the 4 importance regions. Equation 3 is then solved for this first iteration and its result compared with \bar{k}_{∞} . According to the outcome of this comparison, a series of element exchanges between regions is started.

If the discrepancy between the result of Equation 3 and \bar{k}_{∞} is larger than a certain value, the exchanges will be made between a region with a small importance factor and a region with a large importance factor. When the discrepancy becomes smaller than this value, the exchanges are

made between two low or two high importance regions. Figure 12 is an example of how this converging process reaches the assigned value.

A computer program was developed for the ORACLE. This program consists of two sections. The first section computes the burnup produced during the previous cycle and keeps the total fuel inventory up to date. The second section computes the new loading from the inventory by using the method described above.

This program is now in the process of being translated into the language of a CDC 1604 computer because this type of computer will, sometime this summer, replace the ORACLE.

The two previous examples illustrate the types of problems which must be met to maintain these reactors in a balanced steady-state operation and to keep the fuel element inventory from growing too large, and from attaining impractical fuel-distribution shapes.

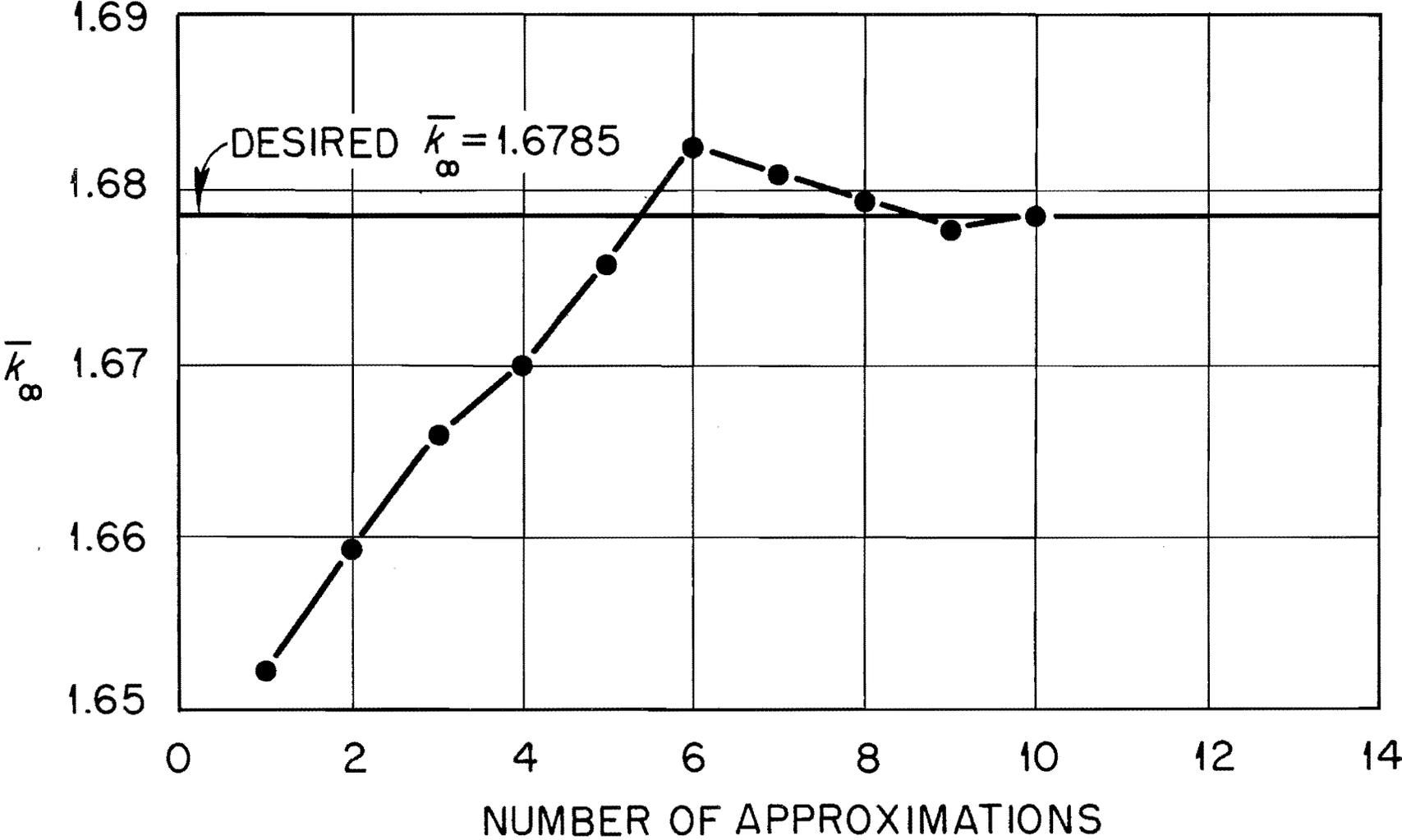
The last question to discuss is: How does one reach a steady state when starting with a new reactor.

This problem is greatly facilitated by the fact that a large amount of experimental data is available for this type of reactor. Operating teams from swimming pool reactors, MTR, LITR, ORR, and others have published a large amount of information concerning critical loadings.

Furthermore, these types of reactors can be investigated successfully using a program like PDQ³ as shown by Silver.⁴

³G. G. Bilodeau et al, PDQ an IBM-704 Code to Solve the Two-Dimensional Few-Group Neutron-Diffusion Equations, WAPD-TM-70 (1957).

⁴Neutron Physics Division Annual Progress Report for Period ending Sept. 1, 1959, ORNL-2842, p 6-15.



- 28 -

Typical Convergence of Exchange Process $\sum_{i=1}^4 k_{\infty_i} I_i \rightarrow \bar{k}_\infty$.

Fig. 12.

With such information, the problem is reduced to a compromise between two extreme solutions--one expensive, the other inconvenient.

One can determine the fuel-weight distribution to maintain the system in steady-state operation and, then, buy a series of fuel elements corresponding to this weight distribution, making some minor allowances for computation inaccuracies. This method will force the system into a steady state at the first cycle but at additional cost. Buying such a fuel inventory will certainly increase the cost, and this must be balanced against the inconvenience of operating for several cycles with an expanding core.

On the other extreme, one could buy only fuel elements containing the amount of U^{235} selected for the new elements during steady-state operation. Then, starting with small loadings, gradually increase the size of the loading until a steady state and the operational loading are reached. This method is cheap from the point of view of fuel fabrication costs but is rather complicated for loading programming.

An intermediate solution will certainly have to be sought according to the conditions prevailing for each particular case.

To conclude, it is quite interesting to observe that although the data used to compute ORR new loading are quite inaccurate, the results are very satisfactory. Not only has the fuel cycle achieved stability, but recent results obtained following the reprocessing of 245 ORR fuel elements indicate that the calculated and observed U^{235} inventory in spent fuel agree to within 0.0775%.

The neutron-flux values used for the burnup computations are certainly inaccurate; and, therefore, the fuel weights attributed to the par-

tially burned elements are uncertain. The reactivity values of the control rods are not much more reliable, especially the interaction factor.

It is quite possible that the accuracy of the results is only due to a fortunate compensation of errors. To understand these effects better, one should try, when operating at steady state, to reach a situation where the same fuel weight is always loaded at the same location.

Once this state is reached, it will be worthwhile to spend the time necessary for a good flux mapping of the core.

Then flux factors, and especially burnup factors, will have a reliable meaning.

DISTRIBUTION

- 1-200. J. A. Cox
 - 201. Document Reference Library
- 202-203. Central Research Library
- 204-206. Laboratory Records
 - 207. ORO
- 208-222. Division of Technical Information Extension

