A two-dimensional perturbation theory computer code, PERT-IV, has been developed that will calculate reactivity coefficients, the delayed neutron fraction, and the neutron generation time. The program uses the output flux and adjoint flux from either a diffusion theory or transport theory program. A discussion and derivation of the perturbation equation and expressions for the effective neutron delayed fraction and the neutron generation time are given. In addition, the reactivity coefficients as calculated by diffusion perturbation theory and $S_N$ transport theory are compared.

The input flux and adjoint flux to PERT-IV can be taken directly from two-dimensional calculations or synthesized (by the code) from one-dimensional calculations. The code is compatible with the one-dimensional DTF-IV and two-dimensional 2DF transport theory codes from Los Alamos and the one-dimensional 1DX.
and two-dimensional 2DB diffusion theory codes from Battelle-Northwest.
PERT-IV, A Two-Dimensional Perturbation Theory Code for Fast Reactors

by

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PERT-IV, A Two-Dimensional Perturbation Theory Code for FastReactors

I. INTRODUCTION

Perturbation Theory in General

Nuclear Engineers often desire to calculate the change in the effective multiplication constant, $k_{\text{eff}}$, caused by small changes in the macroscopic cross sections of a nuclear reactor. This is necessary, for example, in the operational calculations of a reactor such as the Fast Test Reactor (2) where numerous samples will be inserted into the core at various test positions. Another example of the usefulness of computing reactivity coefficients ($\delta k_{\text{eff}}/k_{\text{eff}}$ per mass) is in the analysis of critical assemblies (e.g., ZPR-III, ZPR-VI (12)) where material worths are frequently measured for the purpose of checking cross section data and models.

One method of determining reactivity coefficients is to use a diffusion or transport theory computer code and calculate $k_{\text{eff}}$ twice—with and without the sample. The disadvantage of this method is that a separate computer run is required for each reactivity coefficient at each position. Since a typical five energy group, two-dimensional transport theory problem can take from 15 to 30 minutes of UNIVAC 1108 computer time, the calculation of more than a couple material worths becomes very expensive. Moreover, since the change in $k_{\text{eff}}$ is usually quite
small, the problem must be well converged for accurate worth calculations. Perturbation theory offers the advantage that once the fluxes and adjoint fluxes are known for the unperturbed case, many reactivity coefficients can be calculated very rapidly. A secondary benefit of perturbation theory is that it lends an insight into the causes of the change in $k_{\text{eff}}$ due to a variation in the reactor composition because the perturbation equation gives the reactivity coefficient as the sum of four components: fission, absorption, leakage, and moderation.

It should be mentioned at this point that the use of perturbation theory is not a panacea for all material worth calculations. The main disadvantage is that perturbation theory assumes the variation in the reactor composition is "small". Frequently, the variation is either obviously too large for perturbation theory to work well—the insertion of a high worth control rod, for example—or it is unclear whether the variation is too large. Large perturbations in the flux caused by the variation result in large errors in the worth calculations. Therefore, if very accurate worths are desired for samples big enough to significantly perturb the flux, perturbation theory should not be used.

**PERT-IV, A Two-Dimensional Perturbation Code**

PERT-IV uses flux and adjoint flux output (in the form of punched cards or a binary tape) from diffusion or transport theory codes (e.g., 1DX(7), 2DB(10), DTF-IV(8), 2DF(11)) and calculates
reactivity coefficient traverses for any specified row or column in the reactor using first order perturbation theory based on the multigroup diffusion model. If one-dimensional fluxes are used, both an axial and a radial case must be run for the flux and adjoint; the code then synthesizes the fluxes into a two-dimensional array. Each component of the perturbation equation, in addition to the total reactivity coefficient, is printed at each mesh point in the specified row or column.

Input to PERT-IV, in addition to the fluxes, consists of cross sections, axial and radial mesh structure, fission source data, and cross section mixture information. If the absolute number of neutrons per fission for each fissionable isotope, the delayed fission source, and neutron velocities are included, the neutron generation time, $\Lambda$, and the effective delayed neutron fraction, $\beta_{\text{eff}}$, are also calculated. This is a reasonable addition to a perturbation theory code because the expressions for $\Lambda$ and $\beta_{\text{eff}}$ are similar in many respects to the perturbation equation.

In a 65K memory (and 4 tapes), the code can handle up to 35 materials, 26 energy groups, and a 50 x 50 spatial mesh. The code will calculate up to 20 reactivity coefficient traverses in one run. The running time for a typical run (5 groups) on the UNIVAC 1108 is roughly 20 seconds.
Outline of Thesis

A derivation of the perturbation equation is given in Section II followed by a derivation of the expressions for the neutron generation time and beta effective in Section III. Section IV gives a general description of PERT-IV. Finally, a comparison is given in Section V between worths calculated by first order perturbation theory (PERT-IV), diffusion theory (2DB), and transport theory (DTF-IV and 2DF). Input instructions, a sample problem, and a FORTRAN-IV source deck listing are given in the Appendices.
II. DERIVATION OF PERTURBATION EQUATION

Perturbation theory as applied to reactor physics was developed before the days of high speed computers by Wigner(13) to derive expressions for the change in the pile period due to cross-section changes in the two group equations for a bare, graphite pile. Another early contributor to the perturbation method was Brooks,(4) who derived an expression for the change in reactivity resulting from changes in the cross sections in the multigroup equations for a pile.

To derive the perturbation equation, we shall begin with the familiar multigroup time independent diffusion equation for an unperturbed system.

\[ \nabla \cdot D_i \nabla \phi_i - \Sigma_{a,i} \phi_i = - \sum_{j=i+1}^{N} \Sigma_{(i+j)} \phi_i + \frac{\chi_i}{k} \sum_{j=1}^{N} (\nu \Sigma_t)^j \phi_j \]

\[ + \sum_{j=1}^{i-1} \Sigma_{(j+i)} \phi_j = 0 \]

where:

- \( N \) = number of energy groups,
- \( \phi_i \) = flux in group \( i \) \( (n^{-2} \text{cm}^{-2} \text{sec}^{-1}) \),
- \( \chi_i \) = fission source born in group \( i \) \( (\sum_{i=1}^{N} \chi_i = 1.0) \),
- \( \Sigma_{a,i} \) = macroscopic absorption cross section in group \( i \) \( (\text{cm}^{-1}) \),
\[ D_i = \text{diffusion coefficient for energy group } i \ (\text{cm}), \]
\[ \Sigma(i\rightarrow j) = \text{macroscopic transfer cross section from group } i \]
\[ \text{to group } j \ (\text{cm}^{-1}), \]
\[ k = \text{effective multiplication factor}, \]
\[ (\nu E_f)^i = \text{macroscopic fission source cross section in } \]
\[ \text{group } i \ (\text{cm}^{-1}). \]

Although the flux and cross sections are a function of position, for simplicity, the spatial variables are not explicitly shown.

A similar expression can also be written for a perturbed system.

\[
\begin{align*}
\nabla \cdot D_i \nabla \phi_i^{\prime} &- \sum_{a=1}^{N} \Sigma^a \phi_i^{\prime} - \sum_{j=i+1}^{N} \Sigma'(i\rightarrow j) \phi_i^{\prime} + \frac{\chi_1}{k'} \sum_{j=1}^{N} (\nu E_f)^j \phi_j^{\prime} \\
&+ \sum_{j=1}^{i-1} \Sigma'(j\rightarrow i) \phi_j^{\prime} = 0 ,
\end{align*}
\]

where:  \[ \Sigma^i = \Sigma^i + \delta \Sigma^i \text{ for } \Sigma_a^i, \Sigma_f^i \text{ and } \Sigma'(i\rightarrow j) \ , \]
\[ D_i' = D_i + \delta D_i , \]
\[ \phi_i' = \phi_i + \delta \phi_i , \]
\[ k' = k + \delta k . \]

First, we multiply Equations (2.1) and (2.2) by an arbitrary weighting function \( \psi_i \) (also a function of spatial position), sum over all groups, integrate over the entire volume of the reactor, and subtract the first integral equation from the second. This yields
\[
\int \sum_{i=1}^{N} \psi_1 (\nabla \cdot D_i \nabla \phi_i - \nabla \cdot D_i \nabla \phi_i) dV - \int \sum_{i=1}^{N} \psi_1 (\Sigma^i \phi_i - \Sigma^i \phi_i) dV
\]

\[
- \int \sum_{i=1}^{N} \psi_1 \left( \sum_{j=i+1}^{N} \Sigma^i (i \rightarrow j) \phi_j - \sum_{j=i+1}^{N} \Sigma (i \rightarrow j) \phi_j \right) dV
\]

\[
+ \int \left[ \sum_{i=1}^{N} \frac{\psi X_i}{k^2} \sum_{j=1}^{N} \left( \nu \Sigma \right)^j \phi_j^i - \sum_{i=1}^{N} \frac{\psi X_i}{k^2} \sum_{j=1}^{N} \left( \nu \Sigma \right)^j \phi_j^i \right] dV
\]

\[
+ \int \sum_{i=1}^{N} \psi_1 \left[ \sum_{j=1}^{i-1} \Sigma^i (j \rightarrow i) \phi_j^i - \sum_{j=1}^{i-1} \Sigma (j \rightarrow i) \phi_j^i \right] dV = 0 ,
\]

\[\text{(2.3)}\]

or,

\[I_1 + I_2 + I_3 + I_4 + I_5 = 0 ,\]

\[\text{(2.4)}\]

where \(I_1, I_2, I_3, I_4, \) and \(I_5\) are the individual integrals.

The next step is to expand the primed quantities in Equation (2.3) and consider each integral separately. First order perturbation theory consists of ignoring all terms higher than first order. That is,

\[\Sigma \phi^i = (\Sigma + \delta \Sigma) (\phi + \delta \phi) \approx \Sigma \phi + \phi \delta \Sigma + \Sigma \delta \phi .\]

Expanding the diffusion coefficient and fluxes to the first order and then cancelling the second term in the first integral leaves

\[I_1 = \int \sum_{i=1}^{N} \psi_1 \left[ \nabla \cdot D_i \nabla \phi_i + \nabla \cdot D_i \nabla \delta \phi_i \right] dV .\]

\[\text{(2.5)}\]

Equation (2.5) can also be written as
\[ I_1 = \delta \sum_{i=1}^{N} \psi_i [\nabla \cdot D_i \nabla \psi_i] \, dV \]  

(2.6)

To evaluate the above equation, we will make use of Green's theorem in first and second form.

\[ \int_V D \psi \cdot \nabla \psi \, dV = \int_S \psi \nabla \psi \cdot dS - \int_V \nabla \psi \cdot \nabla \psi \, dV \]  

(2.7)

and

\[ \int_V D \psi \cdot \nabla \psi \, dV = \int_S \psi D \psi \cdot dS - \int_V \psi D \psi \cdot dV \]  

(2.8)

Applying Green's theorem in second form to Equation (2.6),

\[ I_1 = \delta \sum_{i=1}^{N} \psi_i D_i \nabla \psi_i \cdot dS - \delta \sum_{i=1}^{N} D_i \nabla \psi_i \cdot \nabla \psi_i \, dV \]  

(2.9)

Now, since diffusion theory assumes that the current, \( D_i \nabla \psi_i \), is a continuous function across an interface of two dissimilar media, \( D_i \nabla \psi_i \) is constant at the surface of the volume where the perturbation occurs. Therefore, the first term in Equation (2.9) is zero. This assumes, of course, that \( \psi_i \) is also continuous across the interface. To circumvent the case when the perturbation is over the whole reactor and the interface is not between two media, we require that \( \psi_i \) vanish at the reactor boundary.
Taking the variation of the second term in Equation (2.9) gives

$$I_1 = - \int \sum_{i=1}^{N} \delta D_i \overline{\nabla \phi_i} \cdot \overline{\nabla \psi_1} dV - \int \sum_{i=1}^{N} D_i \overline{\nabla \phi_i} \cdot \overline{\nabla \psi_1} dV \ . \quad (2.10)$$

Using Green's theorem in first form in the second term of Equation (2.10),

$$I_1 = - \int \sum_{i=1}^{N} \delta D_i \overline{\nabla \phi_i} \cdot \overline{\nabla \psi_1} dV - \int \sum_{i=1}^{N} \delta \phi_i D_i \overline{\nabla \psi_1} \cdot dS + \int \sum_{i=1}^{N} \delta \phi_i \overline{\nabla} \cdot D_i \overline{\nabla \psi_1} dV \ . \quad (2.11)$$

The first term in Equation (2.9) was found to be equal to zero because the current is continuous across an interface. Similarly, the flux is a continuous function causing the surface integral in Equation (2.11) to also vanish.

Therefore, we conclude that

$$I_1 = - \int \sum_{i=1}^{N} \delta D_i \overline{\nabla \phi_i} \cdot \overline{\nabla \psi_1} dV + \int \sum_{i=1}^{N} \delta \phi_i \overline{\nabla} \cdot D_i \overline{\nabla \psi_1} dV \ . \quad (2.12)$$

The second integral in Equation (2.3) is simply

$$I_2 = - \int \sum_{i=1}^{N} (\psi_i \phi_i \delta \Sigma_a^i) dV - \int \sum_{i=1}^{N} (\psi_i \Sigma^i_a \delta \phi_i) dV \ . \quad (2.13)$$
To evaluate the fourth integral, we note that for small \( \delta k \),

\[
\frac{1}{k} = \frac{1}{k + \delta k} \approx \frac{1}{k} (1 - \frac{\delta k}{k}) . \tag{2.14}
\]

Therefore,

\[
I_4 = \int_V \frac{1}{k} \left[ \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} \phi_j \delta(\Sigma r) \right] dV
\]

\[
- \frac{\delta k}{k} \int_V \frac{1}{k} \left[ \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} (\Sigma r) \phi_j \right] dV
\]

\[
+ \int_V \frac{1}{k} \left[ \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} \delta \phi_j \right] dV . \tag{2.15}
\]

Since the third and fifth integrals are somewhat related, we shall work on them together.

\[
I_3 + I_5 = - \int_V \left[ \sum_{i=1}^{N} \psi_i \sum_{j=i+1}^{N} \phi_i \delta \Sigma (i-j) \right] dV
\]

\[
- \int_V \left[ \sum_{i=1}^{N} \psi_i \sum_{j=i+1}^{N} \Sigma (i-j) \delta \phi_i \right] dV
\]

\[
+ \int_V \left[ \sum_{i=1}^{N} \psi_i \sum_{j=1}^{i-1} \phi_j \delta \Sigma (j-i) \right] dV
\]

\[
+ \int_V \left[ \sum_{i=1}^{N} \psi_i \sum_{j=1}^{i-1} \Sigma (j-i) \delta \phi_j \right] dV . \tag{2.16}
\]
To simplify the above expression, we consider the following double summation,

\[ DS = \sum_{i=1}^{N} \sum_{j=1}^{i-1} \psi_i \Sigma(j-i) . \quad (2.17) \]

If we put \( \psi_i \) inside the second summation, reverse the order of the summation, and note that \( i > j \),

\[ DS = \sum_{j=1}^{N} \sum_{i=j+1}^{N} \psi_i \Sigma(j-i) \phi_j . \quad (2.18) \]

Finally, interchanging the indices and pulling \( \phi_i \) outside the second summation results in the following identity:

\[ \sum_{i=1}^{N} \sum_{j=1}^{i-1} \psi_i \Sigma(j-i) = \sum_{i=1}^{N} \sum_{j=i+1}^{N} \psi_j \Sigma(i-j) . \quad (2.19) \]

The above equation enables us to write Equation (2.16) as

\[ I_3 + I_5 = \int_{V} \sum_{i=1}^{N} \phi_i \sum_{j=i+1}^{N} \delta \Sigma(i-J) \left( \psi_j - \psi_i \right) dV \]

\[ + \int_{V} \sum_{i=1}^{N} \delta \Sigma_i \left( \sum_{j=i+1}^{N} \Sigma(i-J) \psi_j - \sum_{j=i+1}^{N} \Sigma(i-J) \psi_i \right) dV . \quad (2.20) \]
Substituting the values for $I_1$, $I_2$, $I_3$, $I_4$, and $I_5$ in Equation (2.4) and solving for $\delta k/k$, we get

$$\frac{\delta k}{k} = \int \left\{ \frac{N}{k} \sum_{i=1}^{N} \left[ \frac{1}{k} (\nu E_f)^i \phi_i \sum_{j=1}^{N} \chi_j \psi_j - \delta \Sigma^i \phi_i \psi_i \right] 
- \delta D_i (\overline{\nabla \phi_i} \cdot \overline{\nabla \psi_i}) + \sum_{j=i+1}^{N} \delta \Sigma (i-j) \phi_i (\psi_j - \psi_i) 
+ \delta \phi_i \left( \overline{\nabla \cdot D_i \overline{\nabla \psi_i}} - \Sigma^i \psi_i - \sum_{j=i+1}^{N} \Sigma (i-j) \psi_i + \sum_{j=i+1}^{N} \Sigma (i+j) \psi_j \right) 
+ \frac{(\nu E_f)^i}{k} \sum_{j=1}^{N} \chi_j \psi_j \right\} \overline{dV} \int \frac{1}{k} \sum_{i=1}^{N} \chi_i \psi_i \sum_{j=1}^{N} (\nu E_f)^j \phi_j \overline{dV}. \n(2.21)$$

Now, we note that if the coefficient of $\delta \phi_i$ is equal to zero, then the variation in the flux doesn't contribute to the perturbation equation. Therefore, we define the weighting function as

$$\overline{\nabla \cdot D_i \overline{\nabla \psi_i}} - \Sigma^i \psi_i - \sum_{j=i+1}^{N} \Sigma (i-j) \psi_i + \sum_{j=i+1}^{N} \Sigma (i+j) \psi_j 
+ \frac{(\nu E_f)^i}{k} \sum_{j=1}^{N} \chi_j \psi_j = 0 \n(2.22)$$
Observe that this is simply the adjoint equation. The relationship between the diffusion equation, Equation (2.1), and the adjoint equation is easily understood since if one writes the diffusion equation in matrix form,

\[ \bar{M} \bar{\phi} = 0, \]

the adjoint equation can be written as

\[ \bar{M}^T \bar{\psi} = 0, \]

where \( \bar{M}^T \) is the transpose of \( \bar{M} \). It should be mentioned that, unlike the flux, the adjoint flux is dimensionless.

It is easy to show the physical significance of the adjoint flux. To do this, let \( \xi_i \) be the worth of a neutron in energy group \( i \), where we have again not explicitly shown the spatial dependence.

The neutron worth, or total number of neutrons produced by inserting one neutron in energy group \( i \) is

\[
\xi_i = \frac{(\nu \Sigma_f)^i \phi_i + \sum_{j=1}^{N} x_j \xi_j + \sum_{j=i+1}^{N} \Sigma_{i+j} \phi_i \xi_j}{-\nabla \cdot D_i \nabla \phi_i + \sum_{j=i+1}^{N} \Sigma_{i+j} \phi_i},
\]

where the first term in the numerator is the number of neutrons emitted from fission multiplied by the worth of an averaged fission neutron, \( \sum_{j=1}^{N} x_j \xi_j \), and the second term is the scattering into group \( j \)
multiplied by the worth in \( j \). The denominator is just the loss rate.

We now clear Equation (2.23) of fractions, integrate over the volume of the reactor, and use Green's theorem on the leakage term. Since the flux and adjoint flux vanish on the boundary, we can cancel \( \phi_i \) from both sides and write

\[
\frac{\nabla \cdot D_i \nabla \phi_i}{\phi_i} - \sum_{a} \phi_a - \sum_{j=i+1}^{N} \Sigma_{i-j} \phi_i + \sum_{j=i+1}^{N} \Sigma_{i-j} \phi_j + \sum_{j=1}^{N} \chi_j \phi_j = 0.
\]

(2.24)

Comparing Equation (2.24) with Equation (2.22), \( \xi_i = \psi_i \) and the weighting function is simply the neutron worth or adjoint flux.

With this interpretation of the adjoint flux, it is now possible to assign a physical meaning to each of the four components of Equation (2.21), the perturbation equation. Changes in the neutron fission source rate are weighted by the worth of an average fission neutron while changes in the absorption rate are weighted by the neutron worth at the point of capture. The third term is the worth due to a change in the leakage rate.

Note that since \( D = \frac{1}{2F_{tr}} \),
\[ \delta D_i = - \frac{1}{3(\Sigma^i_{\text{tr}})^2} \delta (\Sigma^i_{\text{tr}}), \]

where \( \Sigma^i_{\text{tr}} \) = macroscopic transport cross section in group \( i(\text{cm}^{-1}) \). Finally, as seen in the fourth term, variations in \( \Sigma(i+j) \) are weighted by the worth in group \( i \) minus the worth in group \( j \).
The time dependent diffusion equation for energy group \( i \) with NP delayed precursor groups can be written as

\[
\frac{1}{v_i} \frac{\partial \phi_i}{\partial t} = \nabla \cdot \left( D_i \nabla \phi_i \right) - \Sigma^s_i \phi_i - \Sigma^r_i \phi_i + x_i \sum_{j=1}^{N} \frac{\phi_j}{\Sigma_f^j} \sum_{m=1}^{ND} \beta_m (\nu \Sigma_f^m)^j + x_i \sum_{k=1}^{NP} \lambda_k C_k
\]

\[
+ \sum_{j=1}^{i-1} \Sigma_j \phi_j, \quad (3.1)
\]

where:

\( \phi_i \) = flux in group \( i \) \((n \cdot cm^{-2} \cdot sec^{-1})\),

\( v_i \) = neutron velocity in energy group \( i \) \((cm/sec)\),

\( \Sigma^s_i \) = macroscopic scattering removal cross section in group \( i \) \((cm^{-1})\),

\( \beta_m \) = number of delayed neutrons/fission neutrons for fissionable isotope \( m \),

\( ND \) = number of fissionable isotopes,
NP = number of precursor groups,

\((\nu E_f)_m^i\) = macroscopic fission source cross section in group 

\(i\) for fissionable isotope \(m\) (cm\(^{-1}\)),

\(\chi_i^D\) = delayed fission source born in group \(i\)

\[\sum_{i=1}^{N} \chi_i^D = 1.0,\]

\(\lambda_K\) = decay constant for delayed precursor group \(K\)

\(\text{sec}^{-1}\),

and \(C_K\) = density for delayed precursor group \(K\) (cm\(^{-3}\)).

All other symbols in the above equation were defined after

Equation (2.1) in the previous section. It is obvious from

Equation (3.1) that the flux and precursor density are functions

of both spatial position and time. Note the assumption is made

that \(\chi_i^D\) is independent of the delayed precursor group.

The differential equation describing the time behavior

of the \(k\)th precursor group is

\[
\frac{\partial C_K}{\partial t} = \sum_{j=1}^{N} \phi_j \sum_{m=1}^{ND} \beta_m^K (\nu \Sigma_f)_m^j - \lambda_K C_K,
\]

where:

\(\beta_m^K\) = delayed neutron fraction for precursor group \(K\)

for fissionable isotope \(m\) \((\beta_m = \frac{\sum_{K=1}^{NP} \beta_m^K}{\sum_{K=1}^{NP} \beta_m^K})\).
From Section II, we saw that the time independent adjoint equation for energy group \(i\) is

\[
0 = \nabla \cdot D_i \nabla \psi_i - \Sigma_{a}^{i} \psi_i - \Sigma_{r}^{i} \psi_i + \sum_{j=i+1}^{N} \Sigma(i\rightarrow j) \psi_j \\
+ \frac{(\nu E_f)^i}{k} \sum_{j=1}^{N} x_j \psi_j .
\]  

(3.3)

Equation (3.1) is now multiplied by the adjoint flux and Equation (3.3) by the flux, both equations summed over all groups and integrated over the volume of the reactor. Subtracting the second resulting integral equation from the first,

\[
\int \sum_{i=1}^{N} \psi_i \frac{\partial \phi_i}{\partial t} \, dV = \int \sum_{i=1}^{N} \psi_i \nabla \cdot D_i \nabla \phi_i \, dV - \int \sum_{i=1}^{N} \psi_i \nabla \cdot D_i \nabla \psi_i \, dV \\
- \int \sum_{i=1}^{N} \psi_i \Sigma_{r}^{i} \phi_i \, dV + \int \sum_{i=1}^{N} x_i \psi_i \int_{j=1}^{N} \frac{\nu E_f}{j} \phi_j \, dV \\
- \int \sum_{i=1}^{N} \psi_i \Sigma_{a}^{i} \phi_i \, dV + \int \sum_{i=1}^{N} x_i \psi_i \int_{j=1}^{N} \frac{\nu E_f}{j} \phi_j \, dV \\
\int \sum_{i=1}^{N} \psi_i \int_{j=1}^{i-1} \Sigma(j\rightarrow i) \phi_j \, dV - \int \sum_{i=1}^{N} \phi_i \nabla \cdot D_i \nabla \psi_i \, dV \\
+ \int \sum_{i=1}^{N} \phi_i \Sigma_{a}^{i} \psi_i \, dV + \int \sum_{i=1}^{N} \phi_i \Sigma_{r}^{i} \psi_i \, dV - \int \sum_{i=1}^{N} \phi_i \int_{j=1}^{i+1} \Sigma(i\rightarrow j) \psi_j \, dV \\
- \int \sum_{i=1}^{N} \frac{\nu E_f}{i} \phi_i \int_{j=1}^{N} x_j \psi_j \, dV .
\]  

(3.4)
To simplify Equation (3.4), we shall utilize the following relationships.

Applying Green's theorem,

$$\int_V \sum_{i=1}^N \phi_i \nabla \cdot D_i \nabla \psi_i \, dV = \int_V \sum_{i=1}^N \psi_i \nabla \cdot D_i \nabla \phi_i \, dV$$

$$+ \int_V \sum_{i=1}^N D_i (\phi_i \nabla \psi_i - \psi_i \nabla \phi_i) \cdot d\Omega \quad \text{(3.5)}$$

Since the flux and adjoint flux both vanish at the reactor boundary, the second integral on the right side of the above equation is zero and we can write

$$\int_V \sum_{i=1}^N \phi_i \nabla \cdot D_i \nabla \psi_i \, dV = \int_V \sum_{i=1}^N \psi_i \nabla \cdot D_i \nabla \phi_i \, dV \quad \text{(3.6)}$$

Defining $\rho = \text{reactivity} = \frac{k-1}{k}$, we have

$$\int_V \sum_{i=1}^N x_i \psi_i \sum_{j=1}^N (\nu E_i^2)^j \phi_j \, dV = \int_V \sum_{i=1}^N \frac{(\nu E_i^2)^i \phi_i}{k} \sum_{j=1}^N x_j \psi_j \, dV$$

$$= \rho \int_V \sum_{i=1}^N x_i \psi_i \sum_{j=1}^N (\nu E_i^2)^j \phi_j \, dV \quad \text{(3.7)}$$
Substituting Equations (2.19), (3.6), and (3.7) into Equation (3.4) gives, after cancellation,

\[ \int \sum_{i=1}^{N} \frac{\psi_i}{V_1} \frac{\partial \phi_i}{\partial t} \, dV = \rho \int \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} (\nu \Sigma_f)_{j} \phi_j \, dV - \int \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} \phi_j \sum_{m=1}^{ND} \beta_m (\nu \Sigma_f)_{m} \, dV + \int \sum_{i=1}^{N} x_i \psi_i \sum_{K=1}^{NP} \lambda_K \, dV \]  

(3.8)

We now multiply Equation (3.2) by \( \sum_{i=1}^{N} x_i \psi_i \) and integrate over the volume of the reactor.

\[ \int \sum_{i=1}^{N} x_i \psi_i \frac{\partial C_K}{\partial t} \, dV = \int \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} \phi_j \sum_{m=1}^{ND} \beta_m (\nu \Sigma_f)_{m} \, dV - \int \sum_{i=1}^{N} x_i \psi_i \lambda_K \, dV \]  

(3.9)

What is now desired is to reduce Equations (3.8) and (3.9) to the conventional form of the reactor kinetics equations,

\[ \frac{dT(t)}{dt} = \frac{\rho - \beta}{A} T(t) + \sum_{K=1}^{NP} \frac{\lambda_K C_K}{A} , \]

(3.10) and

\[ \frac{dC_K(t)}{dt} = \frac{\beta}{A} T(t) - \lambda_K C_K . \]

(3.11)
As noted earlier, the flux in Equation (3.8) is a function of both space and time. For a homogeneous reactor, the spatial and time variables are separable and one can speak of the time behavior for the reactor as a whole. For a zoned reactor, however, the variables are not separable and it is necessary to express the flux as the product of two functions.

\[ \phi_i(r,t) = \phi_i(r,t)T(t) \quad (3.12) \]

At first glance, it doesn't appear that this helps much. However, it is assumed that the nonseparable portion of the flux, \( \phi_i(r,t) \) becomes independent of time after a sufficiently large time following the perturbation. That is,

\[ \lim_{t \to \infty} \phi_i(r,t) = \phi_i(r) \equiv \phi_i \quad (3.13) \]

Therefore, \( T(t) \) represents the asymptotic time behavior of the flux. A discussion of the transient neutron flux for a zoned reactor is given by Crawford, (5, p. 41-56) where, using diffusion theory, he expands the flux as a sum of model fluxes and writes the kinetics equations for each mode.

As a consequence of Equation (3.13), we write

\[ \int \sum_{i=1}^{N} \frac{\psi_i}{v_i} \frac{\partial \phi_i}{\partial t} \, dV = \int \sum_{i=1}^{N} \frac{\psi_i}{v_i} \phi_i \frac{dT(t)}{dt} \, dV \]

\[ = \frac{dT(t)}{dt} \int \sum_{i=1}^{N} \frac{\psi_i \phi_i}{v_i} \, dV \quad (3.14) \]
Let

\[ I = \int \sum_{i=1}^{N} x_i \psi_i \sum_{j=1}^{N} (\nu \Sigma_f)^j \phi_j \, dV . \]  

(3.15)

Now, we make the following definitions:

\[ \Lambda = \int \sum_{i=1}^{N} \frac{\psi_i \phi_i}{V} \, dV \]

\[ \Lambda = \frac{\psi_i \phi_i}{V} \, dV \]

(3.16)

\[ \beta_{\text{eff}} = \int \sum_{i=1}^{N} \frac{\psi_i \psi_i}{V} \sum_{j=1}^{N} \phi_j \sum_{m=1}^{N D} \beta_m (\nu \Sigma_f)^j \, dV \]

(3.17)

\[ \beta_{\text{eff}} = \sum_{i=1}^{N} \frac{\psi_i \psi_i}{V} \sum_{j=1}^{N D} \beta_m (\nu \Sigma_f)^j \, dV \]

(3.18)

The next step is to substitute Equations (3.12), (3.13), (3.14), (3.15), (3.16), (3.17) and (3.18) into Equations (3.8) and (3.9). Equation (3.8) reduces to

\[ \lambda I \frac{dT(t)}{dt} = \rho I T(t) - \beta_{\text{eff}} I T(t) + \Lambda I \sum_{K=1}^{N P} \lambda_K C_K(t) \]

\[ + \sum_{j=1}^{N D} \phi_j \sum_{m=1}^{N D} \beta_m (\nu \Sigma_f)^j \left( \sum_{i=1}^{N} \frac{\psi_i \psi_i}{V} - \sum_{i=1}^{N} \frac{\chi_i \psi_i}{V} \right) \, dV . \]  

(3.19)
Equation (3.9) reduces to

\[ \frac{dC_K(t)}{dt} = \beta^K_{\text{eff}} IT(t) - \lambda_K C_K(t) \Lambda_I \]  

(3.20)

The last integral in Equation (3.19) is simply the difference in worth between the delayed neutrons and an equivalent number of prompt neutrons. Since the number of delayed neutrons is small, and the difference in their worths is also small, the product is a very small number and can be dropped for most purposes. Dividing Equations (3.19) and (3.20) by \( \Lambda_I \),

\[ \frac{dT(t)}{dt} = \frac{\rho - \beta_{\text{eff}}}{\Lambda} T(t) + \sum_{K=1}^{NP} \lambda_K C_K(t) \]  

(3.21)

\[ \frac{dC_K(t)}{dt} = \frac{\beta^K_{\text{eff}}}{\Lambda} T(t) - \lambda_K C_K(t) \]  

(3.22)

The expressions for beta effective and neutron generation time in Equations (3.16) and (3.17), therefore, result in equations in the same form as the kinetics equations, Equations (3.10) and (3.11). There are several obvious similarities between the equations for beta effective and neutron generation time and the perturbation equation derived in the previous section. For example, I appears in all of the denominators, and the steady-state flux and adjoint flux appear in all expressions.
## IV. GENERAL DESCRIPTION OF PERT-IV

### Limits on Problem Complexity

PERT-IV is written in FORTRAN-IV for use on the UNIVAC 1107 or 1108. The maximum value of the input variables are as follows:

<table>
<thead>
<tr>
<th>Variable</th>
<th>Definition</th>
<th>Maximum Value</th>
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</thead>
<tbody>
<tr>
<td>IGM</td>
<td>No. of Energy Groups</td>
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<td>IST</td>
<td>No. of Downscattering Terms</td>
<td>10</td>
</tr>
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<td>IM</td>
<td>No. of Radial Intervals</td>
<td>50</td>
</tr>
<tr>
<td>JM</td>
<td>No. of Axial Intervals</td>
<td>50</td>
</tr>
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<td>MCR</td>
<td>No. of Cross Sections from Cards</td>
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</tr>
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<td>MTP</td>
<td>No. of Cross Sections from Tape</td>
<td>25</td>
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<tr>
<td>IZM</td>
<td>No. of Material Zones</td>
<td>35</td>
</tr>
<tr>
<td>MT</td>
<td>No. of Materials Including Mixes</td>
<td>35</td>
</tr>
<tr>
<td>MOL</td>
<td>No. of Mixture Specifications</td>
<td>45</td>
</tr>
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<td>NDELK</td>
<td>No. of Reactivity Traverses</td>
<td>20</td>
</tr>
<tr>
<td>NIBC</td>
<td>No. of Isotopes in $\beta_{\text{eff}}$ Calculation</td>
<td>20</td>
</tr>
<tr>
<td>NACT</td>
<td>No. of Activity Traverses</td>
<td>3</td>
</tr>
</tbody>
</table>

A 65K core and 4 scratch tapes are required for the above specifications. If a smaller core must be used, either reducing the number of radial and axial intervals or the number of groups is an easy way to decrease storage. Running time for a typical problem (5 groups, 20 reactivity traverses) is about 20 seconds on the 1108.
Format of Input Fluxes and Adjoint Fluxes

The input flux and adjoint flux options are:

- Flux and adjoint from 1-D
- Flux and adjoint from 2-D
- Flux from 1-D, adjoint from 2-D
- Flux from 2-D, adjoint from 1-D
- Flux from 1-D, no adjoint
- Flux from 2-D, no adjoint.

When 1-D flux dumps are used, PERT-IV synthesizes these into 2-D arrays by assuming that the fluxes and adjoint fluxes are separable in the two dimensions over the whole reactor. For the last two options, the code assumes a flat adjoint distribution. Flux input can be taken directly from DTF-IV or 2DF, although other diffusion or transport theory codes may be used to provide fluxes and adjoint fluxes.

Flux Normalization Options

Four options are available for normalizing the flux. These are:

- Center point flux
- Center point power density
- Total reactor power
- Denominator of perturbation equation (see Equation 2.21).

The adjoint flux is always normalized such that the fission source times the adjoint, summed over all groups, is unity at the center mesh point.
Reactivity Coefficient Calculations

PERT-IV will calculate reactivity coefficient (in units of $\delta k/k$ per kg) traverses for any specified row or column in R-Z, X-Y, or R-Θ geometry. The contribution of each component of the reactivity coefficient (e.g., moderation, capture) is also given.

Neutron Generation Time and Beta Effective Calculations

Using Equations (3.16) and (3.17), PERT-IV calculates the neutron generation time and the effective delayed neutron fraction. Additional input includes the neutron velocities for each group, the absolute number of neutrons per fission for each fissionable isotope ($= \beta v$), and the delayed fission source. A single delayed fission source distribution is assumed for all delayed groups and isotopes.

Activity Traverses

For any material and cross section (e.g., $\sigma_f$, $\sigma_a$), the code will calculate and print an activity map. A full print gives the activity at each mesh point for each group; a partial print gives the total over all groups at each mesh point. Zone averaged activities are also given.

Other Options

Minor options in PERT-IV include:

- Option to punch either flux or adjoint flux, or both.
• Full print option to print flux, adjoint flux and activity traverses for each group and an extended print out of cross sections.

• Option to read in a value for $k_{\text{eff}}$ to be used in reactivity coefficient calculations—useful when $k_{\text{eff}}$ as calculated by PERT-IV is significantly different from the transport theory value.

**Typical Results**

A plot of the axial worth of Pu239 in a typical fast reactor as calculated by PERT-IV is shown in Figure 4.1. The graph also includes each component to the total reactivity coefficient. As one would expect, the magnitude of the contribution due to fissions, absorptions, and moderations is a maximum in the center of the core and decreases to the edge of the reactor. The leakage component, however, is zero at the center, reaches a maximum at the core-reflector interface, and then decreases to the reactor edge.
Figure 4.1. Axial worth curve of Pu-239 in ZPR-3 Assembly 48
V. COMPARISON BETWEEN DIFFUSION PERTURBATION THEORY AND TRANSPORT THEORY

Since the best way to assess diffusion perturbation theory is to compare results with a two-dimensional transport theory code, reactivity coefficients were computed using both PERT-IV and 2DF for a variety of materials and energy groups for a typical fast reactor core. Such a comparison is necessarily incomplete, however, because of the numerous items to check and the long time required for a 2-D transport theory calculation.

Calculations were performed using R-Z geometry for ZPR-III Assembly 48, *(6, p. 57-64)*, a zero power fast critical assembly located at Idaho Falls. ZPR-III Assembly 48 was a 417 liter, plutonium fueled core, cylindrical in shape, and surrounded by a 30 cm thick reflector in both the axial and radial directions. The reactor dimensions and composition are given in Figure 5.1 and Table 5.1.

![Figure 5.1. Reactor dimensions for ZPR-3 Assembly 48](image-url)
Cross section input consisted of a modified version of the Russian data compilation. The data was punched using FCC, a fundamental mode fast reactor code that will punch an n group Russian cross section set, where $26 \geq n \geq 1$.

The calculated reactivity coefficients in units of $(\delta k/k)$ per kg in the center of ZPR-3 Assembly 48 (R-Z geometry) are shown in Table 5.2. To negate the effect of sample size, most of the 2DF
Table 5.2
Calculated Reactivity Coefficients
in the Center of 2PR-3 Assembly 48

<table>
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<th>Row</th>
<th>Number of Groups</th>
<th>Code</th>
<th>Order of SN Approximation</th>
<th>Pu239</th>
<th>Pu240</th>
<th>U235</th>
<th>Na</th>
<th>Fe</th>
<th>Bi10</th>
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<td>2DF</td>
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<td>2DF</td>
<td>S4</td>
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<td>-</td>
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<td></td>
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<td>-</td>
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<td>S2</td>
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</tbody>
</table>

*Units are (6k/k) per kg x 10³*
reactivity coefficient cases were run twice—with the sample in the test zone for both a positive and negative density and then the two averaged.

Rows 7 and 8 give a direct comparison between 2DF and PERT-IV calculations for several materials using two groups and the S2 option. The discrepancy between transport theory and diffusion perturbation theory is less than 4% for the fuels and the poison; for Na and Fe, however, the error is larger because their worths are the difference between two contributions (absorption and moderation) of the same magnitude. As an interesting sidelight, Los Alamos has written a one-dimensional transport perturbation theory code, DAC, compatible with DTF-IV, and report similar differences in worths calculated by DAC and DTF-IV similar to those between PERT-IV and 2DF.

To determine whether the agreement is a function either of the number of energy groups or the order of the SN approximation, Pu239 worths were calculated using the S2, S4, and S6 options for 1 group, the S2 and S4 options for 2 groups, and the S2 option for 5 groups. A plot of this data can be seen in Figure 5.1. The percent error does not appear dependent upon the number of energy groups; however, the discrepancy decreases somewhat for higher orders of the SN approximation for both the one group and two group worth calculations. It is interesting to note that the 2DF value seems to be converging towards the PERT-IV value.
Figure 5.1. Comparison between PERT-IV, 2DF, and 2DB calculations in ZPR-3 Assembly 48
The worth of Pu239 was calculated using the two-dimensional diffusion theory program, 2DB, and compared with PERT-IV (using 2Db flux dumps). These results are shown in rows 11 and 12. As one would expect, since both codes are based on diffusion theory, the agreement is very good. Therefore, we conclude that inaccuracies in PERT-IV worths when compared to 2DF is because PERT-IV is based on the diffusion model.

The last two rows of Table 5.2 are PERT-IV calculations using one-dimensional DTF-IV fluxes. For this particular core there appears to be good agreement between PERT-IV calculations using DTF-IV fluxes and those using 2DF fluxes. It is not expected that this agreement will be as good for a multi-zoned reactor, however, since the assumption that the fluxes and adjoint fluxes are separable over the whole reactor will be less valid.

Recapitulating the above paragraphs, for the reactor studied, worths calculated using diffusion perturbation theory agree well with transport theory--especially fuel and poison worths. The agreement does not appear to be strongly dependent either on the number of groups or the order of the SN approximation in 2DF. If a two-dimensional diffusion theory code is used, the agreement is very good. Finally, using 1-D transport theory fluxes (radial and axial) in the diffusion perturbation code agrees fairly well with 2-D transport theory fluxes.
VI. SUMMARY AND CONCLUSIONS

This thesis has described a two-dimensional, first order perturbation theory computer code based on the multigroup diffusion model that will compute reactivity coefficient traverses, the neutron generation time, and the effective delayed neutron fraction. The derivations of the perturbation equation and of the expressions for the neutron generation time and effective delayed neutron fraction as used in the code were given. Finally, some comparisons between reactivity coefficients calculated by diffusion perturbation theory and transport theory were shown.

Given the type of errors shown in Table 5.2, one must now ask if PERT-IV is a useful method for calculating reactivity coefficients. One criterion could be that the discrepancy between PERT-IV and 2DF worths should be less than the discrepancy between two 2DF calculations where the number of groups or the order of the SN approximation is varied. For the reactor studied in this report, this criterion is certainly met for Pu239 since there is over a 10% change from a 1 group to a 5 group 2DF calculation and a 5% change from S2 to S4, while the difference between PERT-IV and 2DF worths is less than 4%.


11. Shapiro, Martin. 2DF, a two-dimensional $S^H$ code. Computer code received from Los Alamos Scientific Laboratory, 1965. (Unpublished)


APPENDIX I: INPUT INSTRUCTIONS

The following two pages describe the input data for PERT-IV.

For DTF-IV fluxes and all subsequent data, the cards consist of six data fields of 12 columns each. The last nine columns contain the data associated with the particular field; columns 2-3 contain an integer, N, from 0 (or blank) to 99. The first column of each field must contain:

0 - no effect (N=0),

1 - repeat associated data entry N times,

2 - do N linear interpolations between associated data entry and succeeding data entry,

3 - terminate reading of this array with previous data entry.
**Identification Card**

<table>
<thead>
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<th>ADE</th>
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<th>2</th>
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<th>3</th>
<th>Value from 3</th>
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**Flux and Energy Data**

| ZONE NO | MATERIAL NO | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 | 15 | 16 | 17 | 18 | 19 | 20 | 21 | 22 | 23 | 24 | 25 | 26 | 27 | 28 | 29 | 30 | 31 | 32 | 33 | 34 | 35 | 36 | 37 | 38 | 39 | 40 | 41 | 42 | 43 | 44 | 45 | 46 | 47 | 48 | 49 | 50 | 51 | 52 | 53 | 54 | 55 | 56 | 57 | 58 | 59 | 60 | 61 | 62 | 63 | 64 | 65 | 66 | 67 | 68 | 69 | 70 | 71 | 72 | 73 | 74 | 75 | 76 | 77 | 78 | 79 | 80 | 81 | 82 | 83 | 84 | 85 | 86 | 87 | 88 | 89 | 90 | 91 | 92 | 93 | 94 | 95 | 96 | 97 | 98 | 99 | 100 | 101 | 102 | 103 | 104 | 105 | 106 | 107 | 108 | 109 | 110 | 111 | 112 | 113 | 114 | 115 | 116 | 117 | 118 | 119 | 120 | 121 | 122 | 123 | 124 | 125 | 126 | 127 | 128 | 129 | 130 | 131 | 132 | 133 | 134 | 135 | 136 | 137 | 138 | 139 | 140 | 141 | 142 | 143 | 144 | 145 | 146 | 147 | 148 | 149 | 150 | 151 | 152 | 153 | 154 | 155 | 156 | 157 | 158 | 159 | 160 | 161 | 162 | 163 | 164 | 165 | 166 | 167 | 168 | 169 | 170 | 171 | 172 | 173 | 174 | 175 | 176 | 177 | 178 | 179 | 180

**Geometry**

- **R1**
  - Center Pl. Flux
  - Center M. Power Density
  - Total Power
- **R2**
  - Normalized Perturbation Formula
- **R3**
  - Calculations

**Miscellaneous**

- **I**
  - Partial Print
- **F**
  - Full Print
APPENDIX II: SAMPLE PROBLEM

The following pages show the input data (omitting input fluxes) and computer output for a two-group, three-region test problem in R-Z geometry.
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## PERT-IV TEST CASE

### (2D Flux Input, 2 Groups, 3 Zones)

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*Flux (n/cm^2-sec) or power (MW)*

*Release per fission*
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### PFRT-IV TEST CASE

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Mesh Boundaries:
- R1/Z1 = Radial Points/ Axial Points

- MESH Boundaries (R1/Z1 = Radial Points / Axial Points)

- MESH Boundaries (R1/Z1 = Radial Points / Axial Points)
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Fission Spectrum

Chi 0

.98700-00 .13000-01

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DELAYED FISSION SPECTRUM

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PERTURBATION CALCULATION INFORMATION (MATUK/NROW/NCOL=MATERIAL NUMBER/R04/COLUMN TO BE PERTURBED)

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ACTIVITY TRAVERSE INFORMATION (MA/NX=MATERIAL NUMBER/CROSS SECTION POSITION)

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PERT-IV TEST CASE
(2DF FLUX INPUT, 2 GROUPS, 5 ZONES)
## PERT-IV TEST CASE

### (20F FLUX INPUT, 2 GROUPS, 3 ZONES)

### CROSS-SECTION EDIT

#### GROUP 1 CROSS-SECTIONS

| MAT 1 | 1.72+01 | 1.85+01 | 3.12+01 | 2.69+01 | 3.67+01 | 4.77+01 | 0.001 |
| MAT 2 | 6.97+03 | 4.97+03 | 0.21+01 | 6.34+01 | 5.99+01 | 0.001 |
| MAT 3 | 1.32+01 | 1.51+01 | 3.32+01 | 6.23+01 | 4.87+01 | 0.001 |
| MAT 4 | 1.00+00 | 2.33+03 | 0.28+00 | 6.34+01 | 6.01+01 | 0.001 |
| MAT 5 | 0.000 | 8.34+05 | 0.000 | 2.84+01 | 2.45+01 | 0.001 |
| MAT 6 | 7.13+03 | 0.000 | 3.09+01 | 3.00+01 | 0.001 |
| MAT 7 | 0.000 | 5.92+02 | 0.000 | 2.56+01 | 2.52+01 | 0.001 |
| MAT 8 | 0.000 | 4.54+02 | 0.000 | 2.67+01 | 2.61+01 | 0.001 |
| MAT 9 | 5.05+01 | 0.000 | 3.08+01 | 3.00+01 | 0.001 |
| MAT 10 | 4.31+01 | 0.000 | 3.27+01 | 5.16+01 | 0.001 |
| MAT 11 | 0.000 | 2.05+02 | 0.000 | 2.83+01 | 2.73+01 | 0.001 |
| MAT 12 | 1.31+01 | 1.57+01 | 3.34+01 | 6.75+01 | 5.06+01 | 0.001 |
| MAT 13 | 3.72+01 | 1.90+00 | 1.05+00 | 6.95+01 | 6.66+01 | 0.001 |
| MAT 14 | 0.000 | 6.71+02 | 0.000 | 2.85+01 | 2.60+01 | 0.001 |
| MAT 15 | 0.000 | 4.83+02 | 0.000 | 2.76+01 | 2.67+01 | 0.001 |
| MAT 16 | 0.000 | 1.82+01 | 0.000 | 1.53+01 | 1.31+01 | 0.001 |
| MAT 17 | 3.69+02 | 5.03+02 | 1.08+01 | 1.71+00 | 1.60+00 | 0.001 |
| MAT 18 | 1.60+02 | 7.75+02 | 4.49+02 | 2.95+01 | 2.83+00 | 0.001 |
| MAT 19 | 3.69+02 | 5.03+02 | 1.08+01 | 1.71+00 | 1.60+00 | 0.001 |

#### GROUP 2 CROSS-SECTIONS

<p>| MAT 1 | 2.88+01 | 3.24+01 | 0.50+01 | 1.30+02 | 1.07+02 | 7.90+01 |
| MAT 2 | 2.09+01 | 1.58+01 | 3.79+01 | 1.39+02 | 1.24+02 | 7.49+01 |
| MAT 3 | 3.36+01 | 4.87+01 | 0.15+01 | 1.47+02 | 0.98+01 | 1.14+00 |
| MAT 4 | 9.000 | 5.32+00 | 0.000 | 1.52+02 | 1.26+02 | 0.97+01 |
| MAT 5 | 0.000 | 4.57+17 | 0.000 | 4.69+01 | 4.49+01 | 1.16+00 |
| MAT 6 | 0.000 | 4.23+02 | 0.000 | 4.96+01 | 4.98+01 | 0.99+01 |
| MAT 7 | 9.000 | 2.15+01 | 0.000 | 4.82+01 | 4.80+01 | 0.34+01 |
| MAT 8 | 0.000 | 1.56+01 | 0.000 | 4.72+01 | 4.71+01 | 0.56+01 |
| MAT 9 | 0.000 | 2.04+01 | 0.000 | 9.86+01 | 9.43+01 | 0.42+01 |
| MAT 10 | 0.000 | 3.03+00 | 0.000 | 7.94+01 | 7.59+01 | 0.70+01 |
| MAT 11 | 9.000 | 8.32+02 | 0.060 | 2.17+01 | 2.16+01 | 0.96+01 |
| MAT 12 | 2.12+01 | 2.84+01 | 0.15+01 | 1.23+02 | 0.98+01 | 1.16+00 |
| MAT 13 | 0.000 | 4.05+00 | 0.000 | 1.22+02 | 1.16+02 | 0.12+00 |
| MAT 14 | 0.000 | 1.28+01 | 0.000 | 4.50+01 | 4.49+01 | 0.45+01 |
| MAT 15 | 0.000 | 9.24+02 | 0.000 | 8.41+01 | 8.40+01 | 0.79+01 |
| MAT 10 | 0.000 | 1.69-01 | 1.900 | 7.50+91 | 7.94+01 | 5.46-01 |
| MAT 17 | 3.03-02 | 9.92-02 | 1.12-01 | 3.21-06 | 3.31-00 | 4.91-02 |
| MAT 19 | 1.73-03 | 1.05-01 | 4.23-03 | 5.10-06 | 5.02-00 | 4.83-02 |
| MAT 19 | 3.33-02 | 9.92-02 | 1.20-01 | 3.21-06 | 3.11-00 | 5.91-02 |</p>
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### PERT-IV TEST CASE

#### ONE FLUX INPUT, 2 GROUPS, 3 ZONES

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### Note

- The table provides the average radii and axial coordinates for different PERT-IV test cases.
- The data is divided into groups and zones, indicating a structured approach to the analysis.
## PERT-IV Test Case

(2Dc Flux Input, 2 Groups, 3 Zones)

### Material Inventory (Kilograms) for Each Zone

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**Reference:** (2DF FLUX INPUT, 2 GROUPS, 3 ZONES)

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PERT-IV TEST CASE (2DF FLUX INPUT, 2 GROUPS, 3 ZONES)

BETA EFFECTIVE - - - - - - 4.20547-03
NEUTRON GENERATION TIME - - - - 3.02887-07
### Reactivity Coefficients for Material 1: M(247U,9) Row 1

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### PERT-IV TEST CASE

(20F FLUX INPUT, 2 GROUPS, 3 ZONES)

**Reactivity Coefficients for Material b**

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</tbody>
</table>
APPENDIX III: LISTING OF PERT-IV
REAL I2
DIMENSION A(25,10), A0(50), A1(50), A2(50), ADJ(50, 50), ATW(25),
1 BETA(20), C(15, 26, 35), C0(15, 35), CHI(26), CHIP(26), DUM(2500),
2 HOLN(25), PMHC(26), PHI(50, 50), PHR(50, 50), PHZ(50, 50), R1(51),
3 R4(50), R5(50), SLOPR(50), SLOPRZ(50), SORC1(50, 50), SORC2(50, 50),
4 V0(50, 50), VEL(26), VOL(40), Z1(51), Z4(50), Z5(50), ZACT(40, 3),
5 ZPHI(40), ZPOW(40)
DIMENSION IO(45), I1(45), I2(45), ID(12), L2(40), LDUM(2500),
1 MU(50, 50), M2(40), MA(15), MATDK(20), NBET(20), NCOL(20),
2 NROW(20), NX(15)
COMMON NINP, NOUT, NCR1, NCR2, NCRAT, NCR2, IFFP, IGM, IGE, IGEP, IM-
1 IMJM, IP, IST, ITL, IZM, JM, JP, MOL,
2 MCR, MT, MTP, NACT, ND, NDELT, NFP, NIBC, NPRT
COMMON AK, DENOM, FEF, FLPO, VF, ZKEFF
COMMON A, A0, A1, A2, ADJ, ATW, BETA, C, C0, CHI, CHIP, DUM,
1 HOLN, PHC, PHI, PHR, PHZ, R1, R4, R5, SLOPR, SLOPRZ, SORC1, SORC2,
2 V0, VEL, VOL, Z1, Z4, Z5, ZACT, ZPHI, ZPOW
COMMON IU, IL, I2, ID, L2, LDUM, M0, M2, MA, MATDK, NBET, NCOL,
1 NROW, NX
END
ITC for MAIN, MAIN

INCLUDE ABC

MAIN MAIN PROGRAM SETS UP TAPE UNITS AND CALLS INP1, MAPR, SETUP, GRAM, NORM, ABETA, AND CALC.

INP1 SUBROUTINE TO CONTROL THE READING AND PRINTING OF ALL INPUT DATA. INP1 IS CALLED BY MAIN AND CALLS XSINP, FXINP, REAG2, AND REAI2.

XSINP XSINP READS, CHECKS, PRINTS, AND WRITES THE CROSS SECTIONS TO TAPE. THE SUBROUTINE IS CALLED BY INP1.

FXINP SUBROUTINE TO READ AND PRINT THE INPUT FLUXES. IF FLUXES ARE IN 1-D FORM, THEY ARE COMBINED RESULTING IN THE 2-D FORM. INP1 CALLS FXINP, REAG2, AND REAI2 AND IS CALLED BY MAIN.

REAI2 REAI2 READS ALL INTEGER DATA AFTER CARD 4. IT IS CALLED BY FXINP AND INP1.

REAG2 REAG2 READS ALL REAL DATA AFTER THE CROSS SECTIONS EXCEPT FOR THE 2-D FLUXES. IT IS CALLED BY FXINP AND INP1.

MAPR SUBROUTINE TO PRODUCE A PICTURE PRINT BY ZONE AND BY MATERIAL. MAPR IS CALLED BY MAIN.

PRT SUBROUTINE PRINTS OUT TWO-DIMENSIONAL ARRAYS. IT IS CALLED BY NORM.

ERRO2 SUBROUTINE TO PRINT AN ERROR MESSAGE. ERRO2 IS CALLED BY REAI2 AND REAG2.

SETUP SETUP MIXES THE CROSS SECTIONS AND CALCULATES THE AREAS AND VOLUMES. IT IS CALLED BY MAIN.

GRAM THIS SUBROUTINE CALCULATES THE MASS OF EACH MATERIAL IN EACH ZONE AND ALSO THE ZONE VOLUME. IT IS CALLED BY MAIN.

NORM NORM NORMALIZES THE FLUX AND ADJOINT FLUX AND CALCULATES K EFFECTIVE. IT CALLS PRT AND IS CALLED BY MAIN.

AVG THIS SUBROUTINE CALCULATES ZONE AVERAGED FLUXES, POWERS, AND ACTIVITIES. IT IS CALLED BY NORM.

ABETA ABETA PERFORMS THE NEUTRON LIFETIME AND BETA EFFECTIVE
CALCULATION. IT IS CALLED BY MAIN.

CALC SUBROUTINE TO CALCULATE AND PRINT REACTIVITY COEFFICIENTS. IT IS CALLED BY MAIN.

AK CALCULATED VALUE OF K EFFECTIVE
CAD TOTAL CHI*ADJOINT AT CENTER
CFX TOTAL CENTER POINT FLUX
CPW TOTAL CENTER POINT POWER
DENOM DENOMINATOR OF PERTURBATION FORMULA
FEF AVERAGE ENERGY RELEASED PER FISSION (MEV/FISSION)
FLPO FLUX OR POWER (SEE NFP)
IFP FLUX PUNCH OPTION
IGE TYPE OF GEOMETRY (0/1/2=R-Z/X-Y/R-THETA)
IGEP IGE + 1
IGM NUMBER OF GROUPS ( <= 26)
IGP NUMBER OF GROUPS PLUS 1
IM NUMBER OF RADIAL INTERVALS ( <= 50)
IMJM IM*JM
IP IM + 1
IST NUMBER OF DOWNSCATTERING TERMS ( <= 10)
ITL CROSS SECTION TABLE LENGTH ( <= 15)
IZM NUMBER OF MATERIAL ZONES ( <= 40)
JM NUMBER OF AXIAL INTERVALS ( <= 50)
JP JM + 1
MO1 LENGTH OF MIX VECTORS (10*11*12) ( <= 45)
MCR NUMBER OF MATERIALS FROM CARDS ( <= 25)
MT TOTAL NUMBER OF MATERIALS ( <= 35)
MTP NUMBER OF MATERIALS FROM TAPE
NACT NUMBER OF ACTIVITY TRAVERSES ( <= 3)
NCR1 CROSS SECTION TAPE
ND FLUX INPUT OPTION
NDELK NUMBER OF REACTIVITY COEFFICIENT CALCULATIONS ( <= 20)
NFLUX1 FLUX TAPE
NFLUX2 ADJOINT FLUX TAPE
NFP FLUX NORMALIZATION OPTION
NIBC NUMBER OF ISOTOPES FOR BETA CALCULATION ( <= 20)
NINP LOGICAL INPUT TAPE
NOUT LOGICAL OUTPUT TAPE
NPRT PRINT OPTION
NSCRAT SCRATCH TAPE
POWER TOTAL REACTOR POWER
RABS ABSORPTION RATE
RL RADIAL LEAKAGE
RLEAK LEAKAGE RATE
RSORS SOURCE RATE
TSD (MW-SEC)/(FISSIONS)
VF  VOLUME FACTOR
ZKEFF  KEFF FOR DELTA K CALC. (IF =0, USE KEFF CALC. BY CODE)
ZL  AXIAL LEAKAGE
A(MCR,10)  ID CARD FOR EACH MATERIAL CROSS SECTION SET
A0(IM)  RADIAL AREA ELEMENT
A1(JM)  AXIAL AREA ELEMENT
A2(JM)  AREA AT I=IP BETWEEN J-1 AND J
ACT(IM,JM,NACT)  ACTIVITY TRAVERSE
ADJ(IM,JM)  ADJOINT FLUX
ADJR(IGM,IM)  1-D RADIAL ADJOINT FLUX
ADJZ(IGM,JM)  1-D AXIAL ADJOINT FLUX
ATW(MCR)  MATERIAL ATOMIC WEIGHT
BETA(NIBC)  ABSOLUTE NEUTRON YIELD PER DELAYED FISSION
C(IIM,IGM,MT)  CROSS SECTION ARRAY
C0(IIM,MT)  CROSS-SECTION ARRAY FOR CURRENT GROUP
CHI(IGM)  FISSION SPECTRUM
CHIP(IGM)  DELAYED NEUTRON FISSION SPECTRUM
CX(IIM,IGM)  CROSS SECTION ARRAY FOR MATDK (EQUIVALENCED)
DUM(IMJM)  DUMMY VARIABLE
HOLN(MCR)  MATERIAL NAME
I0(M01)  MIX NUMBERS
I1(M01)  MATERIAL NUMBERS FOR MIX
I2(M01)  MATERIAL DENSITIES FOR MIX
ID(12)  IDENTIFICATION CARD ARRAY
L2(MT)  ZONE NUMBER BY MATERIAL
LDUM(IMJM)  DUMMY VARIABLE
M0(IM,JM)  ZONE NUMBERS
M2(IM)  MATERIAL NUMBERS BY ZONE
MA(IM,JM,NACT)  MATERIAL NUMBER FOR ACTIVITY TRAVERSE
MASS(IM,JM)  MASS OF MATERIAL IN ZONE
MATDK(NDELK)  MATERIAL NUMBER FOR REACTIVITY COEFFICIENT CALCULATION
NBET(NIBC)  MATERIAL NUMBER FOR BETA CALCULATION
NCOL(NDELK)  COLUMN NUMBER FOR REACTIVITY COEFFICIENT CALCULATION
NROW(NDELK)  ROW NUMBER FOR REACTIVITY COEFFICIENT CALCULATION
NX(NACT)  CROSS SECTION POSITION FOR ACTIVITY TRAVERSE
PHC(IGM)  CENTER POINT FLUX
PHI(IM,JM)  FLUX
PHR(IGM,IM)  1-D RADIAL FLUX
PHZ(IGM,JM)  1-D AXIAL FLUX
R1(IP)  RADII VALUES
R4(IM)  AVERAGE RADII
R5(IM)  DELTA-R
SLOA(IM)  D(DADJOINT)/DR (EQUIVALENCED)
SLOAZ(IM)  D(DADJOINT)/DZ (EQUIVALENCED)
SLOPR(IM)  D(DFLUX)/DR
SLOPZ(JM)  D(DFLUX)/DZ
SORC1(IM,JM)  NU*SIGF*FLUX (IN NORM)
SORC2(IM,JM) = CADJ(I+J) CHI*ADJOINT (IN NORM)

TPHI(IM,JM) TOTAL FLUX (EQUIVALENCED)

V0(I,M) VOLUME ELEMENTS

VEL(IGM) NEUTRON VELOCITY

VOL(IJM) ZONE VOLUME

Z1(JP) AXII VALUES

Z4(JM) AVERAGE AXII

Z5(JM) DELTA-Z

ZACT(IJM,NACT) ZONE AVERAGED ACTIVITIES

ZPHI(IJM) ZONE AVERAGED FLUX

ZPOW(IJM) ZONE AVERAGED POWER

DIMENSION JLPTAB(28)

CALL SETIO(9,2)

CALL SETIO(10,1)

CALL SETDR(3,525000,50000,JLPTAB)

CALL SETDR(4,575000,200000,JLPTAB(8))

CALL SETDR(6,775000,200000,JLPTAB(15))

CALL SETDR(7,975000,200000,JLPTAB(22))

REWIND 3

REWIND 4

REWIND 6

REWIND 7

NCR1 = 3

NSCRAT = 4

NFLUX1 = 6

NFLUX2 = 7

NOUT = 9

NINP = 10

CALL INP1

CALL MAPR(DUM)

CALL SETUP

CALL GRAM

CALL NORM

IF(NIBC) 8, 8, 6

CONTINUE

8

IF(NDELK) 20, 20, 10

10

CALL CALC

20

CONTINUE
SUBROUTINE INP1

INCLUDE ABC

C THIS SUBROUTINE CONTROLS THE READING OF ALL INPUT DATA

WRITE(NOUT, 3)
FORMAT(42X*35H * * * P E R T - I V * * * //////////)
READ(NINP, 5) (ID(I), I = 1, 12)

WRITE(NOUT, 6)
FORMAT(20X)
WRITE(NOUT, 20) MCR, MTP, IZM, MT, M01, NPRT

WRITE(NOUT, 30) NDELK, NIBC, NACT, NFP, IGE

READ(NINP, 15) FLPO, VF, FEF, ZKEFF

WRITE(NOUT, 40) NFP
FORMAT (78H NFP FLUX OR POWER NORMALIZATION, /, 1/2/3/4=CENT,12X, 
1/2/3/4=TOTAL,12X,46H POWER/12X,39X*5TH NPRT/)
WRITE(NOUT, 50) IGE
FORMAT (46H IGE GEOMETRY, /, 1/2/3/4=35X*19/)
WRITE(NOUT, 60) FLPO, VF, FEF, ZKEFF

FORMAT (41H FLPO FLUX(N/CM2-SEC) OR POWER(MWT), 1PE10.4/4
VF VOLUME FACTOR
FEF ENERGY(MEV) RELEASED PER FISSION
ZKEFF KEFF FOR DELTA K CALC. IF ZERO, USE KEFF CALCULATION
4ED BY CODE*

1H 33X 1PE10.4/
244H 36X 1PE10.4/
374H 6X 1PE10.4/

FORMAT(1H1, 12A6///)
ITL = IST + 5
C READ, CHECK, PRINT, AND WRITE CROSS SECTIONS TO TAPE
WRITE(NOUT, 64) (ID(I), I = 1, 12)
CALL XSNP
C READ FLUXES
WRITE(NOUT, 64) (ID(I), I = 1, 12)
WRITE(NOUT, 82)
FORMAT(198H INPUT FLUXES (PHR/PHZ/ADJR/ADJZ/PHI/ADJ=RADIAL FLUX/AXIAL FLUX/RADIAL ADJOINT / AXIAL ADJOINT))
CALL FXINP
IP=IM + 1
JP=JM + 1
C READ RADII
WRITE(NOUT, 84)
FORMAT(51HOMESH BOUNDARIES (R1/Z1=RADIAL POINTS/AXIAL POINTS))
CALL REA(6H R1tR1tIP)
C READ ZONE NUMBERS
WRITE(NOUT, 86)
FORMAT(30HOZONE NUMBERS BY MESH INTERVAL)
CALL REAI2(6H MOtLDUMtIMJM)
L = 0
DO 87 J = 1, JM
DO 87 I = 1, IM
L = L + 1
87 MO(I,J) = LDUM(L)
C READ MATERIAL NUMBERS
WRITE(NOUT, 88)
FORMAT(25HOMATERIAL NUMBERS BY ZONE)
CALL REAI2(6H M2tM2tIZM)
C READ FISSION FRACTIONS
WRITE(NOUT, 90)
FORMAT(17HOFISSION SPECTRUM)
CALL REAI2(6H CHItCHItIGMI)
C READ MIXTURE SPECIFICATIONS
WRITE(NOUT, 92)
FORMAT(82HOMIXTURE SPECIFICATIONS (IO/I1/I2=MIX NUMBER/MATERIAL NUMBER/MATERIAL DENSITY))
CALL REAI2(6H MO1tMO1tI0M01)
CALL REAG2 (6H 12*12*M01)
C CHECK FOR LIFETIME AND BETA CALCULATIONS
IF(NIBC ) 120*120* 100
C READ MATERIAL NUMBERS FOR BETAS
100 WRITE(NOUT*94)
94 FORMAT(38HMATERIAL NUMBERS FOR BETA CALCULATION)
CALL REAI2(6H NBET*NBET*NIBC)
C READ BETAS
CALL REAG2(6H BETABETA*NIBC)
C READ DELAYED FISSION SPECTRUM
WRITE(NOUT, 102)
102 FORMAT(25HDELAYED FISSION SPECTRUM)
CALL REAG2(6H CHIP*CHIP*IGM)
C READ VELOCITIES
WRITE(NOUT*104)
104 FORMAT(17HNEUTRON VELOCITY)
CALL REAG2(6H VEL*VEL*IGM)
C CHECK FOR DELTA K CALCULATION
120 IF(NDELK ) 160*160* 140
C READ MATERIAL NUMBERS, ROW NUMBERS, AND COLUMN NUMBERS FOR PERTURBATION CALCULATION
140 WRITE(NOUT*138)
138 FORMAT(9SHOPERTURBATION CALCULATION INFORMATION (MATDK/NROW/NCOL=MATERIAL NUMBER/ROW/COLUMN TO BE PERTURBED))
CALL REAI2(6H MATDK*MATDK*NDELK)
CALL REAI2(6H NROW* NROW* NDELK)
CALL REAI2(6H NCOL* NCOL* NDELK)
C CHECK FOR ACTIVITIES
160 IF(NACT ) 1000*1000* 180
C READ MATERIAL NUMBERS AND XS POSITION FOR ACTIVITIES
180 WRITE(NOUT*178)
178 FORMAT(77HOACTIVITY TRAVERSE INFORMATION (MA/NX=MATERIAL NUMBER/CROSS SECTION POSITION))
CALL REAI2(6H MA*MA*NACT)
CALL REAI2(6H NX* NX* NACT)
1000 CONTINUE
RETURN
END
ITC FOR XSINP
SUBROUTINE XSINP
INCLUDE ABC
C THIS SUBROUTINE READS, CHECKS, PRINTS, AND WRITES THE CROSS SECTIONS TO TAPE
ITEMP = MCR + MTP
10 DO 50 I=1,ITEMP
   READ(NINP, 20) HOLN(I), ATW(I), (A(I,J), J=1, 10)
20 FORMAT(A6, E6.2, 10A6)
   IF(MCR) 35*35, 25
25 DO 40 IIG=1, IGM
40 READ(NINP, 3U) (C(LtIIG, I), L=1, ITL)
30 FORMAT(6E12.5)
   GO TO 50
35 READ(15) ((C(LtIGM, I), L=1, ITL), IIG=1, IGM)
50 CONTINUE
C CHECK ON CROSS SECTION CONSISTENCY AND ORDER
DO 200 J=1,ITEMP
   DO 200 I=1, IGM
      G = C(2, I, J) + C(5, I, J)
      DO 80 K=1, IST
         KK = I + K
         M = 5 + K
         IF(KK > IGM) 70, 70, 80
50 CONTINUE
   IF(ABS((G - C(4, I, J))/C(4, I, J)) - .01) 200, 90, 90
90 WRITE(NOUT, 110) J, I
110 FORMAT(1H/ t16H CHECK MATERIAL I2,5X/ t 7H GROUP 12)
200 CONTINUE
C PRINT CROSS SECTIONS
   IF(NPRT) 210, 210, 250
210 WRITE(NOUT, 220)
220 FORMAT(38H THE FOLLOWING NUCLIDES ARE READ-IN )
   DO 240 K=1,ITEMP
      WRITE(NOUT, 245) K, HOLN(K), (A(K, J), J=1, 10)
240 FORMAT(19, 6X t11A6/)
   GO TO 510
250 DO 480 K=1,ITEMP
480 FORMAT(1H1)
260 FORMAT(1H10, 2113H GROUP SIGF SIGA NUSIGF SIGTR SIGXG G-1X
   3G G-2XG G-3XG G-4XG G-5XG G-6XG/ )
   IF(IST = 6) 290, 290, 265
265 DO 270 I=1, IGM
270 FORMAT(I3 t11F10.3)
270 WRITE(NOUT, 268) I*(C(J*I*K)+J=1*11)
    WRITE(NOUT, 275)
275 FORMAT (44H1GROUP     G-7XG     G-8XG     G-9XG     G-10XG/ )
    DO 279 I=1*IGM
279 WRITE(NOUT, 268) I*(C(J*I*K)+J=12*ITL)
    IF(K -ITEMP) 460, 480, 480
290 CONTINUE
    DO 300 I=1*IGM
300 WRITE(NOUT, 268) I*(C(J*I*K)+J=1*ITL )
350 IF(K -ITEMP) 460, 480, 480
460 WRITE(NOUT, 255)
480 CONTINUE
C WRITE CROSS SECTIONS TO TAPE
510 DO 1160 IIG=1*IGM
    DO 1060 M=1*MT
    DO 1060 L=1*ITL
1060 CO (L*M) = C(L*IIG$M)
1160 WRITE (NCR1) ((CO (L*M), L=1*ITL), M=1*MT)
REWIND NCR1
RETURN
END
SUBROUTINE FXINP

INTEGER

C THIS SUBROUTINE READS FLUXES (REGULAR FLUX AND ADJOINT FLUX) IF
C FLUXES ARE IN 1-D FORM, THEY ARE COMBINED IN SUCH A WAY TO PUT
C THEM IN 2-D FORM

IMJM = IM*JM
JJR = IGM*IM
JJZ = IGM*JM
LLH = 1
NND = IABS(ND)
ND = IABS(ND)

C LLH=1/2=REGULAR FLUX/ADJOINT FLUX
GO TO (1UUT 800, 10U, 800, 100, 800, 800), NND

C FLUX IS IN 1-D FORM

100 CONTINUE
ZFX = 0.0
RFX = 0.0
GO TO (110, 120), LLH

110 CALL REAG2(6H PHR,DUM,JJR)
GO TO 130

120 CALL REAG2(6H ADJR,DUM,JJR)
C IF NND IS POSITIVE, ADJOINT FLUXES ARE REVERSED BY GROUP
IF(NND) 130, 130, 121

121 CONTINUE
L = 0
DO 126 K = 1, IGM
DO 124 I = 1, IM
L = L + 1
KK = IGM + 1 - K
124 PHR(KK; I) = DUM(L)
126 RFX = RFX + PHR(KK; 1)
GO TO 165

130 L = 0
DO 160 K = 1, IGM
DO 158 I = 1, IM
L = L + 1
158 PHR(K; I) = DUM(L)
160 RFX = RFX + PHR(K; 1)
CONTINUE
GO TO (170, 180), LLH

170 CALL REAG2(6H PHZ,DUM,JJZ)
GO TO 190

180 CALL REAG2(6H ADJZ,DUM,JJZ)
C IF(NND) 190, 190, 182
C SWIOTC GROUPS
CONTINUE
L = 0
DO 186 K = 1, IGM
 DO 184 I = 1, JM
  L = L + 1
  KK = IGM + 1 - K
184  PHZ(KK, I) = DUM(L)
186  ZFX = ZFX + PHZ(KK, I)
  GO TO 262
190  L = 0
 DO 260 K = 1, IGM
  DO 258 I = 1, JM
   L = L + 1
258  PHZ(K, I) = DUM(L)
260  ZFX = ZFX + PHZ(K, I)
262  CONTINUE
 DO 300 K = 1, IGM
 DO 280 I = 1, JM
  PHR(K, I) = PHR(K, I) / RFX
 DO 300 I = 1, JM
300  PHZ(K, I) = PHZ(K, I) / ZFX
 DO 400 K = 1, IGM
   PHC(K) = (PHZ(K, I) + PHR(K, I)) * (.5)
 DO 320 I = 2, JM
320  PHR(K, I) = PHR(K, I) / PHR(K, 1)
 DO 360 I = 2, JM
360  PHZ(K, I) = PHZ(K, I) / PHZ(K, 1)
   PHR(K, 1) = 1.0
   PHZ(K, 1) = 1.0
 DO 380 I = 1, JM
 DO 380 J = 1, JM
380  PHI(I, J) = PHR(K, I) * PHZ(K, J) * PHC(K)
380  CONTINUE
 WRITE(NFLUX1) PHI(I, J), I = 1, IM, J = 1, JM
400  CONTINUE
 GO TO 1000
790  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
800  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
820  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
 C FLUX IS IN 2-D FORM
820  CONTINUE
 LWH = IGM * IM JM
 GO TO (810, 815), LLH
810  WRITE(NOUT, 812) LWH
812  FORMAT(6X, 6H PHI(I, J), I = 1, IM, J = 1, JM)
815  CALL NTRAN (14, 8, 1)
DO 840 K = 1, IGM
 IF (ND-6) 819,819,822
819 READ(NINP, 820) ((PHI(I, J), I = 1, IM), J = 1, JM)
820 FORMAT(6E12.6)
 GO TO 825
822 READ (14) ((PHI(I*J)*I=1*IM), J=1, JM)
825 WRITE(NOUT*818) ((PHI(I, J), I = 1* IM), J = 1, JM)
 WRITE(NFLUX1)((PHI(I*J), I=1*IM), J=1, JM)
 CONTINUE
 GO TO (910, 850), LLH
850 CONTINUE
C IF NND IS POSITIVE, ADJOINT FLUXES ARE REVERSED BY GROUP
 IF(NND) 910, 910, 860
860 CONTINUE
 REWIND NFLUX1
 DO 880 K = 1, IGM
880 CONTINUE
 DO 900 K = 1, IGM
 BACKSPACE NFLUX1
 READ(NFLUX1) ((PHI(I*J)*I=1*IM), J=1, JM)
 WRITE(NSCRAT) ((PHI(I, J), I = 1* IM), J = 1, JM)
 BACKSPACE NFLUX1
900 CONTINUE
 REWIND NSCRAT
 REWIND NFLUX1
 TEMP = NSCRAT
 NSCRAT = NFLUX1
 NFLUX1 = TEMP
 CONTINUE
1000 GO TO (1100, 2200), LLH
1100 LLH=2
C SWITCH TAPE DESIGNATIONS
 TEMP = NFLUX1
 NFLUX1 = NFLUX2
 NFLUX2 = TEMP
C READ ADJOINT FLUX
 GO TO (10U, 800, 800, 100, 2000, 2000, 790), ND
C THERE IS NO ADJOINT FLUX INPUT
2000 CONTINUE
 AAA = 1.0
 DO 2100 K = 1, IGM
 WRITE(NFLUX1) ((AAA, I = 1* IM), J = 1, JM)
2100 CONTINUE
2200 REWIND NFLUX1
 REWIND NFLUX2
 IF(ND-6) 2224,2224,2210
2210 REWIND 14
C SWITCH TAPE DESIGNATIONS
2220 TEMP = NFLUX2
     NFLUX2 = NFLUX1
     NFLUX1 = TEMP
RETURN
END
SUBROUTINE REAI2(HOLL, IARRAY, NCOUNT)
DIMENSION IARRAY(10), IV(6), K(6), IN(6)
COMMON NINP, NOUT, NCR1, NFLUX1, NSCRAT,NFLUX2
J=1
10 READ(NINP,20) (K(I), IN(I), IV(I), I=1,6)
20 FORMAT(6(I1, I2, I9))
DO 70 I=1,6
L=K(I)+1
GO TO (30,40,60,80) L
CNO MODIFICATION
30 IARRAY(J)=IV(I)
  J=J+1
  GO TO 70
C
CREPEAT
40 L=IN(I)
DO50 M=1,L
  IARRAY(J)=IV(I)
  J=J+1
50 CONTINUE
GO TO 70
C
CINTERPOLATE
60 CALL ERROR2(6H REAI2)
C
70 CONTINUE
GO TO 10
C
CTERMINATE
80 J=J-1
WRITE(NOUT,90) HOLL, (IARRAY(I), I=1,J)
IF(J-NCOUNT)100,110,100
90 FORMAT(6X,6X/A6,6X/I6/(10I12))
100 CALL ERROR2(6H**REAI2)
110 RETURN
END
SUBROUTINE REAG2(HOLL, ARRAY, NCOUNT)
DIMENSION ARRAY(10), V(12), N(12)
COMMON NINP, NOUT, NCR1, NFLUX1, NSCRAT, NFLUX2
JFLAG = 0
J = 1
10 IF(JFLAG) 20, 40, 20
20 DO 30 JJ = 1, 6
   K(JJ) = K(JJ+6)
   IN(JJ) = IN(JJ+6)
30   V(JJ) = V(JJ+6)
   JFLAG = 0
   GO TO 60
40 READ (NINP, 50) (K(I), IN(I), V(I), I = 1, 6)
50 FORMAT (6(I10, I2, E9.4))
60 DO 140 I = 1, 6
   L = K(I) + 1
   GO TO (70, 90, 100, 110) * L
70   ARRAY(J) = V(I)
    J = J + 1
    GO TO 140
90 CONTINUE
GO TO 140
110 READ (NINP, 50) (K(JJ), IN(JJ), V(JJ), JJ = 7, 12)
JFLAG = 1
120 L = IN(I) + 1
   DEL = (V(I+1) - V(I)) / FLOAT(L)
   DO 130 M = 1, L
      ARRAY(J) = V(I) + DEL * FLOAT(M-1)
      J = J + 1
130 CONTINUE
GO TO 140
C
140 CONTINUE
GO TO 10
C
CTERMINATE
  150  J=J-1
      WRITE (NOUT,160)       HOLL,J       *( ARRAY(I)*I=1*J)
      IF(J - NCOUNT)170,180,170
160  FORMAT(6X,A6,I6/(10E12.5))
170  CALL ERRO2( '6H**REAG',170,1)
180  RETURN
END
ITC FOR MAPR*MAPR

SUBROUTINE MAPR(K)

INCLUDE ABC

DIMENSION K(9)

IX=IM
IY=JN

WRITE (NOUT*6) (ID(I), I = 1, 12)

6 FORMAT(1H1*12A6///)

DO 5 JJ=1,IY

J=IY-JJ+1

5 WRITE (NOUT*4) (MO(I,J),I=1,IX)

4 FORMAT(5H5512)

WRITE(NOUT*15)

15 FORMAT(2H A/2H X/2H I/2H A/2H L/8H RADIAL)

WRITE (NOUT*6) (ID(I), I = 1, 12)

DO 10 JJ=1,IY

J=IY-JJ+1

DO 11 L=1,IX

N=MO(L,J)

11 K(L)=IABS(M2(N))

10 WRITE (NOUT*4) (K(L),L=1,IX)

WRITE(NOUT*15)

RETURN

END
SUBROUTINE PRT(IM, JM, PHI)
COMMON NINP, NOUT, NCR1, NFLUX1, NSCRAT, NFLUX2
DIMENSION PHI(50, 50)
DO 60 I=1*IM, 5
   I1=I
   I2=I+4
   IF(I2-IM) 20, 20, 10
10 I2=IM
20 WRITE (NOUT, 30) (JJ, JJ=I1, I2)
30 FORMAT(5X, 5I20)
DO 50 JJ=1, JM
   J=JJ
   WRITE(NOUT, 40) J, (PHI(K, J), K=I1, I2)
40 FORMAT(5X, 15E15.7, 4E20.7)
50 CONTINUE
60 CONTINUE
RETURN
END
FOR ERROR2

SUBROUTINE ERROR2 (HOL, JSUBR)
COMM NINP, NOUT, NCR1, NFLUX1, NSCRAT, NFLUX2
WRITE (NOUT) HOL, JSUBR
1 FORMAT (2H */9H ERROR IN AT,I6/2H */2H *)
GO TO (3*4)
3 STOP
4 RETURN
END
-ITC FOR SETUP, SETUP
SUBROUTINE SETUP
INCLUDE ABC
C MIX CROSS SECTIONS
ITL = IST + 5
WRITE (NOUT, 65) (ID(I), I = 1, 12)
65 FORMAT(1H1, 12A6///)
WRITE (NOUT*70)
70 FORMAT(19H CROSS-SECTION EDIT)
REWIND NCR1
DO 100 IIG = 1, IGM
READ (NCR1) ((CO(I, J)), I = 1, ITL, J = 1, MT)
DO 92 M = 1, M01
IF (IO(M) = MT), 74, 74
72 CALL ERROR2(6H**S807972, 72)
74 IF (IN(M) = MT), 80, 80
80 N = IO(M)
L = I1(M)
E01 = I2(M)
DO 90 I = 1, ITL
IF (L), 86, 86
86 CO(I, N) = CO(I, N) + CU(I, L) * E01
GOTO 90
88 CO(I, N) = CO(I, N) * E01
90 CONTINUE
92 CONTINUE
WRITE (NOUT, 94) IIG
94 FORMAT(7H GROUP 1, 15H CROSS-SECTIONS/
DO 96 N = 1, MT
96 WRITE (NOUT, 98) N, (CO(I, N), I = 1, ITL)
98 FORMAT(4H MAT, 13, 10E11, 3/*7X, 10E11, 3/)
WRITE (NSCRAT) ((CO(I, J)), I = 1, ITL, J = 1, MT)
100 CONTINUE
REWIND NCR1
C SWITCH TAPE DESIGNATIONS
ITEMP = NSCRAT
NSCRAT = NCR1
NCR1 = ITEMP
101 FORMAT(1H1)
C CALCULATE AREAS AND VOLUMES
C******
C******
PI2 = 6.28318
IGEP = IGE + 1
DO 170 I = 1, IM
R4(I) = (R1(I, 1) + R1(I)) * 0.5
R5(I) = R1(I+1) - R1(I)
130 IF (R5(I)) 130, 130, 140
140 CALL ERRO2 (6H*R5(I), 130, 1)
140 CONTINUE
GO TO (160, 150, 150), IGEP
150 CONTINUE
A1(I) = R5(I)
GO TO 170
160 CONTINUE
A1(I) = P12*R5(I)*R4(I)
170 CONTINUE
DO 200 J=1, JM
Z4(J) = (Z1(J+1)+Z1(J))*0.5
Z5(J) = Z1(J+1) - Z1(J)
IF (Z5(J)) 175, 175, 180
175 CALL ERRO2 (6H*Z5(J), 175, 1)
180 CONTINUE
GO TO (183, 182, 183), IGEP
182 A2(J) = Z5(J)
GO TO 184
183 A2(J) = P12*R1(IP)*Z5(J)
184 DO 195 I = 1, IM
GO TO (190, 185, 190), IGEP
185 VO(I, J) = R5(I)*Z5(J)*VF
GO TO 195
189 VO(I, J) = P12*R5(I)*Z5(J)*R4(I)*VF
190 CONTINUE
DO 200 J=1, JM
Z4(J) = (Z1(J+1)+Z1(J))*0.5
Z5(J) = Z1(J+1) - Z1(J)
IF (Z5(J)) 240, 240, 280
240 WRITE (NOUT, 220) ((I, R1(I), R4(I), Z1(I), Z4(I)), I = 1, IP)
240 WRITE (NOUT, 220) ((I, RR, ZZ, Z1(I), Z4(I)), I = I + IP)
280 IF (IM - JM) 241, 241, 241
241 WRITE (NOUT, 220) ((I, ZZ, ZZ, Z1(I), Z4(I)), I = III + JP)
GO TO 500
WRITE(NOUT, 220) (I, R1(I), R4(I), Z1(I), Z4(I)), I = 1, JP)
JJJ = JP + 1
WRITE(NOUT, 220) (I, R1(I), R4(I), ZZ, ZZ), I = JJJ, JP)
500 CONTINUE
RETURN
END
ITC FOR GRAM

SUBROUTINE GRAM
  INCLUDE ABC
  REAL MASS
  DIMENSION MASS(50, 50)
  EQUIVALENCE (C(1), MASS(1))
C THIS SUBROUTINE CALCULATES THE MASS OF THE VARIOUS MATERIALS
  WRITE(NOUT, 1)(ID(I), I = 1, 12)
  1 FORMAT(1H12A6///)
  WRITE(NOUT, 5)
  5 FORMAT(45H MATERIAL INVENTORY (KILOGRAMS) FOR EACH ZONE / )
  DO 10 K = 1, 12M
      VOL(K) = 0.0
  10 CONTINUE
  DO 15 J = 1, 50
      DO 15 I = 1, 50
          MASS(I, J) = 0.0
  15 CONTINUE
  DO 20 J = 1, 50M
      DO 20 I = 1, IM
          K = M0(I, J)
          KK = M2(K)
          L2(KK) = K
          VOL(K) = VOL(K) + V0(I, J) * .01
  20 CONTINUE
  KK = M0 + M01
  DO 40 N = 1, 12M
      NN = M2(N)
      IF(IO(M1) - NN) 40, 22, 40
  22 L = I1(M1)
      IF(L - KK) 28, 28, 24
  24 NNAA = L
      IF(IO(MAA) - NNAA) 35, 32, 35
  32 L = I1(MAA)
      IF(L) 35, 35, 33
  33 E01 = I2(MAA) * I2(M)
      MASS(N*L) = (E01*ATW(L)*VOL(N))/6023 + MASS(N*L)
  35 CONTINUE
  GO TO 40
  28 IF(L) 40, 44, 30
  30 E01 = I2(M)
      MASS(N*L) = (E01*ATW(L)*VOL(N))/6023 + MASS(N*L)
  40 CONTINUE
  DATA ZONE/6H ZONE /
  DO 90 L = 1, 12M, 5
      LL = L + 4
  90 WRITE(NOUT, 5) LL, VOL(N), MASS(N*L)
IF (LL - IZM) 62, 62, 61
  LL = IZM
  WRITE(NOUT, 72) ((ZONE, K), K=L, LL)
  FORMAT(//26H MATERIAL ATOMIC WT. .3X, 5(A6,12.12X))
  WRITE(NOUT, 73) (VOL(K), K = L, LL)
  FORMAT(25X, 5(F8.2, 7H LITERS, 5X))
  DO 75 K = 1, KK
  WRITE(NOUT, 76) K, HOLN(K), ATW(K), (MASS(I, K), I = L, LL)
  FORMAT(//I3,1X, A6, F13.3, F14.3, 4F20.3)
  IF (LL - IZM) 90, 94, 94
  CONTINUE
  CONTINUE
  RETURN
END
ITC FOR NORM
SUBROUTINE NORM
INCLUDE ABC
DIMENSION TPHI(50, 50), CADJ(50, 50), ACT(50, 50, 3), POW(50, 50)
EQUIVALENCE (SORC1(I, J), TPHI(I, J)), (SORC2(I, J), CADJ(I, J)),
     (C(I, J), ACT(I, J)), (LDUM(I), POW(I, J))
C THIS SUBROUTINE NORMALIZES THE FLUXES (REGULAR AND ADJOINT)

CFX = 0.0
CPW = 0.0
POWER = 0.0
DENOM = 0.0
RL = 0.0
ZL = 0.0
RABS = 0.0
RSORS = 0.0
DO 10 J = 1, JM
    DO 10 I = 1, IM
    SORC1(I, J) = 0.0
    SORC2(I, J) = 0.0
10 CONTINUE
REWIND NFLUX1
REWIND NFLUX2
REWIND NSCRAT
REWIND NCR1
DO 100 KK = 1, IGM
    READ(NCR1), II = 1, ITL, J = 1, MT
    READ(NFLUX1), (PHI(I, J), I = 1, IM), J = 1, JM
    READ(NFLUX2), (ADJ(I, J), I = 1, IM), J = 1, JM
    CFX = CFX + PHI(1, 1)
    K = M0(1, 1)
    K = M2(K)
    CPW = CPW + C0(1, K) * PHI(1, 1)
    DO 20 J = 1, JM
        DO 20 I = 1, IM
            K = M0(I, J)
            K = M2(K)
            POWER = POWER + C0(1, K) * PHI(I, J) * V0(I, J)
            SORC1(I, J) = SORC1(I, J) + C0(1, K) * PHI(I, J)
            SORC2(I, J) = SORC2(I, J) + C0(1, K) * ADJ(I, J)
            RABS = RABS + C0(1, K) * PHI(I, J) * V0(I, J)
            RSORS = RSORS + C0(1, K) * PHI(I, J) * V0(I, J)
20 CONTINUE
PI2 = 6.28318
DO 50 I = 1, IM
    GO TO (35, 35, 25), IGEP
35 SLOPZ(I) = (PHI(I, JM) - PHI(I, JM - 1)) / (Z4(JM) - Z4(JM - 1))
      * PI2 * R4(I)
GO TO 40
SLOPZ(I) = (PHI(I,JM) - PHI(I,JM - 1))/ (Z4(JM) - Z4(JM - 1))

CONTINUE
K = MO(I, JM)
K = M2(K)
ZL = ZL + (SLOPZ(I)*A1(I))/(3.*C0(4,K))

CONTINUE
DO 60 I = 1, IM
GO TO (55+56, 52), IGEF

SLOPZ(I) =-(PHI(I*2) - PHI(I*1))/((Z4(2) - Z4(1))*PI2*R4(I))
GO TO 56

CONTINUE
K = MO(I*1)
K = M2(K)
ZL = ZL + (SLOPZ(I)*A1(I))/(3.*C0(4,K))

CONTINUE
DO 60 J = 1, JM
SLOPR(J) = (PHI(IM,J) - PHI(IM - 1,J))/ (R4(IM) - R4(IM - 1))
K = MO(IM*J)
K = M2(K)
RL = RL + (SLOPR(J)*A2(J))/(3.*C0(4,K))

CONTINUE
RLEAK = -ZL - RL
RSORS = RSORS/VF
RABS = RABS/VF
AK = (RSORS)/(RABS + RLEAK)

T5D = (MW-SEC)/(FISSIONS)
TSD = FEF*1.602*10**(-19)
CAD = SORC2(I, J)
CPW = CPW*TSD*1000.
POWER = POWER*TSD
REWORD NCR1
REWORD NFLUX1
REWORD NFLUX2
DO 200 J = 1, JM
DO 200 I = 1, IM
DENOM = DENOM + (SORC1(I, J) *SORC2(I*1)*V0(I, J))/(SORC2(I*1)*AK)

CONTINUE
GO TO (240, 260, 280, 300), NFP

CONST = FLPO/CFX
GO TO 320

CONST = FLPO/CPW
GO TO 320

CONST = FLPO/POWER
GO TO 320
CONST = FLPO/DENOM
CONTINUE
CFX = CONST*CFX
CPW = CONST*CPW
POWER = CONST*POWER
DENOM = CONST*DENOM
WRITE(NOUT, 328) (ID(IK), IK = 1, 12)
FORMAT(1H1, 12A6/)
RSORS = RSORS*VF*CONST
RABS = RABS*VF*CONST
RLEAK = RLEAK*VF*CONST
RL = -RL*VF*CONST
ZL = -ZL*VF*CONST
AAK = .1*AK
WRITE(NOUT,330) CFX, CPW, POWER, DENOM, RSORS, RABS, RLEAK
1, RL, ZL, AAK
FORMAT(44H CENTER POINT FLUX (N/CM2-SEC) - - - - 1PE12.5//
1 44H CENTER POINT POWER (MWT/LITER) - - - - - - 1PE12.5//
2 44H TOTAL POWER (MWT) - - - - - - - - 1PE12.5//
3 44H DENOMINATOR OF PERT. FORMULA - - - - - - - - 1PE12.5//
4 44H SOURCE RATE - - - - - - - - - - 1PE12.5//
5 44H ABSORPTION RATE - - - - - - - - - - 1PE12.5//
6 44H LEAKAGE RATE (TOTAL) - - - - - - - - 1PE12.5//
7 44H RADIAL LEAKAGE - - - - - - - - 1PE12.5//
8 44H AXIAL LEAKAGE - - - - - - - - 1PE12.5//
9 44H KEFF - - - - - - - F12.5)
DO 350 J = 1, JM
DO 350 I = 1, IM
DO 344 N = 1, 3
ACT(I, J, N) = 0.0
TPHI(I, J) = 0.0
CADJ(I, J) = 0.0
POW(I, J) = 0.0
CONTINUE
DO 400 K = 1, IGM
READ(NFLUX1) (PHI(I, J), I = 1, IM), J = 1, JM)
READ(NFLUX2) ((ADJ(I,J), I = 1, IM), J = 1+ JM)
READ(DCR1) (CUI(I, J), J = 1, ITL), J = 1, MT)
DO 360 J = 1, JM
DO 360 I = 1, IM
KK = MU(I, J)
PHI(I, J) = PHI(I, J)*CONST
ADJ(I, J) = ADJ(I, J)/CAD
TPHI(I, J) = TPHI(I, J) + PHI(I, J)
CADJ(I, J) = CADJ(I, J) + CHI(K) * ADJ(I, J)
POW(I, J) = POW(I, J) + COI(K)*PHI(I, J)*TSD*1000.
IF(NACT) 360, 360, 352
CONTINUE
DO 356 N = 1, NACT
KK = MA(N)
NN = NX(N)
ACT(I, J, N) = ACT(I, J, N) + CO(NN, KK)*PHI(I, J)
CONTINUE
WRITE (NSCRT) ((PHI(I, J), I = 1, IM), J = 1, JM)
WRITE (NSCRT) ((ADJ(I, J), I = 1, IM), J = 1, JM)
GO TO (368, 362, 364) LMN
PUNCH 363, ((PHI(I, J), I = 1, IM), J = 1, JM)
FORMAT(6E12.6)
GO TO 368
PUNCH 363, ((ADJ(I, J), I = 1, IM), J = 1, JM)
IF(NPRT) 376, 376, 370
CONTINUE
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE (NOUT, 372) K
FORM H FLUX FOR GROUP /3/1
CALL PRT(IM, JM, PHI)
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE (NOUT, 374) K
FORM H ADJOIN FLUX FOR GROUP /3/1
CALL PRT(IM, JM, ADJ)
CONTINUE
400 CONTINUE
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE (NOUT, 478)
FORMAT(11H TOTAL FLUX/)1
CALL PRT(IM, JM, TPHI)
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE (NOUT, 480)
FORMAT(28H FISSON SOURCE*ADJOIN FLUX/)1
CALL PRT(IM, JM, CADJ)
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE (NOUT, 482)
FORMAT(26H POWER DENSITY (MWT/LITER)/)
CALL PRT(IM, JM, POW)
IF(NACT) 490, 490, 484
484 DO 488 N = 1, NACT
DO 486 J = 1, JM
DO 486 I = 1, IM
SORC2(I, J, N) = ACT(I, J, N)
WRITE (NOUT, 328) (ID(IK), IK = 1, 12)
WRITE(NOUT, 487) N, KK, NN
FORMAT(9H ACTIVITY, 13, 13H ION POSITION, I3/)
CALL PRT(IM, JM, SORC2)
CONTINUE
CONTINUE
CALL AVG(TPHI, POW, ACT)
C SWITCH TAPE DESIGNATIONS
ITEMP = NSCRAT
NSCRAT = NFLUX1
NFLUX1 = ITEMP
REWIND NSCRAT
REWIND NFLUX1
REWIND NFLUX2
REWIND NCR1
RETURN
END
SUBROUTINE AVG(TPHI, POW, ACT)
    INCLUDE ABC

    THIS SUBROUTINE CALCULATES ZONE AVERAGED FLUXES, POWERS, AND ACTIVITIES

    DIMENSION ACT(50*50*3), TPHI(50*50), POW(50*50)
    DATA CAT/6H, ACT/0.0

    WRITE(NOUT,328) (ID(IK), IK=1,12)

    328 FORMAT(1H1,12A6,1X)

    DO 340 KKK=1,IZM
        ZPHI(KKK) = 0.0
        ZPOW(KKK) = 0.0
    DO 340 N=1,3
        ZACT(KKK,N) = 0.0
    DO 525 I=1,IM
        KKK = MO(I,J)
        ZPHI(KKK) = ZPHI(KKK) + VO(I,J)*.001*TPHI(I,J)
        ZPOW(KKK) = ZPOW(KKK) + VO(I,J)*.001*POW(I,J)
        IF(NACT) 525,525,505
    505 CONTINUE
    DO 508 N=1,NACT
        ZACT(KKK,N) = ZACT(KKK,N) + VO(I,J)*.001*ACT(I,J,N)
    525 CONTINUE
    DO 540 KKK=1,IZM
        ZPHI(KKK)/VOL(KKK)
        ZPOW(KKK)/VOL(KKK)
        IF(NACT) 540,540,535
    535 CONTINUE
    DO 538 N=1, NACT
        ZACT(KKK,N) = ZACT(KKK,N)/VOL(KKK)
    540 CONTINUE
    IF(NACT) 560,560,580
    WRITE(NOUT,565)
    565 FORMAT(1H1,48HZONEAVFLUXAVPOWER//)
    DO 570 KKK=1,IZM
    WRITE(NOUT,575) KKK, ZPHI(KKK), ZPOW(KKK)
    575 FORMAT(1H1,48HZONEAVFLUXAVPOWER//)
    GO TO 600
    WRITE(NOUT,585) ((CAT* N), N=1,NACT)
    585 FORMAT(1H1,48HZONEAVFLUXAVPOWER//)
    13(A6*11,13X))
    DO 590 KKK=1,IZM
    WRITE(NOUT,575) KKK, ZPHI(KKK), ZPOW(KKK), (ZACT(KKK,N)*N=1,NACT)
    590 CONTINUE
    RETURN
END
ITC FOR ABETA

SUBROUTINE ABETA
INCLUDE ABC
DIMENSION BSIGF(50t50)
EQUIVALENCE (PHI(1:1), BSIGF(1:1))

C THIS SUBROUTINE CALCULATES BETA EFFECTIVE AND NEUTRON LIFETIME
WRITE(NOUT,5) (ID(I), I=1,12)
5 FORMAT(1H1,12A6///)
DO 10 J=1,50
DO 10 I=1,50
SORC1(I,J)=0.0
SORC2(I,J)=0.0
PHZ(I,J) = 0.0
BSIGF(I,J) = 0.0
10 CONTINUE
DO 1000 I=1,NMG
READ(NFLUX1) ((PHI(I,J), I=1,IM), J=1,JM)
READ(NFLUX1) ((ADJ(I,J), I=1,IM), J=1,JM)
READ(NCR1) ((CO(I,J), I=1,ITL), J=1,MT)
DO 500 J = 1, JM
DO 500 I = 1, IM
PHZ(I,J) = (PHI(I,J)*ADJ(I,J))/VEL(KK) + PHZ(I,J)
KKK = M0(I,J)
KKK = M2(KKK)
DO 300 L = 1, NIBC
DEN = U.0
DO 120 M=1, MO1
IF(I0(M) = KKK) 120, 110, 120
110 IF(I1(M) = NBET(L)) 120, 115, 120
115 DEN = DEN + I2(M)
120 CONTINUE
NN = NBET(L)
BSIGF(I,J) = BSIGF(I,J) + BETA(L)*DEN*C0(1,NN)*PHI(I,J)
300 CONTINUE
C DELAYED NEUTRON SOURCE*ADJOINT FLUX (SUMMED OVER ENERGY GROUPS)
SORC2(I,J) = CHIP(KK)*ADJ(I,J) + SORC2(I,J)
500 CONTINUE
DO 1000 J = 1, JM
DO 1000 I = 1, IM
BE = BE + SORC2(I,J)*BSIGF(I,J)*VO(I,J)
ALIFE = ALIFE + (PHZ(I,J)*VO(I,J))
2000 CONTINUE
ALIFE = ALIFE/(DENOM*AK)
BEFF = BE/(DENOM*AK)
WRITE(NOUT,2050) BEFF, ALIFE
2050 FORMAT(43H BETA EFFECTIVE - - - - - 1PE11.5//
        43H NEUTRON GENERATION TIME - - - - - 1PE11.5)
RETURN
END
SUBROUTINE CALC
  INCLUDE ABC
  DIMENSION CX(1,526), FLUX(2,265,50,3), SLOAZ(50), SLOAR(50)
  EQUIVALENCE (CX(1,1), PHR(1,1)), (FLUX(1,1), C(1,1)), SLOAR(1), DUM(1)

C THIS SUBROUTINE USES PERTURBATION THEORY TO CALCULATE REACTIVITY COEFFICIENTS
C
DO 9999 III = 1, NDELK
  REWIND NCR1
  REWIND NFLUX1
  REWIND NFLUX2
  REWIND NSCRAT
  NR = NROW(III)
  NC = NCDL(III)
  MAT = MATDK(III)

C CONVERSION FROM ATOMS TO KG
  CT = 602.3/ATW(MAT)
  PI2 = 6.28318
  DO 1 J = 1, 50
    DO 1 I = 1, 50
      SORC1(I, J) = 0.0
      SORC2(I, J) = 0.0
  WRITE(NOUT, 2) (ID(I), I = 1, 12)
  2 FORMAT(1H12A6///)
  DO 1000 K = 1, IM
    READ(NFLUX1) ((PHI(I, J), I = 1, IM), J = 1, JM)
    READ(NFLUX2) ((ADJ(I, J), I = 1, IM), J = 1, JM)
    READ(NSCRAT) ((C0(I, J), I = 1, ITL), J = 1, MT)
    DO 3 II = 1, ITL
      CX(II, K) = C0(II, MAT)
  3 IF(NR = 400, 40, 10)

C DELTA K CALCULATION FOR ROW = NROW
  CONTINUE
  DO 60 I = 1, IM
    IF(NR = 400, 40, 10)
  60 NN = NR + 1
    FLUX(1, K + I, 3) = PHI(I, NN)
    FLUX(2, K + I, 3) = ADJ(I, NN)
  30 IF(NR = 40, 40, 40)
  40 NN = NR + 1
    FLUX(1, K + I, 1) = PHI(I, NN)
    FLUX(2, K + I, 1) = ADJ(I, NN)
  50 NN = NR
    FLUX(1, K + I, 2) = PHI(I, NN)
  40
FLUX(2, K, I, 2) = ADJ(I, NN)
SORC2(I+1) = CHI(K)*FLUX(2, K, I, 2) + SORC2(I+1)
SORC1(I+1) = CU(N, MAT)*FLUX(1, K, I, 2) + SORC1(I+1)
SORC1(I+2) = CO(2, MAT)*FLUX(1, K, I, 2)*FLUX(2, K, I, 2)
1 + SORC1(I+2)

CONTINUE
DO 150 I=1, IM
IF(I-1) 82, 82972

PSR1 = (FLUX(1, K, I+2) - FLUX(1, K, I-1, 2))/(R4(I) - R4(I-1))
ASR1 = (FLUX(2, K, I+2) - FLUX(2, K, I-1, 2))/(R4(I) - R4(I-1))

IF(I=IM) 78, 86*86

PSR2 = (FLUX(1, K, I+1, 2) - FLUX(1, K, I+2))/R4(I+1) - R4(I))
ASR2 = (FLUX(2, K, I+1, 2) - FLUX(2, K, I+2))/R4(I+1) - R4(I))
GO TO 90

SLOPR(I) = 0.0
SLOAR(I) = 0.0
GO TO 92

SLOPR(I) = PSR1
SLOAR(I) = ASR1
GO TO 92

SLOPR(I) = (PSR1 + PSR2)/2.0
SLOAR(I) = (ASR1 + ASR2)/2.0
GO TO 92

IF(NR = 1) 100, 1000 93

GO TO (96, 96, 94)*IGEP

PSZ1 = (FLUX(1, K, I+1, 2) - FLUX(1, K, I+1, 2))/(Z4(NR) - Z4(NR-1))
1 *PI2*R4(I))
ASZ1 = (FLUX(2, K, I+1, 2) - FLUX(2, K, I+1, 2))/(Z4(NR) - Z4(NR-1))
1 *PI2*R4(I))
GO TO 97

PSZ1 = (FLUX(1, K, I+1, 2) - FLUX(1, K, I+1, 2))/Z4(NR) - Z4(NR-1))
1 *PI2*R4(I))
ASZ1 = (FLUX(2, K, I+1, 2) - FLUX(2, K, I+1, 2))/Z4(NR) - Z4(NR-1))
1 *PI2*R4(I))
GO TO 97

IF(NR = JM) 100, 115, 115
GO TO (105, 105, 102)*IGEP

PSZ2 = (FLUX(1, K, I+1, 2) - FLUX(1, K, I+1, 2))/Z4(NR+1) - Z4(NR+1))
1 *PI2*R4(I))
ASZ2 = (FLUX(2, K, I+1, 2) - FLUX(2, K, I+1, 2))/Z4(NR+1) - Z4(NR+1))
1 *PI2*R4(I))
GO TO 106

PSZ2 = (FLUX(1, K, I+1, 2) - FLUX(1, K, I+1, 2))/Z4(NR+1) - Z4(NR+1))
1 *PI2*R4(I))
ASZ2 = (FLUX(2, K, I+1, 2) - FLUX(2, K, I+1, 2))/Z4(NR+1) - Z4(NR+1))
1 *PI2*R4(I))
GO TO 106

IF(NR = JM) 100, 115, 115
GO TO 130

CONTINUE
IF(NR = JM) 120, 115, 115
115  SLOPZ(I) = PSZ1
    SLOAZ(I) = ASZ1
    GO TO 130
120  CONTINUE
    SLOPZ(I) = (PSZ1 + PSZ2)/2.0
    SLOAZ(I) = (ASZ1 + ASZ2)/2.0
130  CONTINUE
    KK = M0(I, NR)
    KK = M2(KK)
    SORC1(I,3) = ((SLOPZ(I)*SLOAZ(I) + SLOPR(I)*SLOAR(I))*C0(49*MAT)/
                  1*(C0(4*KK)*C0(4*KK)*3.0) + SORC1(I,3))
    150 CONTINUE
    GO TO 1000
C  DELTA K CALCULATION FOR COLUMN = NCOL
400  CONTINUE
    DO 460 J=1,JM
        IF(NC = IM) 420,440,440
        420 NN = NC + 1
            FLUX(1*K+J+3) = PHI(NN+J)
            FLUX(2*K+J+3) = ADJ(NN+J)
        430 IF(NC = -1) 450,450,440
        440 NN = NC + 1
            FLUX(1*K+J+1) = PHI(NN+J)
            FLUX(2*K+J+1) = ADJ(NN+J)
450 NN = NC
    FLUX(1*K+J+2) = PHI(NN+J)
    FLUX(2*K+J+2) = ADJ(NN+J)
460 CONTINUE
    DO 550 J=1,JM
        IF(J = 1) 4789,4789,472
        472 GO TO (474, 474, 473), IGEP
        473 PSZ1 = (FLUX(1*K+J+2) - FLUX(1*K+J-1+2))/(Z4(J) - Z4(J-1))
                1*PI2*R4(NC)
        ASZ1 = (FLUX(2*K+J+2) - FLUX(2*K+J-1+2))/(Z4(J) - Z4(J-1))
                1*PI2*R4(NC)
        GO TO 476
474 PSZ1 = (FLUX(1*K+J+2) - FLUX(1*K+J-1+2))/(Z4(J) - Z4(J-1))
        ASZ1 = (FLUX(2*K+J+2) - FLUX(2*K+J-1+2))/(Z4(J) - Z4(J-1))
476 IF(J = JM) 4789,486,486
478 GO TO (480, 480, 479), IGEP
479 PSZ2 = (FLUX(1*K+J+1+2) - FLUX(1*K+J+2))/(Z4(J+1) - Z4(J))
        1*PI2*R4(NC)
        ASZ2 = (FLUX(2*K+J+1+2) - FLUX(2*K+J+2))/(Z4(J+1) - Z4(J))
I *PI2*R4(NC)
GO TO 481
PSZ2 = (FLUX(1*K,J+1,2) - FLUX(1*K,J+2))/Z4(J+1) - Z4(J))
ASZ2 = (FLUX(2*K,J+1,2) - FLUX(2*K,J+2))/Z4(J+1) - Z4(J))
IF(J - 1) 482, 482, 490
SLOPZ(J) = PSZ2
SLOAZ(J) = ASZ2
GO TO 492
SLOPZ(J) = PSZ1
SLOAZ(J) = ASZ1
GO TO 492
SLOPZ(J) = (PSZ1 + PSZ2)/2.0
SLOAZ(J) = (ASZ1 + ASZ2)/2.0
CONTINUE
IF(NC - 1) 508, 508, 495
PSR1 = (FLUX(1*K,J+1,2) - FLUX(1*K,J+2))/R4(NC) - R4(NC-1))
ASR1 = (FLUX(2*K,J+1,2) - FLUX(2*K,J+2))/R4(NC) - R4(NC-1))
IF(NC = IM) 500, 515, 515
PSR2 = (FLUX(1*K,J+3) - FLUX(1*K,J+2))/R4(NC+1) - R4(NC))
ASR2 = (FLUX(2*K,J+3) - FLUX(2*K,J+2))/R4(NC+1) - R4(NC))
IF(NC - 1) 508, 508, 520
SLOPR(J) = 0.0
SLOAR(J) = 0.0
GO TO 530
CONTINUE
IF(NC - 1) 520,515,515
SLOPR(J) = PSR1
SLOAR(J) = ASR1
GO TO 530
CONTINUE
SLOPR(J) = (PSR1 + PSR2)/2.0
SLOAR(J) = (ASR1 + ASR2)/2.0
CONTINUE
KK = M2(NC,J)
SORC1(J,3) = ((SLOPZ(J)*SLOAZ(J) + SLOPR(J)*SLOAR(J))*C0(4,MAT))/
1 (C0(4,KK)*C0(4,KK)*3.0) + SORC1(J,3)
CONTINUE
1000 CONTINUE
1010 IGGM = IGGM + 1
DO 1210 1 = 1, IGGM
IF(NR) 1100, 1100, 1050
1050 DO 1075 1 = 1, IM
DO 1070 1 = 1, IM
N = NN + K
L = 5 + NN
SORC2(I, 2) = SORC2(I, 2) + FLUX(K, I, 2)*CX(L, N)*
1 (FLUX(K, I, 2) - FLUX(N, I, 2))
IF(N = IGX) 1070, 1075, 1075
1070 CONTINUE
1075 CONTINUE
GO TO 1200
1100 DO 1175 J = 1, JM
DO 1170 NN = 1, IST
N = NN + K
L = 5 + NN
SORC2(J, 2) = SORC2(J, 2) + FLUX(K, J, 2)*CX(L, N)*
1 (FLUX(K, J, 2) - FLUX(N, J, 2))
IF(N = IGX) 1170, 1175, 1175
1170 CONTINUE
1175 CONTINUE
1200 CONTINUE
1210 CONTINUE
C SET KEFF EQUAL TO READ IN VALUE
1225 DENOM = DENOM*AK/ZKEFF
AK = ZKEFF
1250 CONTINUE
IF(NR) 3000, 3000, 2000
2000 WRITE(NOUT, 2050) MAT, HLN(MAT), NR
WRITE(NOUT, 2060)
WRITE(NOUT, 2062)
SUM = 0.0
TOT = 0.0
DO 2200 I = 1, IM
C FISSIONS
SORC2(I, 3) = (SORC2(I, 1)*SORC1(I, 1)*CT)/(AK*DENOM)
C ABSORPTIONS
SORC1(I, 2) = (SORC1(I, 2)*CT)/DENOM
C SLOWING DOWN
SORC2(I, 2) = (SORC2(I, 2)*CT)/DENOM
C LEAKAGE
SORC1(I, 3) = (SORC1(I, 3)*CT)/DENOM
C DELTA K
SORC2(I, 4) = SORC2(I, 3) - SORC1(I, 2) - SORC1(I, 3) - SORC2(I, 2)
SUM = SUM + SORC2(I, 4)*V0(I, NR)
TOT = TOT + V0(I, NR)
ST = SUM/TOT
C FISSIONS
C
2050 FORMAT(38HOREACTIVITY COEFFICIENTS FOR MATERIAL 12*6X*A6,
1 11H ROW 12//)
2060 FORMAT(131H0 SK/K PER KG K K K
1 6K 6K 6K AVG RADII
2 AVG AX II 1)
2062 FORMAT (99H, (NUSIGF), (SIGA)
1 (SIGTR), (SIGJXK), (INTG) / )
SORC1(I, 2) = -SORC1(I, 2)
SORC1(I, 3) = -SORC1(I, 3)
SORC2(I, 2) = -SORC2(I, 2)
WRITE(NOUT*, 2480) J, SORC1(I, 4), SORC2(I, 3), SORC1(I, 2),
1 SORC1(I, 3), SORC2(I, 2), ST, R4(I), Z4(NR)
2080 FORMAT(I3, 8F16.4)
2200 CONTINUE
GO TO 3400
3000 WRITE(NOUT*, 3050) MAT, HOLN(MAT), NC
WRITE(NOUT*, 2060)
WRITE(NOUT*, 2062)
SUM = 0.0
TOT = 0.0
DO 3200 I = 1, JM
SORC2(I, 3) = (SORC2(I, 1) + SORC1(I, 1) * CT) / (AK * DENOM)
SORC1(I, 2) = (SORC1(I, 2) * CT) / DENOM
SORC2(I, 2) = (SORC2(I, 2) * CT) / DENOM
SORC1(I, 3) = (SORC1(I, 3) * CT) / DENOM
SORC2(I, 4) = SORC2(I, 3) - SORC1(I, 2) - SORC1(I, 3) - SORC2(I, 2)
SUM = SUM + SORC2(I, 4) * VO(NC*1)
TOT = TOT + VO(NC*1)
ST = SUM / TOT
3050 FORMAT (38MOREACTIVITY COEFFICIENTS FOR MATERIAL I2, 6X, A6, 
11X, COLUMN I2//)
SORC1(I, 2) = -SORC1(I, 2)
SORC1(I, 3) = -SORC1(I, 3)
SORC2(I, 2) = -SORC2(I, 2)
WRITE(NOUT*, 2480) J, SORC1(I, 4), SORC2(I, 3), SORC1(I, 2),
1 SORC1(I, 3), SORC2(I, 2), ST, R4(I), Z4(NR)
3200 CONTINUE
3400 CONTINUE
9999 CONTINUE
REWIND NCR1
REWIND NFLUX1
REWIND NFLUX2
REWIND NSCRAT
RETURN
END