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CALCULATING REACTOR POWER
FROM ACTIVATION TECHNIQUES
AS APPLIED TO AN UNUSUAL FUEL GEOMETRY (ATRC)

J. L. Durney and N. C. Kaufman

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

IDAHO NUCLEAR CORPORATION
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ABSTRACT

Two methods for determining reactor power using neutron activation techniques in a low-power reactor are developed and then applied to a particular reactor, the ATR Critical Facility. In one method, fission rate data are taken between fuel bearing plates, the values are graphically integrated, and reactor power is calculated. In the other method, the fuel region is divided into a number of subregions without regard to fuel plate position, a fission rate is measured at a point in each subregion, and these values are appropriately weighted and summed to obtain reactor power. Both methods have proved reliable, but the latter method is more convenient to use since it requires fewer fission rate detectors and allows the use of a digital computer in performing the calculation of power.

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CALCULATING REACTOR POWER FROM ACTIVATION TECHNIQUES AS APPLIED TO AN UNUSUAL FUEL GEOMETRY (ATRC)

1. INTRODUCTION

The measurement of total reactor power and of the distribution of that power within the reactor is one of the most frequently performed experiments in a critical facility. In the Advanced Test Reactor Critical Facility [1] (ATRC), which is designed to support the operation of a high-power test reactor, these measurements assume particular importance since the neutron population within the experiment irradiation positions is so closely related to the power generated in surrounding fuel elements. In addition, measurements in such reactors are frequently made to determine the extent to which changes in reactor composition have altered the extreme peak-to-average power ratios. Such ratios are, of course, important to the safe operation of the test reactor.

In a critical facility, these power distributions are usually calculated from the activation of thin wires or foils placed in selected positions within the fuel elements. At the time the ATRC began operation, it was common practice at the Test Reactor Area to use gold as a detector material for these reactor power determinations. Such measurements require cadmium ratio [2] measurements on gold and U-235, in addition to bare foil activations, in order to calculate fission rates and reactor power. Due to the variations from point-to-point in the neutron spectrum of the ATRC, a large number of cadmium ratio measurements would be required if gold were used as a detector material. This problem was essentially side-stepped, however, by using the fuel material (U-235) as a detector to obtain the fission rate directly instead of using the more roundabout gold activation technique. Two calculational methods of converting the resulting fission rate data into reactor power have been developed for the ATRC.

The first calculational method developed involves making several U-235 wire activations throughout each fuel element and then using the resulting data to construct fission rate profiles along the fuel plates. These profiles are, in turn, graphically integrated to obtain the average fission rate and, subsequently, the average fuel element power. This method has proved to be quite reliable, but the large number of activations needed requires considerable processing time to obtain reactor power.

The second calculational method developed requires that each fuel element be considered as a collection of subregions, each containing portions of several fuel plates. A U-235 wire activation is obtained in each subregion, and the resulting fission rate data are suitably weighted to obtain the average fission rate from which the average power is easily obtained. The use of this method requires an initially laborious calculational process, but the power calculations are easily adapted for use with a digital computer. The resulting decrease in processing time more than offsets the time required for formulation.

In this report both computational methods described above are derived, and their application to the serpentine fuel geometry of the ATRC is described. However, the pertinent equations are quite general, and both methods would

be applicable to any reactor fuel geometry. In particular, for geometries with significant spatial variations in the fission rate, the use of the second method can result in significant savings of process time.

2. DEVELOPMENT OF POWER CALCULATION EQUATIONS

The power generated by fissions within the fueled region of a reactor is equal to the product of the fission rate in the fuel and the energy released in each of those fissions. Mathematically, this may be described by the equation,

$$P = C \int_{\text{vol}} \int_{E_T} \Sigma_f(\vec{r}, E) \phi(\vec{r}, E) dE d\vec{r} \quad (1)$$

where P is power, C is the energy released per fission, $\Sigma_f(\vec{r}, E)$ is the macroscopic fission cross section as a function of position $\vec{r}$ and energy E , $\phi(\vec{r}, E)$ is the neutron flux also as a function of $\vec{r}$ and E , "vol" indicates the reactor volume, and E_T indicates all neutron energies present. (A list of symbols with definitions and units is given in Table I.) To determine reactor power experimentally, this integral relationship is generally expressed in terms of experimentally measured quantities; each quantity is measured; and the power is calculated. This process of reducing Equation (1) to measurable quantities is described below.

The definition of the macroscopic cross section permits a natural division of Equation (1) into integrals over space and energy. The resulting equation is

$$P = C \int_{\text{vol}} N(\vec{r}) \int_{E_T} \sigma_f(E) \phi(\vec{r}, E) dE d\vec{r} \quad (2)$$

where $N(\vec{r})$ is the atom density of fissile material at position $\vec{r}$ and $\sigma_f(E)$ is the microscopic fission cross section of that material as a function of energy.

In order to evaluate Equation (2) by use of gold foil activation techniques, the integral over the energy is generally divided into two energy intervals, E_1 and E_2 , and is expressed as

$$\begin{aligned} \int_{E_T} \sigma_f(E) \phi(\vec{r}, E) dE &= \int_{E_1} \sigma_f(E) \phi(\vec{r}, E) dE + \int_{E_2} \sigma_f(E) \phi(\vec{r}, E) dE \\ &= \left(1 + \frac{1}{B}\right) \int_{E_1} \sigma_f(E) \phi(\vec{r}, E) dE \end{aligned} \quad (3)$$

In Equation (3), the "constant" B is the ratio of the two energy integrals and will be a function of position if there are spatial variations in the neutron spectrum. The next step is to replace the integral over the region E_1 by an integral involving the gold foil absorption rate; ie, assume

$$\int_{E_1} \sigma_f(E) \phi(\vec{r}, E) dE = H \int_{E_1} \sigma_a(E) \phi(\vec{r}, E) dE \quad (4)$$

TABLE I

LIST OF SYMBOLS
(With a Consistent Set of Units)

Symbol	Description
$\Sigma_f(\vec{r}, E)$	Macroscopic fission cross section as a function of position $\vec{r}$ and energy E (cm^{-1})
$\phi(\vec{r}, E)$	Neutron flux as a function of position $\vec{r}$ and energy E ($\text{n}/\text{cm}^2\text{-sec}$)
C	The energy release per fission (watt-sec/fission)
P	Power generated (watts)
$N(\vec{r})$	The atom density of fissile material as a function of position $\vec{r}$ (atoms/ cm^3)
$\sigma_f(E)$	The microscopic fission cross section as a function energy E (cm^2)
$\sigma_a(E)$	The microscopic absorption cross section as a function of energy E (cm^2)
$M(\vec{r})$	The fission rate per atom of fissile material (fissions/sec-atom)
S	The number of fuel elements in the reactor
Q	The number of fuel plates in each fuel element
$\rho(\vec{r})$	The fissile material density as a function of position $\vec{r}$ (grams/ cm^3)
$F(\vec{r})$	The specific power (power/unit mass of fissile material) as a function of position $\vec{r}$ (watts/gram)
$\bar{F}_i$	The average specific power in the i^{th} fuel plate (watt/gram)
T_J	The total fissile material mass in the J^{th} fuel element (grams)
m_i	The total fissile material mass in the i^{th} plate of a fuel element (grams)
Δa_i	A small unit of area (cm^2)
W_{iJ}	A set of weighting factors for Method 1 calculations
C_{iJ}	A set of weighting factors for Method 2 calculations
$\bar{F}_{im}$	The average specific power at the horizontal midplane of the i^{th} fuel plate (watt/gram).
$F_i(\theta)$	The specific power at the horizontal midplane of the i^{th} fuel plate as a function of the azimuthal angle θ (watt/gram).

where $\sigma_a(E)$ is the microscopic absorption cross section for gold as a function of energy and H is a proportionality constant. With these assumptions P may be evaluated from the expression,

$$P = CH \int_{\text{vol}} \left(1 + \frac{1}{B}\right) N(\vec{r}) \int_{E_1} \sigma_a(E) \phi(\vec{r}, E) dE d\vec{r} \quad (5)$$

provided reasonable measurements or estimates can be obtained for H and B .

A more direct method of evaluating the integral over energy in Equation (2), and indeed the most desirable if B in Equation (3) is position dependent, is to use a detector of the same material as the reactor fuel itself. With such a detector the fission rate per atom of fuel, $M(\vec{r})$, can be measured where

$$M(\vec{r}) = \int_{E_T} \sigma_f(E) \phi(\vec{r}, E) dE \quad (6)$$

and, thus, all spectrum variations will be accounted for in the detector activation. (A detailed description of the use and calibration of such a detector is contained in Reference 3.) Substituting Equation (6) into Equation (2), the relation,

$$P = C \int_{\text{vol}} N(\vec{r}) M(\vec{r}) d\vec{r} \quad (7)$$

is obtained.

Two methods for evaluating Equation (7) to determine the power P from the activation of U-235 wires have been developed for the case in which $\vec{r} = (r, \theta, z)$, the usual cylindrical coordinates. These methods were developed for the curved parallel-plate fuel elements in ATRC but may be easily extended to any reactor fuel geometry. In Method 1 the fuel geometry is considered as a collection of fuel elements, and each element is, in turn, considered as a collection of plates. In Method 2 the fuel geometry is again considered as a collection of fuel elements, but each element is considered as a collection of subregions without regard to fuel plate positions.

2.1 Method 1

If the fuel region of a reactor consists of S fuel elements, each having Q fuel bearing plates, Equation (7) may be applied to each plate. Thus, the total power generated is given by the summation,

$$P = \sum_{J=1}^S P_J \quad (8)$$

with the power generated in the J^{th} fuel element, P_J , given by

$$P_J = \sum_{i=1}^Q C \int_{\text{vol}_i} N_i(\vec{r}) M_i(\vec{r}) d\vec{r} \quad (9)$$

In Equation (9), J denotes a fuel element, i denotes a particular fueled plate in that element, and vol_i , $N_i(\vec{r})$, and $M_i(\vec{r})$ denote the previously defined quantities

applied to the i th plate. Equation (9) may be recast in terms of the specific power (power per unit mass) [a], $F_i(\vec{r})$. This leads to the equation

$$P_J = \sum_{i=1}^Q \int_{\text{vol}_i} \rho_i(\vec{r}) F_i(\vec{r}) d\vec{r} \quad (10)$$

where $F_i(\vec{r}) = \frac{C N_O M_i(\vec{r})}{A_f}$, N_O is Avogadro's number, $\rho_i(\vec{r})$ is the fuel density at $\vec{r}$, and A_f is the atomic mass number of the fissile material. The units of $F_i(\vec{r})$ will be watts per gram in this report. This, of course, is not necessary and depends only upon the units chosen for C , the average energy release per fission. If the average specific power in the i th plate, $\overline{F}_i$, is defined as

$$\overline{F}_i = \frac{\int_{\text{vol}_i} \rho_i(\vec{r}) F_i(\vec{r}) d\vec{r}}{\int_{\text{vol}_i} \rho_i(\vec{r}) d\vec{r}}$$

then Equation (10) may be simplified to

$$P_J = \sum_{i=1}^Q m_i \overline{F}_i \quad (11)$$

where m_i is the mass of fuel in the i th plate.

Assuming that the fuel elements can be conveniently described by cylindrical coordinates and that each fuel plate is uniformly loaded, Equation (11) may be written

$$P_J = \frac{\sin DH}{DH} \sum_{i=1}^Q m_i k_i \overline{F}_{im} \quad (12)$$

In this equation, the average specific power, $\overline{F}_i$, has been separated into axial, radial, and azimuthal components. These components are $\frac{\sin DH}{DH}$, k_i , and $\overline{F}_{im}$, respectively, with the first two terms having no units while the third retains those units determined by the selection of the constant C . This separation of $\overline{F}_i$ is described below.

The uniform fuel loading in each plate results in a cosine distribution of the fission rate in the axial direction which is assumed the same for each fuel plate.

[a] The specific power is just the product of the specific fission rate in fissions per gram per second and the average energy in watt-sec released per fission.

As a result, the average specific power, $\overline{F}_i$, in Equation (11) above may be written

$$\overline{F}_i = \frac{\int_{-H}^H \cos Dz \, dz}{\int_{-H}^H dz} \frac{\int_{r,\theta} F_i(r,\theta) r \, dr \, d\theta}{\int_{r,\theta} r \, dr \, d\theta}$$

where $2H$ is the active core height, D is a constant related to the axial geometric buckling, and $F_i(r,\theta)$ is the specific-power distribution at $z = 0$. Assuming further that $F_i(r,\theta)$ may be separated into radial and azimuthal components (consistent with the uniform loading assumption), the above expression for $\overline{F}_i$ may be written

$$\overline{F}_i = k_i \frac{\sin DH}{DH} \frac{\int_{\theta} F_i(\theta) \, d\theta}{\int_{\theta} d\theta}$$

or

$$\overline{F}_i = k_i \frac{\sin DH}{DH} \overline{F}_{im}$$

The axial dependence of the average specific power $\overline{F}_i$ is just the average-to-peak specific-power ratio in the axial direction and is conveniently obtained experimentally. The radial component k_i is not easily obtained but, in some cases, may be estimated from fission rate depression or disadvantage factor measurements. The azimuthal dependence of the average specific power is determined by a measurement of the azimuthal fission rate at the radial and axial center of each fuel plate.

It is convenient to rewrite Equation (12) to include the total fuel mass T_J in the J^{th} element by defining a set of weighting factors, W_{iJ} , such that $W_{iJ} = \frac{m_i k_i}{T_J}$. This set of weighting factors is just the fractional fuel contained in each plate multiplied by the radial component of the average specific power for the i^{th} plate. With this definition, the fuel element power P_J becomes

$$P_J = T_J \frac{\sin DH}{DH} \sum_{i=1}^Q W_{iJ} \overline{F}_{im} \quad (13)$$

The total core power P may then be expressed as

$$P = \frac{\sin DH}{DH} \sum_{J=1}^S T_J \sum_{i=1}^Q W_{iJ} \overline{F}_{im} \quad (14)$$

Consequently, to obtain the total core power, experimental values for the quantities D , W_{iJ} , and $F_i(\theta)$ are determined, and Equation (14) is evaluated.

2.2 Method 2

In the second method for solving the power equation (Equation 7), the equation is again applied to discrete fuel elements. However, these elements will now be divided into subregions without respect to the fuel plates in contrast with Method 1 which considers the element as a collection of plates. In Method 2 the total power generated in an element may be described by the relation,

$$P_J = C \int_{\text{vol}_J} N_J(\vec{r}) M_J(\vec{r}) d\vec{r} \quad (15)$$

where $N_J(\vec{r})$ and $M_J(\vec{r})$ denote the previously defined quantities now applied to element J . As in Method 1, Equation (15) may be expressed in terms of the power per gram of fuel and the density of that fuel in order to obtain

$$P_J = \int_{\text{vol}_J} \rho_J(\vec{r}) F_J(\vec{r}) d\vec{r} \quad (16)$$

Again, as in Method 1, it is assumed that cylindrical coordinates are convenient and that the fuel elements are uniformly loaded in the axial direction. With these assumptions, Equation (16) becomes

$$P_J = \frac{2 \sin DH}{D} \int_{r,\theta} \rho'_J(r,\theta) F_J(r,\theta) r dr d\theta \quad (17)$$

where, again, $F_J(r,\theta,z) = F_J(r,\theta) \cos Dz$, the total core height is $2H$, and $\rho'_J(r,\theta) = [\text{const. multiplied by } \rho_J(r,\theta)] = \rho_J(r,\theta,z)$. With the additional definition of a quantity $\alpha_J(r,\theta) = \frac{2H}{T_J} \rho'_J(r,\theta)$ (using the definitions previously given), Equation (17) may be written in a form similar to Equation (13) in Method 1; that is,

$$P_J = T_J \frac{\sin DH}{DH} \int_{r,\theta} \alpha_J(r,\theta) F_J(r,\theta) r dr d\theta \quad (18)$$

In order to determine the power in the J^{th} fuel element P_J from Equation (18), the integral in Equation (18) must be evaluated in terms of experimental data. To evaluate this integral, it will first be noted that the fundamental theorem of the calculus enables the expression of the integral as an infinite summation of point values of the integrand throughout the midplane area of the fuel element. This summation may, in turn, be approximated by a finite summation of values of the integrand averaged over finite regions within the fuel element. Provided that the sum of these finite regions is equal to the total area of the fuel bearing portions of the element, any desired degree of approximation may be obtained by including sufficient terms in this finite sum. Then, in order to evaluate element power to within a desired accuracy, it is necessary to use experimental data to estimate the terms of this finite summation.

To obtain the desired estimation of the terms in this finite summation, it will be noted that in the limit, as the small regions approach a point, it is always

possible to find a value for the ratio of the region average of $\alpha_J(r_i, \theta_i)$ $F_J(r_i, \theta_i)$ to the value of $F_J(r_i, \theta_i)$ at a point in the region, such that the ratio is constant for all expected changes in the power distributions. If these invariant ratios can be calculated, the desired terms of the finite summation can be calculated by multiplying these ratios by measured values at the chosen point in the small region.

The process discussed above for estimating the integral of Equation (18) is described mathematically in the following paragraphs:

The integral in Equation (18) may be written as

$$I_J = \int_{r, \theta} \alpha_J(r, \theta) F_J(r, \theta) r dr d\theta = \lim_{n \rightarrow \infty} \sum_{i=1}^n \alpha_J(r_i, \theta_i) F_J(r_i, \theta_i) \Delta A_i$$

provided that $\lim_{n \rightarrow \infty} \Delta A_i = 0$. In this summation, (r_i, θ_i) is a point in the small region ΔA_i . This infinite sum may be approximated by a finite sum I'_J defined as

$$I'_J = \sum_{i=1}^{n'} \frac{\overline{\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)}}{F_J(r_i, \theta_i)} \Delta a_i$$

where $\overline{\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)}$ is the average value over a small but finite region Δa_i of the product $\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)$ at each point (r_i, θ_i) in Δa_i . ($\sum_i \Delta a_i$ = area of the fuel bearing portions of the element.) Clearly as $n' \rightarrow \infty$, $I'_J \rightarrow I_J$ provided that $\lim_{n' \rightarrow \infty} \Delta a_i = 0$; thus, for some $\epsilon_J > 0$, there exists a value n' such that $|I'_J - I_J| < \epsilon_J$. In other words, the summation I_J may be approximated to any desired accuracy by choosing n' sufficiently large.

The terms of the finite summation may be evaluated from experimental data by noting that there exist regions Δa_i for which the ratio,

$$\frac{\overline{\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)}}{F_J(r_i, \theta_i)}$$

is constant for all anticipated power distributions. This is assured since in the limit as Δa_i approaches a point, this ratio is simply $\alpha_J(r_i, \theta_i)$. When n' suitable regions have been selected for which this ratio is sufficiently constant, the terms of the finite summation may be obtained by multiplying this ratio by measured values of $F_J(r_i, \theta_i)$ and by Δa_i . The product of this ratio and Δa_i will be defined as C_{iJ} . That is,

$$C_{iJ} = \frac{\overline{\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)}}{F_J(r_i, \theta_i)} \Delta a_i$$

With this definition, the sum I'_J may be written as

$$I'_J = \sum_{i=1}^{n'} C_{iJ} F_J(r_i, \theta_i) \quad (19)$$

where $F_J(r_i, \theta_i)$ is obtained from fission rate measurements. Since n' was selected such that $|I'_J - I_J| < \epsilon_J$, Equations (19) and (18) may be combined to define

$$P'_J = T_J \frac{\sin DH}{DH} \sum_{i=1}^{n'} C_{iJ} F_J(r_i, \theta_i) \quad (20)$$

where

$$|P'_J - P_J| < \epsilon_J T_J \frac{\sin DH}{DH} = \epsilon'_J.$$

Equation (20) may now be used to calculate the total reactor power as

$$P = \frac{\sin DH}{DH} \sum_{J=1}^S T_J \sum_{i=1}^{n'} C_{iJ} F_J(r_i, \theta_i) \quad (21)$$

This equation is similar to Equation (14) in Method 1. However, it is quite useful in that it requires only point values of the specific power since all averaging is contained in the constants C_{iJ} .

The use of Equation (21) requires the selection of the n' regions and the calculation of the C_{iJ} . To summarize, the selection of the regions Δa_i is governed by three factors:

- (1) $|P'_J - P_J| < \epsilon'_J$
- (2) $\sum_i \Delta a_i$ = the total area of the fuel bearing portion of the fuel element
- (3) The ratio $\frac{\alpha_J(r_i, \theta_i) F_J(r_i, \theta_i)}{F_J(r_i, \theta_i)}$ must be sufficiently invariant during all anticipated changes in the power distribution.

A rather important observation should be made with regard to the restrictions imposed on Δa_i . That is, the regions Δa_i are not restricted to single plates or whole plates but may include portions of one or several plates. Additionally, once values for the C_{iJ} have been determined, the resultant power equations are actually independent of the coordinate system selected. As a consequence of these factors, this method for power determination is easily extended to all geometries. Furthermore, since the calculational procedures are merely sums and products of measured values, this method offers the advantage of rapid data processing; and as will be discussed subsequently, this advantage has proved to outweigh the initially laborious calculation process required to determine values for C_{iJ} .

3. APPLICATION OF POWER CALCULATIONS TO ATRC

The core of the ATRC consists of 40 fuel elements arranged in a serpentine fashion to form flux traps as shown in Figure 1. Each element forms a 45-degree sector of a right circular cylinder and, as shown in Figure 2, consists

FIG. 1 CROSS-SECTIONAL DIAGRAM OF THE ATRC CORE SHOWING THE ARRANGEMENT OF THE CURVED-PLATE FUEL ELEMENTS.

of 19 fuel-bearing plates with coolant channels between each plate. The fully-enriched uranium fuel is contained in a thin meat section within each plate. This meat is a dispersion of U_3O_8 in X 8001 aluminum with a uniformly

Wire No.	Inches from C	Wire No.	Inches from C
1	1.79	13	0.08
2	0.94	14	0.69
3	0.08	15	1.29
4	0.94	16	1.10
5	1.79	17	0.59
6	1.57	18	0.08
7	0.83	19	0.59
8	0.08	20	1.10
9	0.83	21	0.85
10	1.57	22	0.08
11	1.29	23	0.85
12	0.69		

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FIG. 2 CROSS-SECTIONAL DIAGRAM OF THE ATR-ATRC FUEL ELEMENT SHOWING POSITIONS OF WIRES USED TO DETERMINE THE AVERAGE SPECIFIC POWER IN METHOD 1.

dispersed burnable poison in the form of B_4C and is clad with Type 6061-O aluminum. Each element nominally contains 975 grams of U-235 and 4.15 grams of natural boron.

The design of the reactor permits power differences from one flux trap experiment region to another. This capacity as well as the thin annular geometry of the fuel region results in rather large differences in the fission rate contours within the fuel elements. When these fission rate contours vary, it has been established that the ratio of the epithermal and thermal components of the fission rate also vary. This behavior has led to the use of fission rate detectors instead of thermal neutron flux detectors whenever reactor power is measured. Other measurements with gold and dysprosium proved unsuitable since this shift in the epithermal-to-thermal ratio required an additional cadmium ratio measurement at each point.

Both power calibration methods discussed in Section 2 require the use of a number of fission rate monitors at reactor midplane in order that a power may be calculated. The standard detector used is a 0.25-inch-long, 40-mil wire containing about 10 weight percent U-235 in aluminum. These wires are attached to acrylic plastic strips for positioning and are inserted into the coolant channels of the fuel elements.

As a result of several measurements using 10-mil U-235 wires both within and without the fuel plates, it has been determined that there is apparently no fission rate depression in the fuel plates [4]. Therefore, the constant, k_1 , in W_{iJ} [Equation (13)] is taken to be unity in ATRC, and specific power values obtained in the water channels are applied to adjacent fuel plates as shown in Section 3.1 below.

With the uranium wires inserted in the core, the reactor is operated for 20 minutes at approximately 300 watts. The wires are removed after a waiting time of about 2 hours, and the induced activity is then measured by use of the automatic beta counters [5] in the MTR counting room. The reactor power has been chosen to provide sufficient activation to maintain statistical counting errors less than 1 percent. The counting data are then converted to specific powers using constants [3] obtained in a standard pile and thermal column.

The maintenance of an axially uniform fission rate distribution was a major design feature of the ATR and ATRC. This axial distribution has been measured in several positions under various power distributions. The measured axial fission rate distributions have in every case proved to have a near-cosine shape with a peak-to-average ratio of 1.42, the peak occurring at the reactor horizontal midplane,

In the ATRC, the value used for C , the energy release per fission, is 197 MeV per fission, or 3.156×10^{-11} watt-sec per fission. This value is the same as that determined for the Materials Testing Reactor [6] and, as shown in Appendix A, also is applicable to the ATRC. Using the value 1.42 for the peak-to-average ratio of the axial fission rate distribution, one can write the power equations [Equations (14) and (21)] as

$$P = 0.704 \sum_{J=1}^S T_J \sum_{i=1}^Q W_{iJ} \overline{F_{im}} \quad (22)$$

for Method 1 and

$$P = 0.704 \sum_{J=1}^S T_J \sum_{i=1}^{n'} C_{iJ} F_J(r_i, \theta_i) \quad (23)$$

for Method 2. When T_J is expressed in grams and the specific power is expressed in watts per gram, the power is in watts. The application of each of these two equations is described below.

3.1 Application of Method 1

When using Method 1 to determine reactor power, 23 standard U-Al wires are irradiated at the horizontal midplane in the water channels of each fuel element at the positions shown in Figure 2. The activity of each of these fission rate detectors is converted to watts per gram U-235 (specific power) to obtain values of $F(R_i\theta)$ where R_i denotes the mean radius of the i^{th} water channel. (The specific powers are now expressed as a function of the arc length, $R_i\theta$, for convenience.) These specific powers are then plotted as a function of $R_i\theta$ for each of five water channels; and as shown in a typical case in Figure 3, a smooth curve fitted to the points. The average specific power for each of these water channels is obtained by integrating the appropriate curve. These average values are then used to construct a radial profile of the average specific power in the plates of the fuel element. One such profile is shown in Figure 4. From this profile, the average specific power in each of the fuel plates, F_{im} , may be read for all 19 fuel plates. A typical listing of these values is shown in Table II. When the values of F_{im} have thus been obtained, it is only necessary to multiply by the appropriate weighting factor, W_{iJ} , also listed in Table II, and sum the resulting products to obtain the average specific power at the horizontal midplane. In order to complete the power calculation, it is necessary to multiply by $T_J = 975$ grams (U-235 mass per fuel element) and 0.704 (the numerical value of $\frac{\sin DH}{DH}$ in the ATRC). Based on the uncertainty of the wire positioning as well as wire grouping and counting uncertainties, the estimated standard deviation for the fuel element power calculated by this method is ± 5 percent.

3.2 Application of Method 2

In contrast to Method 1 where the coefficients W_{iJ} were comparatively easy to obtain, the determination of the values for C_{iJ} , n' , and Δa_i in Equation (21) required an initial experimental program once an acceptable error criterion ϵ'_J had been established. Values for P_J were determined using Method 1, and an error criterion of ± 2 percent of the value of P_J was considered acceptable. Thus, $|P'_J - P_J| < \epsilon'_J = 0.02 P_J$. Data obtained in two different power distributions were then analyzed to establish values for n' , Δa_i , and C_{iJ} consistent with this criterion.

To limit the complete arbitrariness of these parameters as well as to shorten the selection process, some restrictions were placed on what would be acceptable values for C_{iJ} . It was decided that, if possible, no more than ten different sets of C_{iJ} would be used and, furthermore, that five of these sets were to be mirror images of the other five sets. In particular, with reference to Figure 1, it was decided that the ten sets for Fuel Elements 1 through 10 would be the same as the ten sets for Elements 11 through 20, Elements 21 through 30, and Elements 31 through 40. Furthermore, the five sets for Elements 1 through 5 were to be mirror images about a NE-SW axis of the five sets for Elements 6 through 10. As an illustration of this reflection, using the core diagram in Figure 1 and the wire positions shown in Figure 5, the value for Wire Position 1 in Element 5, or C_{15} , is the same as that for Position 5 in Element 6 or C_{56} , $C_{25} = C_{46}$, $C_{35} = C_{36}$, $C_{45} = C_{26}$, $C_{55} = C_{16}$, $C_{65} = C_{86}$, and so on.

In determining C_{iJ} , n' , and Δa_i , the regions Δa_i were first selected as equal flux-weighted areas surrounding each of the 23 positions used in a calculation with Method 1. Values of C_{iJ} for each region were calculated;

FIG. 3 TYPICAL GRAPHS OF THE MIDPLANE SPECIFIC POWERS IN AN ATRC FUEL ELEMENT.

FIG. 4 TYPICAL AVERAGE SPECIFIC POWER RADIAL PROFILE AT MIDPLANE IN AN ATRC FUEL ELEMENT.

a value for P'_J was calculated; and the difference $|P'_J - P_J|$ was calculated. All values of C_{iJ} and Δa_i were then adjusted in an attempt to decrease n' , the number of regions or positions monitored. After several trials, it was found that n' could be reduced to 17 using values of $F_J(r_i, \theta_i)$ measured at the positions shown in Figure 5. The resulting 10 sets of C_{iJ} , found to remain constant for the two power distributions analyzed, are listed in Table III.

The power calculation is then completed by performing the indicated multiplications and additions in Equation (21) using the C_{iJ} values in Table III, $n' = 17$ and values for $F_J(r_i, \theta_i)$ measured at the positions shown in Figure 5. Recently, a new series of measurements was completed in which the power per element was calculated using both Method 1 and 2. For these data, one standard deviation of the differences in individual fuel element powers computed with Method 1 and 2 was found to be 0.8 percent. The total core power calculated by the two methods differed by 0.5 percent, and the maximum difference in power for a single fuel element was 1.8 percent.

3.3 Computer Calculation

As discussed above, ATRC power measurements require a relatively large number of fission rate measurements. Efficiency dictates the use of a digital computer to perform the required data reduction. Method 2 lends itself very well to this task, because once the values of C_{iJ} have been determined, the power calculation is simply a sum of products of the input specific power data. On the other hand, Method 1 requires an averaging process to be performed

TABLE II

TYPICAL ATRC LISTING FOR METHOD 1 POWER CALCULATION

Plate Number	Weighting Factor W_{iJ} (units of 10^{-2})	Average Specific Power, F_{im} (10^{-2} watts/gram)	Weighted Specific Power, $W_{iJ} F_{im}$ (10^{-4} watts/gram)
1	3.10	2.30	7.13
2	3.72	2.09	7.77
3	3.92	1.89	7.41
4	4.11	1.73	7.11
5	4.31	1.59	6.85
6	4.51	1.48	6.67
7	4.71	1.42	6.69
8	4.91	1.37	6.73
9	5.11	1.34	6.85
10	5.30	1.33	7.05
11	5.50	1.29	7.10
12	5.70	1.27	7.24
13	5.90	1.26	7.43
14	6.10	1.25	7.62
15	6.29	1.25	7.86
16	6.49	1.27	8.24
17	6.69	1.30	8.70
18	6.89	1.37	9.44
19	6.73	1.59	10.70
			144.59 Total

Calculation of P_J is completed as follows:

$$P_J = 0.704 \times 975 \text{ g} \times 144.59 \times 10^{-4} \frac{\text{watt}}{\text{g}} = 9.92 \text{ watt.}$$

on the specific power data. A curve fitting by computer could be performed, but due to the small number of points per curve presently obtained, the fitting would be subject to large errors.

A computer program to calculate power using Method 2 has been written. This program, PPCo. 40.0523, accepts as input the punched paper tape output of the automatic beta counters [5], associates the measured values with core positions, and calculates fission rates, specific powers, and core power. It also calculates individual fuel element and regional power generation. The details of the several output options, input requirements, and a complete

Wire No.	Inches from C	Wire No.	Inches from C
1	1.79	10	0.08
2	0.94	11	1.29
3	0.08	12	0.59
4	0.94	13	0.08
5	1.79	14	0.59
6	0.83	15	0.85
7	0.08	16	0.08
8	0.83	17	0.85
9	1.29		

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FIG. 5 CROSS-SECTIONAL DIAGRAM OF ATR-ATRC FUEL ELEMENT SHOWING POSITIONS OF WIRES USED TO DETERMINE THE AVERAGE SPECIFIC POWER IN METHOD 2.

program description and listing are provided in Appendix B. The program itself is written in FORTRAN IV for use on the IBM-7040 computer. At present, there are still more comparisons to be made to determine appropriate values of C_{ij} , but when a final set is obtained, the computer program will be converted to FORTRAN II for use on the CDC-636 computer associated with the ATR. Use of this computer will further shorten the time required to reduce the data because of its proximity to the beta counters.

TABLE III

LISTING OF THE METHOD 2 ATRC WEIGHTING FACTORS
 C_{iJ} IN INTEGER MULTIPLES^[a] OF $(145.489)^{-1}$

<u>J</u>	<u>i^[b]</u>																
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>	<u>11</u>	<u>12</u>	<u>13</u>	<u>14</u>	<u>15</u>	<u>16</u>	<u>17</u>
1	1	10	6	4	2	13	14	9	12	18	8	13	14	9	2	11	1
2	2	7	9	11	4	9	13	10	9	23	3	13	15	10	1	10	1
3	2	12	8	3	1	13	10	9	10	34	3	10	12	8	2	12	1
4	2	6	4	3	1	13	13	8	8	33	4	15	15	11	2	10	2
5	2	3	3	10	1	12	12	12	6	24	8	10	16	15	2	12	2
6	1	10	3	3	2	12	12	12	8	24	6	15	16	10	2	12	2
7	1	3	4	6	2	8	13	13	4	33	8	11	15	15	2	10	2
8	1	3	8	12	2	9	10	13	3	34	10	8	12	10	1	12	2
9	4	11	9	7	2	10	13	9	3	23	9	10	15	13	1	10	1
10	2	4	6	10	1	9	14	13	8	18	12	9	14	13	1	11	2

[a] The multiplier is simply the least common denominator.

[b] i denotes a wire position within Element J.

4. CONCLUSIONS

The unusual design of the core of the ATRC has led to the need for fission rate measurements at many positions in the fuel region. Two methods have been developed and applied to this reactor for the purpose of calculating power from these measured values. Method 1 requires 23 monitors per fuel element, and all coefficients in the power calculations are easily related to the physical constants of the fuel elements. Method 2 requires only 17 monitors per fuel element, but requires an initial experimental program to determine coefficients for the power calculation. Since power distributions to date have been symmetrical about the center of the core, only ten elements are usually monitored to obtain total core power. The resulting reduction in the number of monitors from 230 to 170 to monitor this quarter core has proved to be highly desirable. Method 2 also has the desirable feature of being easily adapted for use on a high-speed computer.

In the near future, power calculations will be made by both methods until the parameters used in Method 2 are established as adequate for all anticipated power distributions. After that, Method 2 will be used because it requires fewer measurements, and the power calculations can be performed by a computer.

5. REFERENCES

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APPENDIX A
ENERGY RELEASE PER FISSION IN ATRC

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APPENDIX A -- ENERGY RELEASE PER FISSION IN ATRC

Since the ATR, the ETR, and the MTR all have generally the same type of fuel elements, reflectors, and moderators, it is assumed that the energy release per fission is also essentially the same in all three reactors. Due to this similarity then, the energy release per fission in the ATR (and, hence, the ATRC) is obtained below from earlier calculations for the MTR [6].

On the basis of one-group theory, the number of neutrons per fission that leak from the fuel region is given by

$$L = \frac{\nu \Sigma_f - \Sigma_a}{\Sigma_f}$$

where L is the net neutron leakage, ν is the number of neutrons released per fission, Σ_f is the average macroscopic fission cross section, and Σ_a is the average macroscopic absorption cross section of the fuel annulus. Evaluating this expression for ATRC, one obtains 1.04 for the net neutron leakage. Since the number of neutrons produced per fission is 2.44 (neglecting any photo-neutrons produced in the Be reflector), there are 1.40 neutrons per fission absorbed in the fuel and 1.04 neutrons per fission absorbed in the reflector. The history of the neutrons is shown in the following tabulation.

NEUTRON HISTORY (per fission)

Event	Number of Neutrons	
	MTR	ATRC
Leakage from the fuel	0.95	1.04
Absorptions in fuel resulting in fission	1.00	1.00
Absorptions in fuel not resulting in fission	0.52	0.40
TOTAL	2.47 ^[a]	2.44

[a] The accepted value for ν was 2.47 when the MTR estimate was made. This difference has little effect on ATRC results.

The MTR calculations estimate that the $\text{Be}(n-2n)$ reaction contributes 0.08 neutrons per fission which are absorbed in the reflector. The number of neutrons per fission absorbed in the MTR reflector is $0.95 + 0.08$ or 1.03 compared to 1.12 for the ATRC (assuming the same reflector contribution in the ATRC). The energy release per fission in the ATRC reflector then is

$\frac{1.12}{1.03}$ times the energy release in the MTR reflector. In the fuel region of the ATRC, the energy release is $\frac{0.40}{0.52}$ times that in the MTR. These energy releases are summarized below.

GENERATION OF HEAT ENERGY
(per fission)

Event	MTR	ATRC
Energy release in the fuel from non-fission absorptions	2.81 MeV	2.16 MeV
Energy release in the fuel from fission absorptions	190.0	190.0
Energy release from absorptions in the reflector	<u>4.73</u>	<u>5.14</u>
TOTAL	197.5 MeV	197.3 MeV

Since the uncertainty attached to the MTR calculation is ± 6 MeV, the two energy releases are essentially the same in this approximation.

APPENDIX B

A DESCRIPTION OF ATRC FLUX MAP: A COMPUTER PROGRAM TO ANALYZE ATRC FLUX DATA

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APPENDIX B -- A DESCRIPTION OF ATRC FLUX MAP:
A COMPUTER PROGRAM TO ANALYZE ATRC FLUX DATA

1. INTRODUCTION

As discussed in the text of this report, a computer program, PPCo. 40.0523, has been written to calculate the power generated in ATRC fuel elements from neutron activation data. This program, written in FORTRAN IV for use on the IBM-7040, basically converts count data from the automatic beta counters to fission rates, arranges and associates these data with fuel element positions, and calculates fuel element power using the equations developed as Method 2 in this report. This program is called ATRC Flux Map.

The use of this program requires that the following operational steps be completed starting with the activation of U-235 wires in ATRC at some power, usually 300 watts. The reactor is operated at this power for a precisely timed interval, usually 20.0 minutes, which begins when the reactor power reaches $1/e$ times the intended operating power [a]. This interval, in seconds, is designated TEXP. When the interval has been completed, the reactor is shut down, and a stop watch is started. After about 2 hours, the plastic wands holding the U-235 wires are removed from the core. Each wand is identified as to the number of the fuel element position from which it was removed, and each U-235 wire is identified as to its position within that element using the position numbering system illustrated in Figure 5.

When the plastic wands have been unloaded, the U-235 wires must be transferred to numbered plastic planchets or chips for automatic counting. All wands corresponding to any one element are assembled and unloaded as described below. The wire in Position 1 (see Figure 5) is placed in Chip 1, the wire in Position 2 is placed in Chip 2, and so on until the wire in Position 17 has been placed in Chip 17. A second set of wands for some other element may then be assembled and similarly sequentially unloaded into Chips 18 through 35. The process is then continued until all U-235 wires have been transferred to the numbered chips.

The stacks of chips containing wires are then transferred to the MTR beta counter which has three counting channels. The stacks of chips, numbered sequentially from bottom to top, are divided between two of the channels. For example, Chips 1 through 75 may be placed on Channel 2, and Chips 76 through 150 may be placed on Channel 1. Chips containing activated wires not pertinent to the power calculation may then be added to the top of the stack in either channel. One irradiated wire not pertinent to the power calculation must be placed in Channel 3 to provide an automatic decay correction.

The beta counters are then started; and, using the stop watch started at reactor shutdown, the elapsed time from shutdown to the start of the counter is recorded. This time, in hours, is called TWAIT. The beta counter measures the activity of the U-235 wires and automatically provides a decay correction

[a] Beginning the timing interval before reaching full power accounts for activations of the U-235 wires during reactor startup.

by determining the counting time for each sample from a fixed count on the sample in Channel 3. As each sample is counted, the following information pertinent to the power calculation is output on punched tape: the elapsed time since the counter was started, the duration of the counting interval, and the counts recorded during that interval in Channels 1, 2, and 3 (of course, the counts recorded in Channel 3 are always the same). The information on this tape is then transferred to computer cards which have the format described on page 31. The computer processing of these data may then begin.

2. GENERAL DESCRIPTION OF CALCULATIONS PERFORMED

The first stage of ATRC FLUX MAP is a calculation of the fission rate, specific power, fission flux, and thermal flux (if a cadmium ratio, CR, has been supplied). This stage is a modified form of a program written by G. A. Cazier of Phillips Petroleum Company for use by the MTR counting room personnel. Each of these parameters is calculated as shown below, and each is calculated for all samples regardless of whether or not they are to be used in the power calculations.

(1) Fission Rate (fissions/sec/gram U-235):

$$F. R. = K^n \left\{ \left(\frac{C^n}{TCOUNT} \right) \left[1.8 \times 10^{-6} \left(\frac{C^n}{TCOUNT} \right) + 1 \right]^{-6.2} \right\} \frac{1200}{TEXP},$$

where C_n is the number of counts in Channel n, TCOUNT is the count interval for the first sample counted, and K^n is a calibration constant for Channel n based upon TWAIT. The derivation of this equation and the determination of K^n are described in Reference 3.

(2) Specific Power (watts/gram U-235):

$$S.P. = (F.R.) \times 3.156 \times 10^{-11}$$

(3) Fission Flux (neutrons/cm²-sec): F.F. = 0.673 x (F.R.) where 0.673 is a conversion based on standard pile measurements described in Reference 3.

(4) Thermal Flux (neutrons/cm²-sec): T.F. = (F.F.) x (CR-1)/CR which is calculated only when a cadmium ratio (CR) is provided.

The second stage of the program associates a fuel element position and a monitor position with each fission rate and flux. The association presupposes that the wires have been put onto the chips in the order specified above. The program can then begin the Method 2 power calculation.

For each fuel element, the program calculates the average fission rate at midplane and the average power for the fuel element. The average specific power at midplane for the J^{th} fuel element is calculated as

$$S.P.A._{Jm} = \sum_{i=1}^{17} C_{iJ} (S.P.)_{iJ}$$

where $(S.P.)_{iJ}$ is the specific power in watts/g-U-235 for the i th position in Fuel Element J and where C_{iJ} is the input weighting factor for Position i in Element J . The average power in watts for Fuel Element J is calculated as

$$P_J = S.P.A._{Jm} \times 975/1.42$$

where 1.42 is the peak-to-average ratio for the axial flux distribution in a fuel element containing 975 g U-235.

If a quarter-core flux mapping has been done and quarter-core reactor symmetry can be assumed, the program performs other calculations. A quarter-core mapping is defined as a mapping in each of the fuel elements in Fuel Positions 6 through 15, 16 through 25, 26 through 35, or 36 through 5. Quarter-core symmetry implies that the total core power can be accurately given by multiplying the quarter-core power by four. When this symmetry is obtained, the program calculates total core power and extrapolates the fuel element power values to 250 MW. It also calculates point-to-core average ratios for each fuel position. For this latter calculation, each ratio calculated is tested as to whether it is greater than 2.9 or less than 0.7 and flags those values outside of this range. This feature can be very useful in hot-spot searches and in discovering misplaced data entries.

3. PREPARATION OF INPUT INFORMATION

The input cards required are of three general classes: weighting functions, control cards, and data cards. The order of program assembly is (a) program deck, (b) \$ENTRY card, (c) weighting functions, (d) control cards for the first data set, (e) first data set, (f) blank card, (g) control cards for second data set, (h) second data set, (i) blank card, etc. When the last data set has been added, it must be followed by two blank cards. The input information and format for each of the three classes is described below.

3.1 Weighting Functions

Each time the program is used, the weighting functions C_{iJ} must be read in for $J=1$ to 20. When these values have been finalized, they will be incorporated into the object program. However, for the present there must be 20 cards, one for each fuel element, arranged sequentially for Fuel Elements 1 through 20. Since at present there are 17 weighting functions in use, there must be 17 entries per card. These entries must be ordered to correspond with the position numbers shown in Figure 5. The entry for each position must be an integer multiple of the inverse of an overall normalizing factor, 145.489, and the format used is I4. The normalizing factor and the integer values in use were experimentally determined, as discussed in this report, and may change at a later date as more data are accumulated.

3.2 Control Cards

There are five control cards required. These cards are used to control the position ordering, to select the output options, and to provide constants for the calculations. The format for each card is as follows:

Card Number	Column	Entry Description	Format
1	1-5	Any nonzero number	I5
2	1-10	Number of elements monitored	I10
2	11-20	Number of weighting factors used	I10
2	21-30	Number of the first chip in the counting channel not containing Chip 1	I10
2	31-40	Chip number of the last chip pertinent to the power calculation in the counting channel containing Chip 1	I10
2	41-50	Number of the counting channel (1 or 2) containing Chip 1	I10
2	51-60	For a quarter-core map of Positions 6 through 15, insert 0. For a quarter-core map of Positions 16 through 25, insert 10. For a quarter-core map of Positions 26 through 35, insert 20. For a quarter-core map of Positions 36 through 5, insert 30.	I10
3	1-50	Fuel position number in decimal form in the order that the monitors for each element were unloaded onto plastic chips. Ten to a card.	10E5.3
Next card	1-10	1.0 is placed in this space if an output of the power at 250 MW is desired. However, use only if a quarter-core has been mapped. Otherwise no entry.	E10.5
	11-20	2.0 is placed in this space if an output of the fission rate at each monitored point is desired for each element. The fission rates are placed within an element outline. Otherwise no entry.	E10.5
	21-30	3.0 is placed in this space if an output is desired exactly like Option 2 above except a point-wise presentation of specific power in watts/g U-235 is given. Otherwise no entry.	E10.5
	31-40	4.0 is placed in this space for an output exactly like Option 2 except point-wise values for fission flux are given. Otherwise no entry.	E10.5

<u>Card Number</u>	<u>Column</u>	<u>Entry Description</u>	<u>Format</u>
	41-50	5.0 is placed in this space for an output exactly like Option 2 except point-wise values for thermal flux are given. Otherwise no entry.	E10.5
	51-60	6.0 is placed in this space for an output of point-wise values of point-to-core averages for all monitored points of all elements. Also appearing will be a note as to whether any values are > 2.9 or < 0.7. A proper core average will be used only if a quarter-core has been mapped. Otherwise no entry	E10.5
Next card	1-2	01	I2
	3-6	TWAIT expressed as an integer multiple of 1/100 hours.	F4.2
	7-10	TEXP expressed in seconds as an integer	F4.0
	11-12	Cadmium ratio as the nearest integer	F2.0
	13-15	Blank	3X
	16-70	Alpha-numeric problem title	11A6

3.3 Data Cards

There will be one data card for each counting interval on the automatic beta counter. The format for each data card is as follows:

<u>Column</u>	<u>Entry</u>	<u>Format</u>
1-7	Elapsed time in seconds since the counter was started	F7.0
8-10	Blank	3X
11-17	Count time in units of 1/100 seconds	F7.2
18-20	Blank	3X
21-23	Run number	I3
24-26	Blank	3X
27-29	Data card number	I3
30-32	Blank	3X
33-39	Number of counts in channel one	F7.0

<u>Column</u>	<u>Entry</u>	<u>Format</u>
40-42	Blank	3X
43-49	Number of counts in channel two	F7.0
50-52	Blank	3X
52-59	Number of counts in channel three	F7.0
60-70	Blank	11X
71-80	Ten nonzero integers	I10

4. OUTPUT DESCRIPTION

For all wires counted, a value for fission rate, specific power, fission flux, and thermal flux will output in the form presented in page 33. Additionally, a value for the midplane and total power for each fuel element monitored will be output in the form presented on page 34. Further output information will be controlled by the entries selected in the output control card. There are six such options available. For each, a typical page is shown on pages 34 through 37.

5. PROGRAM LISTING

A complete FORTRAN IV listing of ATRC FLUX MAP is given on pages 38 through 42.

10/10/66

40-50-60 POWER SPLIT WITH FTF IN 60MLCBE SHIM REV

PAGE 1
000000

RUN NO.	SAMPLE NO.	CH. NO.	TIME WAIT	FISSIONS GM-SEC	WATTS GM	NEUTRONS CM**2-SEC	THERMAL NEUTS CM**2-SEC	EXPOSURE TIME
1	1	1	3.30	4.851E 08	1.531E-02	3.264E 08	0.	1200
1	1	2	3.30	1.429E 09	4.511E-02	9.619E 08	0.	1200
1	2	1	3.30	5.377E 08	1.697E-02	3.619E 08	0.	1200
1	2	2	3.30	9.820E 08	3.099E-02	6.609E 08	0.	1200
1	3	1	3.30	9.787E 08	3.089E-02	6.586E 08	0.	1200
1	3	2	3.30	9.306E 08	2.937E-02	6.263E 08	0.	1200
1	4	1	3.30	7.538E 08	2.379E-02	5.073E 08	0.	1200
1	4	2	3.30	9.883E 08	3.119E-02	6.652E 08	0.	1200
1	5	1	3.30	1.026E 09	3.238E-02	6.906E 08	0.	1200
1	5	2	3.30	1.407E 09	4.440E-02	9.467E 08	0.	1200
1	6	1	3.30	1.017E 09	3.209E-02	6.844E 08	0.	1200
1	6	2	3.30	5.854E 08	1.847E-02	3.939E 08	0.	1200
1	7	1	3.30	7.361E 08	2.323E-02	4.954E 08	0.	1200
1	7	2	3.30	5.540E 08	1.749E-02	3.729E 08	0.	1200
1	8	1	3.30	6.596E 08	2.082E-02	4.439E 08	0.	1200
1	8	2	3.30	5.847E 08	1.845E-02	3.935E 08	0.	1200
1	9	1	3.30	5.713E 08	1.803E-02	3.843E 08	0.	1200
1	9	2	3.30	7.973E 08	2.516E-02	5.366E 08	0.	1200
1	10	1	3.30	6.414E 08	2.024E-02	4.317E 08	0.	1200
1	10	2	3.30	4.988E 08	1.574E-02	3.357E 08	0.	1200
1	11	1	3.30	4.488E 08	1.416E-02	3.020E 08	0.	1200
1	11	2	3.30	7.791E 08	2.459E-02	5.244E 08	0.	1200
1	12	1	3.30	4.020E 08	1.266E-02	2.706E 08	0.	1200
1	12	2	3.30	6.177E 08	1.949E-02	4.157E 08	0.	1200
1	13	1	3.30	3.684E 08	1.163E-02	2.480E 08	0.	1200
1	13	2	3.30	5.441E 08	1.717E-02	3.662E 08	0.	1200
1	14	1	3.30	5.844E 08	1.844E-02	3.933E 08	0.	1200
1	14	2	3.30	5.834E 08	1.841E-02	3.925E 08	0.	1200
1	15	1	3.30	3.615E 08	1.141E-02	2.433E 08	0.	1200
1	15	2	3.30	1.106E 09	3.489E-02	7.441E 08	0.	1200
1	16	1	3.30	4.941E 08	1.559E-02	3.325E 08	0.	1200
1	16	2	3.30	8.213E 08	2.592E-02	5.527E 08	0.	1200
1	17	1	3.30	4.704E 08	1.485E-02	3.166E 08	0.	1200
1	17	2	3.30	1.131E 09	3.569E-02	7.611E 08	0.	1200
1	18	1	3.30	4.118E 08	1.300E-02	2.771E 08	0.	1200
1	18	2	3.30	1.391E 09	4.389E-02	9.359E 08	0.	1200
1	19	1	3.30	4.003E 08	1.263E-02	2.694E 08	0.	1200
1	19	2	3.30	1.026E 09	3.239E-02	6.907E 08	0.	1200
1	20	1	3.30	9.468E 08	2.988E-02	6.372E 08	0.	1200
1	20	2	3.30	1.042E 09	3.288E-02	7.011E 08	0.	1200
1	21	1	3.30	7.016E 08	2.214E-02	4.722E 08	0.	1200
1	21	2	3.30	1.054E 09	3.325E-02	7.091E 08	0.	1200
1	22	1	3.30	8.190E 08	2.585E-02	5.512E 08	0.	1200
1	22	2	3.30	1.481E 09	4.674E-02	9.967E 08	0.	1200
1	23	1	3.30	5.355E 08	1.690E-02	3.604E 08	0.	1200
1	23	2	3.30	5.748E 08	1.814E-02	3.869E 08	0.	1200
1	24	1	3.30	4.000E 08	1.262E-02	2.692E 08	0.	1200
1	24	2	3.30	5.832E 08	1.841E-02	3.925E 08	0.	1200
1	25	1	3.30	3.559E 08	1.123E-02	2.395E 08	0.	1200
1	25	2	3.30	6.127E 08	1.934E-02	4.123E 08	0.	1200

ELEMENT	AVE FISS RATE AT MIDPLANE	AVE WATT PER GM AT MIDPLANE
26	0,6987739E 09	0,2205330E-01
27	0,7035034E 09	0,2220257E-01
28	0,7319627E 09	0,2310074E-01
29	0,7211030E 09	0,2275801E-01
30	0,6231651E 09	0,1966709E-01
31	0,5824366E 09	0,1838170E-01
32	0,5989133E 09	0,1890170E-01
33	0,5173224E 09	0,1632670E-01
34	0,3495060E 09	0,1103041E-01
35	0,2731456E 09	0,8620475E-02
22	0,7117786E 09	0,2246373E-01
THE PWR IN ELEM 26 IS 0,1514223E 02 WATTS		
THE PWR IN ELEM 27 IS 0,1524472E 02 WATTS		
THE PWR IN ELEM 28 IS 0,1904142E 02 WATTS		
THE PWR IN ELEM 29 IS 0,1562610E 02 WATTS		
THE PWR IN ELEM 30 IS 0,1350381E 02 WATTS		
THE PWR IN ELEM 31 IS 0,1262124E 02 WATTS		
THE PWR IN ELEM 32 IS 0,1297828E 02 WATTS		
THE PWR IN ELEM 33 IS 0,1121023E 02 WATTS		
THE PWR IN ELEM 34 IS 0,7573696E 01 WATTS		
THE PWR IN ELEM 35 IS 0,5918988E 01 WATTS		
THE PWR IN ELEM 22 IS 0,1542404E 02 WATTS		

OUTPUT OPTION NUMBER ONE

ELEMENT	PWR AT 250MW	PERCENT TOTAL CORE PWR
26	0,7530189E 01	0,3012044E 01
27	0,7581075E 01	0,3032430E 01
28	0,7887757E 01	0,3155103E 01
29	0,7770732E 01	0,3108293E 01
30	0,6715336E 01	0,2686134E 01
31	0,6276438E 01	0,2510575E 01
32	0,6453995E 01	0,2581598E 01
33	0,5974797E 01	0,2229903E 01
34	0,3766338E 01	0,1506535E 01
35	0,2943465E 01	0,1177386E 01
22	0,7670250E 01	0,3068100E 01
POWER SPLIT NE = 0,6153935E 08 CTR = 0,5196709E 08 SE = 0,3747711E 08		

OUTPUT OPTION NUMBER TWO

0,1017E 10	0,7361E 09	0,6596E 09	0,5713E 09	0,6414E 09
*				*
*	0,4488E 09	0,4020E 09	0,3684E 09	*
*				*
0,5844E 09		0,3615E 09		0,4941E 09
*				*
*	0,4704E 09	0,4118E 09	0,4003E 09	*
*				*
0,9468E 09		0,7016E 09		0,8190E 09
FISSION RATE MAP				
ELEMENT 33				
AVERAGE VALUE = 0,5173224E 09				

OUTPUT OPTION NUMBER THREE

0,3209E-01	0,2323E-01	0,2082E-01	0,1803E-01	0,2024E-01
*				*
*	0,1416E-01	0,1269E-01	0,1163E-01	*
*				*
0,1844E-01		0,1141E-01		0,1559E-01
*				*
*	0,1485E-01	0,1300E-01	0,1263E-01	*
*				*
0,2988E-01		0,2214E-01		0,2585E-01
MAP OF WATT PER GRAM				
ELEMENT 33				
AVERAGE VALUE = 0,1632670E-01				

OUTPUT OPTION NUMBER FOUR

0,6844E 09	0,4954E 09	0,4439E 09	0,3845E 09	0,4317E 09
	0,3020E 09	0,2706E 09	0,2480E 09	
0,3933E 09		0,2433E 09		0,3325E 09
	0,3166E 09	0,2771E 09	0,2694E 09	
0,6372E 09		0,4722E 09		0,5512E 09

FISSION FLUX MAP

ELEMENT 33

OUTPUT OPTION NUMBER FIVE

0,	0,	0,	0,	0,
	0,	0,	0,	
0,		0,		0,
	0,	0,	0,	
0,		0,		0,

THERMAL FLUX MAP

ELEMENT 33

OUTPUT OPTION NUMBER SIX

0,2490E 01	0,1802E 01	0,1615E 01	0,1399E 01	0,1970E 01
*				*
*				*
*	0,1099E 01	0,9843E 00	0,9020E 00	*
*				*
*				*
0,1431E 01		0,8850E 00		0,1210E 01
*				*
*				*
*	0,1152E 01	0,1008E 01	0,9801E 00	*
*				*
*				*
0,2318E 01	0,1718E 01		0,2005E 01	
MAP OF PEAK TO CORE AVERAGE FISSION RATE				
ELEMENT 33				

10/12/66

ISN	SOURCE STATEMENT	
0	\$IRFTC LIST, NODECK	003
C		004
C	FLUX CALCULATION PROGRAM	005
C	ATRC CORE FLUX MAPS	006
C	N.C. KAUFMAN	007
1	DIMENSION ELEM(40), ROLL(6), WT(40,20), VRT(200), RRT(800), THERM(800)	008
2	DIMENSION FRRT(800), TOTL(40), TOTW(40), ROT(40,45), TOT(40,40)	010
3	DIMENSION ROT(40,20), COT(40,20), AVFR(40), AVRT(40), PKAV(25)	011
4	DIMENSION ALT(12), CH(2), CHK(2), LGN(40)	012
C		013
C	FORMAT STATEMENTS	014
C		015
5	ASSIGN 53 TO K3	016
6	ASSIGN 54 TO K4	017
7	ASSIGN 55 TO K5	018
10	ASSIGN 56 TO K6	019
11	53 FORMAT(5H0 RUN,4X,6HSAMPLE,3X,3MCH,,5X,4HTIME,10X,8HFISSIONS,8X, 1 5HWATTS,10X,8HNEUTRONS,7X,13HTHERMAL NEUTS,4X,8HEXPOSURE)	020
12	54 FORMAT(5H NO,,6X,3HNO,,4X,3HNO,,5X,4HWAIT,10X,8H GM+SEC ,RX, 1 5H GM ,9X,9HCH**2=SEC,7X,13H CM**2=SEC ,4X,8H TIME ,/)	022
13	55 FORMAT(5H0 RUN,4X,6HSAMPLE,3X,3MCH,,5X,4HTIME,10X,9HFLUX-GOLD,8X, 1 9HTH. NEUTS,7X,9HFLUX=DYSP,6X,9HTH. NEUTS,7X,8HEXPOSURE)	023
14	56 FORMAT(5H NO,,6X,3HNO,,4X,3HNO,,5X,4HWAIT,10X,9HCH**2=SEC,8X, 1 9HCH**2=SEC,7X,9HCH**2=SEC,6X,9HCH**2=SEC,7X,6H TIME/)	024
15	61 FORMAT(12,F4,2,F4,0,F2,0,3X,11A6)	025
16	62 FORMAT(F7,0,3X,F7,2,2(3X,13),3(3X,F7,0),11X,110)	026
17	63 FORMAT(2H ,13,5X,13,6X,11,6X,F6,2,8X,1PE10,3,5X,1PE10,3,6X, 1 1PE10,3,5X,1PE10,3,9X,14)	027
20	131 FORMAT(6110)	028
21	132 FORMAT(10E9,3)	029
22	133 FORMAT(6E10,5)	030
23	134 FORMAT(1714)	031
24	116 FORMAT(1H1,5X,8H ELEMENT,6X,14H AVE FISSION RATE,11X,16H AVE WATT PE 1R GM)	032
25	117 FORMAT(1H0,4X,15,9X,E14,7,10X,E14,7)	033
26	118 FORMAT(1H0,5X,32H TOTAL POWER OF ELE MONITORED IS,5X,E14,7)	034
27	119 FORMAT(1H1,5X,8H ELEMENT,6X,13H PWR AT 250MH,12X,23H PERCENT TOTAL 1L CORE PWR)	035
30	121 FORMAT(1H0,12H POWER SPLIT,5X,5H NE =,E14,7,2X,6H CTR =,E14,7,2X, 15H SE =,E14,7)	036
31	122 FORMAT(1H1,E11,5,4X,E11,5,4X,E11,5,3X,E11,5,4X,E11,5,2H +,68X,2H 1 +//1X,2H +,66X,2H +//2X,2H +,9X,E11,5,6X,E11,5,7X,E11,5,9X,2H +// 23X,2H +,62X,2H +//4X,2H +,60X,2H +//5X,E11,5,13X,E11,5,13X,E11,5,5// 36X,2H +,56X,2H +//7X,2H +,54X,2H +//8X,2H +,7X,E11,5,3X,E11,5,3X,E 411,5,6X,2H +//9X,2H +,50X,2H +//10X,2H +,48X,2H +//11X,E11,5,8X,E1 51,5,8X,E11,5)	037
32	123 FORMAT(1H0,20X,17H FISSION RATE MAP//30X,8H ELEMENT,15//20X,16H A 1VERAGE VALUE =,E14,7)	038
33	124 FORMAT(1H0,20X,21H MAP OF WATT PER GRAM//30X,8H ELEMENT,15//20X,1 16H AVERAGE VALUE =,E14,7)	039
34	125 FORMAT(1H0,20X,17H FISSION FLUX MAP//30X,8H ELEMENT,15)	040
35	126 FORMAT(1H0,20X,17H THERMAL FLUX MAP//30X,8H ELEMENT,15)	041
36	127 FORMAT(1H0,20X,41H MAP OF PEAK TO CORE AVERAGE FISSION RATE//30X, 18H ELEMENT,15)	042

ATRC FLUX MAP TEST		FORTRAN SOURCE LIST	10/12/66
ISN	SOURCE STATEMENT		
37	135	FORMAT(1H0,20X,9HTHERE ARE,3X,(2,3X,20HPOINTS OUT OF LIMITS)	056
40	141	FORMAT(1H ,20X,12H AT MIDPLANE,13X,12H AT MIDPLANE)	059
41	142	FORMAT(1H0,16H THE PWR IN ELEM,15,3H IS,E14,7,6H WATTS)	060
42		DO 83 KNM=1,20,1	061
43		READ(5,134)(LGN(NX),NX=1,17)	062
50		DO 84 MAL=1,17,1	063
51		WT(KNM,MAL)=LGN(MAL)	064
52	84	CONTINUE	065
54	83	CONTINUE	066
56	100	READ(5,101) KXL	067
60	101	FORMAT(15)	068
61		IF(KXL,LT,1) GO TO 102	069
64		READ(5,131)(NO,NZ,M,NG,LG,LSTP)	070
77		READ(5,132)(ELEM(N),N=1,NQ)	071
104		READ(5,133)(ROLL(LIN),LIN=1,6)	072
111		READ(5,61)IOP,TWAIT,TEXP,CDR,(ALF(I),I=1,11)	073
117		WRITE(6,61)IOP,TWAIT,TEXP,CDR,(ALF(I),I=1,11)	074
124		IEXP=TEXP	075
125		CALL INITAL(=1,=2,=3,ALF)	076
126		IF(IOP,GT,9)GO TO 90	077
131		KNS=0	078
132		READ(5,62)BEGTM,FCOUNT,IRN,ISN,CH(1),CH(2),CH3,ID	079
136		WRITE(6,62)BEGTM,FCOUNT,IRN,ISN,CH(1),CH(2),CH3,ID	080
137		IF(IOP,GT,5)GO TO 87	081
142		CALL FUNCTK(TWAIT,CHK(1),CHK(2))	082
143	5	KNS=KNS+1	083
C			084
144		DO 91 I=1,2	085
145		FPGS=CHK(I)*((CH(I)/FCOUNT)*(1,8E-6*CH(I)/FCOUNT+1,)=6,2)*	086
	1	1200,0/TEXP	087
146		WATTS=FPGS*3,156E-11	088
147		SNEUTS=0,673*FPGS	089
150		IF(CDR,EQ,0,0)GO TO 40	090
153		TNEUT=((CDR-1,)/CDR)*0,673*FPGS	091
154	7	CALL SPACE(K3,K4)	092
155		WRITE(6,63)IRN,KNS,I,TWAIT,FPGS,WATTS,SNEUTS,TNEUT,IEXP	093
156		IF(I,EQ,LG) GO TO 21	094
161		KJ=M+KNS-1	095
162		GO TO 22	096
163	21	KJ=KNS	097
164		IF(KJ,GT,NG) GO TO 89	098
167	22	CONTINUE	099
170		VRT(KJ)=FPGS	100
171		FRT(KJ)=WATTS	101
172		THERM(KJ)=TNEUT	102
173		FRT(KJ)=SNEUTS	103
174	89	GO TO 91	104
175	40	TNEUT=0,	105
176		GO TO 7	106
177	91	CONTINUE	107
201		READ(5,62)BEGTM,COUNT,IRN,ISN,CH(1),CH(2),CH3,ID	108
205		IF(ID,EQ,0)GO TO 6	109
210		GO TO 5	110
211	90	WRITE(6,64)IOP	111
212	64	FORMAT(37H THE SPECIFIED OPTION IS TOO BIG-IOP=13)	112

ATRC FLUX MAP TEST

FORTRAN SOURCE LIST

10/12/66

ISN	SOURCE STATEMENT	
213	CALL EXIT	113
214	6 LNS=0	114
215	CORF=0.0	115
216	CORPW=0.0	116
217	CORPWR=0.0	117
220	CORFR=0.0	118
221	WRITE(6,116)	119
222	WRITE(6,141)	120
223	DO 24 NN=1,NQ,1	121
224	J=ELEM(NN)	122
225	TOTL(J)=0.0	123
226	TOTW(J)=0.0	124
227	DO 25 JK=1,NZ,1	125
230	LNS=LNS+1	126
231	IF(J,LE,20) GO TO 431	127
234	JLJLR=J-20	128
235	WTTY=WT(JLJLR,JK)	129
236	GO TO 432	130
237	431 WTTY=WT(J,JK)	131
240	432 CONTINUE	132
241	ROT(J,JK)=VRT(LNS)	133
242	TOT(J,JK)=RRT(LNS)	134
243	BOY(J,JK)=FRRT(LNS)	135
244	COT(J,JK)=THERM(LNS)	136
245	TOTW(J)=TOTW(J)+TOT(J,JK)*WTTY	137
246	TOTL(J)=TOTL(J)+ROT(J,JK)*WTTY	138
247	25 CONTINUE	139
251	AVFR(J)=TOTL(J)/145.489	140
252	AVWT(J)=TOTW(J)/145.489	141
253	WRITE(6,117)(J,AVFR(J),AVWT(J))	142
260	24 CONTINUE	143
262	DO 71 LEAP=1,NQ,1	144
263	INK=ELEM(LEAP)	145
264	AVELM=AVWT(INK)*975.0/1.42	146
265	WRITE(6,142)(INK,AVELM)	147
272	71 CONTINUE	148
274	EOBSE=0.0	149
275	EOBNE=0.0	150
276	EOBC=0.0	151
277	DO 26 LL=6,9,1	152
300	LR=LL*LSTP	153
301	IF(LR,GT,40) LR=LR+40	154
304	CORF=CORF+AVFR(LR)*4.0	155
305	CORPW=CORPW+AVWT(LR)*4.0	156
306	EOBNE=EOBNE+2.0*AVWT(LR)*975.0	157
307	26 CONTINUE	158
311	DO 27 LN=10,11,1	159
312	LR=LN*LSTP	160
313	IF(LR,GT,40) LR=LR+40	161
316	CORF=CORF+AVFR(LR)*4.0	162
317	CORPW=CORPW+AVWT(LR)*4.0	163
320	EOBC=EOBC+4.0*AVWT(LR)*975.0	164
321	27 CONTINUE	165
323	DO 28 LK=12,15,1	166
324	LR=LK*LSTP	167

ATRC FLUX MAP TEST		FORTRAN SOURCE LIST		10/12/66
ISN	SOURCE STATEMENT			
325	IF(LR.GT.40) LR=LR+40			166
330	CORF=CORF+AVFR(LR)*4.0			166
331	CORPW=CORPW+AVWT(LR)*4.0			170
332	EOBSE=EOBSE+2.0*AVMT(LR)*975.0			171
333	28 CONTINUE			172
335	CORPWR=CORPW/40.0			173
336	CONF=CORF/40.0			174
337	IF(ROLL(1).EQ.1.0) GO TO 41			175
342	GO TO 42			176
343	41 CONTINUE			177
344	WRITE(6,119)			176
345	DO 29 NR=1,NQ,1			176
346	JJ=ELEM(NR)			180
347	PER=(AVMT(JJ)*975.0/(CORPWR*39000.0))			181
350	PWRMW=PER*250.0			182
351	PPERT=PER*100.0			183
352	WRITE(6,117)(JJ,PWRMW,PPERT)			184
357	29 CONTINUE			185
361	ETRP=250000000.0/(CORPWR*39000.0)			184
362	EOBSE=EOBSE+ETRP			187
363	EOBC=EOBC+ETRP			186
364	EOBNE=EOBNE+ETRP			186
365	WRITE(6,121)(EOBNE,EOBC,EOBSE)			190
372	42 CONTINUE			191
373	DO 31 JLD=1,NQ,1			192
374	JTD=ELEM(JLD)			193
375	DO 57 NNN=2,6,1			194
376	IF(ROLL(NNN).EQ.2.0) GO TO 43			195
401	IF(ROLL(NNN).EQ.3.0) GO TO 44			196
404	IF(ROLL(NNN).EQ.4.0) GO TO 45			197
407	IF(ROLL(NNN).EQ.5.0) GO TO 46			198
412	IF(ROLL(NNN).EQ.6.0) GO TO 47			199
415	GO TO 48			200
416	43 WRITE(6,122)(ROT(JTD,KKK),KKK=1,NZ,1)			201
423	WRITE(6,123)(JTD,AVFR(JTD))			202
430	GO TO 48			203
431	44 WRITE(6,122)(TOT(JTD,KMN),KMN=1,NZ,1)			204
436	WRITE(6,124)(JTD,AVWT(JTD))			205
443	GO TO 48			206
444	45 WRITE(6,122)(BOT(JTD,KMM),KMM=1,NZ,1)			207
451	WRITE(6,125) JTD			208
452	GO TO 48			209
453	46 WRITE(6,122)(COT(JTD,KTM),KTM=1,NZ,1)			210
460	WRITE(6,126) JTD			211
461	GO TO 48			212
462	47 ETPFR=CONF/1.42			213
463	KTM=0			214
464	DO 36 NK=1,NZ,1			215
465	PKAV(NK)=(ROT(JTD,NK)/ETPFR)			216
466	IF(PKAV(NK).LT.0.7) KTM=KTM+1			217
471	IF(PKAV(NK).GT.2.9) KTM=KTM+1			218
474	36 CONTINUE			219
476	WRITE(6,122)(PKAV(LOS),LOS=1,NZ,1)			220
503	WRITE(6,127) JTD			221
504	IF(KTM.EQ.0) GO TO 48			222

ATRC FLUX MAP TEST		FORTRAN SOURCE LIST	10/12/66
ISN	SOURCE STATEMENT		
507	WRITE(6,135) KTM		223
510	48 CONTINUE		224
511	57 CONTINUE		225
513	31 CONTINUE		226
515	87 CONTINUE		227
516	GO TO 100		228
517	102 CONTINUE		229
520	END		230

ATRC FLUX MAP TEST		FORTRAN SOURCE LIST	10/12/66
ISN	SOURCE STATEMENT		
0	SIAPTC SUB		231
1	SUBROUTINE FUNCTK(X,Y1,Y2)		232
C			233
2	$Y1 = -1.6456477E+04 + 7.0044771E+04 * X$		234
3	$Y2 = -1.6456477E+04 + 7.0044771E+04 * X$		235
4	RETURN		236
5	END		237