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AEC Research and Development Report

DRAFT
A COMPUTER CODE FOR THE
CALCULATION OF FISSION
PRODUCT ACTIVITY RATIOS

June 1971

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DRAFT

A COMPUTER CODE FOR THE CALCULATION
OF FISSION PRODUCT ACTIVITY RATIOS

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June 1971

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ABSTRACT

Program DRAFT calculates information concerning fission events such as fuel isotopic concentrations and fission product disintegration rates at selected times during and after irradiation. The code was developed to provide analytical data on fission products for use in the determination of absolute neutron flux and exposure via non-destructive gamma scanning techniques. Equations used in the code are described and detailed input instructions are provided. Use of the code is illustrated with a test case, and both printed and plotted (optional) output are described.

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I. INTRODUCTION

Program DRAFT calculates information concerning fission events such as fuel isotopic concentrations and fission product disintegration rates at selected times during and after irradiation. The code was developed to provide analytical data on fission products for use in the determination of absolute neutron flux and exposure via non-destructive gamma scanning techniques. The codes NOTSFI and NUCON written for this purpose by J.A. Sovka⁽¹⁾ served as models for the present code.

Initial fuel isotopes are presently limited to U^{235} , U^{238} and Pu^{239} . Irradiation history of the fuel is input in the form of a histogram to allow an accurate description of fission product formation. Other input data required include initial concentrations and appropriate fission, capture, and absorption cross sections for each fuel isotope, and decay constants, absorption cross sections and fission product yields from each fuel isotope for each fission product of interest.

Printed output, provided at times specified by the user in the input power history, includes :

- Accumulated exposure time.
- Time averaged flux and accumulated exposure.
- Fuel isotope concentrations.
- Accumulated fissions in each fuel isotope, accumulated fissions in all fuel isotopes, and the fraction of accumulated fissions for each fuel isotope.
- Disintegration rate for each fission product.
- Ratios of selected fission product disintegration rates.

Optional plotted output consists of any or all of the following:

- Ratio of final to initial fuel concentrations vs. exposure.
- Disintegration rate vs. exposure or time, between any two specified times, for any fission product.
- Disintegration rate ratio vs. exposure or time between any two specified times and for any two fission products.

The DRAFT code is written in FORTRAN V for use on the UNIVAC 1108 computer. The time required to process a typical case with plotting output is about one minute on the UNIVAC 1108. The code is available from PNL by request to the author. Equations used in the code, input instructions for the code, and a sample test case are described in the following sections of this report.

II. EQUATIONS USED IN THE DRAFT CODE

The equations used in Program DRAFT are presented in this section. There are four basic equations in the program, each of which is used to calculate the concentrations as a function of time, of a given type of fission product. The calculation of quantities used in these equations and related output data will be discussed first, followed by definitions of and equations for each type of fission product.

A. Miscellaneous Calculations

The power history of the reactor is input in the form of a histogram; i e., for each increment of time, t_i , there is a corresponding value for relative power, P_i . For a specified value of the average flux, $\bar{\Phi}$, the exposure is given by:

$$E = \sum_i \Phi_i t_i = \bar{\Phi} t \quad (1)$$

where Φ_i is the absolute flux corresponding to the relative power (P_i) during the i 'th time interval (t_i) and t is the exposure time, defined as

$$t = \sum_i t_i \quad (P_i \neq 0) \quad (2)$$

i.e., for all t_i during which $P_i \neq 0$. The problem then is to determine the Φ_i which are the quantities used in the fission product equations.

Since flux and power are proportional, we can write all the Φ_i in Equation (1) in terms of any one. Arbitrarily choosing Φ_1 for this normalization we have

$$\Phi_i = \frac{P_i}{P_1} \Phi_1 \quad (3)$$

Substituting this expression for Φ_i into Equation (1) results in

$$\sum_i \frac{P_i}{P_1} \phi_1 t_i = \bar{\phi} \sum_i t_i \quad (4)$$

Solving for ϕ_1 , we get

$$\phi_1 = \frac{\bar{\phi} \sum_i t_i}{\sum_i \frac{P_i}{P_1} t_i} = P_1 \left[\frac{\bar{\phi} \sum_i t_i}{\sum_i P_i t_i} \right] = P_1 Q$$

The remaining values evaluated from Equation (4) are then

$$\begin{aligned} \phi_2 &= \frac{P_2}{P_1} \phi_1 = \frac{P_2}{P_1} P_1 \left[\frac{\bar{\phi} \sum_i t_i}{\sum_i P_i t_i} \right] = P_2 Q \\ \phi_3 &= \frac{P_3}{P_1} \phi_1 = \frac{P_3}{P_1} P_1 \left[\frac{\bar{\phi} \sum_i t_i}{\sum_i P_i t_i} \right] = P_3 Q \\ &\vdots \\ &\vdots \\ &\vdots \\ \phi_i &= \frac{P_i}{P_1} \phi_1 = \frac{P_i}{P_1} P_1 \left[\frac{\bar{\phi} \sum_i t_i}{\sum_i P_i t_i} \right] = P_i Q \end{aligned} \quad (5)$$

The ϕ_i are the fluxes actually used in the computations in the program. Note that if power is input in absolute rather than relative units, then Q represents the absolute flux per unit power. The value of Q is printed out for each input value of $\bar{\phi}$.

Additional information required for the fission product calculations are the fuel atom densities and number of

fissions in each fuel isotope as a function of time. In the case of ^{235}U , the differential equation describing the rate of change of the number of atoms in the fuel is

$$\frac{dN_{25}}{dt} = -N_{25}\sigma_{25}^a\phi \quad (6)$$

where N_{25} is the ^{235}U atom density

σ_{25}^a is the microscopic absorption cross section for ^{235}U and ϕ is the absolute neutron flux during the time increment (t).

Integration of Equation (6) with the initial condition

$$N_{25}(t=0) = N_{25}^0$$

yields

$$N_{25} = N_{25}^0 e^{-\sigma_{25}^a\phi t} \quad (7)$$

Similarly, for ^{238}U we have

$$\frac{dN_{28}}{dt} = -N_{28}\sigma_{28}^a\phi \quad (8)$$

where N_{28} is the ^{238}U atom density and σ_{28}^a is the microscopic absorption cross section for ^{238}U .

Integration of Equation (8) subject to the initial condition

$$N_{28}(t=0) = N_{28}^0$$

yields

$$N_{28} = N_{28}^0 e^{-\sigma_{28}^a\phi t} \quad (9)$$

The differential equation describing the formation of ^{239}Pu from neutron capture in ^{238}U , and loss through fission and neutron capture is

$$\frac{dN_{49}}{dt} = N_{28} \sigma_{28}^c \Phi - N_{49} \sigma_{49}^a \Phi \quad (10)$$

where σ_{28}^c is the microscopic capture cross section for ^{238}U

N_{49} is the ^{239}Pu atom density

and σ_{49}^a is the microscopic absorption cross section for ^{239}Pu .

Substituting the expression for N_{28} from Equation (9) into Equation (10) and integrating with the initial condition

$$N_{49}(t=0) = N_{49}^0$$

we get

$$N_{49} = N_{49}^0 e^{-\sigma_{49}^a \Phi t} + \frac{\sigma_{28}^c N_{28}^0}{\sigma_{49}^a - \sigma_{28}^a} \left[e^{-\sigma_{28}^a \Phi t} - e^{-\sigma_{49}^a \Phi t} \right] \quad (11)$$

The number of fissions in ^{235}U is given by

$$F^{25} = \int N_{25} \sigma_f^{25} \Phi dt \quad (12)$$

where σ_f^{25} is the microscopic fission cross section for ^{235}U .

Substituting the expression for N_{25} from Equation (7) into Equation (12) and integrating we get

$$F^{25} = \frac{\sigma_f^{25}}{\sigma_a^{25}} N_0^{25} \left[1 - e^{-\sigma_a^{25} \Phi t} \right] \quad (13)$$

Similarly for ^{238}U we have

$$F^{28} = \int N_{28} \sigma_f^{28} \Phi dt \quad (14)$$

where σ_f^{28} is the microscopic fission cross section for ^{238}U .

Substituting the expression for N_{28} from Equation (9) into Equation (14) and integrating yields

$$F^{28} = \frac{\sigma_f^{28}}{\sigma_a^{28}} N_{28}^0 \left[1 - e^{-\sigma_a^{28} \Phi t} \right] \quad (15)$$

Finally for ^{239}Pu we have

$$F^{49} = \int N^{49} \sigma_f^{49} \Phi dt \quad (16)$$

where σ_f^{49} is the microscopic fission cross section for ^{239}Pu .

Substituting the expression for N_{49} from Equation (11) into Equation (16) and integrating results in

$$F^{49} = \sigma_f^{49} \left\{ \frac{N_{49}^0}{\sigma_a^{49}} \left[1 - e^{-\sigma_a^{49} \Phi t} \right] + \frac{N_{28}^0 \sigma_c^{28}}{(\sigma_a^{49} - \sigma_a^{28})} \left[\frac{1 - e^{-\sigma_a^{28} \Phi t}}{\sigma_a^{28}} - \frac{1 - e^{-\sigma_a^{49} \Phi t}}{\sigma_a^{49}} \right] \right\} \quad (17)$$

B. Fission Product Equations

The equations in DRAFT, which describe the concentrations of selected fission product nuclides in a reactor, assume that any fission product of interest can be classified as belonging to one of the following four groups:

Group A

These are fission products that are produced directly in the fission process or are products of precursors having half lives much shorter than their own. Examples of Group A nuclides are Cs^{137} , Zr^{95} , Ru^{103} , Ru^{106} , Ba^{140} , and Ce^{144} .

Group B

These are daughters of Group A nuclides that are not in secular equilibrium with their parents and have a negligible direct fission yield. Examples of Group B nuclides are Nb^{95} and La^{140} .

Group C

These are fission products which are produced as a result of neutron capture in stable (i.e., Group D) fission product nuclides. Examples of Group C nuclides are Cs^{134} and Eu^{154} .

Group D

These are stable fission products that are produced directly in the fission process or are daughters of precursors that are produced directly in the fission process. Examples of Group D nuclides are Cs^{133} , Eu^{153} , and Nd^{148} .

Equations for each of the above types of fission products are given below. Symbols used in these equations are defined in Table I in terms of previously defined quantities.

TABLE I

DEFINITION OF SYMBOLS USED IN FISSION PRODUCT EQUATIONS

$$\alpha = \sigma_{25}^a \Phi$$

$$\alpha' = \sigma_{25}^f \Phi Y_A^{25}$$

$$\beta = \sigma_{28}^a \Phi$$

$$\beta' = \sigma_{28}^f \Phi Y_A^{28}$$

$$\delta = \sigma_{49}^a \Phi$$

$$\delta' = \sigma_{49}^f \Phi Y_A^{49}$$

$$\gamma = \sigma_{28}^c \Phi$$

$$\xi' = \sigma_{25}^f \Phi Y_D^{25}$$

$$\theta = \sigma_C^c \Phi$$

$$\eta' = \sigma_{28}^f \Phi Y_D^{28}$$

$$\rho = \sigma_D^a \Phi$$

$$\nu' = \sigma_{49}^f \Phi Y_D^{49}$$

σ_A^a = Microscopic absorption cross section for Type A fission product

σ_B^a = Microscopic absorption cross section for Type B fission product

σ_C^a = Microscopic absorption cross section for Type C fission product

σ_D^a = Microscopic absorption cross section for Type D fission product

σ_D^c = Microscopic capture cross section for Type D fission product

λ_A = Decay constant for Type A fission product, $\lambda_A' = \lambda_A + \sigma_A^a \Phi$

λ_B = Decay constant for Type B fission product, $\lambda_B' = \lambda_B + \sigma_B^a \Phi$

λ_C = Decay constant for Type C fission product, $\lambda_C' = \lambda_C + \sigma_C^a \Phi$

Y_A^{25} , Y_A^{28} and Y_A^{49} = Fission product yields for Type A fission product in ^{235}U , ^{238}U and ^{239}Pu fissions.

Y_D^{25} , Y_D^{28} and Y_D^{49} = Fission product yields for Type D fission product in ^{235}U , ^{238}U and ^{239}Pu fissions.

TYPE A EQUATIONS

The time rate of change of the number of a Type A fission product nuclide is given by

$$\begin{aligned} \frac{dN_A}{dt} = & \text{rate of production by } {}^{235}\text{U fissions} \\ & + \text{rate of production by } {}^{238}\text{U fissions} \\ & + \text{rate of production by } {}^{239}\text{Pu fissions} \\ & - \text{rate of decay} \\ & - \text{rate of depletion by neutron absorption} \end{aligned}$$

In terms of the previously defined symbols

$$\begin{aligned} \frac{dN_A}{dt} = & N_{25} \sigma_{25}^f \phi Y_A^{25} + N_{28} \sigma_{28}^f \phi Y_A^{28} \\ & + N_{49} \sigma_{49}^f \phi Y_A^{49} - N_A \lambda_A - N_A \sigma_A^a \phi \end{aligned} \quad (18)$$

Substituting expressions for N_{25} , N_{28} , and N_{49} from Equations (7), (9), and (11) respectively, and integrating over the time increment, t , we get

$$\begin{aligned} N_A = & \frac{N_{25}^0 \alpha'}{(\lambda_A' - \alpha)} \left[e^{-\alpha t} - e^{-\lambda_A' t} \right] + \frac{N_{28}^0 \beta'}{(\lambda_A' - \beta)} \left[e^{-\beta t} - e^{-\lambda_A' t} \right] \\ & + \frac{N_{49}^0 \delta'}{(\lambda_A' - \delta)} \left[e^{-\delta t} - e^{-\lambda_A' t} \right] + \frac{N_{28}^0 \gamma \delta'}{(\delta - \beta)(\lambda_A' - \beta)} \left[e^{-\beta t} - e^{-\lambda_A' t} \right] \\ & - \frac{N_{28}^0 \gamma \delta'}{(\delta - \beta)(\lambda_A' - \delta)} \left[e^{-\delta t} - e^{-\lambda_A' t} \right] + N_A^0 e^{-\lambda_A' t} \end{aligned} \quad (19)$$

where N_A^0 is the value of N_A at the beginning of the time increment.

TYPE B EQUATIONS

The time rate of change of the number of a Type B fission product nuclide is given by

$$\frac{dN_B}{dt} = \text{rate of production by decay of Type A nuclide}$$

$$- \text{rate of decay}$$

$$- \text{rate of depletion by neutron capture.}$$

In terms of previously defined symbols

$$\frac{dN_B}{dt} = N_A \lambda_A - N_B \lambda_B - N_B \sigma_B^a \Phi \quad (20)$$

Substituting the expression for N_A from Equation (19) and integrating over the time increment, t , we get

$$N_B = \frac{N_{25}^0 \lambda_A \alpha'}{(\lambda_A' - \alpha)(\lambda_B' - \alpha)} \begin{bmatrix} e^{-\alpha t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$- \frac{N_{25}^0 \lambda_A \alpha'}{(\lambda_A' - \alpha)(\lambda_B' - \lambda_A')} \begin{bmatrix} e^{-\lambda_A' t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$+ \frac{N_{28}^0 \lambda_A \beta'}{(\lambda_A' - \beta)(\lambda_B' - \beta)} \begin{bmatrix} e^{-\beta t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$- \frac{N_{28}^0 \lambda_A \beta'}{(\lambda_A' - \beta)(\lambda_B' - \lambda_A')} \begin{bmatrix} e^{-\lambda_A' t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$+ \frac{N_{49}^0 \lambda_A \delta'}{(\lambda_A' - \delta)(\lambda_B' - \delta)} \begin{bmatrix} e^{-\delta t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$- \frac{N_{49}^0 \lambda_A \delta'}{(\lambda_A' - \delta)(\lambda_B' - \lambda_A')} \begin{bmatrix} e^{-\lambda_A' t} & -e^{-\lambda_B' t} \end{bmatrix}$$

$$\begin{aligned}
& + \frac{N_{28}^0 \gamma \lambda_A \delta'}{(\delta - \beta)(\lambda_A' - \beta)(\lambda_B' - \beta)} \begin{bmatrix} e^{-\beta t} & -e^{-\lambda_B' t} \end{bmatrix} \\
& - \frac{N_{28}^0 \gamma \lambda_A \delta'}{(\delta - \beta)(\lambda_A' - \delta)(\lambda_B' - \delta)} \begin{bmatrix} e^{-\delta t} & -e^{-\lambda_B' t} \end{bmatrix} \\
& + \frac{N_{28}^0 \gamma \lambda_A \delta'}{(\lambda_A' - \delta)(\lambda_A' - \beta)(\lambda_B' - \lambda_A')} \begin{bmatrix} e^{-\lambda_A' t} & -e^{-\lambda_B' t} \end{bmatrix} \\
& + \frac{N_A^0 \lambda_A}{(\lambda_B' - \lambda_A')} \begin{bmatrix} e^{-\lambda_A' t} & -e^{-\lambda_B' t} \end{bmatrix} + N_B^0 e^{-\lambda_B' t}
\end{aligned} \tag{21}$$

where N_B^0 is the value of N_B at the beginning of the time increment, t .

TYPE C EQUATIONS

The time rate of change of the number of a Type C fission product nuclide is given by

$$\begin{aligned}
\frac{dN_C}{dt} = & \text{rate of production through neutron capture} \\
& \text{reactions in Type D nuclide} \\
& - \text{rate of decay} \\
& - \text{rate of depletion by neutron absorption.}
\end{aligned}$$

In terms of quantities previously defined

$$\frac{dN_C}{dt} = N_D \sigma_D^C \Phi - N_C \lambda_C - N_C \sigma_C^a \Phi \tag{22}$$

Substituting the expression for N_D from Equation (25) and integrating over the time increment, t , we get

$$\begin{aligned}
N_C = & \frac{N_{25}^0 \theta \xi'}{(\rho - \alpha)(\lambda_C' - \alpha)} \begin{bmatrix} -\alpha t & -\lambda_C' t \\ e & -e \end{bmatrix} - \frac{N_{25}^0 \theta \xi'}{(\rho - \alpha)(\lambda_C' - \rho)} \begin{bmatrix} -\rho t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& + \frac{N_{28}^0 \theta \eta'}{(\rho - \beta)(\lambda_C' - \beta)} \begin{bmatrix} -\beta t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& - \frac{N_{28}^0 \theta \eta'}{(\rho - \beta)(\lambda_C' - \rho)} \begin{bmatrix} -\rho t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& + \frac{N_{49}^0 \theta \nu'}{(\rho - \delta)(\lambda_C' - \delta)} \begin{bmatrix} -\delta t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& - \frac{N_{49}^0 \theta \nu'}{(\rho - \delta)(\lambda_C' - \rho)} \begin{bmatrix} -\rho t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& + \frac{N_{28}^0 \gamma \theta \nu'}{(\delta - \beta)(\rho - \beta)(\lambda_C' - \beta)} \begin{bmatrix} -\beta t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& - \frac{N_{28}^0 \gamma \theta \nu'}{(\delta - \beta)(\rho - \delta)(\lambda_C' - \delta)} \begin{bmatrix} -\delta t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& + \frac{N_{28}^0 \gamma \theta \nu'}{(\rho - \delta)(\rho - \beta)(\lambda_C' - \rho)} \begin{bmatrix} -\rho t & -\lambda_C' t \\ e & -e \end{bmatrix} \\
& + \frac{N_D^0 \theta}{(\lambda_C' - \rho)} \begin{bmatrix} -\rho t & -\lambda_C' t \\ e & -e \end{bmatrix} + N_C^0 e^{-\lambda_C' t}
\end{aligned} \tag{23}$$

where N_C^0 is the value of N_C at the beginning of the time increment, t .

TYPE D EQUATIONS

The time rate of change of the number of a Type D fission product nuclide is given by

$$\begin{aligned} \frac{dN_D}{dt} = & \text{rate of production by } ^{235}\text{U fissions} \\ & + \text{rate of production by } ^{238}\text{U fissions} \\ & + \text{rate of production by } ^{239}\text{Pu fissions} \\ & - \text{rate of depletion by neutron absorption.} \end{aligned}$$

In terms of quantities defined previously

$$\frac{dN_D}{dt} = N_{25}\sigma_{25}^f\phi Y_D^{25} + N_{28}\sigma_{28}^f\phi Y_D^{28} + N_{49}\sigma_{49}^f\phi Y_D^{49} - N_D\sigma_D^a\phi \quad (24)$$

Substituting expressions for N_{25} , N_{28} , and N_{49} from Equations (7), (9), and (11) respectively and integrating over the time increment, t , we get

$$\begin{aligned} N_D = & \frac{N_{25}^0 \xi'}{(\rho - \alpha)} \begin{bmatrix} -\alpha t & -\rho t \\ e & -e \end{bmatrix} + \frac{N_{28}^0 \eta'}{(\rho - \beta)} \begin{bmatrix} -\beta t & -\rho t \\ e & -e \end{bmatrix} \\ & + \frac{N_{49}^0 \nu'}{(\rho - \delta)} \begin{bmatrix} -\delta t & -\rho t \\ e & -e \end{bmatrix} + \frac{N_{28}^0 \gamma \nu'}{(\delta - \beta)(\rho - \beta)} \begin{bmatrix} -\beta t & -\rho t \\ e & -e \end{bmatrix} \\ & - \frac{N_{28}^0 \gamma \nu'}{(\delta - \beta)(\rho - \delta)} \begin{bmatrix} -\delta t & -\rho t \\ e & -e \end{bmatrix} + N_D^0 e^{-\rho t} \end{aligned} \quad (25)$$

where N_D^0 is the value of N_D at the beginning of the time increment, t .

III. INPUT INSTRUCTIONS FOR DRAFT

Following a case title card, all input is in the NAMELIST⁽²⁾ format. In NAMELIST there must be a comma following each variable and no blank spaces between the variable and the comma. There may however, be blanks following the comma. A variable need only be input if the value of that variable is different from that in the previous case. All values of variables remain the same unless specifically changed by new input. All variables needed for the DRAFT input are defined in the following input instructions.

DRAFT INPUT INSTRUCTIONS

The following input need only be specified once for each run.

	<u>FORMAT</u>	<u>VARIABLE NAME</u>	<u>DESCRIPTION</u>
Card 1	(I5,9A6)	NISO TITL	Number of isotopes in this case. (≤ 20) 54 Holerith case identification.
Card 2	(NAMELIST)	In NAMELIST input the first card for each isotope must begin with the name \$DRAFT in column 2. Each card after that must also begin in column 2. As many cards may be used as is necessary for each isotope. The last card for each isotope must end with a \$.	
		IVAR	Must be 2.
		NPHI	Number of average fluxes to be used in the calculations. (≤ 20)
		NTIME	Number of times used in the calculations. (≤ 200)
		NRATIO	Number of ratios to be calculated. (≤ 20)
		FLXAV(I)	Average fluxes for which the calculations will be made. (neutrons/cm ² -sec) (NPHI of these)
		POWR(I)	Reactor power at each input time step. Can be absolute or relative power or flux. (NTIME of these)
		TIME(I)	Input times. (seconds)

ANO25(1)	U-235 initial concentration. (nuclei/barn-cm)
ANO28(1)	U-238 initial concentration. (nuclei/barn-cm)
ANO49(1)	Pu-239 initial concentration. (nuclei/barn-cm)
SIGA25(1)	Sigma A for U-235. (barns)
SIGA28(1)	Sigma A for U-238. (barns)
SIGA49(1)	Sigma A for Pu-239. (barns)
SIGC28(1)	Sigma C for U-238. (barns)
SIGF25(1)	Sigma F for U-235. (barns)
SIGF28(1)	Sigma F for U-238. (barns)
SIGF49(1)	Sigma F for Pu-239. (barns)

PLOTTING OPTIONS

IPL0T	0 = No plots of decay rates or N25, N28, N49. 1 = Plot decay rates only. 2 = Plot N25, N28, N49 only. 3 = Plot both decay rates and N25, N28, N49.
RPLT	0 = No plots of ratios. 1 = Plot ratios.
CYCLE(1)	Total length of X-axis for decay rate plots. (inches)
CYCLE(2)	Height of one cycle on Y-axis for decay rate plots. (inches)
CYCLE(3)	Total length of X-axis for N25, N28, N49 plots (inches)
CYCLE(4)	Total height of Y-axis for N25, N28, N49 plot. (inches)
CYCLE(5)	Total length of X-axis for ratio plots. (inches)
CYCLE(6)	Height of one cycle on Y-axis for ratio plots. (inches)
NANE1(1) NANE2(1)	U-235 will be plotted vs. exposure from time step NANE1(1) to time step NANE2(1).
NANE1(2) NANE2(2)	U-238 will be plotted vs. exposure from time step NANE1(2) to time step NANE2(2).
NANE1(3) NANE2(3)	Pu-239 will be plotted vs. exposure from time step NANE1(3) to time step NANE2(3).
NLE1(I) NLE2(I)	Decay rates vs. exposure will be plotted for isotope I from time step NLE1(I) to time step NLE2(I). (NISO of these) If NLE1(I)=0 then the decay rates for this isotope will not be plotted vs. exposure.

NLT1(I) NLT2(I)	Decay rates vs. time will be plotted for isotope I from time step NLT1(I) to time step NLT2(I). (NISO of these) If NLT1(I)=0 then the decay rates for this isotope will not be plotted vs. time. If NLT1(I) >1, then time NLT1(I) will be subtracted from all TIME(I) values for purposes of plotting the decay rate data for the I'th isotope.
NRE1(I) NRE2(I)	Ratios vs. exposure will be plotted for ratio I from time step NRE1(I) to time step NRE2(I) (NRATIO of these). If NRE1(I)=0 then this ratio will not be plotted vs. exposure.
NRT1(I) NRT2(I)	Ratios vs. time will be plotted for ratio I from time step NRT1(I) to time step NRT2(I). (NRATIO of these) If NRT1(I)=0 then this ratio will not be plotted vs. time. If NRT1(I)>1, then time NRT1(I) will be subtracted from all time (I) values for purposes of plotting the ratio data for the I'th ratio.

The following input must be specified for each isotope.

NTYPE	Type of isotope. (see program description) 1=type A, 2=type B, 3=type C, 4=type D. If isotope(i) is a type "B" fission product then isotope (i-1) must be the corresponding type "A" fission product. Similarly if isotope (i) is a type "C" fission product then isotope (i-1) must be the corresponding type "D" fission product isotope.
ALMDA	Decay constant for type A isotope. (sec^{-1})
ALMDB	Decay constant for type B isotope. (sec^{-1})
ALMDC	Decay constant for type C isotope. (sec^{-1})
SIGAA(1)	Sigma A for type A isotope, (barns)
SIGAB(1)	Sigma A for type B isotope. (barns)
SIGAC(1)	Sigma A for type C isotope. (barns)
SIGAD(1)	Sigma A for type D isotope. (barns)
SIGCD(1)	Sigma C for type D isotope. (barns)

YELD25(1)	U-235 fission yield for this isotope. (atoms/fission)
YELD28(1)	U-238 fission yield for this isotope. (atoms/fission)
YELD49(1)	Pu-239 fission yield for this isotope. (atoms/fission)
TNAME(I)	6 Holerith name assigned to each isotope. (NISO of these).

\$ LAST CARD OF INPUT FOR EACH ISOTOPE

If ratios are desired then the last card(s) of the case is as follows--

(26I3)	NRAT(I,1) NRAT(I,2)	The code will calculate the ratios of the NRAT(I,1)'th isotope to the NRAT(I,2)'th isotope. (NRATIO of these).
--------	------------------------	--

IV. SAMPLE TEST CASE

Input data for a test case is shown in Table II. The data in this table are listed in the same order as indicated in the input instructions. Note that repeated entries may either be input individually or as a block by writing

$$\text{VARIABLE}(i) = N * \text{VALUE}$$

This will enter the numerical value VALUE into the i'th thru (i+N)'th storage locations set aside for the variable whose name is VARIABLE.

The printed output for each case begins with a listing of the input data. This is followed by a tabulation of total exposure time (seconds) at each of the input times. Then the absolute flux ($\text{n/cm}^2\text{-sec}$) and accumulated exposure (neutrons/kilobarn) are given at each input time for each input value of average flux. Fuel isotope (^{235}U , ^{238}U , and ^{239}Pu) concentrations (nuclei/barn-cm), the accumulated number of fissions in each fuel isotope and the total accumulated number of fissions (fission/cm^3), and the fraction of fissions in each fuel isotope are then tabulated at each input time and for each input value of the average flux.

Data for the first of the specified fission products is given next. The input data, specifying the type, cross sections, yields, and decay constant is printed first. This is followed by a tabulation of the decay rate ($\text{decays/cm}^3\text{-sec}$) at each input time for each input value of the average flux. The same information is printed for each of the other fission products in the order in which they were specified. After all of the decay rate information is printed the fission product activity ratios are given. Each ratio is tabulated at each input

time for each input value of the average flux and in the order in which the ratios were specified in the input.

Typical computer plots of the output are shown in Figures 1 thru 6. The various plotting options available are described in the input instructions.

TABLE IILIST OF INPUT CARDS FOR DRAFT TEST CASE

12 GENERAL DRAFT TEST CASE (12 ISOTOPES AND 12 RATIOS)
 \$DRAFT IVAR=2, NPHI=3, NTIME=99, NRATIO=12,
 FLXAV(1)=1.0+14, 3.0+14, 5.0+14,
 POWR(1)=20*362.9, 3*0.0, 13*481.2, 8*0.0, 15*536.1, 40*0.0,
 TIME(1)=0.0, 2.0+6, 4.0+6, 6.0+6, 8.0+6, 1.0+7, 1.2+7, 1.4+7,
 1.6+7, 1.8+7, 2.0+7, 2.2+7, 2.4+7, 2.6+7, 2.8+7, 3.0+7,
 3.2+7, 3.4+7, 3.6+7, 3.8+7, 4.0+7, 4.2163+7, 4.6+7,
 5.0+7, 5.3568+7, 5.6+7, 5.8+7, 6.0+7, 6.2+7, 6.4+7, 6.8+7,
 7.0+7, 7.2+7, 7.4+7, 7.6+7, 7.8+7, 8.0+7, 8.2944+7, 8.6+7,
 9.0+7, 9.4+7, 9.8+7, 1.02+8, 1.06+8, 1.10+8, 1.14826+8,
 1.16+8, 1.18+8, 1.20+8, 1.22+8, 1.24+8, 1.26+8, 1.28+8,
 1.30+8, 1.32+8, 1.34+8, 1.36+8, 1.38+8, 1.40+8, 1.42+8,
 1.4403+8, 1.446348+8, 1.452396+8, 1.458444+8, 1.464492+8,
 1.470540+8, 1.476588+8, 1.482636+8, 1.488684+8, 1.494732+8,
 1.500780+8, 1.506828+8, 1.512876+8, 1.518924+8, 1.524972+8,
 1.531020+8, 1.537068+8, 1.561260+8, 1.585452+8, 1.609644+8,
 1.633836+8, 1.658028+8, 1.682220+8, 1.706412+8, 1.730604+8,
 1.754796+8, 1.803180+8, 1.851564+8, 1.899948+8, 1.948332+8,
 1.996716+8, 2.069292+8, 2.383788+8, 2.698284+8, 3.012780+8,
 3.0370+8, 3.0663+8, 3.0776+8, 3.1156+8, 3.142368+8,
 ANO25(1)=7.59330-4, ANO28(1)=2.13020-2, ANO49(1)=0.0,
 SIGA25(1)=34.30, SIGA28(1)=0.73916, SIGA49(1)=63.04,
 SIGC28(1)=0.6637,
 SIGF25(1)=27.29, SIGF28(1)=0.07546, SIGF49(1)=61.94,
 IPLOT=3, RPLT=1, CYCLE(1)=12.0, 3.0, 12.0, 12.0, 12.0, 3.0,
 NANE1(1)=1,0,1, NANE2(1)=59,0,59,
 NLE1(1)=0,0,0,0,0,0,1,0,0,0,0,0, NLE2(1)=0,0,0,0,0,0,59,0,0,0,0,0,
 NLT1(1)=0,0,0,0,0,0,1,0,0,0,0,0, NLT2(1)=0,0,0,0,0,0,99,0,0,0,0,0,
 NRE1(1)=0,0,0,0,0,1,0,0,0,0,0,0, NRE2(1)=0,0,0,0,0,59,0,0,0,0,0,0,
 NRT1(1)=0,0,0,0,0,1,0,0,0,0,0,0, NRT2(1)=0,0,0,0,0,99,0,0,0,0,0,0,
 NTYPE=1, ALMDA=1.225-07, SIGAA(1)=0.0,
 YELD25(1)=0.0620, YELD28(1)=0.0580, YELD49(1)=0.0520,
 TNAME(1)=6HZR--95
 \$
 \$DRAFT NTYPE=2, ALMDB=2.292-07, SIGAB(1)=0.0,
 ALMDA=1.225-07, SIGAA(1)=0.0,
 YELD25(1)=0.0620, YELD28(1)=0.0580, YELD49(1)=0.0520,
 TNAME(2)=6HNB--95
 \$
 \$DRAFT NTYPE=1, ALMDA=2.031-07, SIGAA(1)=0.0,
 YELD25(1)=0.030, YELD28(1)=0.058, YELD49(1)=0.0560,
 TNAME(3)=6HRU-103
 \$
 \$DRAFT NTYPE=1, ALMDA=2.18-8, SIGAA(1)=0.0,
 YELD25(1)=0.00399, YELD28(1)=0.028, YELD49(1)=0.044,
 TNAME(4)=6HRU-106
 \$
 \$DRAFT NTYPE=4, ALMDC=1.074-8, SIGAC(1)=91.2, SIGAD(1)=19.20,
 SIGCD(1)=19.20,

TABLE II (cont'd)

YELD25(1)=0.0669, YELD28(1)=0.055, YELD49(1)=0.066,
TNAME(5)=6HCS-133

\$

\$DRAFT NTYPE=3, ALMDC=1.074-8, SIGAC(1)=91.2, SIGAD(1)=19.20,
SIGCD(1)=19.20,
YELD25(1)=0.0669, YELD28(1)=0.055, YELD49(1)=0.066,
TNAME(6)=6HCS-134

\$

\$DRAFT NTYPE=1, ALMDA=7.322-10, SIGAA=0.0,
YELD25(1)=0.062, YELD28(1)=0.063, YELD49(1)=0.0656,
TNAME(7)=6HCS-137

\$

\$DRAFT NTYPE=1, ALMDA=6.268-07, SIGAA(1)=0.0,
YELD25(1)=0.0630, YELD28(1)=0.0600, YELD49(1)=0.0533,
TNAME(8)=6HBA-140

\$

\$DRAFT NTYPE=2, ALMDB=4.787-06, SIGAB(1)=0.0,
ALMDA=6.268-07, SIGAA(1)=0.0,
YELD25(1)=0.0630, YELD28(1)=0.0600, YELD49(1)=0.0533,
TNAME(9)=6HLA-140

\$

\$DRAFT NTYPE=1, ALMDA=2.468-07, SIGAA(1)=0.0,
YELD25(1)=0.0610, YELD28(1)=0.0550, YELD49(1)=0.0575,
TNAME(10)=6HCE-141

\$

\$DRAFT NTYPE=1, ALMDA=2.825-8, SIGAA(1)=0.0,
YELD25(1)=0.054, YELD28(1)=0.043, YELD49(1)=0.0379,
TNAME(11)=6HCE-144

\$

\$DRAFT NTYPE=4, SIGAD(1)=1.150,
YELD25(1)=0.0167, YELD28(1)=0.0193, YELD49(1)=0.0168,
TNAME(12)=6HND-148

\$

1 2 1 7 2 7 3 7 4 7 6 7 8 7 9 7 10 7 11 7 8 9 10 11

U-235

FIGURE 1
RATIO OF FINAL TO INITIAL ATOMS OF ^{235}U VS. NEUTRON EXPOSURE

FIGURE 2

^{239}Pu CONTENT OF FUEL (ATOMS/BARN-CM) VS. NEUTRON EXPOSURE

EXPOSURE (NEUTRONS/KILOBARN)

FIGURE 2

^{239}Pu CONTENT OF FUEL (ATOMS/BARN-CM) VS. NEUTRON EXPOSURE

FIGURE 3

^{137}Cs DECAY RATE VS. TIME

FIGURE 4

^{137}Cs DECAY RATE VS. EXPOSURE

FIGURE 5
RATIO OF ^{134}Cs TO ^{137}Cs ACTIVITIES VS. TIME

GENERAL DRAFT TEST CASE (12 ISOTOPES AND 12 RATIOS)

FIGURE 6

RATIO OF ^{134}Cs TO ^{137}Cs ACTIVITIES VS. NEUTRON EXPOSURE

V. REFERENCES

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