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# ACDOS1: A COMPUTER CODE TO CALCULATE DOSE RATES FROM NEUTRON ACTIVATION OF NEUTRAL BEAMLINES AND OTHER FUSION REACTOR COMPONENTS 

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ACDOSI: A COMPUTER CODE TO CALCULATE DESE RATES FROM NEUTRON ACTIVATION OF NEDTRAL BEAMLINES AND OIHER FUSION-REACTOR COMPONENTS

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(M.S. thesis)

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ACDOS1: A Computer Code to Calculate Dose Rates From Neutron Activation of Neutral Beamlines and Other Fusion-Reactor Components<br>By Gregory 5. Keney<br>> (M.S. Thesis)<br>Accelerator and Fusion Research Division Lawrence Berkeley Laboratory University of California Berkeley, California, 94720

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## ABSTRACT

A computer code has been written to caiculate neutron induced activaition of neutral-beam injector components and the corresponding dose rates as a function of geometry, component composition, and time after shutdown. The code, ACDOSl, was written in Fortran IV to calculate both activity and dose rates for up to 30 target nuclides and 50 nevtron groups. Sufficient versatility has also been incorporated into the code to make it applicable to a variety of general activation problems due to neutrons of energy less than 20 MeV .

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## Introduction

This project has originated from a need in the fusion community to calculate dose rates from neutron-induced activation of fusion components. In particular, the subject matter of this work revolves around the calculations required to determine the dose rates from neutral-beam injector components as a function of geometry, component composition, and time after shutdown.

The methodology for solving the overall problem is broken down into essentially five major steps. The first is that of determining a mathematical model for the production of neutrons by the neutral-beam injector. In practice, neutron production is not uniform but occurs during a succession of evenly spaced pulses of short time duration. During the pauses, no neutrons are produced. An effective steady neutron source term is generated by utilizing an appropriate duty factor over the time duration of the pulses. The neutron production rate is then calculated on the basis of the energy and current of the neutral beam. The second step required writing and solving the pertinent ordinary differential equation that describes the system undergoing activation and then algebraically moaifying the standard solution to make it amenable to the arbitrary injector test schedule and adapting the results to multigroup calculations. With the unity-normalized
average group fluxes assumed to be available from a previous calculation, the third step involves calculating the flux weighted group cross sections by making use of a library containing microscopic neutron cross sections. Prerequisite to this calculation is the determination of a weighting flux for use in averaging cross section data over the appropriate energy groups. This was accomplished by assuming a room-temperature Maxwell-Boltzmann distribution for the thermal region, a $1 / E$ function for intermediate energies, and an exponential function for the fast neutron groups. The forms of the equations are all known. Arbitrary constants in the weighting fiuxes are determined by integral and/or boundary conditions on the known flux for the particular energy group in question. With the weighting flux a determinable function of enei:gy, the required flux weighted group cross sections can be evaluated. The fourth step involves a calculation of the resultant activities given the neutron production rate, the flux weighted group cross sections, target nuclide mass and type, and the specific times after shutdown at which induced activities are desired. The routine which calculates activity interrogates the cross section library to find the atomic weight of the specific target nuclide and in addition, the product nuclide and its halflife. The fifth and final step is a calculation of the dose rates as a function of time and geometry given the
previously calculated activities. This last routine interrogates a decay library and extracts the necessary gamma ray energy and intensity information. The progran incorporates a gamma-flux-to-dose-rate function for several simple geometries. The user specifies the type of geometry to be used in the actual calculations-point, spherical, or cylindrical-on-axis, in addition to the distance from the activated component at which the dose rate is desired. The result of the five steps is a code called ACDOSl which performs the activation and dose rate calculations given the required input information. The code can handle from one to fifty neutron groups with up to thirty target nuclides and has an cption which allows the user to substitute a neutron source term other than that from a neu-tral-beam injector, thereby increasing the versatility of the code. In addition, the entire code is written in ANSI FORTRAN IV which shoula minimize compiler compatability problems at other user sites. Interested users should submit a written request for additional information. Address requests to: Head, Neutral Beam Development Group, Bldg. 5, Lawrence Berkeley Laboratory, Berkeley, California, 94720.

The end result of this project, then, is a code that is tailored to solve injector activation problems, but contains enough versatility to be useful in solving a variety of general activation problems produced by neutrons of energy less than 20 MeV .

## Neutron Production Model

The first majoz step in solving the overall activation problem is to determine a mathematical model for the production of neutrons by the injector assembly. As previously mentioned, neutron production is not uniform but occurs during an arbitrary succession of evenly spaced injector pulses of short time duration. Neutron production coincides with the injector pulses. Figure 1 depicts schematically how this pulse-pause nature of neutron production might be viewed. INSNPS is the instantaneous number of neutrons produced per second. Tl is the length of time the short pulses and pauses are occurring and $T 2$ is the length of time in which no pulsing-pausing occurs. It is assumed throughout the activation calculations that $T 1$ and $T 2$ do not vary during the operation of the injector.

The neutrons originate from two sources within the injector system. Gaseous deuterium that leaks from the neutralizer and ion-source adsorbs on the surface of the copper target located in the ion-dump and neutral-target sections of the injector. A ( $d, n$ ) reaction occurs at the far end of the injector as $D^{\circ}$ impinges upon the deuterium. The reaction is:

$$
{ }^{2} \mathrm{H}+{ }^{2} \mathrm{H} \longrightarrow 3^{3} \mathrm{He}+\mathrm{l}_{\mathrm{n}}+\mathrm{MeV}
$$

In addition, a similar reaction occurs as $\mathrm{D}^{+}$impinges upon deuteriun adsorbed on the copper target located in the iondump section.

The calculation done to determine the instantaneous number of neutrons produced per second is empirical in nature and taken under accelerator test conditions. (1) The expression used is:

$$
\begin{gathered}
\text { INSNPS }=8.64 \times 10^{4} \frac{\text { neutrons }}{\mu \text { coulomb }} \times \mathrm{A} \times 1.0 \times 10^{6} \frac{\mu \text { coulomb }}{\sec } \times \\
\text { F } \times \text { CF }
\end{gathered}
$$

where the variables are defined as below:
INSNPS: the instantaneous number of neutrons produced per second

A: current in amperes
F: fraction of the beam that is monatomic
CF: yield correction factor for voltages different from 150 keV

The yield correction factor for voltages different than 150 keV is taken from the calculation ${ }^{(2)}$ reproduced in Figure 2. To facilitate the calculation of INSNPS during execution of the program, a power fit of the form $\mathrm{y}=\mathrm{a}^{\mathrm{x}}$ was applied to the points read from the curve to determine an analytical expression for CF. Three different fits were used to insure a high coefficient of determination. These expressions are incorporated into subroutine SOURCE which


Figure 1

Figure 2

is the subprogram which calculates INSNPS.
In modeling the beam pulses, INSNPS is multiplied by the fraction of the time the beam is on (the duty factor DF) to obtain the average number of neutrons produced per second during time T1, AVENPS. The short successive pulses are now represented by a single pulse (now referred to as a "test") with the same time length $T 1$ but reduced in magnitude as shown in Figure 3. This type of pulse modeling is used throughout the project in connection with activation calculations. Typically, Tl is several hours. (T2, which represents an outage period as required by maintenance or overnight shutdowns, is also several hours in duration). The quantities, Tl and T 2 , and N , the total number of tests, are variables and must ke specified by the user prior to execution of the program.

## Figure 3



Derivation \& Modification of the Activation Equation

The balance equation for the activation of target nuclides in the injector system is of the following form:
rate of change of $=$ rate of rate of a product nuclide $=$ production ${ }^{-}$decay .

In differential form, this equation becomes

$$
\frac{d B(t)}{d t}=\sigma_{A} g_{A}-B(t) \lambda_{B}
$$

where the variables are defined as below:
$B(t)$ : number of atoms of the product nuclide $B$
$\boldsymbol{\lambda}_{B}$ : decay constant for the product nuclide $B$ $\sigma_{\bar{A}}$ : group activation cross section for target nuclide A

A: number of atoms in the target nuclide A $\phi$ : group neutron flux

The solution is:

$$
B(t)=\frac{\sigma_{A} \phi_{A}}{\lambda_{B}}\left[1-e^{\lambda_{B} t}\right]
$$

This is the basic differential equation that is used in
subsequent activation calculations. This form, nowever, must be algebraically modified to take into account pauses of length $T 2$ in the injector test schedule as shown in Figure 4. Tl is the length of the test, $T 2$ is the length of the pause, $t=0$ is the time at which shutdown occurs, and tl and t 2 are arbitrarily selected times after shutdown where an activity is required. $N$ is the total number of tests. It is desired to calculate the amount of an activated nuclide at t=0 due to an arbitrary neutron testing history. The method of approach is to calculate the amount at $t=0$ due to each test prior to $t=0$. The total amount at $t=0$ will theil be the sum of the contriblations from each test. For some time $t$ after shutdown, the amount of nuclide B will be given by:

$$
\begin{aligned}
& B(t)=\frac{A \sigma A}{\lambda_{B}}\left[1-e^{-\lambda_{B} \cdot T 1}\right] \\
& {\left[\sum_{i=1}^{N} e^{\left.-\{[N-i][T 1+T 2]\} \lambda_{B}\right] e^{-\lambda_{B} t}}\right.}
\end{aligned}
$$

This expression, however, represents only I group of neutrons. Tc generalize the relation for up to $\boldsymbol{n}$ neutron groups, the calculation for $B(t)$ above must be done for each group, that is:

## Figure 4



$$
\begin{aligned}
& B(t)=\sum_{K=1}^{n} \frac{A \sigma_{A K} \phi}{\lambda_{B}}\left[1-e^{-\lambda_{B} \cdot T 1}\right] e^{-\lambda_{B} t} \\
& {\left[\sum_{i=1}^{N} e^{\left.-\{[N-i][T 1+T 2]\} \lambda_{B}\right]}\right.}
\end{aligned}
$$

For the general case of the reaction $A(n, b) B$, the activity of $B$ in $B q, t$ seconds after shutdown will be:

$$
\begin{align*}
& B(t) \lambda_{B}=\begin{array}{c}
\text { Activity of } \\
\text { Product Nuclide } \\
B
\end{array}= \\
& \sum_{K=1}^{n} A_{A_{K}} \phi_{K}\left[1-e^{-\lambda_{B} \cdot T 1}\right] e^{-\lambda_{B} t} \\
& {\left[\sum_{i=1}^{N} e^{-\left\{[N-i][T 1+T 2] \lambda_{B}\right]}\right.} \tag{1}
\end{align*}
$$

Since the amount of a target nuclide ( $A$ in this case) is entered into computer memory in terms of kilograms, the number of target atoms "A" is computed by multiplying the input mass by the quantity:

$$
\frac{(1000 \mathrm{~g} / \mathrm{kg}) \quad\left(6.023 \times 10^{23} \text { atoms } / \mathrm{mol}\right)}{\text { atomic weight in } \mathrm{g} / \mathrm{mol}}
$$

Equation 1 is the activation equation which is used in ACDOSl for calculating total activities. Note that the equation is applicable to situations where the flux is constant as opposed to a "pulsing" nature by the disapm pearance of the second bracketed term when $\mathrm{N}=1$.

Determination of the Weighting Flux
\& Flux Weighted Group Cross Sections

In order to average cross section data over the appropriate energy groups to acquire flux weighted group cross sections, a weighting flux is required. The flux weighted group cross section $\overline{\boldsymbol{\sigma}}$, can then be evaluated from the expression:

$$
\bar{\sigma}=\int_{\Delta E} \sigma(E) \phi(E) d E / \int_{\Delta E} \phi(E) d E
$$

The problem was solved by assuming that the thermal region couid be represented by a classical Maxwell-Boltzmann distribution with $k T=0.025 e V$, intermediate energies by a $1 / E$ function and fast groups by an exponential function, with all equations known except for arbitrary constants.

The decision to use this approach came in part from results of slowing-down theory, and, in addition, from results of the Nonte Carlo calculations which produced the data for the sample problem. (The latter involves a neutral beam injector which is surrounded by thick concrete walls.) The group fluxes calculated ${ }^{(3)}$ for this case are given in Table 1. To verify the above assumptions, column 3 (neutron/cm ${ }^{2} \cdot \mathrm{eV}$.source neutron) was plotted as a function of the arithmetic average of the group boundaries found in column 2 for groups 1 through 19. See Figure 5.

NOTE: The convention will be adrpted that Group 1 represents the highest energy group.

| GP | Energy Interval (ev) | $\begin{aligned} & \text { Neutrons/ } \\ & \mathrm{cm}^{2} \cdot \mathrm{ev} \text {. } \end{aligned}$ <br> Source Neutron | Neutrons/ $\mathrm{cm}^{2}$. <br> Source Neutron |
| :---: | :---: | :---: | :---: |
| 1 | $2.385 \mathrm{E}+6$ to 2.307E+6 | $1.013 \mathrm{E}-10$ | 7.901E-6 |
| 2 | $2.307 \mathrm{E}+6$ to $1.827 \mathrm{E}+6$ | 2.879E-13 | 1.382E-7 |
| 3 | $1.827 \mathrm{E}+6$ to 1.108E+6 | 1.959E-13 | $8.833 \mathrm{E}-8$ |
| 4 | $1.108 \mathrm{E}+6$ to $5.502 \mathrm{E}+5$ | $1.896 \mathrm{E}-13$ | $1.058 \mathrm{E}-7$ |
| 5 | $5.502 \mathrm{E}+5$ to $1.576 \mathrm{E}+5$ | $3.510 \mathrm{E}-13$ | 1.378E-7 |
| 6 | $1.576 \mathrm{E}+5$ to $1.111 \mathrm{E}+5$ | $6.111 \mathrm{E}-13$ | 2.842E-8 |
| 7 | $1.111 \mathrm{E}+5$ to $5.248 \mathrm{E}+4$ | $8.172 \mathrm{E}-13$ | 4.790E-8 |
| 8 | $5.248 \mathrm{E}+4$ to $2.479 \mathrm{E}+4$ | 1.394E-12 | $3.860 \mathrm{E}-8$ |
| 9 | $2.479 \mathrm{E}+4$ to $2.188 \mathrm{E}+4$ | 2.550E-12 | $7.420 \mathrm{E}-9$ |
| 10 | $2.188 \mathrm{E}+4$ to $1.033 \mathrm{E}+4$ | 3.049E-12 | 3.522E-8 |
| 11 | $1.033 \mathrm{E}+4$ to $3.355 \mathrm{E}+3$ | $6.552 \mathrm{E}-12$ | 4.570E-8 |
| 12 | $3.355 \mathrm{E}+3$ to $1.234 \mathrm{E}+3$ | 1.896E-11 | $4.021 \mathrm{E}-8$ |
| 13 | $1.234 \mathrm{E}+3$ to $5.829 \mathrm{E}+2$ | 4.673E-11 | $3.043 \mathrm{E}-8$ |
| 14 | $5.829 \mathrm{E}+2$ to $1.013 \mathrm{E}+2$ | $1.502 \mathrm{E}-10$ | $7.234 \mathrm{E}-8$ |
| 15 | $1.013 \mathrm{E}+2$ to $2.902 \mathrm{E}+1$ | 6.481E-10 | $4.684 \mathrm{E}-8$ |
| 16 | 2.902E+1 to 1.06BE+1 | 1.642E-9 | $3.011 \mathrm{E}-8$ |
| 17 | $1.068 \mathrm{E}+1$ to $3.059 \mathrm{E}+0$ | $5.184 \mathrm{E}-9$ | $3.951 \mathrm{E}-8$ |
| 18 | $3.059 \mathrm{E}+0$ to $1.125 \mathrm{E}+0$ | $1.787 \mathrm{E}-8$ | $3.456 \mathrm{E}-8$ |
| 19 | $1.125 E+0$ to $4.140 \mathrm{E}-1$ | 4.327E-8 | $3.076 \mathrm{E}-8$ |
| 20 | $4.140 \mathrm{E}-1$ to $1.000 \mathrm{E}-5$ | $2.002 \mathrm{E}-6$ | $8.288 \mathrm{E}-7$ |

Table 1


Figure 5


Figure 5


Inspection of the curve does suggest a $1 / E$ type of behavior for groups 4 through 19 since a quantity that varies as a constant/E will plot as a straight line on log-log paper and will have a slope of -1 . Calculation of the slope at various points along the line did reveal a slofe close to -1. At higher energies (above $2 \times 10^{4} \mathrm{eV}$ ) the curve departs somewhat more from $1 / E$ behavior and finally feaches a region just below fast energies where the curve is neither 1/E nor exponential. Even though this region just below the exponential part of the curve deviates from $1 / E$ behavior, it is assumed, for ease of calculations, that it does behave as $1 / E$. Since neutrons are "born" with a narrow group of discrete energies in the injector as contrasted with a fission spectrum in a reactor, one would intuitively expect the fast region to appear more as a "spike" then a "spread nut" distribution. The graph definitely illustrates this behavior and for this reason the assumption of an exponential form for the fast groups appears justified. As stated previously, the Naxwell-Boltzmann distribution is used for describing the thermal region. At this point, the forms of the equations for describing the weighting flux are known except for arbitrary constants. To evaluate the constants, each analytic expression for the weighting flux is integrated over the appropriate energy group and set equal to the numerical value of the corresponding group quantities in column 4 of Table 1, which are normalized
total group fluxes. For example, for the thermal region, group 20:

$$
\int_{1.0 \times 10^{-5}}^{4.14 \times 10^{-1}} c \sqrt{E} e^{-\frac{E}{0.025}} d E=8.288 \times 10^{-7}
$$

therefore,

$$
c=2.36 \times 10^{-4}
$$

For an intermediate neutron group, group 17

$$
\int_{3.059}^{10.68} \mathrm{C} / \mathrm{E} \quad \mathrm{~d} E=3.951 \times 10^{-8}
$$

therefore,

$$
c=3.160 \times 10^{-8}
$$

The determination of the constants associated with fast neutron groups, however, is somewhat more involved. Since the assumed form for the fast exponential is

$$
\phi(E)=A e^{\frac{E 1-E 2}{B}}
$$

where El and E2 are knowns and
E2 is the highest energy of that group, two equations are required to evaluate the two constants. One equation results from the usual integral condition that
(2) $\int \begin{aligned} & E 2 \\ & E l \\ & \text { Ae } \\ & \frac{E 1-E 2}{B} \\ & d E\end{aligned}=\begin{gathered}\text { total flux for } \\ \text { that group }\end{gathered}$
while the other condition results from a continuity of flux requirement at the boundary of the two groups. Equation 2 already represents one equation. A secu:cs equation resulting from the continuity requirement is:

$$
C / E 1=A e^{\frac{E L-E 2}{B}}
$$

where $C$ is a known constant from the integral condition placed on the last $1 / E$ group. (Analysis is always done starting with the lowest energy group). See Figure 6. With two equations and two unknowns, the constants can be determined by solving the resultant transcendental equations. If there is another fast group, the same method is applied. One equation results from the required integral condition on that group, and a second equation results from a continuity of flux requirement at the boundary with the preceding group. The result of this method is that the first exponential group is always matched to the last $1 / E$ group, and any additional fast groups are always matched to the p:receding fast groups. In the case of only two groups, one thermal and one fast, a match is required at the boundary between the groups to allow evaluation of the two constants in the assumed exponential function following the Maxwellian. For $1 / E$ groups, only one equation is re-


A continuity of flux requirement is used at energy
El to obtain another equation required to evaluate the two unknown constants in the assumed exponential function for group 2. This equation is:

$$
\mathrm{C} / \mathrm{El}=\mathrm{A} \mathrm{e}^{\frac{\mathrm{El}-\mathrm{E} 2}{\mathrm{~B}}}
$$

See appendix $A$ for the forms and solutions to these transcendental equations.
quired as there is only one constant to be evaluated. Since l/E constants are calculated solely from the integral condition placed on the equation for that group, discontinuities sometimes result in the flux at the group boundary between two adjacent $1 / E$ groups or between the thermal and the first $1 / E$ group. This is not a serious problem because the discontinuities are not significantly large. The important consideration is that the integral of the weighting flux function over the group be equal to the total flux for that group so that the correct averaging of the flux over that same interval is maintained. Table 2 shows the results of this approach to the weighting flux function determination. The integral of these functions over their proper energy range always results in the numerical value of the total flux for that group as it should. A plot of the data in Table 2 (Groups l-19) is shown in Figure 7. As previously mentioned, slight flux discontinuities at boundaries between $1 / E$ groups exists. Note that the discontinuities are greater in the region just below the two fast groups. This is not surprising since that region deviates somewhat from $1 / E$ but was assumed to be $1 / E$ in weighting flux calculations. Comparison of Figure 5 with Figure 7 shows a close resemblance as one would definitely want and expect.

During the execution of ACDOSl, subroutine WTFLUX will determine the values of the constants associated with these

Table 2
Flux Densities for the Sample Problem

| Group | Group Boundaries (ev) | Equation $\phi(E)$ |
| :---: | :---: | :---: |
| 20 | $1.0 \times 10^{-5}$ to 0.414 | $\begin{aligned} & 2.36 \times 10^{-4} \text { Eexp } \\ & \left(-\frac{E}{0.025}\right) \end{aligned}$ |
| 19 | 0.414 to 1.125 | $3.077 \times 10^{-8} \mathrm{E}^{-1}$ |
| 18 | 1.125 to 3.059 | $3.455 \times 10^{-8} E^{-1}$ |
| 17 | 3.059 to 10.68 | $3.160 \times 10^{-8} E^{-1}$ |
| 16 | 10.68 to 29.02 | $3.012 \times 10^{-8} E^{-1}$ |
| 15 | 29.02 to 101.3 | $3.747 \times 10^{-8} \mathrm{E}^{-1}$ |
| 14 | 101.3 to 582.9 | $4.134 \times 10^{-8} E^{-1}$ |
| 13 | 582.9 to $1.234 \times 10^{3}$ | $4.057 \times 10^{-8} \mathrm{E}^{-1}$ |
| 12 | $1.234 \times 10^{3}$ to $3.355 \times 10^{3}$ | $4.020 \times 10^{-8} \mathrm{E}^{-1}$ |
| 11 | $3.355 \times 10^{3}$ to $1.033 \times 10^{4}$ | $4.064 \times 10^{-8} E^{-1}$ |
| 10 | $1.033 \times 10^{4}$ to $2.188 \times 10^{4}$ | $4.693 \times 10^{-8} E^{-1}$ |
| 9 | $2.188 \times 10^{4}$ to $2.479 \times 10^{4}$ | $5.942 \times 10^{-8} E^{-1}$ |
| 8 | $2.479 \times 10^{4}$ to $5.248 \times 10^{4}$ | $5.147 \times 10^{-8} E^{-1}$ |
| 7 | $5.248 \times 10^{4}$ to $1.111 \times 10^{5}$ | $6.387 \times 10^{-8} E^{-1}$ |
| 6 | $1.111 \times 10^{5}$ to $1.576 \times 10^{5}$ | $8.129 \times 10^{-8} E^{-1}$ |
| 5 | $1.576 \times 10^{5}$ to $5.502 \times 10^{5}$ | $1.102 \times 10^{-7} \mathrm{E}^{-1}$ |
| 4 | $5.502 \times 10^{5}$ to $1.108 \times 10^{5}$ | $1.511 \times 10^{-7} \mathrm{E}^{-1}$ |
| 3 | $1.108 \times 10^{6}$ to $1.827 \times 10^{6}$ | $1.666 \times 10^{-7} E^{-1}$ |
| 2 | $1.827 \times 10^{6}$ to $2.307 \times 10^{6}$ | $\begin{aligned} & 6.62 \times 10^{-13} \mathrm{exp} \\ & \left(\mathrm{E}-2.307 \times 10^{6} /\right. \\ & \left.2.42 \times 10^{5}\right) \end{aligned}$ |
| 1 | $2.307 \times 10^{6}$ to $2.385 \times 1.0^{6}$ | $\begin{aligned} & 7.07 \times 10^{-10} \mathrm{exp}^{\left(\mathrm{E}-2.385 \times 10^{6}, 11,184\right)} \end{aligned}$ |

Figure 7



assumed functions. The particular forms of the weighting functions chosen depend upon the number of neutron groups in any given activation problem as follows:

| Number of Neutron Groups | Group Energy Designation | Form (s) of Weighting Function(s) |
| :---: | :---: | :---: |
| 1 | Thermal | Maxwellian |
| 2 | $\begin{aligned} & 1 \text { Thermal } \\ & \text { I Fast } \end{aligned}$ | Maxwellian Exponential |
| 3 | ```l Thermal l Intermedi- ate l Fast``` | Maxwellian $1 / E$ <br> Exponential |
| 4 to 50 | 1 Thermal <br> 1 or more <br> Intermediate <br> 1 or more Fast | Maxwellian <br> $1 / E$ <br> Exponential |

For four or more neutron groups, the user must specify the number of fast groups, NF, so that the correct apportionment of $1 / E$ groups can be made.

With the weighting flux function known for each group, the flux weighted group cross section can be calculated from the expression:

$$
\bar{\sigma}=\int_{\Delta E} \sigma(E) \phi(E) d E / \int_{\Delta E} \phi(E) d E
$$

where $\boldsymbol{\phi}(E)$ is the previously determined weighting flux function. During execution of the subroutine that calculates activation, ACTVAT, the cross section information for the target nuclide in question is read from the ACTLMFE library, which is a subset (without fission cross sections) of the ACTL library. (4) The data consist of two numbers, an energy in MeV and an associated cross section for that energy in barns. To calculate $\bar{\sigma}$, a subroutine, SMOOTH, first merges the group boundary values into the energy-cross-section pair data and performs a linear interpolation to calculate the cross section at the group boundaries. (5) The result is two arrays, one with energy values and the other with cross section values. There is always a one to one correspondence. See Figure 8 (less caption). GP (J) and GP (J+1) are arbitrary group boundaries which initially had no cross section value associated with them. The value is obtained by a linear interpolation using the first value on each side of a group boundary assuming that cross sections are linearly interpolable in energy. For two arbitrary cross sections, $\sigma(I)$ and $\sigma(I+1)$ whose values lie within the group boundaries GP(J) and GP (J+I), an analytical expression is found for the line connecting them, using the common two point formula:

$$
y-Y_{I}=\frac{Y_{2}-Y_{1}}{X_{2}-X_{I}} \quad\left(X-X_{1}\right)
$$



Figure 8

Straight lines are connected between two cross section points using the two point formula. The resulting equation for the line is used as the analytical expression for $\sigma(E)$ between $E(I)$ and $E(I+1) . \sigma(E)$ is used in the evaluation of

$$
\Delta \bar{\sigma}=\int_{E(I)}^{E(I+1)} \sigma(E) \phi(E) d E \iint_{E(I)}^{E(I+1)} \quad \phi(E) d E
$$

where $\Delta \bar{\sigma}$ is an arbitrary fraction of the total flux weighted group cross section, $\bar{\sigma}$, for the group in question.

In this case $y$ is $\sigma^{-(E)}$ and $x$ is energy $\Sigma$ in MeV. The equation takes the form:

$$
\sigma(E)-\sigma(I)=\frac{\sigma(I+1)-\sigma(I)}{E(I+1)-E(I)}[E-E(I)]
$$

so that $\sigma(E)$ for $\sigma(I) \leq E \leq \sigma(I+1)$ is

$$
\sigma(E)=\left\{\frac{\sigma(I+1)-\sigma(I)}{E(I+I)-E(I)}[E-E(I)]+\sigma(I)\right\}
$$

See Figure 8.
Now that an analytical expression exists for $\boldsymbol{\sigma}^{( }(E)$ and
(E), the product of these two functions is numerically integrated between $E(I)$ and $E(I+1)$ and the value kept in a running sum. This process is continued, point by point, until the upper boundary is reached. The particular form of $\phi(E)$ used depends upon the energy range in which the specific group boundaries lie. For example, in the thermal region where $G P(J)=1 \times 10^{-5} \mathrm{eV}$ and. $G P(J+1)=0.414 \mathrm{eV}$, $\phi(E)=2.360 \times 10^{-4} \sqrt{E^{-1}} \exp (-(E / 0.025)$ from Table 1 , so that the expression for $\sigma(E) \phi(E)$ is:

$$
\begin{aligned}
& \left\{\frac{\sigma(I+1)-\sigma^{\prime}(I)}{E(I+I)-E(I)}[E-E(I)]+\sigma(I)\right\} \\
& \left\{2.360 \times 10^{4} \sqrt{E} e^{-\frac{E}{0.025}}\right\}
\end{aligned}
$$

It is this expression that is numerically integrated between each thermal $E(I)$ and $E(I+1)$. When all numerical
integrations are completed for a specific group, the running sum is divided by the total flux for that group. The quotient, $\bar{\sigma}$, is the desired quantity--the flux weighted group cross section for that group. This entire process is repeated until all energy groups have been addressed. Several test cases have been run to acquire results for comparison purposes with another "averaging program which originated at LLNL. (5) The values obtained by this method are remarkably close to those obtained by the LLNL code for fast and intermediate neutron groups. For the thermal region, however, the above mentioned procedure produced better results than the LLNL code.

Two further comments are in order considering the method of averaging cross sections. First, for the case of nuclear reactions which are threshold oriented, a zero is substituted for each group cross section whose energy range is below the threshold energy. Calculations begin only when the energy at which a cross section value was measured equals or exceeds the threshold energy. Typically, that first cross section value is located somewhere within the group boundaries at which the calculations begin. The contribution to that particular group cross section comes solely from evaluation of the pertinent guantities for energies greater than or equal to the first cross section value and less than or equal to the first encountered group boundary. This is shown in Figure 9.

Figure 9


In this example, the reaction has a threshold located between the arbitrary group boundaries, GP(J+3) and $G P(J+4)$. The group cross section for group (J+3) results strictly from the evaluation of the pertinent quantities from the threshold value, $T v$, to the first encountered group boundary, $G P(J+4)$. Even if there is only one cross section value located in group (J+3), the calculation proceeds since there is a cross section value associated with GP(J+4) as a result of the initial merging of group boundary energies into the energy-cross-section data. In any case, however, zeros are substituted for group cross sections in group (J) through (J+2) in accordance with the above discussion.

The second comment concerns the calculation of the thermal group cross section. Since no provision is made. for interpolating cross section values at the first group boundary, averaging calculations begin at the first encountered cross section value above the energy of the first group boundary as shown in Figure lo. As a result, a very small portion of the Maxwellian (generally below $1.15 \times 10^{-10} \mathrm{eV}$ ) is not accounted for. However, this is unimportant since the contribution from this part of the Maxwellian is exceedingly small.


Figure 10

## Program to Calculate Activities

The fourth major step in the solution of the problem required writing a standard FORTRAN program to calculate activities given the necessary information. The code is written in a structured format form using one executive program to call up a number of subroutines. Variably dimensioned arrays are used where possible to conserve on memory. The name and purpose of each subroutine is discussed below.

1) INPUT: for entering all "variable" input data into computer memory. Error traps are provided to check input data. All data is printed out for user verification.
2) ARAYIN: for entering all "array" data. Array data is printed out for user verification.
3) SOURCE: given the current, voltage, beam fraction, and duty factor, calculates the instantaneous and average number of source neutrons produced per second during a test. This subroutine is bypassed if the user wishes to use his own source term.
4) GPFLUX: uses the calculated values of the average number of source neutrons
produced per second during a test or the user supplied source term and calculates the average flux for each group by taking the product of the unitynormalized total group iflux and source neutron term.
5) WTFLUX: determines the constants for the assumed weighting flux functions by applying integral and/or group boundary constraints to the pertinent equations. Also prints out the integrals of the weighting flux functions over thear appropriate energy intervals for user verification.
6) AVRAGE: determines an analytical expression for $\sigma(E)$ over a specific $\Delta E$ and numerically integrates the product of $\sigma(E)$ and $\phi(E)$, the weighting function, over the group energy interval. It then takes the sum of the integrated products over the unity normalized total flux for the same group to calculate the flux weighted group cross section, $\sigma$.
7) SMOOTH: first determines if the energy associated with the first energy-cross-section pair read from the ACTLMFE library
is below the greatest group boundary. If so, the routine then merges the group boundary snergies into the en-ergy-cross-section data read from the ACTLMFE library and linearly interpolates to find the values of the cross sections at the group boundaries. If not, the particular reaction is skipped and the next one consideled.
8) POSITN: for positioning the file marker in the ACTLMFE library.
9) POSIT2: same as POSITN.
10) ACTVAT: takes the following parameters and calculates activities according to Equation 1.
a) number of tests
b) length of the tests
c) length of the pauses
d) number of kilograms of a particular nuclide
e) specific times after shutdown
f) average group fluxes
g) flux weighted group cross sections

The result of each activation calculation is printed out. This allows the user to determine what reaction is the most significant for a particular target nuclide.

Moreover, two running sums are maintained in order to print out the total activity from the activation of a particular target nuclide which is subject to many different reactions and, in addition, the total system activity when all target nuclides have been addressed. If, in the case of threshold reactions, the energy read from the first energy-cross-section pair is greater than the highest group boundary, the reaction is bypassed and interrogation continues. Only one pass through the entire ACTLMFE library is required since the nuclide data is arranged in order of increasing $\mathrm{ZA}(1000 \mathrm{Z}+\mathrm{A})$ as is the user specified target nuclide information. The accuracy of the calculations, of course, depend directly on the reliability of the cross section data and the flux calculations. At the termination of subroutine ACTVAT, all activation results will have been stored in memory for future dose rate calculations.

## Program To Calculate Dose Rates

The fifth major and final step required writing a program to calculate dose rates as a function of geometry and time after shutdown. This subprogram, DOSRTE, accomplishes the desired calculation by performing essentially four major tasks:

1) interrogation of the decay library, LEVDEC, which is a subset of ENSL ${ }^{(6)}$, for the energy and multiplicity associated with each applicable gamma ray produced by a specific nuclide,
2) determination of an effective particle flux that yields a unit absorbed dose in soft tissue,
3) calculation of the dose rate using the previously calculated activities, user specified geometry, applicable multjplicities, and appropriate effective particle fluxes, and
4) interrogation of the LEVDEC library, on a second pass, for radioactive daughters.

Provisions are made in the program for keeping two running sums of dose rate values, one for a particular target nuclide and one for the system as a whole. The first הllows che user to determine which of the particular target nuclides is contributing to the highest dose rate values. The second, of course, gives the dose rate for the entire system. After all parent and daughter
nuclides have been addressed, the results are printed out and the ACDOSl object program terminates execution.

Prior to performing the calculations, parent ID numbers are rearranged in order of increasing $2 \mathrm{~A}, 1000$ $\mathrm{Z}+\mathrm{A}$, to match the format of nelides listed in the library. The associated target ID numbers and parent activities and half-lifes are also rearranged at the same time to maintain the correct association of data. All rearrangement occurs automatically during the execution of DOSRTE and is a prerequisite for efficient interrogatron of the library.

Since the same nuclides (parents or daughters) may result from two different target nuclides by different reactions, a temporary holding array is utilized to store the decay information when two or more successive parents have the same ID numbers. This allows repeated interrogation of the information stored in the holding array when the routine finds parents with the same ID number. The use of the array in conjunction with the initial rearrangement of parent ID numbers makes it possible to address all parents on one pass through the LEVDEC library. Dose rates due to each parent are calculated and then subsequently stored in memory with the correct target affiliation prior to addressing any potential radioactive daughters.

After calculating dose rates for applicable parents, radioactive daughters are considerea. Since the
identity of any daughters is not known in advance, a list of such potential candidates (read from the LEVDEC library) is compiled and stored in memory with the proper parent ID number and parent activity affiliation. After all parents have been addressed on the first pass through LEVDEC, and the resulting dose rates calcu!ated, a second pass is made to search the library fore previously compiled daughters. Since the same interrogation methodology is employed as with the parents, only one pass through the library is required.

After finding a particular nuclide in the decay library, parent or daughter, the first consideration prior to calculating any dose rates is the determination of an effective particle flux that yields a unit absorbed dose in soft tissue. For each applicable gamma ray energy read from the LEVDEC library, an effective particle flux is calculated. This calculation was accomplished by first fitting 13 curves to the data (7) shown in Table 3. Twelve of the curves are of the form $a x^{b}$ and one of the form $a e^{b x}$. A large number of curves were chosen so as to accurately reproduce the data over such a wide energy range. (As a result, all coefficients of determination arising from the curve fitting process are in excess of 0.99.) Table 4 shows the results of the curve fittings. The appropriate equation for calculating the effective particle flux, FLUXE, is chosen according to the value

| E(MeV) | Dose Rate of $2.5 \mathrm{mrad} / \mathrm{h}$ corresponds to $\phi_{0}$ gUANTA/ $\mathrm{cm}^{2}$.S |
| :---: | :---: |
| 0.01 | 956 |
| 0.015 | 2,310 |
| 0.020 | 4,320 |
| 0.030 | 9,980 |
| 0.040 | 17,400 |
| 0.050 | 23,100 |
| 0.060 | 25,200 |
| 0.080 | 23,200 |
| 0.100 | 18,600 |
| 0.150 | 10,800 |
| 0.300 | 5,020 |
| 0.400 | 3,660 |
| 0.500 | 2,920 |
| 0.600 | 2,440 |
| 0.800 | 1,880 |
| 1.00 | 1,550 |
| 1.25 | 1,330 |
| 1.50 | 1,130 |
| 2.0 | 912 |
| 3.0 | 686 |
| 4.0 | 559 |
| 5.0 | 480 |
| 6.0 | 420 |
| 8.0 | 339 |
| 10.0 | 284 |


| Energy Range (MeV) | Particle Flux Equation |
| :---: | :---: |
| 0.010 to 0.020 | $\begin{aligned} \text { FLUXE }= & 21495330.64 * E * * \\ & 2.1759447 \end{aligned}$ |
| 0.020 to 0.040 | $\begin{aligned} \text { FLUXE }= & 11460233.22 \star E * * \\ & 2.0136022 \end{aligned}$ |
| 0.040 to 0.050 | $\begin{aligned} \text { FLUXE }= & 1036918.975 * \mathrm{E} * * \\ & 1.2698653 \end{aligned}$ |
| 0.050 to 0.060 | $\begin{gathered} \text { FIUXE=}=96497.38032 * E * * \\ 0.4772399 \end{gathered}$ |
| 0.060 to 0.080 | $\begin{aligned} \text { FLUXE }= & 11225.12221 \mathrm{E}^{*} \mathrm{E}^{*} \\ & (-0.2874410) \end{aligned}$ |
| 0.080 to 0.150 | $\begin{aligned} & \text { FLUXE }= 55482.78326 * E X P \\ &(E *(-10.9132069)) \end{aligned}$ |
| 0.150 to 0.400 | $\begin{aligned} F L U X E= & 1330.569205 * E^{* *} \\ & (-1.1036085) \end{aligned}$ |
| 0.400 to 0.600 | $\begin{aligned} \text { FLUXE }= & 1462.198966^{\star} E^{\star} \star \\ & (-1.0004513) \end{aligned}$ |
| 0.600 to 1.000 | $\begin{aligned} \text { FLUXE }= & 1546.997798 * E * * \\ & (-0.8891058) \end{aligned}$ |
| 1.000 to 1.500 | $\begin{aligned} \text { FLUXE }= & 1549.091320{ }^{*} E^{* *} \\ & (-0.7797676) \end{aligned}$ |
| 1.500 to 3.000 | $\begin{aligned} \text { FLUXE }= & 1508.205757 \star E^{\star *} \\ & (-0.7188784) \end{aligned}$ |
| 3.000 to 6.000 | $\begin{aligned} F L U X E= & 1488.362190 * E^{*} * \\ & (-0.7051622) \end{aligned}$ |
| 6.000 to 10.000 | $\begin{aligned} \text { FLUXE }= & 1656.852874 * E^{\star *} \\ & (-0.7649757) \end{aligned}$ |

Table 4
of $E$ read from the decay library for a particular gamma ray.

The second consideration is that of geometry. Three choices are possible. The first is a point source approximation given by the following equation for the dose rate:

$$
\operatorname{DOSRAT}=\frac{S_{O}}{4 \pi D^{2}}
$$

$S_{o}$ is the source strength divided by the flux-per-unit-dose-rate as given by the formula

$$
S_{0}=\frac{\text { MULT } \times 2.5 \times \mathrm{ACT}}{\text { FLUXE }}
$$

where MULT is the multiplicity of the gamma ray, and FLUXE is the effective photon flux corresponding to a dose rate of $2.5 \mathrm{mrem} / \mathrm{h}$ in soft tissue. $D$ is the radial distance in cm to the point where the dose rate is desired. The second shoice is that of cylindrical-on-axis geometry subject to the condition that the cylinder be non-absorbing. The equation ${ }^{9}$ that is used to calculate the dose rate as a function of the distance from one end is:

$$
\begin{aligned}
& \operatorname{DOSRAT}=\frac{S V}{4}\left\{( H T + D ) \left[\ln \left(1+\frac{R^{2}}{(H T+D)^{2}}\right)+\right.\right. \\
& \frac{2 R}{H T+D} A R C T A N \\
&\left.\frac{H T+D}{R}\right]-D\left[\operatorname { l n } \left(1+\frac{R^{2}}{D^{2}}+\right.\right. \\
&\left.\left.\frac{2 R}{D} \text { ARCTAN } \frac{D}{R}\right]\right\}
\end{aligned}
$$

where $S_{v}$ is the ratio of $S_{o}$ to the volume in $\mathrm{cm}^{3}$, $D$ is the distance in cm , and R and $H T$ are the radius and height in cm. The third and final choice, is that of spherical geometry also subject to the condition that the sphere be non-absorbing. The equation ${ }^{9}$ is:

$$
\operatorname{DOSRAT}=\frac{S_{v}}{4 d}\left[2 R d-\left(d^{2}-R^{2}\right) \quad \ln \left(\frac{R+d}{d-R}\right)\right]
$$

where $S_{V}$ is defined as above, $R$ is the radius in $\mathrm{cm}_{\mathrm{p}}$ and $d=D+R$ is the distance from the center of the sphere in cm.

In addition to the constaninc that the cylinder and sphere be non-absorizing, the following assumptions apply to all dose rate calculations:

1) any potential radioactive daughters result only from $\boldsymbol{P}^{-}$decay,
2) the daughter decays from the grounc state,
3) the daughter radiations appear coincidently wj 'h those of the parents, i.e. have zero half lives,
4) the daughter of the daughter is stable, and
5) dose rates due to parents and daughters with half-lives less than one second can be neglected.

After all parents and radioactive daughters have been located and their contributions determined, the results are printed out starting with the lowest target ID number and progressing upward in ascending order. For each target nuclide, the dose rate is printed out for each reaction that occurs. and then, for the target as a whole. This is, of course, just the sum of the dose rates due to all of the reactions generated by that target nuclide. Finally, after all target nuclides are addressed, the dose rates for the ertire system are printed out and the ACDOSI object program terminates execution.

Code Description

1. ACTLMFE - Description
2. LEVDEC - Description
3. Code Input Description for ACDOSI Data Deck
4. Code Output Description for ACDOS1
5. Program Flowchart
6. Subroutine Level Chart
7. Description of ACDOSI Tape Structure

## ACTLMFE Description



The above pattern is repeated for each reaction. The last reaction in ACTLMFE is for $\mathrm{U}^{240}$, ( $\mathrm{N}, \mathrm{Y}$ ).

## LEVDEC Description

The LEVDEC library contains 62,221 lines. The format for the first record of each set is:
$\begin{array}{ll}\text { Parameter } & \text { Format } \\ 2 A(1000 Z+A) & I 6\end{array}$
Level (MeV) Ell.4
Parity F4.1
Spin F5.l
Half-Life(s) Ell.4
Number of Decay Modes to Follow (NDMODE) I3

2nd et seq. records to NDMODE

Parameter Format
Blank 38X
Mode of Decay I2
2A of the
Daughter
I 7
Level of
Daughter Ell.4
Probability
of Decay to
that Level
El2.4

Two comments are in order concerning the LEVDEC library. First, of the twelve possible modes of decay shown on Table 5, only three are considered in dose rate calculations, $\gamma, \beta^{+}$, and $\mathcal{F}^{-}$

| Decay | Identifier | Mode of Decay |
| :---: | :---: | :---: |
|  | 1 | neutron |
|  | 2 | proton |
|  | 3 | deuteron |
|  | 4 | triton |
|  | 5 | $\mathrm{He}^{3}$ |
|  | 6 | $d$ |
|  | 7 | $\gamma$ |
| 8 | 8 | ${ }^{+}$ |
| 9 | 9 | $\beta^{-}$ |
| 10 |  | EC |
| 18 |  | unresolved $\mathrm{EC}+\mathrm{P}^{+}$ |
| 99 |  | no decay- <br> stable ground <br> state |

Table 5

During the interrogation of the LEVDEC library for a particular parent or radioactive daughter nuclide, subroutine DOSRTE will check the decay identifier. If the identifier is a 7,8 , or 9 , the reaction is considert. $\therefore$. If not, the reaction is skipped and interrogation continues. Secondly, when the identifier is 8 , (which is $\boldsymbol{P}^{+}$decay) two annilihation $\boldsymbol{Y}$ 's of 0.511 Mev each will automatically be included in the dose rate calculations with the appropriate multiplicities.

| $\begin{aligned} & \text { Card } \\ & \text { No. } \\ & \hline \end{aligned}$ | Variable Description | Columns | Format |
| :---: | :---: | :---: | :---: |
| 1 | Any information to be printed out regarding problem | 1-80 | 10(A8) |
| 2 | $A, V, \dot{F}, ~ D F, T 1, T 2, N:$ current (amperes), voltage (KV), fraction of beam that is monatomic, duty factor, length of test (h), length of pause ( $h$ ), number of tests. <br> NOTE: If the user specifies his own neutron source term, $A, V, F$, and DF will not be needed. In this case: | 1-75 | $\begin{gathered} (6(E 10.3, \\ -2 \mathrm{X}), \mathrm{I} 3) \end{gathered}$ |
| 2 | Tl, T2, N: length of test (h), length of pause (h), number of tests. | 1-27 | $\begin{aligned} & (2(E 10.3, \\ & 2 X), I 3) \end{aligned}$ |
| 3 | NOEGPS-number of energy groups | 1-2 | I2 |
| 4 | NOPAS-number of points after shutdown | 1-2 | I 2 |
| 5 | R, D, HT, IGEOM: radius of sphere or cylinder (m), distance from point source or surface of sphere or cylinder (m), height of cylinder (m), geometry designation: l-point source, 2-spherical, 3-cylindrical-onaxis. | 1-37 | $\begin{aligned} & (3 \text { (Elo. } 3, \\ & 2 \mathrm{X}, \mathrm{I} 1) \end{aligned}$ |
| 6 | NONUCL-number of nuclides, 30 maximum | 1-2 | I2 |
| 7 | NF-number of fast neutron groups; use 0.if there are none | I-2 | I2 |


| 8 | (BFLUX(I), $I=1$, NOEGPS: unit normaliaed input fluxes | 1-72 | $\begin{aligned} & \text { (6(Ell. } 4, \\ & \text { 1X)) } \end{aligned}$ |
| :---: | :---: | :---: | :---: |
| 9 | (STAFS (I), I=1, NOPAS;: specific time after shut Jown (h) | 1-72 | $\begin{aligned} & (6(\text { E10. } 3, \\ & 2 \mathrm{X})) \end{aligned}$ |
| 10 | ZNAME (I), IDNO (I), | 1-25 | (A8, 1X, I5, |
| to as | MASS (I): nuclide name, |  | IX, El0.3) |
| many | ID number-1000z+A, mass |  |  |
| as needed | of nuclide (kg). Maximum of 30 target nuclides per run |  | - - |
| from | (GP (I), I=1, NG) : | 1-72 | (6)(E10.3, |
| above | group boundaries in NeV |  | 2X) ${ }^{\text {( }}$ |
| to as | in ascending order |  |  |
| many |  |  |  |
|  |  |  |  |

Code Output Description for ACDOSI

Prior to the execution of activation calculations, all input information is printed out for user verification under the following heading: "THE FOLLOWING DATA HAS BEEN ENTERED INTO MEMORY." This information is:
AMPERES KILO-VOLTS BEAM FRACTION DUTY FACTOR
T1 T2 N

NUMBER OF ENEIRGY GROUPS-NOEGPS
NUMBER OF TARGET NUCLIDES-NONUCL
NUMBER OF POINTS AFTER SHUTDOWN-NOPAS
RADIUS $R$ OF SPHERE OR CYIINDER (M)
HEIGHT HTOF CYLINDER (M)
DISTANCE D FROM POINT SOURCE OR SURFACE OF SPHERE OR CYLINDER (M)

NUMBER OF GROUP BOUNDARIES-NG
NUMBER OF FAST NEUTRON GROUPS-NF
SPECIFIC TIMES AFTER SHUTDOWN (H)
UNIT NORMALIZED FLUXES
GROUP BOUNDARIES (MEV)
NAME ID-NUMBER MASS(KG)

Quantities calculated by ACDOSI are printed out under the heading: "THE FOLLOWING HAS BEEN CALCULATED BY ACDOSI." The quantities are:

INSTANTANEOUS NUMBER OF NEUTRONS PRODUCED PER SECOND

AVERAGE NUMBER OF NEUTRONS PRODUCED PER SECOND

INTEGRALS OF THE FITTING FUNCTION OVER THE GROUP INTERVALS

Calculated activities are printed out under the general heading: "the followine output data ake calculated actiVITIES IN BQ." Three sub-headings are used to display the activation results. The first is: "TARGET PRODUCT ACT (T1) ACT(T2)...ACT(T12)." The second is: "ThARGET ACT (T1) ACT(T2)...ACT(T12)." The third is: "SYSTEM ACT (T1) ACT (T2)...ACT (T12)." Above each of these three subheadings, the times after shutdown in hours, are printed out to expedite review of the activation results. Calculated dose rates are printed out in exactly the same format as activation results. They appear under the general heading: "the following output data are calculated dose rates IN MRAD/H." The corresponding three sub-headings are:

1) "TARGET PRODUCT DSR(T1) DSR(T2)...DSR(T12)"
2) "TARGET DSR(T1) DSR(T2) DSR(T3)...DSR(T12)"
3) "SYSTEM DSR(T1) DSR(T2) DSR(T3)...DSR(T12)"

## General Program Flow Chart*





*Not in detail-only to show the gene :al flow is logic.


## Description of the ACDOSl Tape Structure

Tape ACDOSI has 5 files:

| File \# | Name | Physical Records | $\begin{aligned} & \text { Logical } \\ & \text { Records } \end{aligned}$ |
| :---: | :---: | :---: | :---: |
| 1 | ACDOS1 Source Lisiing (KNSI FORTRAN IV) |  | - |
| 2 | ACTLMFE Library | 449 | 20.177 |
| 3 | LEVDEC 1 | 551 | 24,772 |
| 4 | LEVDEC 2 | 544 | 24,476 |
| 5 | LEVDEC 3 | 289 | 12,973 |

Tape Characteristics
Type: 9 track, 1600 bpi
Blocking Factor: 45
Format: EBSCDIC

## Sample Problem

Determine the dose rates resulting from activation of the copper target in the target section of the injector. Composition-8201bs Cu

Chosen Geometry-Spherical
From tae "Chart of the Nuclides", 9th Edition, 1966

| Element | lbs | Isotope | Wtq | \# of $\mathrm{kg}^{\prime} \mathrm{s}$ |
| :---: | :--- | :--- | :--- | :--- |
| Cu | 820 | Cu 63 | 69.1 | 257.55 |
|  |  | $\mathrm{Cu}^{65}$ | 30.9 | 115.17 |

Volume of a sphere $=\frac{4}{3} \pi R^{3}$. Therefore, the volume of copper is: $\frac{3.727 \times i 0^{5} \mathrm{gm}}{8.96 \mathrm{gm} / \mathrm{cm}^{3}}=4.160 \times 10^{4} \mathrm{~cm}^{3}$

$$
\begin{aligned}
& R=\sqrt[3]{\frac{3 \cdot V O L}{4 \pi}} \\
& R=\sqrt[3]{\frac{3 \cdot 4 \cdot 160 \times 10^{4} \mathrm{~cm}^{3}}{4 \pi}} \\
& R=\sqrt[3]{9931.27 \mathrm{~cm}^{3}} \\
& R=0.215 \mathrm{~m}
\end{aligned}
$$

## The following sample problem parameters apply:

| A | current (amperes) | 65 |
| :--- | :--- | :--- |
| V | voltage (kilovolts) | 170 |
| F | beam fraction | 1.0 |
| DF | duty factor | 0.10 |
| T1 | length of test (h) | 8 |
| T2 | length of pause (h) | 16 |
| N | number of tests | 7 |
| NOEGPS | number of energy groups | 20 |
| NOPAS | number of points after shutdown | 12 |
| R | radius of sphe | 0.215 |
| D | distance from surface of sphere (m) | 2 |
| IGEOM | geometry designator (sphere) | 2 |
| NONUCL | number of nuclides | 2 |
| NF | number of fast neutron groups | 2 |

$$
G P(I), I=1
$$ NG (MeV)

BFLUX (I), I=1, NOEGPS (neutrons/cm².sec/Source Neutron)
$1 \times 10^{-11}$
$4.14 \times 10^{-7}$
$1.125 \times 10^{-6}$
$3.059 \times 10^{-6}$
$1.068 \times 10^{-5}$
$2.902 \times 10^{-5}$
$1.013 \times 10^{-4}$
$5.829 \times 10^{-4}$
$1.234 \times 10^{-3}$
$3.355 \times 10^{-3}$
$1.033 \times 10^{-2}$
$2.188 \times 10^{-2}$
$2.479 \times 10^{-2}$
$5.248 \times 10^{-2}$
$1.111 \times 10^{-1}$
$1.576 \times 10^{-1}$
$5.502 \times 10^{-1}$
1.108
1.827
2.307
2.385
$8.288 \times 10^{-7}$
$3.076 \times 10^{-8}$
$3.456 \times 10^{-8}$
$3.951 \times 10^{-8}$
$3.011 \times 10^{-8}$
$4.684 \times 10^{-8}$
$7.234 \times 10^{-8}$
$3.043 \times 10^{-8}$
$4.021 \times 10^{-8}$
$4.570 \times 10^{-8}$
$3.522 \times 10^{-8}$
$7.420 \times 10^{-9}$
$3.860 \times 10^{-8}$
$4.790 \times 10^{-8}$
$2.842 \times 10^{-8}$
$1.373 \times 10^{-7}$
$1.058 \times 10^{-7}$
$8.333 \times 10^{-8}$
$1.382 \times 10^{-7}$
$7.901 \times 10^{-6}$

Results of Sample Problem

Inspection of the printout shows a dose rate of $2.67 \mathrm{mrad} / \mathrm{h}$ at time $\mathrm{t}=0$ after shutdown. The distance from the surface of the sphere is 2 m . Of this system dose rate, the majority of the contribution is coming from Copper-64 which was produced by the ( $n, \mathcal{Y}$ ) reaction on Copper-63. Due to the 12.9 hour half-life of Copper64, one would need to exercise some caution for a period of several hours after shutdown if maintenance were to be performed near the beam dumps. \{At 10 cm from the spherically modeled source, the corresponding sose rate is about 129 mrad/h due to Copper-64 alone.) It is interesting to note that the total activities of Copper-64 and copper-66 only differ by a Factor 3 , while the corresponding dose rates diffe, by a Factor of 7. This difference, of course, is due to the energy and decay probability considerations specific to the two different nuclides.

Although an activity was calculated for Nickel-63, there was no contribution to the dose rate. The longlived state of Nickel-63 decays to Copper-63 by $\mathbf{\beta}^{-}$emission. Since Copper-63 is formed in the ground state, no dose rate results.

## Project Conclusions

The objective of this project was to determine the dose rates associated with neutron activation of neutral beam injector components. This objective has been accomplished by creating a Fortran IV program to calculate dose rates as a function of geometry, component composition and amount, and time after shutdown.

The work was successfully completed by performing five major tasks:

1) determination of a suitable mathematical model for the production of neutrons by the injector,
2) modification of the differential equation that describes the system undergoing activation (to reflect the pulse-pause nature of neutron production) and adapting the results to multigroup calculations,
3) development of a program to average microscopic cross-section data over the appropriate energy range (to calculate flux weighted group cross sections),
4) creation of a program to calculate activation given the injector testing history, pertinent beam parameters, component composition and amount, and desired times after shutdown, and finally,
5) development of a program to calculate dose rates given the previously calculated activities, user
specified geometry, and distance from the activated component.

The five tasks manifest themselves through 14 subroutine subprograms and one executive calling program. Together, they constitute the program, ACDOS1, which performs the desired dose rate calculations given the necessary information.

Although ACDOSl is a complete program for calculating dose rates due to neutron activation, there are some areas where additional work could improve the code from the standpoint of completeness and accuracy. Coupling of a neutron transport code to the "front end" of the program would elimincte the need for the user to supply the input fluxes. With ACDOSl, the user must obtain the input fluxes from a separate calculation. This addition would make the program more self-contained computationally and could also reduce the input data required to run the program. Also, other geometry options could be added to increase the flexibility in modeling unusual source distributions that might arise in non-injector applications. Addition of these options could be made by incorporating the appropriate equation into subroutine CALC which is the subprogram that calculates the dose rates. (Any additional arguments would have to be passed in the calling statement.)

To produce more accurate dose rate calculations, three immediate improvements could be made to the program:

1) a more realistic treatment of radioactive daughters,
2) inclusion of self absorption in the geometrically modeled sources, and 3) a within-cavity geometry option. Currently, radioactive daughters are assumed to have a half life short compared to that of the parent. Although computa亡ionally expedient, this is not always realistic. The improvement would involve incorporating a more complex activation equation into the program; and in addition, some major changes in software logic. Inclusion of self absorption in the dose rate calculations wruld remove a major conservatism inherent in the present treatment of the calculations. Although modification of the dose rate equation would be a relatively simple matter, there would have to be some provision made for generating or acquiring $\gamma$ ray attenuation coefficients since they vary strongly with composition and energy. An additional library containing such data would probably be the best solution. Another routine would also be necessary for interpolating $\gamma$ attenuation coefficients between data points. Finally, a within-cavity geometry option could be utilized to include dose rates from activation of the concrete walls and/or floor surrounding the injector. Presently, there is no way to include any contribution to the dose rate from the walls of the injector housing.

At this time a second version of ACDOSl, ACDOS2, is being written and will incorporate many of the above men-
tioned improvements. These modifications will remove many of the approximations used in ACDOSl, and should yield more accurate dose rate calculations.

A final comment involves a paper ${ }^{10}$ that was written at ORNL concerning the calculation of dose rates in the Tokomak Fusion Test Reactor Test Cell. Basically, it is a comparison of resultant dose rates in the test cell with and without a neutral beam injector present--the objective being to determine the contribution to the test cell dose rates due to the presence of the injector and its penetration. Although no code details are presented in the article, a comparison of the similarities, differences, and general methodologies used would offer a perspective from which to evaluate or "bench mark" the ACDOSl code work since both projects have similar objectives but different origins.

A tabular comparison of the two projects is shown in Table 6.

$$
\text { Table } 6
$$

ORNL Project
Photon transport calculations are carried out using Monte Carlo methods.

ACDOSI Project
No photon transport calculations are done. Photon flux at a dose point is determined from standard geometric equations. No self attenuation or buildup is taken into account in geometrically modeled sources.

ORNL Project
Concrete walls and ceiling of the test cell were considered.

Uses cylindrical geometry to calculate spatially averaged dose rates.

Considers contributions from parents, daughters, and grand-daughters.

Allows for identification of those nuclides that are contributing to significant dose rates.

Arbitrary pulse sequences are allowed.

Calculates dose rates for a variety of times after shutdown.

Neglects burnout of activated nuclei.

Activation photons produced inside the igloo which stream into the injector through the penetration were considered.

Appears to be specific to neutral beam activation and dose rate calculations.

## ACDOSI Project

Doesn't consider effects due to activation of the concrete. Considers only activation of injector components inside the injector shield.

Choice of three different geometries. Dose rates are point values rather than spatial averages.

Considers contributions from parents and daughters. Grand-daughters are assumed to be stable.

Allows for identificatior of those nuclides that are contributing to significant dose rates.

Arbitrary pulse sequences are allowed.

Calculates dose rates for a variety of times after shutdown.

Neglects burnout of activated nuclei.

No provisions for including this contribution to dose rates.

Can be used for non-injector activation and dose rate calculations.

Review of Table 6 illuminates the similarities and differences. Some of these differences will most likely be eliminated with the second version of ACDOS1, ACDOS2.

Inclusion of self attenuation and buildup in dose rate Calculations along with dose contributions from concrete injector walls will probably be addressed. When these modifications are mede to ACDOSl, calculated dose rates will probably be more in line with those resulting from a transport treatment of the $\gamma$ photons. The ACDOSl code was never intended to consider activation photons produced in an igloo anc which stream into the injector through the penetration. This results from the nature of the design work being done at LBL and the forthcoming upgrade program which will require prolonged testing periods at higher beam currents and duty factors.

In general terms, the URNL calculation is probably more accurate due to the photon transport approach to dose rate calculation:. ACDOSI, however, appears to be more versatile. With upcoming improvements to ACDOSI as mentioned above, improved accuracy coupled with the code's versatility should make it a useful tool for injector design studies and non-injector applications. Acknowledgements

I wish to acknowledge the assistance given to me by the consulting staff of the Lawrence Berkeley Laboratory Computer Center during the development of several programs required to solve the problems associated with this project. Their help and guidance was greatly appreciated.

Also, I would like to thank Mr. Robert J. Howerton of LLNL who expended considerable time and effort in providing both data libraries, preliminary flux-weighted group cross sections, and advice and consultation throughout this endeavor. This work was supported in part by the Office of Fusion Energy, U. S. Department of Energy, under Contract No. $W-7405-E N G-48$.

## Appendix A

Solutions to Solving Transcendental Equations

For problems involving two or more neutron groups, transcendental equations will result due to the assumed form of the fast group exponential used for representing the weighting flux function at higher energies. The exponential is:

$$
\frac{E-E l}{B}
$$

Ae
where $A$ and $B$ are constants to be determined by applying the appropriate boundary and integral constraints on the pertinent equations, and El is the highesi energy in the particular fast group being evaluated--the upper boundary for that group. There are three cases where transcendental equations will result:

1) Matching the thermal flux to a fast flux
2) Matching a $1 / E$ flux to a fast flux, and
3) Matching a fast flux to a fast flux The match is always required at the boundary between the two groups.

Consider Case 1. It is required to match the Maxwellian to the fast group at the koundary between the two groups as shown in Figure 11. $\phi$ (E) is the fast weighting flux function and $G P\langle J\rangle, J=1,2,3$ are the group boundaries. In addition, it is required that the integral of $\phi$ (E) over group 2 be equal to the total flux as


Figure 11
discussed on page . Since the constant in the Maxwellian ( $C \sqrt{E} \exp (-(E / \cup .025)$ ) has already been evaluated, the flux at the boundary (FLUXBD) i.s known. It is:

FLUXBD $=\operatorname{CONST}(1) * \operatorname{SQRT}(\operatorname{GP}(2)) * \operatorname{EXP}(-(\operatorname{GP}(2) / 0.02 .5))$
where CONST(1) is the constant associated with the maxwellian and GP is in eV. The first equation is:

$$
\begin{equation*}
\text { FLUXBD }=A e^{\frac{G P(2)-G P(3)}{B}} \tag{1}
\end{equation*}
$$

in accordance that the flux match at the group boundary GP(2). The second equation is:

$$
\int_{G P(2)}^{G P(3)} A e^{\frac{E-G P(3)}{B}}
$$

dE=total flux for that group=BFLUX(NG)
where $N G=2$ is the total number of groups and BriUX is an array holding the total flux values. Evaluating this integral:

$$
\begin{align*}
& \mathrm{ABe} \frac{(\mathrm{E}-\mathrm{GP}(3))}{\mathrm{B}} \left\lvert\, \begin{array}{l}
\mathrm{GP}(3) \\
\mathrm{Cr}(2)
\end{array}=\mathrm{BFLUX}(\mathrm{NG})\right. \\
& A B-i=\mathrm{Be} \frac{\mathrm{GP}(2)-\mathrm{GP}(3)}{\mathrm{B}}=\mathrm{BFLUX}(\mathrm{NG}) \\
& A B\left(1-e^{\frac{G P(2)-G P(3)}{B}}\right)=\operatorname{BFLUX}(\mathrm{NG}) \tag{2}
\end{align*}
$$

From equation (1), $A=$ FLUXBD e $e^{-\left(\frac{G P(2)-G P(3)}{B}\right)}$

Equating this to equation (2), we get

$$
\text { FLUXBD } e^{-\left(\frac{G P(2)-G P(3)}{B}\right)} \cdot B\left(1-e^{\left(\frac{G P(2)-G P(3)}{B}\right)}\right)=B F L U X(N G)
$$

As a result,

$$
\begin{equation*}
B=\frac{\operatorname{BFLUX}(N G)}{F L \mathcal{X B D}}\left\{e^{\frac{\operatorname{GP}(3)-\operatorname{GP}(2)}{B}}-1\right\}^{-1} \tag{3}
\end{equation*}
$$

This is the above mentioned transcendental equation. Once $\dot{B}$ is found, $A$ can be determined by substituting B into equation (2). Subroutine WTFLUX, which evaluates the constants associated with the assumed wtighting functions, used the Newton-Raphson Method to silve equation (3). The numerator in equation (3), BFLUX(NG)/ FLUXBD, is always a known quantity and is of the same order of magnitude as the numerator in the exponential, GP(3)-GP(2). Therefore, BFLUX(NG)/FLUKBD is used as an initial gues; to start the iterative routine which solves the equation.

The second case involves matching a l/E flux to a
fast flux as illustrated in Figure 12. The same type of method is used as before. There are two requirements: (1) continuity of flux at the boundary, GP (NG), and (2) an integral constraint on the fast group. From the first requirement,

$$
\begin{equation*}
\text { FLUXBD }=\frac{\operatorname{CONST}(I C O U N T)}{G P(N G)}=A e^{\frac{G P(N G)-G P(N G+1)}{B}} \tag{4}
\end{equation*}
$$

where CONST(ICOUNT) is the known constant associated with the previous $1 / \mathrm{E}$ group. ICOUNT is a neutron group counter. From equation (4), A can be solved in terms of B:

$$
\begin{equation*}
A=\frac{\operatorname{CONST}(I \operatorname{COUNT})}{G P(N G)} e^{-\left(\frac{G P(N G)-G P(N G+2)}{B}\right)} \tag{5}
\end{equation*}
$$

From the second requirement:



Figure 12

$$
\begin{align*}
& A B\left(1-e^{\left.\frac{G P(N G)-G P(N G+1)}{B}\right)}=B F L U X(N G)\right. \\
& A=\frac{B F L U X(N G)}{B}\left\{1-e \frac{G P(N G)-G P(N G+1)}{B}\right\}^{-1} \tag{6}
\end{align*}
$$

equating equations (5) and (6)

$$
\begin{equation*}
B=\frac{G P(N G) B F L U X(N G)}{\operatorname{CONST}(I C O U N T)}\left\{e^{\frac{G P(N G+1)-G P(N G)}{B}}-1\right\}^{-1} \tag{7}
\end{equation*}
$$

As before, the numerator in this equation is always known and is of the same order of magnitude as the exponential. Therefore, GP (NG) *BFLUX(NG)/CONST (ICOUNT) is used as an initial guess to start the Newton-Raphson iterative routine.

The thira case involves matching a fast flux to a fast flux as shown in rigure 13. Again, the same method applies. The requirements yield two equations. From the continuity of flux requirement at $G P(I C O U N T+1)$ :

$$
\text { FLUXBD }=\operatorname{CONST}(I C C U N T)=A e^{\frac{G P(I C P 1)-G P(I C P 2)}{B}}
$$



Figure 13
where for brevity, $I C P 1=I C O U N T+1$ and $I C P 2=I C O U N T+2$. Note that the exponential associated with CONST (ICOUNT) equals 1 when evaluated at GP(ICPI). Thereiore,

$$
-\left(\frac{\mathrm{GP}(\mathrm{ICP} 1)-\mathrm{GP}(\mathrm{ICP} 2)}{\mathrm{B}}\right)
$$

$$
\begin{equation*}
A=\text { CONST (ICOUNT) e } \tag{8}
\end{equation*}
$$

The integral constraint requires that



$$
\begin{align*}
& \left.A B\left(1-e \begin{array}{c}
\frac{G P(I C P 1)-G P(I C P 2)}{B} \\
A=\frac{B F L U X(I C P I)}{B}
\end{array}\right\} 1-e^{\frac{G P(I C P 1)-G P(I C P 2)}{B}}\right\}^{-1}
\end{align*}
$$

Equating equation (8) and (9),
$B=\frac{B F L U X(I C P 1)}{\operatorname{CONST}(I C O U N T)}\left\{e^{\frac{G P(I C P 2)-G P(I C P 1)}{B}}-1\right\}$
As before, the numerator (BFLUX(ICPI)/CONST(ICOUNT) is used as an initial guess. A is then found from equation (8).

In all three cases, the methodology employed to solve the equations is the same. Since the numerators in the pertinent equations are all known quantities when they are required, the use= does not have to enter an initial guess. Subroutine WTFLUX performs all of the calculations required to determine the constants associated with the assumed weighting functions. Furthermore, all of the calculated constants are storsd in memory for future use in subroutine AVRAGE which calculates the flix weighted group cross sections. The A's are stured in array CONST(I) and the B's in array FCONST(I).

One word of caution is appropriate. It is entirely possible that converging problems may occur during the execution of the routine that solves the transcendental equations.

## The Newton-Raphson Method

This method is used to solve the transcendental equations that result when two or more neutron groups are uned. The interative expression is:

$$
x_{i+1}=x_{i}-f\left(x_{i}\right) / f^{\prime}\left(x_{i}\right)
$$

where $x_{i}$ is the initial guess.
The parameter B (see Appondix A) is always given by the following form:

$$
B=K N O W N \text { QUANTITY }\left\{e^{\frac{G P(N+1)-G P(N)}{B}}-1\right\}
$$

where $G P(N+I)$ is a group boundary higher in energy than GP (N). Therefore,

$$
f(\mathrm{~B})=\text { KNOWN QUANHITYY }\left\{\mathrm{e}^{\frac{\mathrm{GP}(\mathrm{~N}+1)-\mathrm{GP}(\mathrm{~N})}{\mathrm{B}}}-1\right\}_{-\mathrm{B}}^{-1}
$$

and,

$$
I^{\prime}\left(B ;=\frac{K N O W N \text { QUANTITY }(G P(N+1)-G P(N)) e^{\frac{G P(N+I)-G P(N)}{B}}}{B^{2}\left[\frac{G P(N+1)-G P(N)}{B}-1\right.}\right.
$$

These two expressions are incorporated into subroutine WTFLUX for evaluating the constants associated with the exponential weighting functions. When the difference between $B_{i+1}$ and $B$ is less than 0.1 , the iteration terminates.

Appendix C






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    O mif1TEI6,12)
```



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        STCP
    IS kN[7E{6,I4]
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        HFCHM MLST EE INTEGEF ANE LESS THAN OR EQUAL TO LZORECHECK THES DAT
        *)
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        -mUST AE INTEGER AND LESS THAN OR EQUAL TO. 30:
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    30. MeITE(t,孛2)
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    37 HE{TE{6,53)
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    _ THAA O.D.AND_LESS IHAA CR EDUML TO_1,O-RECHECK THIS DATA)
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44 HRITE{4,5, )
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        STOP
    49 WRI TE 16 %001
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        ETCP
    77 WRITE(6,7E)
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        #GCAL TC O.G-RECGECK THIS DATAJ
        sTcP
    67 HFITE16.68:
```



```
        *CK THIS OATAI
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        valTE (6,20)
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                                    DUITEA
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\Sigma2 FOSMATILH, 35HMLMEER, QF ENERGY_GNOUPS-ACEGPS
                    3____[12,13\ldots
    wFITE{(0.75) NGRUCL
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    DAITE{E,231 NOPAS
```



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    WilEit,25: R
```



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    MFITE(6,79) mT
```



```
    WRITE(6,26)D
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```
    + CTLIMOER {M| +E{0.3.13
    NG=NLEGPS +1
    GRITE16,72 NG
```



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76.FGFMAT:1H ,39FRUPEEP CF FAST NEUTRCN GODUPS=NF_._12:1)
```




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    *C.4.1
        FETURN
    ENC
```



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        -A5,MG)
    C
CTHIS FRCGFAY IS USEL FGN ENTER ING 'MRRAY' DATA INTO MEMORY
G
C
C-__ AEGUMERTS_IN-STAES_NEEGPS_NNONUCL_NOPAS&NG
    ARGUEMAT5 CUT-EFLUX, 2AAME,IDNC,MASS,GP
```



```
    DIMENSICN BFLUHINCEGPSI, ZAANEINCNUCLIEICNCINCHUCL.D,MASSINONUCL D.ST
    +AFSINOPAS),GP(NGi
        RESL pASS
        READ(5,4)(BFLUA({1,I=1,NOEGPS)
        FEAD(S,4)(STAFSII), IEELENOPASI
    FCFMAT(t(ELO.3.2x)
    LC ? IEL,MONLCL
    FEA[{5,8) INAmEIII,ICNC(I),MASS(I]
```



```
    7 CCNT INUE
    KE=0;5,41_GP(II)IE1_AG)
    MR [TE (6,24)
    24. FEFMATU,IH, 33FSPECIFIC_TIMES AFTER SHUTOOWN_(H),/I
    OC 34 I=1,NOPAS
    *R[TE{&,3E: L&FJAFSU1)
    35 FCFFAT (1H P4HT IME,IX,12,10K,E10.3)
    34 CCNTINLE
    GRITE{6,27)
```



```
    DC 32t=1,NGEGFS
    I|=ACECFS:1-1
    &RITEIE.2日) I, EFI'U\(II)
```



```
    32 CCNTINNE
    *RITE(E,73)
    73 FGRMATI//f,1H 22MGROUP DOUMOARIES (MEVI,/\prime
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```
    14 FCRMATIIH,BHBCLNOARY,IX,15,3X,EIL,41
    WRITE(6,29)
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```
!..-- [L 30,I=1,NCNLCL
    mPITE!:, Z1) 2N&ME(1), IDND(I1,MASS(1)
-3) FC&MAT(1X,A8,5x, 15 & X,EB,3)
    30.cCNTINUE.
        KETLRN
    ENE
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subrcutime scufcelavenpst


* SLERCUTINE EFFLLXIAYEAPS,NCEGPS,GPFLUX, BFLUXI** SGeRCGTIAE GRFIUX(PVEAPS, NGEGFS, GPFLUX, EFLUX)





```
    **SUEGUT INE WTFLUXCOMFLUX,GP,CONST, FCONST,NDEGPS,NF,NGI#*
        xS̈Giplaxsi&\-F\StBI/FPRIME
        C|FF|ABS!XSALPI=GUESS1)
        IF (DIFF.LT.O.1) GC TC 5
        ITER=1TER +1
        IFIITEF.GT.1001 GO TC 25
        GLESSI ExSBIP2
        GE TE 6
    75.EITESta.7e!
    7% FCFMBT(IH ,1\4&*###FRCGRAM ABCATED****THE TRANSCENDENTAL EOUATION
        +LSED TG MATGH THE THERMAL GROUP TO FAST GRCUP IS NGT CONUERGINGI
        GFITE\O,27)
        STC
        5 E=xSEIFL
        A=8FhUXIACEGPS // (R#1人-ExP(1GR12)=GP(3)1/B)!)
        CENSTI[CDLNT]=A
        FCONSTIICOUNT)=&
        GC TC 1000
    150. IFINF &T.II GS TO 1&0
CTTHE I GRCUP FAST SPEGTRU* WILL BE MATGHED TD THE LAST L/E GROUP
    ITER=1
    NGPIENCEGPS+I
    ZNUMER= GUESS IEIEELUM (NOEGPS )*GP{NOEGPS)/CONSTLIC OUNTI:
    SL FXSUEI= ZAGMEFIGEXPIIGPINGPII-GPINOEGNSIMGUESSII-I I-GUESSI
    _- - xSUBI =GUESSI
```




```
        KSEIP 1= XSUQ I-FI SUB I/FPR IME
        CIFF=ABSIXSEIPI=GUESS II
        IFIDIFF.1T.O.1) GC TC 92
        ITER=ITER +1
        IF\ITER.GT.100\ GL TC 25
        GLESSIEx 쥬:IP1
        GC TC 91
        42. Ex xSBIP1
```



```
    CCMSIINCEGPS IEA
    F[CAST\NCEGP5]=e
    60 TO 100C
    160 ICPI=ICCINTT+1
CTHE FIRST QF 2 GR MGRE.FASI GRCUPS WILL_EE MATCHEO_TO THE LAST_I/E
GEFCUP
    ICR2=ICCURT +2
    ITER=1
    TNUMER=GUES5IE{BFLUXITCPIIWGPGIGPII/CONSTLICOUNTI)
    2E FXSLBI=2KUNER/(EXFI{GF\ICP2I-GP(ICPIII/GUESSII-1]=GUESSI
        \SLSI=6LE SSI
```




```
        H__+...----
```



```
            C1FF=A睢(XS日[P1=GUESSI)
            IFIDIFF.ET.0.1: GE TL 30
            ITER=1TER+1
            TFIITER,GT., OOS 60 TN 25
            GLESSI=xSE!P!
            GO }702
        25 hEIIE(6.26:
        Z6 FCKMATTILH ,IIGH**** FFEGMAM AGCHTED##**THE TRANSCENOENTML EQUAT IGNS
        *NHICH MATCH IIE TO FAST QR FAST TO FASTGGOUPS ARE NOT GCNVERGING)
```



```
        |RIFEGG27%
    27. ERFPATILE, 75HRCST UKKLY PROBLEM IS UNRESLISIIC CR UNUSAL_ELUX
        *INO GRCUP GCUMOARY DATA)
        stop
    3C E=x501P1
```



```
        ICCUNT= IC CNAT +1
        15RL=1CCLATEL
        dCPz=IC CURT* 2
        CCNSTIICCUNT I=&
        FCCASTIICCUNTI=B
        IF(ICOUNT.EQ.NOEGPS) GO TO LOCO
        ZAUMER=QFLUXIICPLI/CCNST(ICOUNT)
        ITER=1
        C0 }102
    .1000 HE1TE (6,11)
    II fCPMGTI/f,IH, C3H INTEGRMLS OF THE FITTING FUNCTIDN OV ER THE
        *RCUP IN IERVALSJ
        MFITEIS,12:
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```
        +AL CALCULATED INTEGRAL,/I
'.CCALCLLATE THE THERMAL IATEGEAL
        ICCLNT=1
        SumIAT=O-C
        HICTH=1GP(2)-GF\IT1/200.0
        zINCA=WIOTH/2.C
        CPYARGFP11+Z1ACR
        OC 21 I=1,2ct
```



```
        SUR LNT = SUM INT +FARSUM
        DFVARODPVIR+GIIIH
        21 CONTINUE
        JFIIECLNT.EG.ACEGFSI GE TC 109
        IFIICELNT.EG22!GC IC 48
            JJ=NG-ICOUNT
        HRITE(G.22) JJ.EPIICSUNT),GPIICCUNT +1): EPUXX ICOUNTH. SUMINT
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```
    ICCUNT: IT CUNT&I
CEALCLLATE THE 1/E IRTEGFALISI
    45 SumINT=0.0
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```
        ZLMCR=nIDTH/2.C
        DPYAR=GPIICGLNT)+2 INER
            C5 23 {=1,200
            PARSUNF(CENSTIICGLRTI/DPYARJFWIOTH
            SUM IN T= SJMINT+PAR SL.A
```



```
    Z3 CCNTINUE
        Jj=NG-jCOUZNT
        HRITE 15,221 JJ,GPIICCUNT 1,GPIICOUNT+11, AFLUXIICDONT1. SUMINT
        ICCUNT=IC CUNT+1
        IF(ICOUNT LENMEPTH+1) GC TD_45
        GC TC 16
        46 JJ=2
```



```
        IF(NGG.ES.2) ICCUNT=ICCUNT +1
    16 SUM[N7=C.C
CCALCULATE TRE FAST INTEEFMIS!]
```



```
    HICTH=\GP{ICCURT*1\-GP\ICDUMTII/200.0
    ZNFREMICTH/2,C
    DPVAR=GP{ICOLNT}+2INGR
    CC 24 [=1,200
```



```
    +r
    SLFIATFSLNINT +FIRSUP
    OPYAR ROPYAR+HIOIH
24 CCNTINUE
    - JJERG-ICCLHT
```



```
    ICCUAT=ICEUNT +1
    IFIICELAT.LE.ACEGFS) GETE 16
    GF TC 15
109 JJ#1
```



```
    15 00 105 JC = 10NG
    GR{JC,=GF\JC1/1_0000E6
IGS CCNTINGE
    RETURN
    EAD
```



```
        SLERCLTINE ACTIATITIMACT,SYSACT,SLLMACT,GPFLUX* PRCDNU, PRODHL GPXSEC
```



```
C
CTHIS PRCERAM CMLCLLATES ACTIVITIES DUE TG NEUTRCN INDUCED REACTICNS
C
C-GFGUMENTS_LN-GPELUX,GPXSEG_LDNO_HASS_STAFS_GP_CONSILECONSTLBELUXL
    ARGUFEKTS IT-NCKUTLIREEGPS,NCPIS,NG,NF
    AFGUMENTS CUT-TIMACT,SYSACT,PRCDNU,PRCDHL
    DIMENSICN FIMACT(NONLEL,13,NOPAS), SYSACT(ALPAS),SUMACT (NOPAS 1,GPFL
```



```
    + S) , CCNST (NCEGFS), FCCAST (NCEGPS) , BFLUX (NCEGPS I, IONO (NONUCL I, MASSINO
    +NUCLD,STAESINOPASI
        CLPPCA A,V,F,LF;T1,TZ,N,R,D,HT,OPTION,IGEDM
    REAL MASS
    mRITE(E,ICZ)
```



```
    TIESIN BO)
    I=1
    J=0
CINITIALLZE.FLAGS_AND_AREAYS
    CD 40 F=1, NCMUCL
    OC, 40 MM=1.l3
    PRCDNG{M,MM\:0.0
    FRCCHL(M,NM)=0.0
    4O CEATIALE
    OC 41 Hal,NCBLCL
    CC 41 PN=1.13
    OC 4l PMM=1,MCFAS
    T1MACTIM, MY, MMMI I=0.0
    41_CCATINE
    OC 22 JN=1,NGPAS
    SYSACTIJNIUSL.MACTIJNI=0.O
    22 CEATIAGE
    KFLAGELFLAGEC
CPFINT HEACINGS
    WRITETG ,37)(STAFS(KI,Km1,NCPAS)
```



```
    kRITE|EO\!
```




```
    *&T\T121%
    2% FEACIT,2SI ITAFETARM&S
    2S FCRMAT(IG,7X,EIL - %)
    IFCITAPE:EQ.IONO(1)1_GO TO 7
    1FIITAPE.GT.ICAC\III GO TO 44
    CALL PCSITCA
    GC TO 24
CTHE AECVE STATEMERTS SEARCF THE TAPE FRR_A HATCHENG TARGET ID NUMBER
    4+ WFITE(E,45) IDAC(I)
    45 FDRMAT (///%/, 1t, 21HATTENTICN-TARGET NLCLIDE NUMEER,IG,8GHDCES NCT
    GEXIST TN CITA LIBRARY-THEREFORE-NO CDNTRIDUTION FROM THIS NUCLIDE
```

```
            *SUERCUT INE ACTVATGTIMACT,SYSACT, SUMACT,GPFLUX,PRMNUPPRCDHL,GPXSEC*L
```




```
    I=\+1
    CALL PCSITCN
CPRINT HEADIRGS
    WFITE{E,37){STAFSGK),K=2,NOPAS)
    GAITEIt,II
    CT.TU_24
46 I-141
CPRINT HEADINGS
        #R[TE(6,3THSTAFSIX],N=1PNGPASI
        &FlTE(6,l)
        7 J=J+1
        REAC[7.8) PRORNULIENH, PROOH2 (1.0J)
        PEACI7,GG1 AP
    5.9 FCRHATT1131
        IF(ITAPE&EN. IOAC\NONUCL.)I LFL AG=1
        IFIPRCDHLIL,JI.OT.O.1000E+501 GG TQ 50
    A FORMAT (20X,E1z.4,12X,E12.41
CLAJL AVFAGE YC CALCULAYE GROUP CROSS SECIIONS
    [ALL A YRAGEIGP)SEC;CEAST,FCCNST,BFLUX,GP,NCEGPS,NG,NF,NP}
    IF(GP)XSEC{1),GT_C.ECCE+49} .OS_TO,103
CIMITIARI2E SUMIA INE SUMZA
    SLPlA=SUM2A=C.C
CEEGIN ACTIVATION CALEULATIONS
    OCNH_NLEN
```



```
        +1)
            SLPWA=SUMIA4TEFFS
    11.CONTINLE
        CC 12 L=I;NOEGPS
```



```
        SUM2A=5UV2A+TENF5
    12. CCRT_INUE
CCALCGLATE ACTIGITY TT TOO
```



```
    4|/F市[DHL(T,J)|l
CCALGHATE ACTIYITY AS A FLACTION OF TIME
    CC 13 K=1,NOPAS
```



```
CSUN ACTIVITIES
    SUMACTIK I=SUMACT(K)&TIMACTAIENGKI
    13 CCNTINLE
CPRINTACTIVITIES
    103 WRITE(6,201 IDND(1), IFIXIPRODNUII,N)I,ITIMACT{I,J,KJ,K=1,NEPASI
```



```
C\EGIN IATERFCGATICN CF THE ACTIMFE LIEPIRY
    KEAD(7.2E1 1 FAPE.TARMAS
        IF!ECF!T! NE.O! KFLAGM!
        IFIKFLAG.EG.11 EC TC 27
        IFICLAG.EO. -ANE -ITAFE,NE.IDNDTNDNGCLI GOTTC 27
        IFIITAPE.EG.ICNG(NDNUCLIILELAG=I
        IF(ITAPE,NE,TORCIIJ)GCTTE 2T
        CD }70.
CPR\AT 子EACIRGS
    <7 WRITE{E,37IISTAFSIKI,&N1,NCPAS\
```

```
            **SUGRCUTINE ACTVAT IT IHAET, SYSACT, SUMAET,GPFLUX, PAODNU_PAODNL,EPXSEC**
            mRITE ( G,'#̈4
```



```
            4.ACT(TS) ACT(TGI ACTITTI ACT (TBS ACTITSI ACTITLO\ ACTITLI)
            *ACTIT12!)
CPGINT SUN OF ACTIVITIES DUE ṪO"A PARTICULAR TARGETT NLCLIDDE
            WRITE{6,15) IDNC{II, TSUMACT (K),K=I,NCPAS)
    15 FCRMAT{1X,5x,IS,5x,1CIEB, 3,1 X], 2x,2(EB.3,2xI)
            DC 1C K=1,NCPAS
            SYSACT (KIESYSACTIK j+SÜMACT (K)
    1: CCATIRUE
            IF(1TIPE.ES.1DNO(1+1)) GO TO 32
            IFGFLAG-EG,IIGO TC 16
    #2 DO 23 JN=1,NCPAS
            SUMACT(JN)=2,O
    23 CLATIALE
            IFIHTAPE.EG_IDACII+11)_GG_TO._31
            IF(ITAPE.ET.JDNO(I+1))GO TD 47
            I=1+1
            CALL PCSI TCN
CPRINT FEACIRGS.
```



```
    WRITE:(6,1)\
        J=0
        GC TC 24
    47 4R\TE{0,45) IDNC(1+1)
        CALL ECSITEN
        I=1+2
        IF{ITAPE.ET.IDNO(13)_WRITEIG.451 IDNCIIN
    IFIJTAPE=CT. ICNEIII! I= I+1
CPAINT HEADINGS
    HR\TEIG,37)ISTAFSTK\,K=6,NOPASI
    RRITE{6,1)
    J=6
    GOTC 24
EPRIMT FEADINGS
    Z1 bRITE(E,37)(STAFS(K),K*1,NCPAS)
        WRITE[6,1)
        l=1}+
        J=6
        GC TC?
CPRINT HEADINGS
    16 HRITE(E,371(STAFSIK),K=1,NOPAS)
        HFITE(6,21)
    Z1FERMATILH_125H SYSTEM ACTIT1I ACTIT2I ACTIT3I ACTITG)
        * ACT(TSG ACTITGJ ACTIT7) ACT(T8) ACTIT9) ACTITIO) ACTITNI\
        +ACT:T12!)
C̈PRINT SGETEM ACTTIGITY
        WГ ITE (B,51)!SYSACT(X),K=1, NOPASI
    51 FCFMMATIIX,15X,10&EB,3, 2XJ, 2X,2{E8.3,2XI)
        IFIMFLAGOEG,G) GE TC 42
        RETURN
    42.CALL PCSITCN
    43 FEMD(7,251 17AFE,TARMAS
    IF(EGFI7I.NE.CI KETLRN
```

[^0]

```
    *#SURROUT INE AYRAGEGGPXSEC,CONST,FCONST,AFLUX,GP,NCEGPS,NG ;NF,MPJ##
    2CCE SLNOEL*O.C
GAYERAGE MLCROSGOPIC CRCSSGECTLON DATA OYER INTEAMERLATEGBERGLES
```



```
        Z1NCR=hIOTH/Z.C
        DPVAR =EI([J+ZINCR
        CC 2001 KK=1 p200
```




```
        SLPCEL*SLPDEL+GARSUN
        DP YAR =DP VAAFHID TH
    2COI CONT INUE
```




```
        -5.13
    ce te }5
    3CEC SLPDEL=O.O
CAVERAGE MICAOSLOPIC CROSS_SECILGN_DATA_QVER_EAST_ENERGIES
```



```
    2INCR=*1.07H.\ell2..6
    DPVAR=E \|||HINCR
    IC 3001 KKEl-200
```




```
        SLPCEL=SUNDEL+FARSUN
        DPYAR *DPYAR+GIOIH
    30C1 CONTINUE
        SNN|TESUN[AT+SUMCEL
        I=1+1
        45 JC.52
    1C2 GPXSECIIJ=1.DE+50
    106 READI7,5).10UME
    5 FEFMAT (7IX,111
        RETLAN
        END
```



*SUERCUTINE PLSITCne*
S゙BRCLTINE PCSITCA




+AESYREDHLEDSAUC:
$\stackrel{C}{C}$
 CICN, GECMETRY,AAD TIME AFTER SHUTOOLN
$c$
$\stackrel{c}{6}$
C ARGUENTS IN-TJMACT.PRODNL,NONLGG,NCPAS,DCSSUH,SYSDES, IDNC,STAFS
C AFGUHERTS [W-AFCOHL, COSNUC
c afglegnts Cli-hche


 +Ci. ICEKMi 30). IDALGI 30), DLE VL( 301, PRES (3C)

REAL MLLT
-RITELENLCJI
ICO FGFPATI/// $2 \mathrm{LH}, 61 \mathrm{hTFE}$ FCLLDING LUTPUT DATA ARE CALCULATED DOSE RA + TES. 1 N. MRAO/H. 11
GJMITIALIZE ARRAYS
DC $1 \quad \mathrm{~m}=\mathrm{l}$, hCPAS
SY50C51M1=6.
CSS $\mathrm{MUC}(\mu)=0,0$
$D C S S L M(M)=C . C$
1.CCNTINUE

CTATTIALIZE ARAOYS
$0 C 101[=1,35 C$
AT (I)=NPTIT=MOCTIXC

101 CONTINUE
CIAITLALIZE ARQAYS
CC 2 In , RICKUCL
DC. 2-JE 213
$k=(1-1)+23+5$

H(K) $=$ PR $\mathrm{COHL}(I, J)$
AC!KIETHACTIEJ, 1$)$
ATIKISIDN(\{I)
2 CONITNUE
CAFFANGE TARGET ID NUMEERS, PARENT ACTIVITTESAND PARENT HALFELIFES IN
ZCRDER CF IKCRESSIMG PARERT CO MUMBER PRICR IE SEARCMING LELDEC FDR
CPARENT CECAY INFORHATIDN
$A C \equiv A C N U C L$
OC $3 \mathrm{~J}=\mathrm{INO}$
NTEMPL=NP(J+1)
ATEPP2 ant $1+1$ :
2 TE MP3ョAC $1+1+1$
2T:
OC. $\& K=2, J$
$1=J+1-k$
IFIATEPP1 GEAMFII\| GOTO. 5
$A P(i+1) \exists \mathrm{AF}(t)$
MT(1+1) $=+111)$
ACII+1)AC(I)

```
        * FSUSRLUTYNE DOSRTETTIMACT,PRODNU,NONUCL,NOPAS,OOSSUM,SYSDCS,IDNO,ST***
        F[I*! )= (i゙d
        4CFNTINLE
        I=C
    5 AF{[+1)EATEFP}
        AT(I+1) =ANTEHP2
        A(13+1)=2TEMP3
        H(1+1)=2TEMP4
    3.CFNIINLE
CSEEFCR TRE NP ARRAY FOR THE FIRST NON-ZEKG PIRENT IO NUMEER
        L=1
    7FINPILI.NE.CI GO 70 205
        L*l*!
        GC TC 7
GMARK THE EEGINNING OF THE PARENT_NGLLLDE_UST-THIS LLSI_IS_IN_ASGENOLN
CG CREEF
    <C5IEASEBL
CEEGIR INTERROGATILN DF THE LEVDEC LIBRARY FCR A PARTICULAR PARENT ID
GHUPBEF
    2CO PEAD( i,IE) l TAPE,HALFLF,ADMCDE
    16 FCRMATIL6,2OX,EN1,4.131
        IFIECF{7I,NE,O) GC TC 200
            IFIITAPE.ED.AP(H)SGC_TC_17
        IFIITAPEOGT.NP(LII GO TO 2E0
        CALL ACVANIACMCDE!
        GC 75 2ce
    28J CALL ACYENMNCMCEE,
    2E MRITE{C,2与1 hf(l)
```



```
        *T EXIST IN GECAY LIERARY-THEKEFORENO CONTRIBUTION FROG THIS NLCL
        *IDE IM OCEE (ALLS)
        MRITE(\epsilon, 37) NTIL)
    37. FCEH&TIIF,19HTHE TARGET ATCM HAS.15%/////1
        L=L+1
CLISCCNTINUE IF THE LAST PARENT NUCLLDE HAS BEEN IDDRESSED
        IFINP(L).EG.O) GCTC BLI
        IFIITAPE,GT,AP(LI)GE TL 28
        CE TC 200
CCHECK IC SEE IF NF(L) ECUALS NP (6+1]-THIS OCCURS HHEN THE SAMEPPARENTS
GAFE PRCDUCEC FRIM OIFFERENT TARGET NUCLIDES. IA THIS CASE, THE CECAY
C[AFGRFATICN IS STCREC IN VARICUS HCLDING ARRAYS AND IS INTERROGATED AS
cmany times as there are mdditIchal like parents. this hegates having
CTC REREAC LEYCEC FCR EACH LIKE PARENTA
    27 1F(NP(2).E6.AFILHIJ GCTC 800
CECMP的E MALF LIFES
    313 1FIPILJ/HALFLF,LT.0.8.OR.H(LJ/HALFLF.GT.1.2I GO TC 19
```



```
        CMÖDÉ= C
    3C2 1MSCE# IMCCE+I
CBEGIN IMTER&CGATICN"CF THELEVDEC LJERAAY
    REACC(7, SCGI LOEKMO, IDAUGM,DLEVEL, PHLT
    36C =cFPAT136x,12,17,E11,4,E12,41
CIF OECAY IS BY BEIA PINLS,STCRE DAUGHTER ID NUROER
        [F[ICEKMO.EG.g] NDTLISIDAUGA
        IFIIEEKMC.EG.7.CR.ICEKAC,EGGI GO TO 301
        IFTIDENMKEO.81 GC TC 310
        303 LFIIMCDE,EC.NCPCDEI OD TO 3G4
        GC TC 302
    301 JFIDLEVEL.EO.C.CJ_GC IO 303
```

```
                        -
    GALL CURYECDEVELIFLLEES
```



```
    6C TE 303
    E10 1FICLEVEL. EQ.0.0) 60 10311
        CALL CLFVEIOLEVEL, FLUXEI
        CALL CALCIAC(LI, OCSPRO(L), PULT,FIUXEI
Cin dny Gase ado in ThC C.SLl mev anntlihation ganmas
_ELLCE 312 I.EETAEL2
        [ALL CLRVETS.511,FLLXE)
        ©GL CALCSACILI, OOSPRD(LHtMULTFFLUXE)
    ミ12 CCATIMLE
    GC IC 3C3
        1c CALL ACVARINC̈MCDEJ
CREAD THE NEXT LEYCEC ALCLIDE
    204 KEAD (7, Ib) I TAPE, HALFLF, NDMODE
        IFIECE171, NE O I GO ID 304
        IFIITAPEEGORP(L)I GC TC 313
        IFIITAPE. EQ .NP(L+1)) GD TO. 21
CEISEEATINUE IF THE LAST RUG IDE HAS BEEN ADDRESSEO
    1F[BP(L+1),EC.C) GC IC B:3
    CALL ACVAN(NCMODE)
    \(L=L+1\)
    6C ic 2CC
    \(21 \mathrm{~L}=\mathrm{L}+1\)
    Gt TC 17
CLQED OECAY INFGRMATICA INTC THE HCLOING ARRAYS
    800 [C 801 IL= 1 , RDNCDE
```



```
    BGI CCNTINUE
CPAEKIPE BEGIARINGCELBKEPARENT NUCLIDES IN THELIST
    IBEGIN=L
    Ecz \(1=1+1\)
    JFINPILJ.EG.NP(L+1) GO TO 802
CMARK IHE END OF LLKE PAREAT NUCLIDES IN THE LIST
    IENC=
    ELO OC 日SS IPEIBEGIA,IEAO
COETERMINE IF Á PARERJ ALCLIDE MILL EE COÃSIDERED FCR DOSE RATE CALCS
```



```
        -50.AMD.AC(1P). AE.0.01 GL TE 804
            GC TE EAF
        804 LC 220 1L 1 , MCPCOE
CBEGIA JATERFLGATICA CF TRE KCLDING ARRAYS
```




```
    IFIDEKATILI EG.BS GL TE 510
    E T
```



```
    CALL. CLRYE (OLE YLII!), FLUXE)
```



```
    GC. TC 025
    EIC IFIDLEVLILI.EG. J.G) GE 7C 511
    (ALL CUPVE(DLEVL (ILI, FLUXE)
    CALL CALCACIIFI,ECSPRGIPI, PROEITLI, FUXEI
    511 DO E22 1BETA=1\&
    CALL CUFVEIO -EII, FLUXE
    CALL CALE (AC IIF), CESPRCIIPI, PRCE (ILI, FLUXE)
    Eえ」 CCNTINUE
    820 CCNTIME
```



```
    E65 CCATINLE
CEETERMINE IF THE NEAILEVQEC NUCLIDE_IS_THESSNE AS THE PREYICUS OAE
    306 FESCIT,16: [TAFE,FALFLFTNOHOOE
            IFIECF(71.NE.C) GC_. IC 806
            IF(ITAPE.EQ.NP(IBEGINI) GO TO BCB
            GC I5 807
        EC日 DO ECG IL=L,NDPCDE
            FEAC(7,300)_IDEK:GHL_JOALGGILI_OLEVLULL,PROBCLLI
    gCg rCNTINLE
            GC TC ElO
creterrine lf tre next leveec nucl ide is equal to the next parent id
CALMSEF fCLLEmING THE.LIST GF LIKE PARENT NUCLIOES
    EG7 IF\ITAPE.EG.NP{IEND+1)\ GO TO 81!
    CALL ACYAN(G)ECCE:
    L=1END+1
    IFIITAPE.GT.NPIIENQ&LULGQTO &12
    GC TC 200
    _ el1 LElENO+1.
    CC 70 17
    EL2.*RIIE(6,29) AP(L)
            MRITE\6.37) NT(L)
            LEl+l
```



```
            GC 10 200
    E 13 FEL1NC }
CMCYE THE_ELLE MARKEF THREUGH IHE ACTLAFF LIERARY AND POSITION AT THE
CEEGINNING OF LEVDEC
    49 FEAE(7,47) ITIFE
    47 FORMAT(I S)
            IFIEDF(7).NE O1_GO_TO 700
            CALL PESITEN
            60 7C 45
CMIRK TFE ENC OF THE LIST DF ARRGMGED PARENT NUCLIOES
    _7SC I$TCP=L
    L-18A5E
GEEGIA INTEFRCGATLCN OE THELEVOEG LIBRARY FGR POSSIBLE RADIDAE TIVE
CDALGHTERSGRAD UNTIL THE IG NUMBER IS MATCHEE
    7C1. [F(NO(L),NE.O) GO TO 702
    L=L+1
    IF{LEEC.{STEF+1) GC TC 913
    GD TC 701
_7C2 EEAC[7,16) [TAFE,HALALFMDMCOE
    IF(ECF(7).NE,C) GC TC 702
    IFIITAPE.EQ.ND(LLI)GO TO 703
    IFIIT&PE.GT *NCILJ& GE TC 925
    CALL_ADVANGMPCDES
    CC TO 702
    526 L=L+1
    IFGLOEO.15TOP+ H1 GC TC S13
    IFIITAPE.ECOND(L))GO TO 703
    CALL ACVAA(NCNCOE)
    GC TO 7E1
CCRECKTHE FALF LIFE CF THE CAUGHTEAR
7CzJF\HALFLF.GT.L.GE+4.8) GC TC 704
    GC Ti 705
    TCG CALL ACYAMINEMCEE:
    cc4 L=L+1
    IFGL.EG.{STOPP1\GGO 10..513
```



```
            G% TIT 702
    745 1%CDE=0
    5C2 1MCDE =1MODE+1
    CEEGIN IMTEFFGGATIGN CF THE LEVCEC LIGRARY FOR DAUGHTER DECAY INFO
        REAC(7.7E7) IDEKME,CLEVEL,NULT
    7C7 FCRMAT(3&),I2,7X,E11,4,E12,4)
        IFIILERNC.EQ.7.CR.ILEKMO.EQ.9) GO゙TO゙O goi
        IFIIDENME.ES&B| G5 1C 910
        SJ2 JF(IMOCE.ES.NDMDDES GO TO SO4
        GC TC POZ
    SC: Jf(DLEVEL.EJ.E.C.C)G̈r TC 933
        CALL CURVESOLEVEL,FLUXE:
        CALL CALC(AC(L),OCSCAU(L),MULT,FLUXE)
        GC TC S03
    910 IFILLEVEL .EG.0.0) GG TO 91L
    CALL_CLHYFIOLEYGL,FLUXE)
    CALL CALC (AC(L),DOSDAL(L), MULT,FLUXE)
CIR ANY CASE ACO IN THJ C.5.LI HEV ANNHLIHATIOM GMMMAS
    Gll OC SI2 IEETAElv2
```



```
    CALG CALC{ACVITOCSCAU|LI,MULT, FLUXES
    s.1% CCATINLE
    GC TD 5C 3
    S13_EE&GACT
    FOCNCNLC下"$3
    CO }315\mathrm{ I=1.NONUCL
CPRINT FEADINGS
```




```
    HRITESS,400:
    4CE FCFMATIIH DI25FTAFGEJ PRCLUCT CSRITI\ OSRIT2\ OSRIT3\ OSR{T4]
    + LSR(TS) DSR(TG) OSR:TT) OSR(TGI OSR(TSI DSR(TLO) DSRITIL)
    +CSF(TL2M)
        DC 日18 l=IEASE,AC
            IFINTCLI,EQ. LDNOCII) GO TO 8LT
            CC 7R 818
CSLH THE DCEF RATES OLE TC PAAENTS ANO DALGHTERS
    BIT CE 819 ka l, NCPAS
```



```
            *XP(- (STAFS(K)*C.6§3*3tCO.OAH(L)])
            CESsum{ki=0CSSUM{K\+EOSNUC{X)
            CYSOCS(K)=SYSOCS(K)+DCSNUC (K)
    EIS CCNTINUE
CEFIAT CCSE RATES
```




```
    818 CCNT INUE
CPRIATHEADINGS
            HRITE{B, EICSTAFSIK1,K=1,NCPAS)
            mflTElG,LOI
```



```
    + DSR(T5) OSR(TG) DSR(17) OSR(T8) OSR(T9) OSRET10) WR\TII)
    +ESF(T12))
    mFtTE(0,23) [CAC{Id,IDCSSUM{K),K=),ACPAS)
```




```
        OC E22 K=1,NDPAS
        CCSSUM(K)=0.0
        E2z CCNTIMLE
    E15 CCMTINLLE
```



```
CPRINT SYSTEF HEAD:NCS
        WRITE(E,2E)
```




```
        *DSP\il21)
CFRIAT SYSTEN EGSE
        HFITE(E,S1H(SYSDCSIK),Kal,MCPAS)
```



```
        RETURN
        ENC
```

```
- -- SSURCUTINE CALCVAGT,OSPRD,MLLT,FLUXES***
    El&RCLTINE* CALCIA゙CT,ISPRD,MULT,FLUXXEJ
    CJHIS PPOGRAF: CALCULATES DCSERATE FGR 3 CIFFERENT .GECMEFRIES
    6
    C ARGLMENTS [N-AT,T,FLLT,FIUXE
    ARGUMENTS OUT-USPRD
C... COMFCA A,V,F,CF,T1,TZ,N,R,D,HT,OPTION,IGEDM
    , BEAL MULI
    GO TD(31,32,33) 1GEGM
```



```
    DSFRD=ESPRD+DCSRAT
    PETURN
    CTFE AECVE dCSEPATE CALCULATIDN IS FOR A POINT SOLGGE
```



```
        -
            S=(上CT*MULT*2.5 //i VOL*FLUXE*4.0)
```



```
            +1R+OI#LCO.al
            CSPRCm [SPRD+OOSRAT
    FETLRN
CTHE ARCVE DQSERATE CALCLLATICN_IS FCR_A HCMGGENEOUS_NON-ABSORGING
CSFPERE
```



```
    S=(ACTMMULT*2*5) /(NOL#FLUXE*4.9)
```




```
    [SPRD=CSPRD+QOSRAT
    FETUFA
CTHE AEOUE CALCULAILCM IS FCR A HCMOGENECMS NCN-ABSORBING CYLINCER WHERE
CTHE OESERYER FACES THE AXIS
    END
```

```
    * OSLSRCLTINE CLRVEIE,FLUXEI***
    sGRClGTME CURGEIE,FUUXEI
C
CGIVEA THE ENEPGY CF THE GAMMA gAY,THIS PRCGRMM CALCULATES THE PARTICLE
CFLUXIUUANTA/CMm*&SECI THAT CORRESPGNDS TG ACSE RATE CF 2.5 HRAHR-
CFLUXE
C
C
C ARGLMEFTS OLT-FLUXE
C. JFIE.EE.C.CIC.ANO.E:LE.O.OZOV GC TE I
```



```
    IFIE,GT.0.040.AND.E.LE.0.050) GD TE 3
    IFIE,GT.G.D5G.AND.E.LE.O.O6OD GO TC. }
    IFTE.GT.0.060.DND.E.LE.C.DAOI CO TO 5
    IFIE.GT,G.GEC.AND.E.LE.O.15OI,GD TC.G
    IFIE.GT.C.15C.AND.E.LE.0.40J1 GO TO }
    IFSE,GL.O.400.AND.EELE.O.6001 GOTTO_8
    JFJE.GT.0.60CT.AND.E.LE.1.0001 GD TC 9
    IFIE.GT._I UCO .AND.E.LE.1. 500)_GO TO 10
    IFIE.GT.1.500.ARD.E.LE.3.0001 GC TO II
```



```
    IF\E.GT .G.000.AND.E.LE.10.000) GO TO 13
    F6LXE=1_=0 E+99
    RETURN
    1.FIUKE=21495330.244毛早年2,1759447
        PETLRN
    2 FIUXE=11460233_22*E*42.C136022
        FETURA
```



```
        RETURM
        FLUXE=96497.38032*E**0.4772399
        RETLRN
    5_FLUXE=11225.12E210E**(-0.2B144101
        RETLRA
    G FLLXE=554E%,7032的EXPiE* (-10.51320651)
        RETURA
    7FLLXE=1330.569205年E果(1-1:1036085)
        KETURN
    8_FLUXEE1462.158GE6&E**1-1.00045131
        RETLRN
    9.FLUXE=154\epsilon.5c.7T58*E**(-G.8.091058)
        FEfURA
    1CFLLXEE154G.C5132C*E*#(-0.7797676)
        GETURM
    11 FLLXE=150日,205757*E**\-0.7188784!
        RETLRN
    12 FLUXE=1468.366150中E䩒(-0.7051622)
    RETLAN
    13 FLLXE El656.852ET4*E*#(-0% 7649757)
        FETUFA
        ERD
```

the feltohiag cat́a has been entered into hemgry


Specific times after shutctha irl

... LAIt ACRPALIEEC FLLXES



THE FOLLOWTNG HAS BEEN CALCULATED EY ACDOSI

INSTAMTANEOUS NUMBER GF XELTRENS PRCCUCEC PEASECCND——7645E＋13

| ERAGE ALPAER CF AELTRCAS FRGLUCEE PER SECENL ．7E45E＋12 |
| :---: |

IMTEGFALS CF THE FITIIAG FURGTICN CVER THE GRCUP INTERVALS GRGUP LCHER UOUNOARY GPPER ROLADARY REGERED INTEGRAL CALGULATED INTEGRAG－

| 2 C | －ICEE－C4 | －414E＋C0 | ． $829 \mathrm{E}-0.5$ | ．820 E－0 |
| :---: | :---: | :---: | :---: | :---: |
| 19 | ．414E＋CO | $.113 \mathrm{E}+01$ | －3CEE－07 | ． 3 CaE－37 |
| 1 I | $.113{ }^{\circ} 801$ | $.366 \mathrm{E}+\mathrm{O}$ | －34EE－07 | －346E－67 |
| 17 | － $30 \in E+C 1$ | ． $107 \mathrm{E}+02$ | .395 E－07 | －395E－07 |
| 16 |  | －250E「スて | －3C1E－⿹1 | －301 E＝07 |
| 15 | ．290E－02 | － 101 Etc 3 | ．46EE－07 | ．468E－07 |




| If ME(H) |  | - 0 | -1 OOE -03 | . $200 \mathrm{E}+01$ | -300E+01 | -400E:01 | - 590E:01 | -600E 4 C | -700c+01 |
| :---: | :---: | :---: | :---: | :---: | :---: | :---: | :---: | :---: | :---: |
| T ARGET | PPDCUCT | ACTITII | 2CIT121 | ACTIT3l | AC.7 (T4) | ACTITS) | ACTITS) | ACTIT ${ }^{\text {d }}$ | actital |
| 29063 | 29062 | . 0 | 0 | - 0 | . 0 | . 0 | . 0 | . 0 | . 0 |
| 29463 | zecte | - 0 | - 0 | . 0 | . 0 | . 0 | . 0 | -0 | . 0 |
| 29063 | 27c59 | 8 | . 5 | 0 | 0 | . 0 | 0 | 0 | 4 |
| 29663 | 28063 | -176E 404 | .176E404 | -176E 404 | . $1765^{+04}$ | 176E+04 | +176E+04 | +176E+04 | -176E+04 |
| 29063 | $276 \pm 0$ | - 0 | . 0 | . 0 | ¢ 0 | . 0 | - 0 | - 0 | . 0 |
| 29063 | 27660 | . 0 | . 0 | . 0 | . 0 | . 0 | - 0 | . 0 | . 0 |
| 29063 | 29064 | -431E!10 | $.409 \mathrm{E}+10$ | $.387 E+10$ | $.367 E+10$ | $.347 E+10$ | .329E+10 | . 312 t +10 | . 29SE+ 10 |


| - 900F+01 | $\begin{aligned} & \text { 900ㅌ } 91 \\ & \text { ACT (TIO) } \end{aligned}$ | $.100 \mathrm{E}+02$ <br> ACTIT11 | $.200 E+02$ <br> ACTIt12) |
| :---: | :---: | :---: | :---: |
|  | - | $\stackrel{0}{0}$ | ${ }^{0}$ |
| - 0 | - 0 | - 0 | - 0 |
| -2. | 0 | 0 | . 0 |
| $.176 E+04$ | .176E+04 | .176E404 | .176E404 |
| - 0 | 40 | - 0 |  |
| - 0 | . 0 | . 0 | . 0 |
| -280E+10 | .265E+10 | -25ifilo | -146E*:9 |
| -900Et01 | $\text { P90 } f+Q 1$ | 2005:03 | 2006902. |
| ACTI9) | ACTITIO: | ACTMII: | ACT 1121 |
| 1-280E+10 | -265E+10 | - $751 \mathrm{E}+10$ | . $1465+10$ |


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| TIMEIHI <br> taRget frcduct | $0$ | $\begin{gathered} \text { 100E+01 } \\ \text { CSROT } 1 \end{gathered}$ | $\begin{aligned} & \text { - 200E+C1 } \\ & \text { DSAIY3i } \end{aligned}$ | $\begin{aligned} & -3 \operatorname{cog} 4.51 \\ & \operatorname{csR}(141 \end{aligned}$ | $\begin{aligned} & =400 E+01 \\ & \text { DSRTTSi } \end{aligned}$ |  | $\begin{aligned} & \text { - } \mathrm{BOOE}+\mathrm{OL} \\ & \text { DSR(T7) } \end{aligned}$ | $\begin{aligned} & .700 \mathrm{E}+01 \\ & \text { DSRIthi } \end{aligned}$ |
| :---: | :---: | :---: | :---: | :---: | :---: | :---: | :---: | :---: |
| -29063-27655 | - 0 | 0 | $\stackrel{C}{C}$ | ¢ 6 | - 0 | - 0 | . 0 | - 0 |
| $29063-27060$ | ${ }^{0}$ | . 0 | .0 | . 0 | . 0 | -0 | .0 | - 0 |
| 29063 27CE0 | . 0 | . 0 | . 0 | . 0 | . 0 | .0 | . 0 | . 0 |
| 29063 2e062 | . 0 | ${ }_{. c}{ }^{\text {c }}$ | - 0 | .0 | .0 | -0 | . 0 | -3 |
| 2966328063 | 0 | .0 | . 0 | -0 | .0 | . 0 | . 0 | : 0 |
| 29063 25C62 | . 0 | - C | . 0 | .0 | . 0 | -0 | - 0 | -0 |
| 29063 25C64 | . 234 E 401 | . 22 lf+ 01 | . 2106-01 | . 199E+ 01 | . 1ase* OL | .178 E -01 | -169E+01 | -160Eal 1 |
|  |  |  |  |  |  |  |  |  |
| 1:mefil | - 0 | - ICOE 01 | - $2 \mathrm{CCE}+\mathrm{Cl}$ | . 3 COEF 01 | . $400 \mathrm{c}+01$ | - 500E+0t | .600E.01 | .700Es0! |
| 1 ARGEt | O5P1T1) | 054[121 | 054(13) | OSf 174 | DSRITS1 | DSR(76) | DSAETII | 0SR(18) |
| $29 C 63$ | . $234 \mathrm{E}+\mathrm{Cl}$ | -221E+0t | -2IOE 01 | -199E+01 | -180E+04 | -178E+01 | . $1695+C 1$ | . $1608 \pm 01$ |


| $\begin{aligned} & \text { MOOE401 } \\ & \text { OSRIT9) } \end{aligned}$ | $\begin{aligned} & .900 E \subset 01 \\ & \text { DSRIT10) } \end{aligned}$ | $\begin{aligned} & .100 E+02 \\ & 05 R E T 11 i \end{aligned}$ | $\begin{aligned} & -200 E+02 \\ & \text { WRitiz } \end{aligned}$ |
| :---: | :---: | :---: | :---: |
| . 0 | . 0 |  | 10 |
| -0 | . 0 | - 0 | - 0 |
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| . 0 | . 0 | . 0 | -0 |
| $\begin{aligned} & 1.0 \\ & 1.151 E+D I \end{aligned}$ | - 0 | . 0 | - 0 |
|  | $.143 \mathrm{E}+01$ | . 136E*01 | . 791E*00 |
|  |  |  |  |
|  | -900E 01 | . 100E+02 | -200E+02 |
|  | DSREI $10:$ | OSRITIII | OSR (TET) |
|  | .143E* 01 | -136E+0L | . $791 \mathrm{E}+00$ |



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    CALL PCSITIDN
    GC-LC 3
    50 CALL PCSIT2 (AP)
    CC 70103
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