

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

A PROPOSED FUEL ASSAY REACTOR

by

F. H. Martens and H. A. Sandmeier

Reactor Engineering Division

August 1960

Operated by The University of Chicago
under
Contract W-31-109-eng-38

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	2
INTRODUCTION	2
I. THEORETICAL CONSIDERATIONS OF A FUEL ASSAY REACTOR	4
II. EXPERIMENTAL ASPECTS	8
Pile Period Measurements	8
Pile Oscillator	8
Automatic Flux Control	9
Precision and Accuracy	10
CONCLUSION	14
ACKNOWLEDGEMENT	15

A PROPOSED FUEL ASSAY REACTOR

F. H. Martens and H. A. Sandmeier

ABSTRACT

There is a need for a special type of nuclear reactor that can test fuel elements nondestructively to determine the total amounts and the distributions of both fissionable and parasitically absorbing materials in completed, unirradiated fuel elements. Such a reactor, called the Fuel Assay Reactor, is proposed here.

Theoretical considerations are presented to show the reactivity changes that result from the insertion of both fissionable and absorbing material, into a central test hole. The choice of moderator depends upon the size of material to be tested. For testing small pellets of fissionable material, a light water-moderated assembly is advantageous because of its small critical mass; for large, completed fuel elements a graphite or heavy water-moderated system is more suitable. An internal reflector gives increased sensitivity in the center and allows enough space surrounding the test element so that spectral changes in a nonfuel region may be produced in order to discriminate between fissionable and absorbing materials.

The factors determining maximum sensitivity are discussed, as are several reactivity measurement methods suitable for testing fuel elements. The limits of accuracy are governed by the statistical variation of the neutron population within the reactor core. The theoretical limits of accuracy of all methods have the same order of magnitude, but the automatic flux level method appears to have advantages over the pile period and oscillator methods for this purpose.

INTRODUCTION

At the end of World War II, when work on the peaceful applications of the atom began, all significant amounts of fissionable material were in the hands of the governments of the U.S.A., the U.S.S.R. and Great Britain. Since then tremendous changes have taken place, especially after the first

Atoms for Peace Conference in Geneva in 1955. Now fissionable material is available to any nation that wants to build reactors, and the time is coming soon when reactor facilities will be scattered all over the world.

However, since the fabrication of fuel elements requires a tremendous capital investment, only a few nations plan to have fabrication facilities, especially during the initial phases of the atomic power development programs. The reactor installations in other nations therefore have to import their fuel elements. As the fuel technology advances and becomes less expensive, this situation may change to a great extent, but for the present, the few existing fabrication facilities must serve all reactors.

Clearly, with a free and frequent exchange of fissionable materials between nations, there must be safeguards to control and monitor the transfers of nuclear materials. For quantitative material control we must have a device to measure the fissionable material content in an assembled fuel element. It would indeed be desirable to have such a facility to measure the contents of an irradiated element, but the problems of such a facility will not be discussed here.

Some of the first reactive assemblies ever built were used to measure the absorption cross sections of fissionable and moderating materials. This early work, under the direction of Enrico Fermi, resulted in the selection of sufficient quantities of high-quality fuel and moderator to achieve the first successful nuclear chain reaction in 1942. Test reactors have been used ever since to measure the characteristics of new fuel assemblies.

Now, with modern experimental techniques and equipment that have been developed, it is possible to design a special low-power "Fuel Assay Reactor" capable of measuring very precisely the amount and distribution of both fuel and absorber in practically any type of completed fuel assembly. The basis for the design of such a reactor would be extremely valuable for several reasons. There are still certain discrepancies between experimental and theoretical results. Part of the trouble is due to the lack of accuracy in determining the content and distribution of fuel in the elements. The manufacturing of fuel elements requires many different processes, some of which cause nonuniform fuel distribution within the element. Also, neutron poisons may be introduced into a fuel element inadvertently. For example, there have been cases where the welding materials have introduced high cross section poisons into elements. Accurate postassembly assay of the fuel and absorbers within fuel assemblies would help immensely in making accurate reactivity calculations.

There are two other very important reasons for having such a reactor. One concerns economics. The transfer of responsibility from complete government control of all fissionable material to partial control by private

industries poses financial accountability problems. A system for assaying the fuel content of assembled elements would eliminate one troublesome source of discrepancies. In this sense, such a system would perform in the nuclear industry the same function as an assay office in the precious-metals trade.

The ability to make such an accurate assay may also allow the nebulous problems of international safeguards and controls to be reduced to more manageable commercial considerations.

There are a number of assemblies presently used in the U.S.A. for assaying materials used in reactors. All of them, however, are used in extremely specialized cases, such as the testing of components designed for insertion into particular production reactors. But a versatile assembly to assay accurately the amount and distribution of fuel and absorbers in a wide variety of shapes, sizes and enrichments presents a much greater problem than the construction of one of these specialized units.

I. THEORETICAL CONSIDERATIONS OF A FUEL ASSAY REACTOR

The choice of design for the Fuel Assay Reactor must be one which will provide the accuracy and versatility required to measure all the test variables: namely, how much fissionable material and how much absorbing material are contained in the test fuel element, and how the two substances are distributed throughout the element. One must first determine a configuration that will have a maximum change of reactivity due to a change in the amount of fuel in the central test position. Secondly, the design must provide enough space within the assembly so that there will be significant changes in the internal flux patterns. It is desirable to concentrate these changes in a region that does not contain fuel. These changes are necessary because there are two unknowns, fissionable and absorbing materials. Consequently, one must make two tests on each element to separate the two effects.

Because of certain limitations imposed by theoretical considerations, the testing system is based upon the comparison of stock fuel assemblies with a standard. This system requires a standard fuel element for each design that is tested. The exact composition of the standard must be known and every stock assembly must conform to the same manufacturing specifications as were used for the standard.

Let us first obtain a mathematical expression for the sensitivity of a bare cylindrical reactor, Fig. 1, to a perturbation in absorption, $\delta \Sigma_a$, and a perturbation in fission, $\delta \nu \Sigma_f$, in an axial perturbation cylinder of volume

δV_p . This reactor will respond in a very definite manner to the insertion of foreign material, either fissionable or absorbing. Two simple cases illustrate the problems that arise in designing a fuel assay reactor.

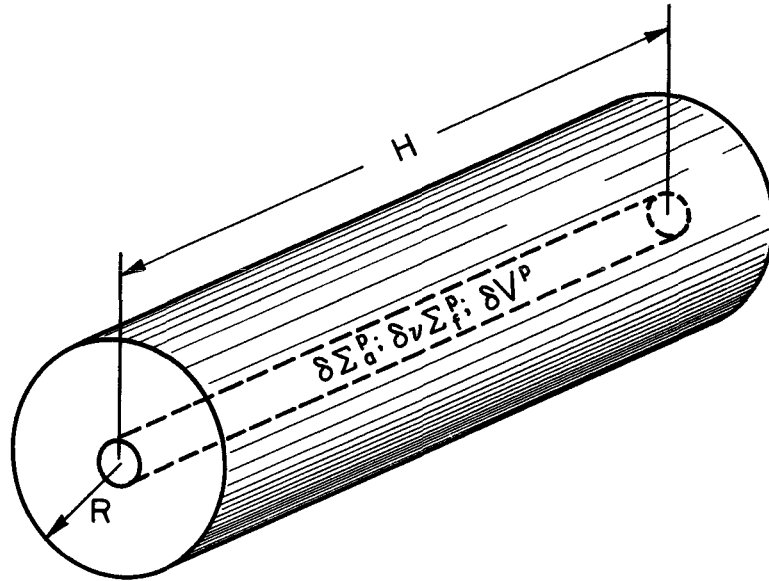


Fig. 1 Bare Cylindrical Reactor with Line Perturbation in Axial Center Line

First, consider the insertion of a purely absorbing sample into the reactor. Assume that the sample is small enough so that the flux pattern and energy spectrum are not changed. This assumption is required in order that the perturbation theory can be applied. According to this theory the change in reactivity due to a perturbation in absorption is

$$\frac{\Delta k}{k} = \frac{- \int_{\text{perturbation}} \delta \Sigma_a \Phi_{\text{th}} \Phi_{\text{th}}^* dV}{\int_{\text{reactor}} \nu \Sigma_f \Phi_{\text{th}} \Phi_{\text{th}}^* dV}, \quad \text{I-1}$$

where

$\delta \Sigma_a$ = change in absorption,

Φ_{th} = thermal flux,

Φ_{th}^* = adjoint thermal flux (importance of thermal flux),

$\nu \Sigma_f$ = fission term,

$\frac{\Delta k}{k}$ = change in reactivity.

If, for the moment, we assume that the dimensions and composition of the reactor are fixed, we can evaluate the denominator of Eq. I-1, and denote its value by a constant, $1/B$. The effect of a disturbance applied to the center of the reactor in the volume element, δV_p , as shown in Fig. 1, is then given by

$$\frac{\Delta k}{k} /_{\text{abs}} = -B \delta \Sigma_a \Phi_{\text{th}}(o) \Phi_{\text{th}}^*(o) \delta V_p \quad . \quad \text{I-2}$$

Here the index (o) refers to the center of the reactor. In an analogous way we get for a perturbation in the fission term,

$$\frac{\Delta k}{k} /_{\text{fission}} = B \delta \nu \Sigma_a \Phi_{\text{th}}(o) \Phi_{\text{f}}^*(o) \delta V_p \quad , \quad \text{I-3}$$

where $\Phi_{\text{f}}^*(o)$ is the importance function of the fast neutrons at the center position.

In the case of a sample of a fissionable isotope, Eq. I-2 and Eq. I-3 are combined to give

$$\frac{\Delta k}{k} /_{\text{fiss+abs}} = B [\delta \nu \Sigma_f \Phi_{\text{th}}(o) \Phi_{\text{f}}^*(o) - \delta \Sigma_a \Phi_{\text{th}}(o) \Phi_{\text{th}}^*(o)] \quad . \quad \text{I-4}$$

Equation I-4 is an expression in two unknowns, $\delta \Sigma_a$ and $\delta \nu \Sigma_f$, which are, respectively, measures of the amounts of absorbing and fissionable material in the sample. In order to solve this explicitly for $\delta \Sigma_a$ and $\delta \nu \Sigma_f$, two measurements must be made with the relative values of $\Phi_{\text{th}}^*(o)$, $\Phi_{\text{th}}(o)$, $\Phi_{\text{f}}(o)$, and $\Phi_{\text{f}}^*(o)$ altered. In this case we have

$$\frac{\Delta k}{k} /_1 = B_1 [\delta \nu \Sigma_f \Phi_{\text{th},1}(o) \Phi_{\text{f},1}^*(o) - \delta \Sigma_a \Phi_{\text{th},1}(o) \Phi_{\text{th},1}^*(o)]$$

and

I-5

$$\frac{\Delta k}{k} /_2 = B_2 [\delta \nu \Sigma_f \Phi_{\text{th},2}(o) \Phi_{\text{f},2}^*(o) - \delta \Sigma_a \Phi_{\text{th},2}(o) \Phi_{\text{th},2}^*(o)] \quad .$$

In order to get the maximum discrimination between $\delta \Sigma_a$ and $\delta \nu \Sigma_f$, one must alter the assembly so as to get the greatest difference between the values of the ratios $\Phi_{\text{f},1}^*(o)/\Phi_{\text{th},1}^*(o)$ and $\Phi_{\text{f},2}^*(o)/\Phi_{\text{th},2}^*(o)$. Note that this alteration will change the value of B, that is, the denominator of Eq. I-1, and the real thermal fluxes, $\Phi_{\text{th},1}(o)$ and $\Phi_{\text{th},2}(o)$.

To illustrate how one can change the importance functions, let us consider surrounding the test fuel element with a boron solution. This poison will decrease both the thermal flux and its importance. The fast neutron importance is decreased only slightly. There will also be a change in the denominator of Eq. I-1, that is, B in I-5, when the reactor is brought back to critical under the new condition.

In general, the sensitivity to either fission or absorption is increased if there is a dominant peaking of the thermal flux at the center of the reactor. This may be achieved by having an internal region which is free from fuel, but which has good moderating properties. Such an internal reflector is often called a "flux trap."

Two general questions might be: what will maximize the perturbation term for absorption (Eq. I-1), and what will maximize the fission term (Eq. I-3)? For a bare cylindrical reactor the following can be shown theoretically. The absorption term is maximized if the total absorption in the volume of the reactor is a minimum. The fission term is maximized if the critical mass of fissionable material is a minimum. The problem of achieving maximum sensitivity in the central test hole, for both uniform and nonuniform fuel distribution, is discussed in Ref. (1).

General statements concerning reactors for reactivity measurements are difficult to make. A large number of considerations govern the choice of moderator and geometry. If only very small samples are to be tested, the choice of type would probably be the light water-moderated, highly enriched uranium reactor, because criticality could be achieved with a smaller critical mass than with other types. However, if the samples are very large, the choice might be either a graphite or a heavy water-moderated assembly. Not only would the critical mass in a large reactor be smaller with either graphite or heavy water as compared to light water, but the parasitic absorption of the moderator would also be much less. Thus the sensitivity to both fission and absorption perturbations would be increased. Beryllium has not been considered because it appears that if a solid moderator should be desired, graphite would have certain advantages, especially with regard to cost.

The equations indicating the sensitivity to fission and absorption changes do not take into account the fuel distribution within the reactor. Although reactors are generally designed to have a uniform fuel distribution, it could be that nonuniform distribution may increase the sensitivity.

(1) H. A. Sandmeier, submitted to Nucl. Sci. & Eng., Feb., 1960.

II. EXPERIMENTAL ASPECTS

The success in testing fuel elements will depend greatly on the accuracy with which the reactivity is measured. Thus, the known methods for measuring reactivity and their limitations and accuracy, as related to this particular application, should be considered. At present there are three methods in general use for measuring reactivity changes in nuclear reactors. In each of these methods the reactivity changes must be kept small so that the requirements noted in the section on theory concerning small perturbations are fulfilled.

Pile Period Measurements

One method, called the "pile period" method, was historically the first method used to determine changes in reactivity. A pile is brought to criticality, then a perturbation is introduced. The reactivity is computed from the rate of increase of the neutron flux using the well-known kinetics equations.

When this method is used for very accurate measurements, the data from the flux level detectors must be fed into an electronic analogue or digital computer to obtain results with speed and precision. Any extraneous neutron source will introduce errors into the measurement.⁽²⁾ The time required for each measurement is long, since it is necessary to wait for all transients to die out after introducing the change. With proper instrumentation, relative reactivity measurements as precise as $10^{-6} \Delta k/k$ have been made by this method.

Pile Oscillator

In the second method, the "pile oscillator" technique, a sample whose reactivity effect is to be measured is alternately inserted and retracted from the core region, either sinusoidally or stepwise. The change in flux amplitude and phase is then recorded. These quantities are functions of the nuclear properties of the sample. By comparison with a standard, the cross section of the unknown is found. This method has the advantage of strongly reducing the effect of any long-term drift in the power level of the reactor. If the flux level of the reactor is increasing linearly, the effect of the increase may be eliminated by integrating the data over the proper interval.

⁽²⁾B. J. Toppel, Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements, Nucl. Sci. & Eng., Vol. 5, No. 2, Feb., 1959.

Automatic Flux Control

The third method used is the "automatic flux control" method⁽³⁾ (AFC). In this method the flux level of the reactor is kept constant by means of a servo system which regulates a lightly absorbing control rod. Any change in the reactivity of the pile will result in a change in the position of the rod. The magnitude and direction of this position change will be a measure of the magnitude and sign of the reactivity perturbation. As with the oscillator method, the cross sections are obtained by comparison with a standard. In this method the perturbation is usually inserted as a step function.

The AFC method appears to have certain advantages for the Fuel Assay Reactor. In both the oscillator and AFC methods one must replace a standard with a stock fuel element while making the measurements. The initial or zero condition of the reactor is determined with the standard fuel element in place; then the perturbation is introduced by alternating the stock and the standard elements. The oscillator method requires that data be taken continuously. A channel containing the standard and the test elements has to be moved forwards and backwards in a periodic fashion. Not only is this system mechanically complex, but it introduces large perturbations caused by the end fittings, which are usually made of aluminum or zirconium. These end effects may well overshadow the small perturbation caused by changes in fuel and poison contents in the active sections of the two elements.

In the AFC method it is not necessary to accumulate data during the sample-transfer period; in fact, often the data accumulation must be stopped during this period to allow the reactor to regain equilibrium after a change is made.

For safety, the insertion of reactivity must not exceed a certain rate. Since the rate of increase of the neutron flux is directly dependent on the rate of reactivity insertion, the test element must be inserted at a certain speed consistent with the maximum expected deviation of reactivity from the standard element. This causes no difficulties in the AFC method. In the oscillator method, however, one might get a very low-frequency response from which one may not be able to extract the pertinent information. The pile period method is very unlikely to be of advantage, since it is quite dependent on the reactor remaining extremely stable for a long period of time. A convenient schedule for the transfer of large elements would make reactor stability difficult to maintain.

⁽³⁾D. R. de Boisblanc, E. E. Burdick and G. L. Smith, The RMF Pile Oscillator, 3rd Annual Meeting of the American Nuclear Society, June 1957, Pittsburgh, Pennsylvania.

According to Bennett, the signal-to-noise ratio for any single measurement is,⁽⁴⁾

$$S/N = C \frac{\Delta\rho}{\sqrt{0.8 f}} \quad , \quad \text{II-1}$$

where

$\Delta\rho$ = reactivity change,

f = total number of fissions during testing time,

C = constant,

= $\sqrt{3/4}$ = 0.433 for pile period method,

= $\sqrt{2/\pi}$ = 0.449 for oscillator method,

= $1/2$ = 0.500 for AFC method.

Equation II-1 will show that for the same signal-to-noise ratio the AFC method is about 10 to 15 per cent more sensitive than the others. If there were good reasons for choosing one of the other methods, this advantage would really not be significant. However, with all of the other reasons set forth, we conclude that the AFC method is the best reactivity measuring technique for this type of experiment.

Precision and Accuracy

The expected precision of the reactor measurements is extremely high. There is a theoretical limit to the precision of reactivity measurements because of the statistical variation of the neutron population within the reactor core. This population is generally considered to remain constant, but in reality it varies about a mean value according to well-known statistical processes. It has been shown⁽⁵⁾ that the standard deviation, σ , in the value of the reactivity, ρ , is given by

$$\sigma = 0.89/\sqrt{f} \quad , \quad \text{II-2}$$

where f is the total number of fissions during the testing time. Since there are 3×10^{10} fissions per second in a reactor operating at a power of one watt, we may rewrite Eq. II-2 as

$$\sigma = \frac{5.1 \times 10^{-6}}{\sqrt{wt}} \quad \text{II-3}$$

(4)E. F. Bennett, Argonne National Laboratory, Private Communication.

(5)C. E. Cohn, A Simplified Theory of Pile Noise, submitted to Nucl. Sci. & Eng., January, 1960.

where w is the power in watts, and t is the testing time in seconds. From Eq. II-3 we observe that the standard deviation from the exact value of ρ will be small if either the power or the testing time is large.

As in many other engineering problems, there are restrictions placed on both variables. The reactor cannot be operated at too high a power because the amount of activity induced in the sample must be kept low. Furthermore, it is desirable to eliminate all effects of temperature upon reactivity. Most reactivity measuring facilities operate at a level of approximately 10 watts. A reasonable time for a single measurement is approximately 100 seconds.

The error of a single measurement under these conditions is obtained from Eq. II-3:

$$\sigma = \frac{5.1 \times 10^{-6}}{\sqrt{10 \times 100}} = 1.6 \times 10^{-7} \Delta k/k \quad . \quad \text{II-4}$$

Since one must make two measurements - one with the standard and one with the test element - the error or standard deviation of the difference between those two measurements is $\sqrt{2}$ times the single error, or $2.3 \times 10^{-7} \Delta k/k$.

It is not practical to try to decrease the experimental error below the statistical limit expressed in Eq. II-2. The largest error in any measurement is the root mean square of all errors. If we assume that the experimental and statistical errors total $10^{-6} \Delta k/k$, we can obtain the precision with which a measurement may be made.

To see what a total error of $10^{-6} \Delta k/k$ means in terms of the error in determining the grams of fissionable material in the test element, let us rewrite Eq. I-4 in a way that relates $\Delta k/k$ and the mass of fissionable material m :

$$\frac{\Delta k}{k} = \alpha \frac{\Delta m}{m} \quad , \quad \text{II-5}$$

where

$\frac{\Delta k}{k}$ = the change in the reactivity of the reactor when a perturbation changes the critical mass m by an amount dm ,

α = constant depending on the sensitivity to the perturbation of the test reactor. (1)

For a reactor designed to make reactivity measurements, the value of α will be of the order of unity. Using unity as the value of α and a total expected error of $10^{-6} \Delta k/k$, we obtain for the error in detectable fissionable material, Δm , from Eq. II-5.

$$\Delta m = m \frac{\Delta k}{k} = 10^{-6} m \quad . \quad \text{II-6}$$

Thus, the error is one milligram for each kilogram contained in the reactor; five milligrams for a reactor with five kilograms of critical mass. This precision will remain constant regardless of the size of the sample which is inserted. Actually, the precision will be far greater than is necessary for this type of test.

An error in the content of a standard would be a systematic error, and consequently would affect the accuracy of the test. Therefore it is necessary to obtain elements whose fuel content is known to a fraction of one per cent. This problem is difficult but not impossible. The standard must have the identical dimensions and composition as the test elements, but it need not be fabricated exactly as the test elements were. Since the standard element will not be placed in a hot liquid bath, the fuel-cladding bond is unimportant. Because there will be no heat generation, there will be no demand for good heat-transfer characteristics. No physical strain will be placed on the element, so welding may be eliminated if a simpler method of keeping the element mechanically intact can be found. It may even be possible to substitute foils of uranium and alloying metals for a fuel alloy. With these allowances it should be possible to construct excellent standards.

Other factors that tend to increase the total error are positioning variations, spurious reactivity shifts within the reactor, and instrumental errors. To exchange alternately test and standard elements of the type used in present-day reactors means shifting and positioning a heavy load in the test region. Even with a well-designed mechanical system, there will undoubtedly be some vibration and shifting of elements within the reactor core. These shifts will cause minute reactivity changes that affect the results. If the element is not placed in precisely the same position each time a change is made, there will be some error.

Spurious reactivity shifts are frequently noticed in low power reactors. The three most common causes are mechanical, temperature, and barometric effects. We have just touched briefly on mechanical effects, showing that they may be reduced, but not eliminated, by proper design. Temperature effects may likewise be reduced by proper design. The overall temperature coefficient of the system should be zero at room temperature, but it is probably not feasible to obtain a zero temperature coefficient in the core and still meet the requirements set by the theory.

However, certain schemes make it possible to approach ideal conditions. A column of a neutron-absorbing liquid may be inserted into the core. A temperature increase would expand the liquid, forcing some of the poison out of the core, thus increasing the reactivity to compensate for the natural negative coefficient. By varying the size of the column and the poison concentration, a zero temperature coefficient may be achieved. But even so, temperature changes could still have an effect on the reactivity because the temperature coefficient at all points in the reactor will be different. If, for example, the internal reflector changes temperature with respect to the other parts of the reactor, there will be a small reactivity shift. It is therefore necessary to keep the reactor at a constant temperature to eliminate these shifts as much as possible. Just as important is the requirement that the elements be brought to exactly the same temperature as the reactor, since the insertion of a warm or cold element would disturb the temperature balance of the reactor.

The barometric pressure effect will be quite serious on any graphite-moderated assembly. Because graphite absorbs large quantities of almost all gases, a change in barometric pressure will cause two effects: the first, a very short-term effect due to the change in the nitrogen content in the small cracks and crevices between blocks of graphite; the second, a longer term effect due to the release or absorption of gases by the graphite pores. These effects may be eliminated from a graphite assembly by encasing the entire assembly with an envelope into which a small flow of CO_2 is fed. Since the absorption cross section of CO_2 is several orders of magnitude less than that of the nitrogen in the air, the barometric effect would be reduced to a negligible quantity. There are no barometric effects in water-moderated reactors.

Another factor affecting the accuracy of the measurement is the self-shielding effect of the uranium in the fuel element. The uranium in the outer regions of an element absorbs thermal neutrons, leaving a smaller flux in the central region of the element. Since the reactivity effect due to a particular amount of material is directly proportional to the flux at its position, the effect of the fuel inserted in the center will be less than that of fuel in the outer regions. This self-shielding effect may be overcome to some degree by immersing the fuel element in a trough of water which will fill the regions between the plates or rods of the element. Then fast neutrons will be thermalized between the plates or rods, raising the value of the thermal flux. The flux in this case would be more uniform through the element, which is the condition that is desired.

There may also be virtue in distributing the fuel through the length of the reactor in such a way that the flux distribution along the axis of the

reactor is uniform. Goertzel⁽⁶⁾ has shown that such a flux distribution requires a lower critical mass than does uniform fuel distribution, and probably increases the sensitivity of the reactor. In addition, the effect of neutron scattering by the fuel element would be eliminated. The equations in the theoretical section of this report neglected the effect of scattering by the test element. This assumption is true only at a point of zero flux gradient. With an element that reaches almost the entire length of the reactor, such a zero gradient condition does not exist, and differences in the scattering properties would appear to be differences in the fuel or poison content. If a uniform flux distribution were achieved, these effects would be eliminated. And further, the requirements concerning the axial positioning of the fuel element would be greatly decreased.

By changing the importance functions of the thermal and fast neutrons in the central region, a discrimination can be made between the lack of fuel and the inclusion of poison. Theoretically it is possible to distinguish between several different types of poison by making several different tests with several different importance function ratios. However, in practice it will probably be possible only to distinguish between the lack of fuel and the presence of poison. In this case the inclusion of U^{238} would appear as a neutron poison and the total amount of U^{238} would not be measurable in this system. The method cannot distinguish between the various types of fissionable material, Pu^{239} , U^{235} or U^{233} . The system is only sensitive to the quantity $\nu\Sigma_f$. Consequently, it would be possible to substitute the proper amount of other types of fissionable material and balance the known fission absorption by the inclusion of other poisons. At the present time this problem appears to be purely academic, since the amounts of Pu^{239} and U^{233} available are still extremely small. At some later time this problem may become serious.

In order to determine the fuel distribution in the test element in the axial direction, a set of fast and slow neutron detectors must be placed along the length of the reactor near the test position. By reading the output of these detectors accurately, the neutron flux distribution may be determined with sufficient accuracy to tell whether there is any fuel variation along the length of the fuel element.

CONCLUSION

The concept of a Fuel Assay Reactor is most certainly feasible. There are no great theoretical problems which remain unsolved that would prevent the design of such a system. The state of the art has advanced sufficiently so that such a system could be built with a reasonable effort and

(6)Gerald Goertzel, Minimum Critical Mass and Flat Flux, J. Nucl. Energy, Vol. 2, pp. 193-201, 1956.

investment of materials. The information that could be gained with such a reactor would certainly help the atomic energy program at the present time and may be mandatory in the not too distant future.

ACKNOWLEDGEMENT

The authors are indebted to Dr. B. I. Spinrad, who originated many of the concepts presented in this paper.