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ISODIL
A COMPUTER CODE FOR PROCESSING
ISOTOPIC DILUTION MEASUREMENTS
FROM SPENT FUEL SAMPLES

R. P. Matsen

January 1971

AEC RESEARCH &
DEVELOPMENT REPORT

BNWL-1555

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UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

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ISODIL
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MEASUREMENTS FROM SPENT FUEL SAMPLES

By

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January 1971

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ISODIL
A COMPUTER CODE FOR PROCESSING ISOTOPIC DILUTION
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ABSTRACT

The ISODIL code processes data from isotopic dilution measurements from an irradiated fuel sample into information descriptive of the nuclear transformations that have occurred within the sample. The code analyzes data from either plutonium or uranium-plutonium fuel samples. In addition to printed output, other output features include plots of the measured data and punched cards for input to the DBUFIT least squares burnup code which is used to obtain cross section information from the data. Equations used in the code are described and use of the code is illustrated with several sample cases.

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I. INTRODUCTION

Program ISODIL processes data from isotopic dilution measurements from irradiated fuel samples into information descriptive of the nuclear transformations that have occurred within the samples. It was developed to partially satisfy requirements in the analysis of spent fuel from thermal reactors. Either plutonium or uranium (including uranium-plutonium) fuels may be analyzed by using different versions of the code. The normal output for each sample consists of a printed page listing the measured input, uranium and plutonium isotopic concentrations before and after irradiation, and the fissions that have occurred for each of the fissionable isotopes. Optional output features include plots of measured data from a series of samples (these plots are often useful for detecting anomalous data) and punched input cards for the DBUFIT⁽¹⁾ least squares analysis code. Other codes with some similarities to ISODIL are REBUF⁽²⁾ and BURNUP⁽³⁾.

Isotopic dilution is a technique whereby the amount of an element within a sample can be determined from mass spectrometric measurements made upon (a) the sample, (b) a spike, and (c) a mixture of the sample plus spike. The spike is a solution containing accurately known isotopic concentrations (in atoms/ml) of the element or elements under investigation in the sample solution. At Pacific Northwest Laboratory (PNL) the amounts of uranium, plutonium, and neodymium-148 in a sample are usually measured using slight variations of American Society for Testing and Material (ASTM) standard methods^(6,7). In many laboratories, the fission product, ^{148}Nd , has largely replaced ^{137}Cs as a burnup monitor

but the option of using results from ^{137}Cs measurements⁽⁸⁾ is still retained in the code. An alternate method of obtaining the amount of uranium in the fuel has also been incorporated in ISODIL. It consists of calculating the initial weight of the fuel in the sample based on the weight of the post irradiation spent fuel (mixed uranium and plutonium oxides plus fission products) which equals the initial oxide fuel weight by assuming that the fuel was a stoichiometric oxide.

The ISODIL code is written in FORTRAN V for use on the UNIVAC 1108 computer. It consists of two versions, a uranium-plutonium version of 667 cards and a plutonium version of 433 cards, and three subsidiary subroutines with a total of 156 cards. The time required to process the data from one sample is about three seconds on the UNIVAC 1108. The code is available from PNL by request to the author. Equations used to reduce the data, input instructions for the code, and several sample test cases are described in the following sections of this report. A list of the constants that are used in the code are given in Appendix A.

II. EQUATIONS USED IN THE ISODIL CODE

Many of the equations that are used to reduce the measured data from each sample are applicable to both versions (uranium-plutonium and plutonium fuels) of the code. For instance, subroutines ATMFRFC and NEODYM are used in both versions. The basic equations used in each of the two versions are presented in this section. Generally, these are discussed in the order that they appear in the code.

Many equations in the ISODIL code are concerned with the calculation of one standard deviation uncertainties or errors for each of the computed variables. The equations which are used to derived these uncertainties are based upon propagation-of-error techniques. Due to their length they have not been derived or recorded in this report. These equations are easily recognized since they include variables that begin with a V (for variance) or E (for error). For example, VND8 and END8 are respectively the fractional variance and the one standard deviation error of the variable ND8.

A. Equations for Uranium-Plutonium Fuel Samples

The ISODIL results for each uranium-plutonium fuel sample is normalized to one gram of the U+Pu initially in the sample, i.e., $U_o + Pu_o = 1$ gram, in the following equations. The initial plutonium

content by weight (ISODIL variable $PUBOL = Pu_O / (U_O + Pu_O)$) and the initial uranium plutonium atom ratios are part of the input data. The atom ratios are converted to isotopic atom fractions (ISODIL variables (AF) and weight fractions (WF)) using subroutine ATMFRFC. However, before the plutonium atom fractions and weight fractions are calculated, the $^{241}Pu/^{239}Pu$ atom ratio is corrected for ^{241}Pu decays that have occurred between the time that the ratio was measured and the time that irradiation of the sample was begun. The initial uranium isotopic content by weight is

$$U_O^i = (1 - PUBOL) (WF)_O^i \quad i = ^{234}U, ^{235}U, ^{236}U, ^{238}U$$

and the plutonium isotopic content by weight is

$$Pu_O^j = PUBOL (WF)_O^j \quad j = ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu$$

The number of atoms in each isotope per original gram of U+Pu is obtained by multiplying the appropriate U_O^i or Pu_O^j by Avogadro's number ($N_A = 6.0225 \times 10^{23}$) and dividing by the atomic weight of the isotope. Thus

$$UAT_O^i = U_O^i N_A / W^i \quad \text{and} \quad PuAT_O^j = Pu_O^j N_A / W^j$$

where the W^i and W^j are the appropriate atomic weights.

The mass spectrometric isotopic ratios from the spent fuel sample are also converted to atom fractions and weight fractions by using subroutine ATMFRFC. If the isotopic ratios are designated as X^i

($X^{234} = {}^{234}\text{U}/{}^{238}\text{U}$, $X^{235} = {}^{235}\text{U}/{}^{238}\text{U}$, $X^{236} = {}^{236}\text{U}/{}^{238}\text{U}$, and $X^{238} = {}^{238}\text{U}/{}^{238}\text{U} = 1$ for uranium and $X^{238} = {}^{238}\text{Pu}/{}^{239}\text{Pu}$, $X^{239} = {}^{239}\text{Pu}/{}^{239}\text{Pu} = 1$, $X^{240} = {}^{240}\text{Pu}/{}^{239}\text{Pu}$, $X^{241} = {}^{241}\text{Pu}/{}^{239}\text{Pu}$, and $X^{242} = {}^{242}\text{Pu}/{}^{239}\text{Pu}$ for plutonium), then the

$$\text{atom fractions} = \text{AF}^i = X^i / \sum X^i$$

$$\text{and weight fractions} = \text{WF}^i = W^i X^i / \sum (W^i X^i).$$

Before these equations are used to convert to the plutonium isotopic ratios however, the ${}^{241}\text{Pu}/{}^{239}\text{Pu}$ ratio is corrected for ${}^{241}\text{Pu}$ decays occurring since the sample was removed from the reactor. This corrected ratio is calculated according to the formula

$$({}^{241}\text{Pu}/{}^{239}\text{Pu}) \exp (TOUT \ln 2 / T_{1/2}^{241})$$

where TOUT is the time in years from the end of the sample's irradiation until the ${}^{241}\text{Pu}/{}^{239}\text{Pu}$ ratio was measured in the laboratory and $T_{1/2}^{241} = 14.5$ years is the ${}^{241}\text{Pu}$ half life. The atom fractions (multiplied by a factor of 100) are shown under the atom percent heading (A/0) in the printed output.

The amount of U+Pu initially in the fuel per milliliter of dissolved fuel solution can be determined by two methods. One method is to weigh the oxide fuel (it is assumed that the weight of the spent fuel, including fission products, is equal to the weight of the initial fuel), measure the volume of the dissolver solution, and assume that all of the fuel was initially stoichiometric oxide. In this case the amount of U+Pu originally in the sample per milliliter of dissolver solution is

$$\text{HEVMET} = (\text{U}_0 + \text{Pu}_0)/\text{ml} = \frac{(\text{Fuel weight/solution volume})}{1. + 32. \left[\sum_i (\text{U}_0^i/\text{W}^i) + \sum_i (\text{Pu}_0^i/\text{W}^i) \right]}$$

The other method of determining $(\text{U}_0 + \text{Pu}_0)/\text{ml}$ is to calculate the ^{238}U atoms per milliliter of dissolver solution from the results of uranium isotopic dilution measurements made on the sample. The method for converting ^{238}U atoms/ml into $(\text{U}_0 + \text{Pu}_0)/\text{ml}$ is described in a subsequent section.

1. Uranium and Plutonium Isotopic Dilution Equations

The isotopic dilution equation used to determine the ^{238}U and/or ^{239}Pu atoms per milliliter of dissolver solution is easily derived. First, define the following terms.

$r_i \equiv$ total atoms of isotope i in unspiked sample.

$s_i \equiv$ total atoms of isotope i in spike.

$\alpha \equiv$ index associated with the major isotope of the unspiked sample.

$\beta \equiv$ index associated with the major isotope of the spike.

$R_{i/j}$, $S_{i/j}$, $M_{i/j} \equiv$ ratio of isotope i to isotope j for the unspiked sample, spike, and sample plus spike respectively. These ratios are measured on a mass spectrometer.

By definition

$$M_{\alpha/\beta} = \frac{r_{\alpha} + s_{\alpha}}{r_{\beta} + s_{\beta}}$$

or

$$r_{\alpha} = s_{\beta} \frac{M_{\alpha/\beta} - S_{\alpha/\beta}}{1 - R_{\beta/\alpha} M_{\alpha/\beta}}$$

where use has been made of the definitions $S_{\alpha/\beta} = s_{\alpha}/s_{\beta}$ and

$R_{\beta/\alpha} = r_{\beta}/r_{\alpha}$. For uranium, $\alpha = {}^{238}\text{U}$, $\beta = {}^{233}\text{U}$ and $R_{3/8} = 0$ so that

$${}^{238}\text{U atoms/ml of sample} = ({}^{233}\text{U atoms/ml of spike}) \left(\frac{\text{ml of spike}}{\text{ml of sample}} \right) (M_{8/3} - S_{8/3})$$

where the subscript 8/3 is an abbreviation of 238/233.

For plutonium, $\alpha = {}^{239}\text{Pu}$ and $\beta = {}^{242}\text{Pu}$ so that

$${}^{239}\text{Pu atoms/ml of sample} = ({}^{242}\text{Pu atoms/ml of spike}) \left(\frac{\text{ml of spike}}{\text{ml of sample}} \right) \left(\frac{M_{9/2} - S_{9/2}}{1 - R_{2/9} M_{9/2}} \right)$$

The Pu/U ratio (by weight) of the irradiated sample is

$$\text{Pu/U} = \frac{({}^{239}\text{Pu atoms/ml of sample } W^{239}/(\text{WF})^{239})}{({}^{238}\text{U atoms/ml of sample } W^{238}/(\text{WF})^{238})}$$

2. Neodymium Isotopic Dilution Equations

The number of fissions that have occurred within a sample are determined by measuring one of the fission products. Either ${}^{148}\text{Nd}$ or ${}^{137}\text{Cs}$ can be used in the ISODIL code to compute total fissions. Various conversion constants are then used to convert the number of total fissions into burnup in units of MWD/MIM (megwatt days per metric ton of heavy metal).

The equations used to determine ${}^{148}\text{Nd}$ from the neodymium isotopic dilution measurements are quite different from those used for ${}^{238}\text{U}$ and ${}^{239}\text{Pu}$. The primary difference is that the shielded

isotope, ^{142}Nd (a shielded isotope is an isotope that is not produced either directly during the fission process or by decay afterwards) is used to correct for any contamination of the spiked or unspiked samples by natural neodymium. Derivation of the neodymium isotopic dilution equations employs the following notation.

Subscripts: 0 $\equiv$ ^{150}Nd , 2 $\equiv$ ^{142}Nd , 8 $\equiv$ ^{148}Nd

f_i $\equiv$ Nd atoms of the i^{th} isotope in the sample because of fissions

s_i $\equiv$ Nd atoms of the i^{th} isotope in the spike

c_i $\equiv$ Nd atoms of the i^{th} isotope in the sample plus spike solution because of the contamination by natural neodymium

c'_i $\equiv$ Nd atoms of the i^{th} isotope in the sample solution because of contamination by natural neodymium

$$C_{2/0} = c_2/c_0 = c'_2/c'_0 = 4.825 \text{ and } C_{8/0} = c_8/c_0 = c'_8/c'_0 = 1.020$$

$$S_{2/0} = s_2/s_0 \text{ and } S_{8/0} = s_8/s_0$$

$R_{0/2}$ and $R_{0/8}$ are the mass spectrometer ratios measured for the unspiked sample

$M_{2/0}$ and $M_{8/0}$ are the mass spectrometer ratios measured for the sample + spike (i.e., mixture) solution

$$E_{0/8} \equiv f_0/f_8$$

By definition

$$R_{0/8} = \frac{f_0 + c'_0}{f_8 + c'_8}$$

and since ^{142}Nd is a shielded isotope

$$R_{0/2} = \frac{f_0 + c'_0}{c'_2} .$$

Solving these two equations yields

$$E_{0/8} = \frac{f_0}{f_8} = \frac{(C_{2/0} R_{0/2} - 1) R_{0/8}}{C_{2/0} R_{0/2} - C_{8/0} R_{0/8}}$$

Furthermore, the ^{148}Nd to ^{150}Nd ratio for the sample plus spike is

$$M_{8/0} = \frac{s_8 + f_8 + c_8}{s_0 + f_0 + c_0}$$

and again, since ^{142}Nd is a shielded isotope

$$M_{2/0} = \frac{s_2 + c_2}{s_0 + f_0 + c_0}$$

Solving these two equations for the ratio of ^{148}Nd in the sample to ^{150}Nd in the spike yields

$$F_8 = f_8/s_0 = \frac{(C_{2/0} - S_{2/0})M_{8/0} - (C_{8/0} - S_{8/0})M_{2/0} - (C_{2/0} S_{8/0} - C_{8/0} S_{2/0})}{C_{2/0} (1 - E_{0/8} M_{8/0}) - (1 - E_{0/8} C_{8/0}) M_{2/0}}$$

Therefore,

$$\begin{aligned} ^{148}\text{Nd atoms from fission/ml of sample} &= (^{150}\text{Nd atoms/ml of spike}) \\ &\times \left(\frac{\text{ml of spike}}{\text{ml of sample}} \right) F_8 \end{aligned}$$

The contamination of the unspiked sample

$$\frac{c'_8}{f_8} = \frac{C_{8/0} R_{0/8}}{C_{2/0} R_{0/2} - C_{8/0} R_{0/8}}$$

and the contamination of the sample plus spike solution

$$\frac{C_8}{F_8} = \frac{S_{2/0} \left((1-E_{0/8}) M_{8/0} \right) - \left((1-E_{0/8}) S_{8/0} \right) M_{2/0}}{\left((1-E_{0/8}) C_{8/0} \right) M_{2/0} - C_{2/0} \left((1-E_{0/8}) M_{8/0} \right)} \left(\frac{C_{8/0}}{F_8} \right)$$

by natural neodymium are printed in the output as "ND CONTAMINATION".

All of these neodymium calculations are performed in subroutine NBODYM. No correction for the burnout of ^{148}Nd is applied to the data. This correction is less than one percent for burnups below 20,000 MWD/MTM but amounts to several percent for burnups in the 30,000 to 50,000 MWD/MTM range. Also, no correction is made for the formation of ^{142}Nd by the decay of ^{142}Pr formed by way of an (n, γ) reaction on the fission product ^{141}Pr . The code assumes that all ^{142}Nd atoms in the sample are there because of contamination by natural neodymium.

3. Equations for ^{137}Cs

When ^{137}Cs is the fission indicator, its measured value (reported in units of ^{137}Cs atoms per milliliter of dissolver solution) is corrected for decays that have occurred since the sample was removed from the reactor by multiplying by the term

$$\exp \left(\frac{\text{TOUTCS} \ln 2}{T_{1/2}^{137}} \right)$$

where $T_{1/2}^{137} = 30.0$ years is the ^{137}Cs half life and TOUTCS is the time in years since the sample was removed from the reactor until it was analyzed for ^{137}Cs . Another correction for ^{137}Cs decays while

the sample was in the reactor is also made. This is accomplished by multiplying the ^{137}Cs measurement by the input variable COR1. If the rate of production of ^{137}Cs during irradiation is constant then

$$\text{COR1} = \frac{T \ln 2 / T_{1/2}^{137}}{1 - \exp \left(T \ln 2 / T_{1/2}^{137} \right)}$$

where T is the time that the sample was in the reactor.

4. Burnup Calculations

The next part of the code consists of equations for the number of fissions of each of the fissionable isotopes and also for the depletion of uranium. These two quantities are considered together since they are partially interrelated. An added complication occurs because the uranium depletion, ΔU , is required in some of the equations that are used to calculate ΔU itself. Therefore, this part of the code is enclosed in a loop which is terminated when the change in ΔU from one iteration to the next is suitably small. Because ΔU is unknown at the beginning of the first iteration, it is initially set equal to zero. The calculations are performed in units of atoms per original gram of uranium plus plutonium since the atom is the incremental unit of the transmutation process.

The ^{235}U , ^{238}U , and ^{241}Pu fissions are each calculated separately and are then subtracted from the total fissions in order to determine the ^{239}Pu fissions that have occurred. The ^{235}U fissions are computed from the measured change in ^{235}U and ^{236}U isotopic concentrations. The ^{241}Pu and ^{238}U fissions (which are a small fraction of the total

fissions) are estimated by applying appropriate nuclear parameters to the data. The details of these calculations are presented in this section.

The number of ^{235}U fissions is calculated as the loss of ^{235}U minus the gain in ^{236}U including losses through ^{236}U neutron captures.

$$^{235}\text{U}_{\text{fis}} = \Delta^{235}\text{U} - \Delta^{236}\text{U} - \text{CAP236}$$

with

$$\Delta^{235}\text{U} = N_A (\text{WF})^{235} (1-\text{PUBOL}) U/W^{235} - \text{UAT}_O^{235}$$

$$\Delta^{236}\text{U} = N_A (\text{WF})^{236} (1-\text{PUBOL}) U/W^{236} - \text{UAT}_O^{236}$$

where

$$U = (1-\text{PUBOL} + \Delta U)/(1-\text{PUBOL})$$

is the ratio of end-of-life to beginning-of-life uranium. The variable CAP236 represents the loss of ^{236}U by neutron capture. It is determined from the equation

$$\text{CAP236} = \frac{\sigma_a^{236}}{\sigma_a^{235}} \left[\frac{\Delta^{236}\text{U}}{\Delta^{235}\text{U}} \left(N_O^{235} - N^{235} + \ln \frac{N^{235}}{N_O^{235}} \right) - N_O^{236} \ln \frac{N^{235}}{N_O^{235}} \right]$$

where $N^{235} = N_A (\text{WF})^{235} (1-\text{PUBOL}) U/W^{235}$, $N_O^{235} = \text{UAT}_O^{235}$, $N_O^{236} = \text{UAT}_O^{236}$,

and the ratio of the flux averaged absorption cross sections,

$\sigma_a^{236}/\sigma_a^{235}$, is part of the input (ISODIL variable CON236). This equation can be derived by solving equations of the $^{235}\text{U} \rightarrow ^{236}\text{U} \rightarrow ^{237}\text{Np}$ transmutation chain.

These equations are

$$\begin{aligned} dN^{235} &= -\sigma_a^{235} \phi N^{235} dt \\ dN^{236} &= (\sigma_c^{235} \phi N^{235} - \sigma_a^{236} \phi N^{236}) dt \approx \sigma_c^{235} \phi N^{235} dt \\ dN^{237} &= (\sigma_a^{236} \phi N^{236} - \sigma_a^{237} \phi N^{237}) dt \approx \sigma_a^{236} \phi N^{236} dt \end{aligned}$$

where ϕ is the neutron flux and t is the time. Dividing the last two equations by the first yields

$$\begin{aligned} \frac{dN^{236}}{dN^{235}} &\approx - \frac{\sigma_c^{235}}{\sigma_a^{235}} \\ \frac{dN^{237}}{dN^{235}} &\approx - \frac{\sigma_a^{236}}{\sigma_a^{235}} \frac{N^{236}}{N^{235}} \end{aligned}$$

The solution to these equations is

$$N^{236} = N_o^{236} + \frac{\sigma_c^{235}}{\sigma_a^{235}} (N_o^{235} - N^{235}) \text{ or } \frac{\sigma_c^{235}}{\sigma_a^{235}} = \frac{N^{236} - N_o^{236}}{N_o^{235} - N^{235}} = - \frac{\Delta^{236}U}{\Delta^{235}U}$$

and

$$\begin{aligned} CAP^{236} &= \int dN^{237} = - \frac{\sigma_a^{236}}{\sigma_a^{235}} \int_{N_o^{235}}^{N^{235}} \frac{N^{236}}{N^{235}} dN^{235} \\ &= \frac{\sigma_a^{236}}{\sigma_a^{235}} \left[\frac{\Delta^{236}U}{\Delta^{235}U} \left(N_o^{235} - N^{235} + \ln \frac{N^{235}}{N_o^{235}} \right) - N_o^{236} \ln \frac{N^{235}}{N_o^{235}} \right] \end{aligned}$$

An estimate the number of ^{241}Pu fissions that have occurred is made as follows.

$$^{241}\text{Pu}_{\text{fis}} = ^{241}\text{Pu}_{\text{cap}} / \alpha^{241} = (\Delta^{242}\text{Pu} + \text{CAP242}) / \alpha^{241}$$

This relatively small fraction of the total fissions is based on the definition of the capture to fission ratio, $\alpha^{241} = 0.365$, for ^{241}Pu . The term CAP242 is the sum of the ^{243}Am and ^{242}Cm contents of the sample and

$$\Delta^{242}\text{Pu} = (^{239}\text{Pu atoms/ml of solution}) \times ^{242}/\text{HEVMET} - \text{PuAT}_0^{242}.$$

If HEVMET is determined from the sample's weight and solution volume, the formula explained earlier is used. If, however, HEVMET is determined from uranium isotopic dilution measurements, the following formula is used.

$$\text{HEVMET} = (^{238}\text{U atoms/ml of solution}) W^{238} / [N_A(\text{WF})^{238} (1 - \text{PUBOL}) \text{U}]$$

where U is the ratio of end-of-life to beginning-of-life uranium.

With the aid of these preparatory calculations, the number of ^{239}Pu fission can be determined from the equation

$$\text{Total fission product atoms} = ^{239}\text{Pu}_{\text{fis}} Y^{239} + ^{235}\text{U}_{\text{fis}} Y^{235} + ^{241}\text{Pu}_{\text{fis}} Y^{241} + ^{238}\text{U}_{\text{fis}} Y^{238}$$

where the Y^i are the fission yields of the fission product (either ^{148}Nd or ^{137}Cs) that was measured and the number of ^{238}U fissions is

determined from the equation

$$^{238}\text{U}_{\text{fis}} = \text{DEL39 } ^{239}\text{Pu}_{\text{fis}} + \text{DEL35 } ^{235}\text{U}_{\text{fis}} + \text{DEL41 } ^{241}\text{Pu}_{\text{fis}}$$

The parameters DEL39, DEL35, and DEL41 are the number of fissions induced in ^{238}U by the fast neutrons from a nearby fission in ^{239}Pu , ^{235}U , or ^{241}Pu respectively. The values of DEL39, DEL35, and DEL41 that are in the code are those that are applicable to samples cut from PRTR (Plutonium Recycle Test Reactor) fuel elements. If burnup data from other reactors are to be processed through the ISODIL code, these numbers which are embedded in the code should be changed to the values appropriate for that core.

In order to calculate ΔU , the total weight change in all the uranium, the following computations are performed. The number of end-of-life Pu atoms is

$$\text{ATPU} = (^{239}\text{Pu atoms/ml of solution}) / ((\text{AF})^{239} \text{HEVMET})$$

and therefore the plutonium generated during the irradiation is

$$\text{PUGEN} = (\text{ATPU} - \sum_j \text{PuAT}_O^j) + ^{239}\text{Pu}_{\text{fis}} + ^{241}\text{Pu}_{\text{fis}} + \text{CAP242}.$$

The change in ^{238}U is

$$\Delta^{238}\text{U} = -\text{PUGEN} - ^{238}\text{U}_{\text{fis}}$$

and the total change in all uranium atoms is

$$\Delta\text{UAT} = \Delta^{238}\text{U} + \Delta^{235}\text{U} + \Delta^{236}\text{U} - \text{CAP236}.$$

When converted to grams this becomes

$$\Delta\text{U} = [\Delta^{238}\text{U} W^{238} + \Delta^{235}\text{U} W^{235} + (\Delta^{236}\text{U} + \text{CAP236}) W^{236}] / N_A$$

and the ratio of end-of-life uranium to beginning-of-life uranium is

$$\text{U} = \frac{1 - \text{PUBOL} + \Delta\text{U}}{1 - \text{PUBOL}}$$

With this new value of U the iterative loop containing the fission and ΔU equations is re-entered. The loop is continued until ΔU changes by less than one part in 10^6 from one iteration to the next. Upon exiting the iterative loop, the many quantities that were calculated in atoms/g ($U_o + Pu_o$) are converted to grams/g ($U_o + Pu_o$) before being printed as part of the output.

In addition, the fissions of the several fissioning isotopes are converted to burnup in units of MWD/MIM as follows:

$$\begin{aligned} (BU)^{235} &= (^{235}U_{fis} \text{ in g/g } (U_o + Pu_o)) (N_A) (\text{MEV/FIS})^{235} \\ &= (1.602 \times 10^6 \text{ erg/MEV}) (10^{-17} / 8.64 \text{ MWD/erg}) (10^6 \text{ g } (U_o + Pu_o)) / \text{MIM/W}^{235} \\ &= (1.117 \times 10^6) (\text{MEV/FIS})^{235} (^{235}U_{fis} \text{ in g/g } (U_o + Pu_o)) / W^{235} \end{aligned}$$

and similarly for ^{239}Pu , ^{241}Pu , and ^{238}U . These burnup values are also part of the printed output.

Several other items are also calculated. These are the capture to fission ratio or alpha value for ^{235}U and ^{239}Pu . For ^{235}U

$$\alpha^{235} = (\Delta^{236}U + \text{CAP236}) / ^{235}U_{fis} = (\Delta^{236}U + \text{CAP236}) / (-\Delta^{235}U - \Delta^{236}U - \text{CAP236})$$

For ^{239}Pu

$$\alpha^{239} = ^{239}Pu_{cap} / ^{239}Pu_{fis}$$

where the number of (n, γ) reactions in ^{239}Pu is

$$\begin{aligned} ^{239}Pu_{cap} &= \Delta^{240}Pu + \Delta^{241}Pu + \Delta^{242}Pu + ^{241}Pu_{fis} = \Delta^{240}Pu + \Delta^{241}Pu + \\ &\quad (1 + 1/\alpha^{241}) \Delta^{242}Pu \end{aligned}$$

5. Calculations for Plotted and Punched Data

Provision has been made for plotting some of the more important measured and calculated variables. The measured variables are the uranium and plutonium atom ratios. Calculated variables are the Pu/U ratio and burnup in GWD/MIM (one GWD/MIM = 1000 MWD/MIM). Plots of these variables from a number of initially identical samples with different exposures are often used to check the validity of the data. A poorly measured data point will fall off of the line described by the other samples. Each plot variable is assigned an index so that the variables may later be plotted in any combination desired. The indices assigned to the uranium-plutonium data are shown in Table I.

Table I
Indices Assigned to the Uranium-Plutonium Plot Data

<u>Index</u>	<u>Data</u>
1	$^{235}\text{U}/^{238}\text{U}$ atom ratios
2	$^{236}\text{U}/^{238}\text{U}$ atom ratios
3	$^{240}\text{Pu}/^{239}\text{Pu}$ atom ratios
4	$^{241}\text{Pu}/^{239}\text{Pu}$ atom ratios (corrected for ^{241}Pu decay)
5	$^{242}\text{Pu}/^{239}\text{Pu}$ atom ratios
6	$^{238}\text{Pu}/^{239}\text{Pu}$ atom ratios
7	Pu/U ratios (by weight)
8	Burnup (GWD/MIM)

A final set of computations is done if the data is to be punched on cards as input to the DBUFIT-I⁽¹⁾ least squares fitting program. The DBUFIT-I code is used to extract integral cross section information from isotopic burnup data by fitting isotopic transmutation equations that describe the burnup process to the isotopic data. The cross section information obtained with the aid of this least squares code is useful for verifying analytical methods commonly used in the design of nuclear reactors.

The uranium and plutonium isotopic contents as well as the ^{239}Pu fissions and burnup are punched in a format acceptable to the DBUFIT-I code. The isotopic data is in units of either 10^{20} atoms/g (U + Pu) or 10^{-4} atoms/b-cm depending upon the value that was entered for the input variable, DEN. If $\text{DEN} \leq 0$ then the data is in units of 10^{20} atoms/g (U + Pu). If the density (in g/cc) of the U + Pu initially in the fuel is entered into the program via the variable DEN, then the data prepared for DBUFIT is in units of 10^{-4} atoms/b-c. The elements of the variance-covariance matrix required by the DBUFIT program are also prepared at the same time. The americium and curium data can be obtained if desired.

B. Equations for Plutonium Fueled Samples

Many of the equations used for the analysis of plutonium samples are the same as those discussed above for uranium-plutonium samples. All of the isotopic dilution and atom percent equations are the same except, of course, the uranium equations are omitted. Other parts of the analysis are significantly different, however.

The printed output is normalized to one beginning-of-life atom of plutonium instead of one gram of plutonium. With this normalization, the atom fractions of the pre-irradiated fuel are equivalent to the beginning-of-life isotopic concentrations. The plutonium fissions are also determined differently than for uranium-plutonium samples. The equations that are used are:

$$\begin{aligned}
 \text{Pu atoms/ml} &= (^{239}\text{Pu atoms/ml})/(\text{AF})^{239} \\
 \text{Pu}_0 \text{ atoms/ml} &= \text{Pu atoms/ml} + \text{Pu}_{\text{fis}} \text{ atoms/ml} \\
 \text{Pu}_{\text{fis}} \text{ atoms/ml} &= ^{239}\text{Pu}_{\text{fis}} \text{ atoms/ml} + ^{241}\text{Pu atoms/ml} \\
 &= \frac{\text{Fission product atoms/ml} - Y^{241} \text{ } ^{241}\text{Pu}_{\text{fis}} \text{ atoms/ml}}{Y^{239}} \\
 &\quad + ^{241}\text{Pu}_{\text{fis}} \text{ atoms/ml}
 \end{aligned}$$

where the fission product may be either ^{148}Nd or ^{137}Cs , and

$$\begin{aligned}
 ^{241}\text{Pu}_{\text{fis}} \text{ atoms/ml} &= [(\text{AF})^{242} \text{ Pu atoms/ml} - (\text{AF})_0^{242} \text{ Pu}_0 \text{ atoms/ml} \\
 &\quad + ^{243}\text{Am atoms/ml} + ^{244}\text{Cm atoms/ml}]/\alpha^{241}
 \end{aligned}$$

Therefore the beginning-of-life plutonium atoms per milliliter of dissolver solution are

$$Pu_0 \text{ atoms/ml} = \frac{Pu \text{ atoms/ml} + \frac{\text{Fission product atoms/ml}}{Y^{239}} + \frac{(1-Y^{241}/Y^{239})}{\alpha^{241}} Q}{1 + (AF)_O^{242} \frac{1-Y^{241}/Y^{239}}{\alpha^{241}}}$$

where $Q = Pu \text{ atoms/ml } (AF)^{242} + {}^{234}\text{Am atoms/ml} + {}^{244}\text{Cm atoms/ml}$.

These equations are converted from atoms/ml to atoms per original atom of plutonium by dividing by the expression for Pu_0 atoms/ml. Beginning with the definition for the initial amount of plutonium, the results of this conversion are shown below. These quantities are part of the printed output for each sample.

$$Pu_0 = 1$$

$$j_{Pu_0} = (AF)_O^j Pu_0 = (AF)_O^j \quad j = 238, 239, 240, 241, 242$$

$$Pu = \frac{Pu \text{ atoms/ml}}{Pu_0 \text{ atoms/ml}}$$

$$j_{Pu} = (AF)^j Pu \quad j = 238, 239, 240, 241, 242$$

$${}^{239}\text{Pu}_{\text{fis}} = \frac{\text{Fission product atoms/ml} - Y^{241}({}^{241}\text{Pu}_{\text{fis}} \text{ atoms/ml})}{Y^{239} (Pu_0 \text{ atoms/ml})}$$

$${}^{241}\text{Pu}_{\text{fis}} = \frac{{}^{241}\text{Pu}_{\text{fis}} \text{ atoms/ml}}{Pu_0 \text{ atoms/ml}}$$

$$\beta = 1 - Pu = {}^{239}\text{Pu}_{\text{fis}} + {}^{241}\text{Pu}_{\text{fis}}$$

where β , the fraction of plutonium atoms that have fissioned, is used in place of MWD/MIM as a measure of burnup. Finally, the capture to fission ratio for ^{239}Pu is

$$\alpha^{239} = [(AF)_0^{239} - (AF)^{239} \text{ Pu}] / ^{239}\text{Pu}_{\text{fis}} - 1$$

The indices assigned to the plutonium data for plotting purposes are shown in Table II.

Table II
Indices Assigned to the Plutonium Plot Data

<u>Index</u>	<u>Data</u>
1	$^{240}\text{Pu}/^{239}\text{Pu}$ atoms ratios
2	$^{241}\text{Pu}/^{239}\text{Pu}$ atom ratios (corrected for ^{241}Pu decay)
3	$^{242}\text{Pu}/^{239}\text{Pu}$ atom ratios
4	$^{238}\text{Pu}/^{239}\text{Pu}$ atom ratios
5	Burnup (fraction of Pu atoms that have fissioned)

The data can be prepared for the DEUFIT code in the same manner as the uranium-plutonium data. If the input variable DEN = 0, then the data will be in units of atoms per initial atom of plutonium. If however, DEN equals the number of beginning-of-life atoms per barn-centimeter, then the data will be in units of atoms per barn-centimeter.

III. INPUT INSTRUCTIONS FOR ISODIL

The input for ISODIL falls into three categories. The first category includes input that is the same for a number of samples and is therefore entered just once rather than repeating it for each sample. Included are program options, data on the initial composition of the fuel, and data about the spike solutions that are used in the isotopic dilution analysis. The second input category contains the measured data for each individual sample. The category 2 data for samples having the same category 1 information are stacked sequentially. The third input category contains plot information when the option to plot is exercised. The data to be entered on each of the input cards along with their format is listed in the following sections.

A. Input for Uranium Plutonium Samples

1. Data for a Set of Samples

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	IOP, INPUN, IPLOT, PUBOL, EPUBOL, TIN, CON236, DEN	3I2,4X,6E10.5
	IOP = 0	Neodymium analysis performed.
	IOP = 1	Neodymium analysis not performed.
	IPUN = 0	Bypass calculations for DBUFIT punched cards, punch no cards.
	IPUN = 1	Perform calculations for DBUFIT punched cards, punch no cards.
	IPUN = 2	Perform calculations for DBUFIT punched cards, punch cards for accumulated samples. Omit americium and curium punched cards.
	IPUN = 3	Same as IPUN = 2 except include americium and curium punched cards.
	IPLOT = 0	Bypass accumulation of data for plots, make no plots.
	IPLOT = 1	Accumulate data for plots, make no plots.
	IPLOT = 2	Accumulate data for plots, plot accumulated data.
	PUPOL =	Beginning of life Pu/(U+Pu) by weight.
	EPUBOL =	Uncertainty of PUBOL
	TIN =	Time (in years) from loading of fuel until measurement of initial Pu atom ratios (i.e. TIN is negative if loading occurs after the atom ratio measurement).

$$\text{CON236} = \sigma_a^{236} / \sigma_a^{235}.$$

If DEN = Density (in g/cc) of U+Pu initially in fuel then data for DBUFIT is in units of 10^{-4} atoms/b-cm.

If DEN = 0 then data for DBUFIT is in units of 10^{20} atoms/g (U+Pu).

<u>Card</u>	<u>Variables</u>	<u>Format</u>
2	BUX(1), EBUX(1), BUX(2), EBUX(2), BUX(3), EBUX(3) BUX(1) = $^{234}\text{U}/^{238}\text{U}$ atom ratio for initial fuel. EBUX(1) = uncertainty of BUX(1). BUX(2) = $^{235}\text{U}/^{238}\text{U}$ atom ratio for initial fuel. EBUX(2) = uncertainty of BUX(2). BUX(3) = $^{236}\text{U}/^{238}\text{U}$ atom ratio for initial fuel. EBUX(3) = uncertainty of BUX(3).	8E10.5
3	BPX(1), EPBX(1), BPX(2), EPBX(2), BPX(3), EPBX(3), BPX(4), EPBX(4) BPX(1) = $^{240}\text{Pu}/^{239}\text{Pu}$ atom ratio for initial fuel. EPBX(1) = uncertainty of BPX(1). BPX(2) = $^{241}\text{Pu}/^{239}\text{Pu}$ atom ratio at the time the initial fuel was measured. EPBX(2) = uncertainty in BPX(2). BPX(3) = $^{242}\text{Pu}/^{239}\text{Pu}$ atom ratio for initial fuel. EPBX(3) = uncertainty in BPX(3). BPX(4) = $^{238}\text{Pu}/^{239}\text{Pu}$ atom ratio for initial fuel. EPBX(4) = uncertainty in BPX(4).	8E10.5

<u>Card</u>	<u>Variables</u>	<u>Format</u>
4	ATU3, EATU3, SU83, ESU83, ATP2, EATP2, SP92, ESP92 ATU3 = atoms of ^{233}U /ml in the uranium spike material. EATU3 = uncertainty in ATU3. SU83 = $^{238}\text{U}/^{238}\text{U}$ atom ratio in the uranium spike material. ESU83 = uncertainty in SU83. ATP2 = atoms of ^{242}Pu /ml in the plutonium spike material. EATP2 = uncertainty in ATP2. SP92 = $^{239}\text{Pu}/^{242}\text{Pu}$ atom ratio in the plutonium spike material. ESP92 = uncertainty in SP92.	8E10.5
5	ATN50, EATN50, S20, ES20, S80, ES80 Note: This card is omitted if IOP = 1. ATN50 = atoms of ^{150}Nd /ml in the neodymium spike material. EATN50 = uncertainty in ATN50. S20 = $^{142}\text{Nd}/^{150}\text{Nd}$ atom ratio in the neodymium spike material. ES20 = uncertainty in S20. S80 = $^{148}\text{Nd}/^{150}\text{Nd}$ atom ratio in the neodymium spike material. ES80 = uncertainty in S80.	8E10.5

2. Data for Each Sample

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	IROD, NDAKTR,MAC,NORM,ISM,TOUTPU,FUELML,CS,ECS,COR1,TOUTCS	5I2,6E10.5
	<p>IROD = Ring of 19 rod PRTR cluster from which sample was cut.</p> <p>IROD = -1 Center rod of cluster.</p> <p>IROD = 0 Middle (6 rod) ring of cluster.</p> <p>IROD = 1 Outer (12 rod) ring of cluster.</p> <p>NDAKTR = 0 Fission product = ^{148}Nd.</p> <p>NDAKTR = 1 Fission product = ^{137}Cs.</p> <p>MAC = 0 Omit ^{243}Am and ^{244}Cm input data, skip ^{243}Am and ^{244}Cm analysis.</p> <p>MAC = 1 Include ^{243}Am and ^{244}Cm input data, include ^{243}Am and ^{244}Cm analysis.</p> <p>NORM = 0 Sample results normalized to isotopic dilution results</p> <p>NORM = 1 Sample results normalized to fuel weight per unit volume of dissolver solution.</p> <p>ISM = Reference number of symbol to be plotted for this sample (see CALCOMP manual for definition of symbols).</p> <p>TOUTPU = Time (in years) from the end of fuel irradiation until the plutonium atom ratios are measured.</p> <p>FUELML = Weight of the dissolved oxide fuel per milliliter of dissolver solution -- this variable is required if NORM = 1.</p>	

<u>Card 1 (continued)</u>	<u>Variables</u>	<u>Format</u>
	CS = ^{137}Cs atoms/ml of dissolver solution.	
	ECS = uncertainty in CS.	
	COR1 = Correction for incore decay of ^{137}Cs (COR1 = 1.0 if no correction is made).	
	TOUTCS = Time (in years) from the end of fuel irradiation until ^{137}Cs is measured. These last four variables must be included if NDAKTR = 1.	
2 & 3 G(I), I = 1,32		16A5
	An alphanumeric title heading for the case is written in the first 132 columns of these two cards.	
4 UX(1),EUX(1),UX(2),EUX(2),UX(3),EUX(3)		8E10.5
	UX(1) = $^{234}\text{U}/^{238}\text{U}$ atom ratio for irradiated fuel sample. EUX(1) = uncertainty in UX(1). UX(2) = $^{235}\text{U}/^{238}\text{U}$ atom ratio for irradiated fuel sample. EUX(2) = uncertainty in UX(2). UX(3) = $^{236}\text{U}/^{238}\text{U}$ atom ratio for irradiated fuel sample EUX(3) = uncertainty in UX(3).	
5 SPVLU, BSPVLU, SMVLU, ESMVLU, M83, EM83		8E10.5
	Note: This card may be left blank if NORM = 1. SPVLU = volume (in milliliters) of the spike solution in the uranium sample plus spike solution. ESPVLU = uncertainty in SPVLU.	

<u>Card 5 (continued)</u>	<u>Variables</u>	<u>Format</u>
	SMVLU = volume (in milliliters) of the sample solution in the uranium sample plus spike solution.	
	ESMVLU = uncertainty in SMVLU.	
	M83 = $^{238}\text{U}/^{233}\text{U}$ atom ratio on the uranium sample plus spike solution.	
	EM83 = uncertainty in M83.	
6	PX(1),EPX(1),PX(2),EPX(2),PX(3),EPX(3),PX(4),EPX(4) PX(1) = $^{240}\text{Pu}/^{239}\text{Pu}$ atom ratio for the irradiated fuel sample. EPX(1) = uncertainty PX(1). PX(2) = $^{241}\text{Pu}/^{239}\text{Pu}$ atom ratio at the time the atom ratios of the irradiated fuel sample were measured. EPX(2) = uncertainty in PX(2). PX(3) = $^{242}\text{Pu}/^{239}\text{Pu}$ atom ratio for the irradiated fuel sample. EPX(3) = uncertainty in PX(3). PX(4) = $^{238}\text{Pu}/^{239}\text{Pu}$ atom ratio for the irradiated fuel sample. EPX(4) = uncertainty in PX(4).	8E10.5
7	SPVLP,ESPVLP,SMVLP,ESMVLP,M92,EM92 SPVLP = volume (in milliliters) of the spike solution in the plutonium sample plus spike solution. ESPVLP = uncertainty in SPVLP.	8E10.5

<u>Card 7 (continued)</u>	<u>Variables</u>	<u>Format</u>
	SMVLP = volume (in milliliters) of the sample solution in the plutonium sample plus spike solution. ESMVLP = uncertainty in SMVLP. M92 = $^{239}\text{Pu}/^{242}\text{Pu}$ atom ratio in the plutonium sample plus spike solution. EM92 = uncertainty in M92.	
8	SPVLN,ESPVLN,SMVLN,ESMVLN,M80,EM80 Note: This card is omitted if NDAKTR = 1. SPVLN = volume (in milliliters) of the spike solution in the neodymium sample plus spike solution. ESPVLN = uncertainty in SPVLN. SMVLN = volume (in milliliters) of the sample solution in the neodymium sample plus spike solution. ESMVLN = uncertainty in SMVLN. M80 = $^{148}\text{Nd}/^{150}\text{Nd}$ atom ratio in neodymium sample plus spike solution. EM80 = uncertainty in M80.	8E10.5
9	M20,EM20,R02,ER02,R08,ER08 Note: This card is omitted if NDAKTR = 1. M20 = $^{142}\text{Nd}/^{148}\text{Nd}$ atom ratio in the neodymium sample plus spike solution. EM20 = uncertainty in M20. R02 = $^{150}\text{Nd}/^{142}\text{Nd}$ atom ratio in the irradiated fuel sample. ER02 = uncertainty in R02.	8E10.5

<u>Card 9 (continued)</u>	<u>Variables</u>	<u>Format</u>
	R08 = $^{150}\text{Nd}/^{148}\text{Nd}$ atom ratio in the irradiated fuel sample.	
	ER08 = uncertainty in R08.	
10	AM243,EAM243,CM244,ECM244	8E10.5
	Note: This card is omitted if MAC = 0.	
	AM243 = amount of ^{243}Am (in g/ml) in the irradiated fuel sample.	
	EAM243 = uncertainty in AM243.	
	CM244 = amount of ^{244}Cm (in g/ml) in the irradiated fuel sample.	
	ECM244 = uncertainty in CM244.	

B. Input Data for Plots

1. Data for a set of plots, all with the same X-axis.

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	XVAR,YVAR(1),YVAR(2),...,YVAR(9)	10I2
	XVAR = Variable to be plotted on the X-axis.	
	YVAR(1) = Variable to be plotted on the y-axis of the first plot.	
	YVAR(2) = Variable to be plotted on the y-axis of the second plot.	
	etc.	
2	B(L,1) L = 1,6	6A6,4X,3E10.5,I10
	Alphanumeric title to be printed on the X-axis.	
	(Maximum of 36 characters).	

2. Data for each plot within a set.

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	K1,K2,XMIN,DELX,XLIM,YMIN,DELY,YLIM,FAC <p>K1 = number of title cards for the plot (K1 = 2 if titles are limited to the X and Y axis.)</p> <p>K2 = 0 Plot symbol only. K2 = 1 plot symbol and error bars. K2 = 2 plot error bars.</p> <p>XMIN = value of X variable at beginning of X scale.</p> <p>DELX = incremental change in X scale in X units per inch.</p> <p>XLIM = length of X-axis in inches.</p> <p>YMIN = value of Y variable at beginning of Y scale.</p> <p>DELY = incremental change in Y scale in Y units per inch.</p> <p>YLIM = length of Y-axis in inches.</p> <p>FAC = factor by which plot is expanded or reduced.</p> <p>If FAC = 0 it is reset to FAC = 1.</p>	2I2,6X,7E10.5
2	B(L,2) L = 1,6 <p>Alphanumeric title to be printed in the Y-axis (maximum of 36 Characters).</p>	6A6,4X,3E10.5,I10
3-30	((B(L,K),L=1,6),XB(K),YB(K),HB(K),CEN(k),K=2,K1) <p>Note: These K1-2 cards are used to write extra information on the plot.</p> <p>They are included only if K1 is greater than two.</p>	6A6,4X,3E10.5,I10

3-30 $B(L,K), L=1,6$ = alphanumeric information

(maximum of 36 characters) to be printed on the graph
at location $(XB(K), YB(K))$.

$XB(K)$ = X coordinate in inches at which $B(2,K)$ begins
to be printed.

$YB(K)$ = Y coordinate (in inches) at which $B(L,K)$
begins to be printed.

$HB(K)$ = height (in inches) of the $B(L,K)$ characters

$CEN(K)$ = an integer

IF $CEN(K) > 0$ then $B(L,K)$ is printed out

IF $CEN(K) \leq 0$ then the center referenced symbol
corresponding (see CALCOMP manual) to the absolute
value of $CEN(K)$ is printed out.

C. Input Data for Plutonium Samples

The input data for plutonium samples is a subset of the input data for uranium-plutonium samples. Therefore only the variables and their formats are listed here. Definitions of the variables (except for the variable, DEN) are the same as for the uranium-plutonium input data and may be looked up in subsection A. Input data for plutonium plots are the same as for uranium-plutonium plots as described in section B.

1. Data for a set of samples

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	IOP, IPUN, IPLOT, TIN, DEN	312, 4X, 6E10.5
	If DEN = atoms/b-cm of the plutonium initially in the fuel, then the data for DBUFIT will be in units of atoms/b-cm.	
	If DEN = 0 then data for DBUFIT will be in units of atoms per initial atom of plutonium.	
2	BPX(1), EPBX(1), BPX(2), EPBX(2), BPX(3), EPBX(3), BPX(4), EPBX(4)	8E10.5
3	ATP2, EATP2, SP92, ESP92	8E10.5
4	ATN50, EATN50, S20, ES20, S80, ES80	8E10.5

Note: This card is omitted if IOP = 1.

2. Data for each sample

<u>Card</u>	<u>Variables</u>	<u>Format</u>
1	NDAKTR, MAC, ISM, TOUTPU, CS, ECS, COR1, TOUTCS	312, 4X, 6E10.5
2&3	G(I), I=1, 32	16A5
4	PX(1), EPX(1), PX(2), EPX(2), PX(3), EPX(3), PX(4), EPX(4)	8E10.5
5	SPVLP, ESPVLP, SMVLP, ESMVLP, M92, EM92	8E10.5
6	SPVLN, ESPVLN, SMVLN, ESMVLN, M80, EM80	8E10.5
	Note: This card is omitted if NDAKTR = 1.	
7	M20, EM20, R02, ER02, R08, ER08	8E10.5
	Note: This card is omitted if NDAKTR = 1	
8	AM243, EAM243, CM244, ECM244	8E10.5

Note: This card is omitted if MAC = 0

D. Program Control

Program control is maintained by two methods. One is accomplished through various control variables that are part of the input. The variables IPLOT and IPUN control the accumulation of data for plotting and punching. They also determine when the accumulated data shall be plotted or punched. Other variables such as NDAKTR, NORM, MAC, IOP, and IROD determine which alternative methods shall be used to reduce the data when more than one choice is available. The other method of control is the "99" card which consists of nines punched in the first two columns of a card.

An illustration of how the 99 card is used with the input data is shown in Table III. The 99 card is used to signal the end a group of stacked samples and is also used to signal the end of plot data. Successive 99 cards are used to terminate the computer run.

TABLE III

Illustration of the 99 Card

Data Set #1	[	Input data common to first set of samples
		Input data for samples of the first set.
		99 (Signals end of stacked cases)
		Input data for plots.
		99 (Signals end of plot data)
Data Set #2	[	Input data common to second set of samples
		Input data for samples of the second set
		99 (Signals end of stacked cases)
		99 (Signals end of run)

The user is warned that since the input data for successive cases is stacked, extra or missing cards within an input deck throw all remaining cards out of sequence and therefore destroy the meaning of the input data from the point of the error onwards.

IV. SAMPLE TEST CASE

A. Uranium-Plutonium Samples

Input data for a uranium-plutonium test case is shown in Table IV. The first ten cards as well as the last card of this input deck are control cards for the local UNIVAC 1108 computer. The card 'N XQT,U ISODIL (line 10) calls the uranium-plutonium version of the ISODIL code. If the letter following the XQT command had been P instead of U, the plutonium version would have been called.

The measured data is punched into cards according to the instructions given in Section III of this report. First comes a set of UO_2 - PuO_2 data (four samples) followed by plotting instructions for these samples. The category I data for the four samples are on lines 11-15 of Table IV. Input data for each of the four samples are on lines 16-24, 25-33, 34-40, and 41-50 respectively. These data are followed by a 99 card to signal the end of the data for this group. The next nineteen cards are plot input data for the four samples. These plot data is also terminated with a 99 card. The next fourteen cards are for a sample of a different type than the first four samples. Again the first five cards are used to enter the "category 1" data followed by nine cards of input data for the sample. A 99 card signals the end of the data and since there is to be no plots for this sample the run is terminated by an additional 99 card.

The printed output (one page for each sample) is shown at the end of this subsection. The heading at the top of a page is from the input cards. The next two lines are input data, the first of these being decay times for ^{241}Pu decay corrections. The latter line is input data for the

fission product used in the analysis whether it be ^{148}Nd or ^{137}Cs . A third line is included if ^{243}Am and ^{244}Cm are used in the analyses.

In the next part of the output entitled "ISOTOPIC DILUTION ANALYSIS," all columns except the last are input data. As is done throughout the rest of the output, each quantity is followed by its one standard deviation uncertainty which is enclosed in parentheses. The numbers listed under SPIKE and MIXTURE (i.e., sample plus spike) are the $^{239}\text{Pu}/^{242}\text{Pu}$ atom ratio (PU9/2), the $^{238}\text{U}/^{235}\text{U}$ atom ratio (U 8/3), and the $^{148}\text{Nd}/^{150}\text{Nd}$ atom ratio (ND8/0). One line below the isotopic dilution results is reserved to print out the beginning-of-life uranium plus plutonium in units of grams per milliliter and also state the method by which it was calculated. If ^{148}Nd has been determined by isotopic dilution, the percentage correction that has been applied to the ^{148}Nd result because of the contamination of natural neodymium is also recorded on this line.

The next section of the output is labeled ISOTOPIC INVENTORY. The uranium and plutonium isotopes as well as their totals are listed. The abbreviations listed in the headings are

- BOL = beginning-of-life, meaning at the beginning of the irradiation
- BOL = end-of-life, meaning at the end of the irradiation
- GM = gram
- (A/0) = atom percent.

Notice that the total beginning-of-life uranium plus plutonium is one gram. All of the other quantities that are in units of grams are normalized to this initial one gram of fuel. When ^{243}Am and ^{244}Cm are

Table IV

Listing of the Input Cards for
the Uranium-Plutonium Test Case

```

'Q RUN      8,CUR210,,90,4000      RP MATSEN ROB II  948-2448
'X ASG      A=U3451
' ASG,W     H=(,NH,AT)
'XQT CUR
PEF A
TN A
TRI A
' MSG CUR210 PLOT TAPE H  400 PAPER  FLUID INK PEN
' MSG CUR210 PLOT RP MATSEN  ROB II  3000 AREA
'N XQT,U ISODIL
  2      .02      .0002      2.33      .2      8.315
.5742 -04 .5 -06 .7256 -02 .7 -05
.909 -01 .45 -03 .1123 -01 .6 -04 .879 -03 .2 -04 .291 -03 .7 -05
.51458+19 .5 +17 .2404 -02 .12 -04 .4987 +17 .5 +15 .1031 -03 .66 -05
.784 +16 .78 +14 .8056 -02 .24 -03 .8679 -02 .16 -03
1 1 1.378 .2731
      ELEMENT 6065      ROD FS12(EXTERIOR)      BATCH CORE      THE SAMPLE WAS
CUT 10.4 INCHES FROM THE TOP OF THE FUEL
.54 -04 .2 -05 .6835 -02 .3 -04 .081 -03 .3 -05
.07503 .37 -03 .742 -03 .37 -05 .126 .3 -02
.1326 .4 -03 .2175 -01 .11 -03 .161 -02 .2 -04 .418 -03 .8 -05
.14995 .75 -03 .742 -03 .37 -05 .9777 .29 -02
.39969 .2 -02 .02494 .125 -03 .1446 .14 -02
.01433 .15 -03 17.24 .17 .5531 .28 -02
1 1 1.378 .2731
      ELEMENT 6066      ROD FS16(EXTERIOR)      BATCH CORE      THE SAMPLE WAS
CUT 9.6 INCHES FROM THE TOP OF THE FUEL
.52 -04 .25 -05 .6393 -02 .3 -04 .161 -03 .3 -05
Blank Card
.1798 .0009 .3420 -01 .20 -03 .3077 -02 .31 -04 .0720 -02 .14 -04
.24997 .00125 .9905 -03 .5 -05 .7378 .37 -02
.5997 .003 .7428 -02 .37 -04 .07205 .2 -03
.772 -02 .4 -04 46.62 .23 .5487 .002
1 1 1.378 .2658 .1905 +18 .38 +16 1.0096 1.268
      ELEMENT 6066      ROD FS16(EXTERIOR)      BATCH CORE      THE SAMPLE WAS
CUT 28.9 INCHES FROM THE TOP OF THE FUEL
.51 -04 .25 -05 .5955 -02 .3 -04 .237 -03 .35 -05
.14995 .00075 .9905 -03 .5 -05 .7530 .0023
.2301 .0012 .5088 -01 .25 -03 .5543 -02 .42 -04 .1002 -02 .20 -04
.24997 .00125 .9905 -03 .5 -05 .6654 .33 -02
1 1 1.038 .2713
      ELEMENT 6067      ROD FS23(EXTERIOR)      BATCH CORE      THE SAMPLE WAS
CUT 29.1 INCHES FROM THE TOP OF THE FUEL
.49 -04 .25 -05 .5494 -02 .275 -04 .315 -03 .45 -05
.14995 .00075 .9905 -03 .5 -05 .7555 .0023

```

Table IV (Cont'd)

.2903	.0015	.8971	-01 .55	-05 .7558	-02 .70	-04 .1413	-02 .28	-04
.24997	.00125	.9905	-03 .5	-05 .8097	.50	-02		
.5997	.003	.8723	-02 .55	-04 .1244	.4	-03		
.828	-02 .4	-04 27.35	.20	.5508	.003			
.347	-05 .14	-06 .70	-06 .56	-07				
99								
1 2 3								
U235/U238 ATOM RATIO								
3 1	.0074	-.0004	6.	0.0	.00005	8.		
U236/U238 ATOM RATIO								
ISODIL TEST DATA				3.	1.	.14		1
3 1	.0074	-.0004	6.	0.0	.04	8.		
PU240/PU239 ATOM RATIO								
ISODIL TEST DATA				3.	1.	.14		1
3 4 7 8								
PU240/PU239 ATOM RATIO								
3 1	.05	.05	6.	0.0	.01	8.		
PU241/PU239 ATOM RATIO								
ISODIL TEST DATA				3.	1.	.14		1
3 1	.05	.05	6.	.016	.001	8.		
PU/U								
ISODIL TEST DATA				3.	1.	.14		1
3 1	.05	.05	6.	0.0	1.	8.5		
EXPOSURE(GWD/MTM)								
ISODIL TEST DATA				3.	1.	.14		1
99								
.5742	-04 .5	-06 .7256	-02 .7	-05	.2	8.315		
Blank Card								
.51458	+19 .5	+17 .2404	-02 .12	-04 .49805	+17 .5	+15 .103	-03 .66	-05
.784	+16 .78	+14 .8056	-02 .24	-03 .8679	-02 .16	-03		
-1	1	1.833	.3128					
NATURAL UO2 SAMPLE 62.9 INCHES FROM THE BEGINNING OF FUEL AT TOP OF ROD 1330 IRRADIATED IN THE CENTER POSITION OF PRTR ELEMENT 1037								
.42	-04 .2	-05 .193	-02 .2	-04 .83	-03 .1	-04		
Blank Card								
.4351	.13	-02 .0893	.4	-03 .03489	.35	-03 .05834	-01 .58	-04
.20005	.001	.2463	-02 .123	-04 .4990	.15	-02		
.39969	.002	.4925	-02 .25	-04 .1741	.5	-03		
.01496	.8	-04 11.03	.03	.4908	.001			
99								
99								
FIN								

X

included in the analysis, their results are reported below the plutonium totals.

In the next part of the output the fissions and burnup are recorded. The fission product used to obtain these results is indicated in the headings. Also listed are the constants (FISSION YIELDS, DEL, and MEV/FISSION) used to obtain these results. For a given isotope, DEL is the number of fissions in ^{238}U from fast neutrons of a fission of the given isotope.

Finally, results from a number of miscellaneous calculations are reported. They are the fractional depletion of ^{235}U (U235/U235BOL); the capture to fission ratios for ^{235}U and ^{239}Pu (ALPHA 235 and ALPHA239); the amount of ^{239}Pu that has been generated from neutron captures in ^{238}U (U238 INTO Pu239); and the loss of ^{236}U by neutron capture (U236 INTO NP237).

Some of the test data are plotted in Figures 1-5 in order to demonstrate various plot combinations that can be employed. The ISODIL code prepares the data in a format acceptable for the 736 CALCOMP plotter. The data points, indicated by symbols, have one standard deviation uncertainties attached to them. The symbol assigned to a data point is designated by the input variable ISM. The initial data point, indicated by a circle, has been drawn on the plots by hand. All of the irradiated samples began at this point. These data (except for the next to last point where ^{137}Cs was used in place of ^{148}Nd as a burnup indicator are marked by a square.

Observe that the data in each plot define a curve or path. These burnup paths are the result of the transmutation and fission processes that occur in the fuel during irradiation. A data point will deviate from the path defined by the other data points if

- it is incorrectly measured
- the sample's initial composition was different than that of the other samples
- it was irradiated in a region of the reactor where the spectral environment was different from that of the other samples.
- or the quantity being measured has migrated.

If the latter three items can be ruled out, the data point is probably mismeasured and should be redone.

The only obvious incorrect data point in Figures 1-5 is the next to last data point in Figure 5. The burnup for this point was determined from ^{137}Cs while the burnup of the other points were determined from ^{148}Nd . A probable explanation for the low burnup of this point is that some of the ^{137}Cs migrated from the sample during its irradiation since it is known that its burnup as determined from ^{148}Nd falls on the path described by the other samples.

ELEMENT 6065 ROD FS12(EXTERIOR) BATCH CORE THE SAMPLE WAS CUT 10.4 INCHES FROM THE TOP OF THE FUEL

INPUT DATA---- TIME OUT OF REACTOR=1.000YRS BOL PU241 DECAY CORRECTION TIME= 2.330YRS
SPIKE--ND42/50 .8056-02(.24-03) ND50/42 .1724+02(.17-00)--SAMPLE--ND50/48 .5531-00(.28-02) MIXTURE--ND42/50 .1433-01(.15-03)

I S O T O P I C D I L U T I O N A N A L Y S I S

ATOMS/ML IN SPIKE		SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .4987+17(.50+15)	PU9/2	.1031-03(.66-05)	.1499-00(.75-03)	.7420-03(.37-05)	PU9/2 .9777-00(.29-02)	PU239 .9868+19(.12+18)
U-233 .5146+19(.50+17)	U 8/3	.2404-02(.12-04)	.7503-01(.37-03)	.7420-03(.37-05)	U 8/3 .1126+01(.30-02)	U-238 .5846+21(.76+19)
ND150 .7840+16(.78+14)	ND8/0	.8679-02(.16-03)	.3997-00(.20-20)	.2494-01(.12-30)	ND8/0 .1446-00(.14-20)	ND148 .1837+17(.18+15)

BOL U+PU=.2376(.0031)GM/ML AS CALCULATED BY ISOTOPIIC DILUTION

.68% SAMPLE---ND CONTAMINATION---MIXTURE 1.08%

I S O T O P I C I N V E N T O R Y

ISOTOPE	BOL (GM)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	NET PRODUCTION (GM)
U-234	.5493-04(.487-06)	.5400-04(.200-05)	.0054(.0002)	.5163-04(.195-05)	-.3300-05(.201-05)
U-235	.6971-02(.691-05)	.6835-02(.300-04)	.6788(.0029)	.6563-02(.290-04)	-.4081-03(.299-04)
U-236	.0000 (.000)	.8100-04(.300-05)	.0080(.0003)	.7810-04(.292-05)	.7810-04(.292-05)
U-238	.9730-00(.199-03)	1.0000	99.3078(.0398)	.9724-00(.428-03)	-.5395-03(.379-03)
TOTALS	.9800-00(.200-03)			.9791-00(.200-03)	-.8699-03(.378-03)

ISOTOPE	BOL (A/O)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	BOL (GM)	NET PRODUCTION (GM)
PU238	.0263(.0006)	.4180-03(.800-05)	.0361(.0007)	.6861-05(.174-06)	.5244-05(.137-06)	.1617-05(.222-06)
PU239	90.5182(.0375)	1.0000	86.3973(.0312)	.1648-01(.275-03)	.1810-01(.181-03)	-.1613-02(.329-03)
PU240	8.2281(.0374)	.1326-00(.400-03)	11.4563(.0306)	.2195-02(.372-04)	.1652-02(.181-04)	.5429-03(.414-04)
PU241	1.1478(.0070)	.2175-01(.110-03)	1.9712(.0101)	.3792-03(.663-05)	.2314-03(.271-05)	.1478-03(.716-05)
PU242	.0796(.0018)	.1610-02(.200-04)	.1391(.0017)	.2687-04(.559-06)	.1611-04(.396-06)	.1076-04(.685-06)
TOTALS				.1909-01(.319-03)	.2000-01(.200-03)	-.9101-03(.376-03)

F I S S I O N S B Y N D 1 4 8

ISOTOPE	FISSIONS (GM)	BURNUP (MWD/MTM)	FISSION YIELD	DEL	MEV/FISSION
U-235	.3298-03(.3005-04)	316.1	.0169(.0001)	.0491(.0049)	201.7
U-238	.9295-04(.9211-05)	90.9	.0193(.0010)		208.5
PU239	.1333-02(.4797-04)	1306.9	.0170(.0003)	.0563(.0056)	210.7
PU241	.2937-04(.2005-05)	29.0	.0189(.0005)	.0604(.0060)	213.8
TOTALS	.1785-02(.3797-04)	1743.0			

U235/U235BOL=.941(.004) ALPHA235=.2372(.0241) ALPHA239=.5454(.0421) U238 INTO PU239=.4484-03(.38-03)GM U236 INTO NP237=.476-06GM

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

ELEMENT 6066 ROD FS16(EXTERIOR) BATCH CORE THE SAMPLE WAS CUT 9.6 INCHES FROM THE TOP OF THE FUEL

INPUT DATA---- TIME OUT OF REACTOR=1.378YRS BOL PU241 DECAY CORRECTION TIME= 2.330YRS OXIDE FUEL= .2731GM/ML
SPIKE--ND42/50 .8056-02(.24-03) ND50/42 .4662+02(.23-00)--SAMPLE--ND50/48 .5487-00(.20-02) MIXTURE--ND42/50 .7720-02(.40-04)

I S O T O P I C D I L U T I O N A N A L Y S I S

ATOMS/ML IN SPIKE	SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .4987+17(.50+15)	PU9/2 .1031-03(.66-05)	.2500-00(.12-02)	.9905-03(.50-05)	PU9/2 .7378-00(.37-02)	PU239 .9305+19(.12+18)
ND150 .7840+16(.78+14)	ND8/0 .8679-02(.16-03)	.5997-00(.30-02)	.7428-02(.37-04)	ND8/0 .7205-01(.20-03)	ND148 .4177+17(.54+15)

BOL U+PU=.2407(.0024)GM/ML AS CALCULATED BY FUEL WEIGHT AND SOLUTION VOLUME .25% SAMPLE---ND CONTAMINATION---MIXTURE -.02%

I S O T O P I C I N V E N T O R Y

ISOTOPE	BOL (GM)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	NET PRODUCTION (GM)
U-234	.5493-04(.487-06)	.5200-04(.250-05)	.0052(.0002)	.4964-04(.243-05)	-.5289-05(.248-05)
U-235	.6971-02(.691-05)	.6393-02(.300-04)	.6351(.0029)	.6129-02(.290-04)	-.8419-03(.299-04)
U-236	.0000 (.000)	.1610-03(.300-05)	.0160(.0003)	.1550-03(.292-05)	.1550-03(.292-05)
U-238	.9730-00(.199-03)	1.0000	99.3437(.0398)	.9709-00(.430-03)	-.2040-02(.381-03)
TOTALS	.9800-00(.200-03)			.9773-00(.199-03)	-.2729-02(.380-03)

ISOTOPE	BOL (A/O)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	BOL (GM)	NET PRODUCTION (GM)
PU238	.0263(.0006)	.7200-03(.140-04)	.0590(.0011)	.1100-04(.281-06)	.5244-05(.137-06)	.5757-05(.313-06)
PU239	90.5182(.0375)	1.0000	81.9588(.0623)	.1534-01(.255-03)	.1810-01(.181-03)	-.2753-02(.313-03)
PU240	8.2281(.0374)	.1798-00(.900-03)	14.7362(.0630)	.2770-02(.481-04)	.1652-02(.181-04)	.1118-02(.514-04)
PU241	1.1478(.0070)	.3420-01(.200-03)	2.9938(.0179)	.5652-03(.100-04)	.2314-03(.271-05)	.3338-03(.104-04)
PU242	.0796(.0018)	.3077-02(.310-04)	.2522(.0025)	.4780-04(.930-06)	.1611-04(.396-06)	.3170-04(.101-05)
TOTALS				.1874-01(.312-03)	.2000-01(.200-03)	-.1263-02(.370-03)

F I S S I O N S B Y N D 1 4 8

ISOTOPE	FISSIONS (GM)	BURNUP (MWD/MTM)	FISSION YIELD	DEL	MEV/FISSION
U-235	.6856-03(.3006-04)	657.1	.0169(.0001)	.0491(.0049)	201.7
U-238	.2089-03(.2042-04)	204.4	.0193(.0010)		208.5
PU239	.3024-02(.8914-04)	2965.0	.0170(.0003)	.0563(.0056)	210.7
PU241	.8648-04(.3486-05)	85.3	.0189(.0005)	.0604(.0060)	213.8
TOTALS	.4005-02(.8527-04)	3911.8			

U235/U235BOL=.879(.004) ALPHA235=.2281(.0112) ALPHA239=.5165(.0260) U238 INTO PU239=.1839-02(.38-03)GM U236 INTO NP237=.204-05GM

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

ELEMENT 6066

ROD FS16(EXTERIOR)

BATCH CORE

THE SAMPLE WAS CUT 28.9 INCHES FROM THE TOP OF THE FUEL

INPUT DATA----

TIME OUT OF REACTOR=1.378YRS

BOL PU241 DECAY CORRECTION TIME= 2.330YRS

CS137=.1905+18(.38+16)ATOMS/ML TIME OUT OF REACTOR=1.268YRS IN REACTOR DECAY CORRECTION=1.0096 HALF LIFE(ASSUMED)=30.00YRS

ISOTOPIC DILUTION ANALYSIS

ATOMS/ML IN SPIKE	SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .4987+17(.50+15)	PU9/2 .1031-03(.66-05)	.2500-00(.12-02)	.9905-03(.50-05)	PU9/2 .6654-00(.33-02)	PU239 .8404+19(.11+18)
U-233 .5146+19(.50+17)	U 8/3 .2404-02(.12-04)	.1499-00(.75-03)	.9905-03(.50-05)	U 8/3 .7530-00(.23-02)	U-238 .5847+21(.84+19)

BOL U+PU=.2380(.0034)GM/ML AS CALCULATED BY ISOTOPIC DILUTION

ISOTOPIC INVENTORY

ISOTOPE	BOL (GM)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	NET PRODUCTION (GM)
U-234	.5493-04(.487-05)	.5100-04(.250-05)	.0051(.0002)	.4868-04(.243-05)	-.6243-05(.248-05)
U-235	.6971-02(.691-05)	.5955-02(.300-04)	.5918(.0029)	.5709-02(.291-04)	-.1262-02(.299-04)
U-236	.0000(.000)	.2370-03(.350-05)	.0236(.0003)	.2282-03(.341-05)	.2282-03(.341-05)
U-238	.9730-00(.199-03)	1.0000	99.3796(.0398)	.9709-00(.460-03)	-.2040-02(.415-03)
TOTALS	.9800-00(.200-03)			.9769-00(.199-03)	-.3078-02(.414-03)

ISOTOPE	BOL (A/O)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	BOL (GM)	NET PRODUCTION (GM)
PU238	.0263(.0006)	.1002-02(.200-04)	.0776(.0015)	.1398-04(.378-06)	.5244-05(.137-06)	.8739-05(.402-06)
PU239	90.5182(.0375)	1.0000	77.4600(.0741)	.1401-01(.255-03)	.1810-01(.181-03)	-.4082-02(.313-03)
PU240	8.2281(.0374)	.2301-00(.120-02)	17.8235(.0765)	.3238-02(.612-04)	.1652-02(.181-04)	.1586-02(.639-04)
PU241	1.1478(.0070)	.5088-01(.250-03)	4.2095(.0215)	.7680-03(.145-04)	.2314-03(.271-05)	.5366-03(.148-04)
PU242	.0796(.0018)	.5543-02(.420-04)	.4294(.0033)	.7866-04(.155-05)	.1611-04(.396-06)	.6255-04(.160-05)
TOTALS				.1811-01(.330-03)	.2000-01(.200-03)	-.1888-02(.386-03)

FISSIONS BY CS137

ISOTOPE	FISSIONS (GM)	BURNUP (MWD/MTM)	FISSION YIELD	DEL	MEV/FISSION
U-235	.1030-02(.3013-04)	987.2	.0628(.0003)	.0491(.0049)	201.7
U-238	.2588-03(.2591-04)	253.2	.0620(.0040)		208.5
PU239	.3519-02(.1477-03)	3449.9	.0674(.0014)	.0563(.0056)	210.7
PU241	.1707-03(.6063-05)	168.4	.0660(.0017)	.0604(.0060)	213.8
TOTALS	.4978-02(.1505-03)	4858.8			

U235/U235BOL=.819(.004) ALPHA235=.2252(.0077) ALPHA239=.6658(.0366) U238 INTO PU239=.1789-02(.41-03)GM U236 INTO NP237=.471-05GM

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

ELEMENT 6067 ROD FS23(EXTERIOR) BATCH CORE THE SAMPLE WAS CUT 29.1 INCHES FROM THE TOP OF THE FUEL

INPUT DATA---- TIME OUT OF REACTOR=1.038YRS BOL PU241 DECAY CORRECTION TIME= 2.330YRS OXIDE FUEL= .27136M/ML
SPIKE--ND42/50 .8056-02(.24-03) ND50/42 .2735+02(.20-00)--SAMPLE--ND50/48 .5508-00(.30-02) MIXTURE--ND42/50 .8280-02(.40-04)
AM243=.347-05(.14-06)GM/ML CM244=.700-06(.56-07)GM/ML

ISOTOPIC DILUTION ANALYSIS

ATOMS/ML IN SPIKE	SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .4987+17(.50+15)	PU9/2 .1031-03(.66-05)	.2500-00(.12-02)	.9905-03(.50-05)	PU9/2 .6097-00(.30-02)	PU239 .7716+19(.10+18)
ND150 .7840+16(.78+14)	ND8/0 .8679-02(.16-03)	.5997-00(.30-02)	.6923-02(.35-04)	ND8/0 .1244+00(.40-03)	ND148 .8424+17(.11+16)

BOL U+PU=.2392(.0024)GM/ML AS CALCULATED BY FUEL WEIGHT AND SOLUTION VOLUME .43% SAMPLE---ND CONTAMINATION---MIXTURE .13%

ISOTOPIC INVENTORY

ISOTOPE	BOL (GM)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	NET PRODUCTION (GM)
U-234	.5493-04(.487-06)	.4900-04(.250-05)	.0049(.0002)	.4666-04(.244-05)	-.8262-05(.248-05)
U-235	.6971-02(.691-05)	.5494-02(.275-04)	.5462(.0027)	.5255-02(.267-04)	-.1716-02(.276-04)
U-236	.0000 (.000)	.3150-03(.450-05)	.0313(.0004)	.3026-03(.438-05)	.3026-03(.438-05)
U-238	.9730-00(.199-03)	1.0000	99.4176(.0398)	.9687-00(.442-03)	-.4317-02(.395-03)
TOTALS	.9800-00(.200-03)			.9743-00(.199-03)	-.5740-02(.394-03)

ISOTOPE	BOL (A/O)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	BOL (GM)	NET PRODUCTION (GM)
PU238	.0263(.0006)	.1413-02(.280-04)	.1028(.0020)	.1802-04(.466-06)	.5244-05(.137-06)	.1278-04(.486-06)
PU239	90.5182(.0375)	1.0000	72.7629(.0820)	.1281-01(.213-03)	.1810-01(.181-03)	-.5289-02(.279-03)
PU240	8.2281(.0374)	.2903-00(.150-02)	21.1231(.0863)	.3733-02(.649-04)	.1652-02(.181-04)	.2082-02(.674-04)
PU241	1.1478(.0070)	.6971-01(.350-03)	5.3303(.0269)	.9460-03(.165-04)	.2314-03(.271-05)	.7147-03(.167-04)
PU242	.0796(.0018)	.9358-02(.700-04)	.6809(.0051)	.1214-03(.221-05)	.1611-04(.396-06)	.1052-03(.225-05)
TOTALS				.1763-01(.293-03)	.2000-01(.200-03)	-.2374-02(.355-03)
AM243				.1451-04(.603-06)		
CM244				.2927-05(.236-06)		

ISOTOPE	FISSIONS (GM)	BURNUP (MWD/MTM)	FISSION YIELD	DEL	MEV/FISSION
U-235	.1406-02(.2795-04)	1347.7	.0169(.0001)	.0491(.0049)	201.7
U-238	.4238-03(.4130-04)	414.6	.0193(.0010)		208.5
PU239	.5951-02(.1710-03)	5834.9	.0170(.0003)	.0563(.0056)	210.7
PU241	.3345-03(.1091-04)	330.0	.0189(.0005)	.0604(.0060)	213.8
TOTALS	.8116-02(.1711-03)	7927.2			

U235/U235BOL=.754(.004) ALPHA235=.2206(.0058) ALPHA239=.5435(.0219) U238 INTO PU239=.3910-02(.39-03)GM U236 INTO NP237=.895-05GM

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

NATURAL UO2 SAMPLE 62.9 INCHES FROM THE BEGINNING OF FUEL AT TOP OF ROD 1330 IRRADIATED IN THE CENTER POSITION OF PRTR ELEMENT 1037

INPUT DATA---- TIME OUT OF REACTOR=1.833YRS BOL PU241 DECAY CORRECTION TIME=-.000YRS OXIDE FUEL=.3128GM/ML
SPIKE--ND42/50 .8056-02(.24-03) ND50/42 .1103+02(.30-01)--SAMPLE--ND50/48 .4908-00(.10-02) MIXTURE--ND42/50 .1496-01(.80-04)

ISOTOPIC DILUTION ANALYSIS

ATOMS/ML IN SPIKE	SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .4981+17(.50+15)	PU9/2 .1030-03(.66-05)	.2000-00(.10-02)	.2463-02(.12-04)	PU9/2 .4990-00(.15-02)	PU239 .2054+19(.26+17)
ND150 .7840+16(.78+14)	ND8/0 .8679-02(.16-03)	.3997-00(.20-02)	.4925-02(.25-04)	ND8/0 .1741-00(.50-03)	ND148 .1140+18(.14+16)

BOL U+PU=.2757(.0028)GM/ML AS CALCULATED BY FUEL WEIGHT AND SOLUTION VOLUME .95% SAMPLE--ND CONTAMINATION--MIXTURE .97%

ISOTOPIC INVENTORY

ISOTOPE	BOL (GM)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	NET PRODUCTION (GM)
U-234	.5605-04(.496-06)	.4200-04(.200-05)	.0042(.0002)	.4059-04(.199-05)	-.1546-04(.206-05)
U-235	.7113-02(.690-05)	.1930-02(.200-04)	.1925(.0020)	.1873-02(.199-04)	-.5240-02(.211-04)
U-236	.0000 (.000)	.8300-03(.100-04)	.0828(.0010)	.8090-03(.996-05)	.8090-03(.996-05)
U-238	.9928-00(.692-05)	1.0000	99.7206(.0023)	.9830-00(.193-03)	-.9791-02(.193-03)
TOTALS	.1000+01(.000)			.9856-00(.000)	-.1435-01(.191-03)

ISOTOPE	BOL (A/O)	EOL MASS RATIO	EOL (A/O)	EOL (GM)	BOL (GM)	NET PRODUCTION (GM)
PU238	.0000(.0000)	.5834-02(.580-04)	.3708(.0037)	.1718-04(.325-06)	.0000 (.000)	.1718-04(.325-06)
PU239	100.0000(.0000)	1.0000	63.5606(.0580)	.2957-02(.477-04)	.0000 (.000)	.2957-02(.477-04)
PU240	.0000(.0000)	.4351-00(.130-02)	27.6552(.0607)	.1292-02(.212-04)	.0000 (.000)	.1292-02(.212-04)
PU241	.0000(.0000)	.8930-01(.400-03)	6.1957(.0301)	.2906-03(.492-05)	.0000 (.000)	.2906-03(.492-05)
PU242	.0000(.0000)	.3489-01(.350-03)	2.2176(.0218)	.1045-03(.199-05)	.0000 (.000)	.1045-03(.199-05)
TOTALS				.4661-02(.753-04)	-.0000 (.000)	.4661-02(.753-04)

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FISSIONS BY ND148

ISOTOPE	FISSIONS (GM)	BURNUP (MWD/MTM)	FISSION YIELD	DEL	MEV/FISSION
U-235	.4304-02(.2330-04)	4125.2	.0169(.0001)	.0636(.0064)	201.7
U-238	.6504-03(.6195-04)	636.3	.0193(.0010)		208.5
PU239	.4245-02(.1790-03)	4161.6	.0170(.0003)	.0826(.0083)	210.7
PU241	.2850-03(.8887-05)	281.2	.0189(.0005)	.0855(.0085)	213.8
TOTALS	.9484-02(.1761-03)	9204.3			

U235/U235BOL=.263(.003) ALPHA235=.2175(.0030) ALPHA239=.4619(.0210) U238 INTO PU239=.9179-02(.19-03)GM U236 INTO NP237=.131-03GM

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

FIGURE 1. $^{236}\text{U}/^{238}\text{U}$ vs $^{235}\text{U}/^{238}\text{U}$ for $\text{UO}_2\text{-PuO}_2$
Test Data

FIGURE 2. $^{240}\text{Pu}/^{239}\text{Pu}$ vs $^{235}\text{U}/^{238}\text{U}$ for $\text{UO}_2\text{-PuO}_2$
Test Data

FIGURE 3. $^{241}\text{Pu}/^{239}\text{Pu}$ vs $^{240}\text{Pu}/^{239}\text{Pu}$ for $\text{UO}_2\text{-PuO}_2$
Test Data

FIGURE 4. Pu/U vs $^{240}\text{Pu}/^{239}\text{Pu}$ for $\text{UO}_2\text{-PuO}_2$ Test Data

FIGURE 5. Burnup vs $^{240}\text{Pu}/^{239}\text{Pu}$ for $\text{UO}_2\text{-PuO}_2$ Test Data

B. Plutonium Sample

The input for a plutonium sample is shown in Table IV. Notice that a P follows XQT in the 'N XQT,P ISODIL card. This calls in the plutonium version of the code.

The printed output shown at the end of this subsection has many similarities with the uranium-plutonium output. However, there are major differences. It is the differences that will be stressed here. After the heading at the top of the page, input data and constants used in the analysis are printed on the left and right hand sides of the page respectively. Quantities listed under ISOTOPIC DILUTION ANALYSIS are much the same as for the uranium-plutonium case. The isotopic inventory is normalized to one beginning-of-life plutonium atom rather than to one gram as was done in the uranium-plutonium case. Also, the number of fissions is included with this part of the output rather than in a section of its own. Finally, the capture of fission ratio for ^{239}Pu is printed.

Table V

List of the Input Cards for the Plutonium Test Case

```

ID RUN      8,CUB210,,90,4000      RP MATSEN ROB 11  240-2448
IX ASG      A=U3451
I ASG,W     H=(,NH,AT)
I XQT CUR
PEF A
IN A
TRI A
I MSG CUB210 PLOT TAPE H 400 PAPER FLUID INK PEN
I MSG CUB210 PLOT RP MATSEN ROB 11 3000 AREA
IN XQT,P ISODIL
      1.8201
.20293 .00124 .02825 .00032 .00248 .67 -04
.25575+17 .25 +15 .1074 -03 .86 -05
.784 +16 .784 +14 .8056 -02 .4 -04 .8679 -02 .43 -04
1 2.85 .247 +16 .49 +14 1.01005 2.61
      HXO PRTR PUAL SAMPLE W16-1 ELEMENT 5111 1.3 INCH
FS FROM THE TOP OF THE ROD
.40163 .00016 .06427 .32 -03 .01073 .11 -03
.09993 .0005 .00495 .25 -04 .2219 .0013
.23 .0012 .6 .003 .2 .001
.01 .0005 50. .25 .57 .003
.9264 -08 .4632 -10 .317 -09 .1585 -11
99
99
I FIN
      X

```

HXO PRIOR PUAL SAMPL WIR-1 ELEMENT 5111

1.3 INCHES FROM THE TOP OF THE ROD

INPUT DATA

PU SPIKE ND239/ND242=.1074-03(.86-05) AM243=.906-08(.46-10)G/ML
 TIME PU OUT OF REACTOR=2.850YR CM244=.317-09(.16-11)G/ML
 BOL PU241 DECAY CORRECTION TIME=-.000YR
 ND50/42 .5000+02(.25-00)--SAMPLE--.5700-00(.30-02) ND50/48
 SPIKE--.0056-02(.40-04)--ND42/50--.1000-01(.50-03)--MIXTURE

ASSUMPTIONS

PU241 HALF LIFE=14.5(.4)YR PU239 FISSION YIELD=.017(.003)
 ALPHA241=.365(.009) PU241 FISSION YIELD=.019(.005)

ISOTOPIC DILUTION ANALYSIS

ATOMS/ML IN SPIKE	SPIKE	SPIKE VOL(ML)	SAMPLE VOL(ML)	MIXTURE	ATOMS/ML IN SAMPLE
PU242 .2550+17(.25+15)	PU9/2 .1074-(3(.86-05)	.9993-01(.50-03)	.4950-02(.25-04)	PU9/2 .2219-00(.13-02)	PU239 .1143+1(.15+16)
ND150 .7840+18(.78+14)	ND8/0 .8679-02(.43-04)	.2300-00(.12-02)	.6000-00(.30-02)	ND8/0 .2000-00(.10-02)	ND148 .6470+15(.89+13)

BEGINING OF LIFE PLUTONIUM=.2084+18(.24+16)AT MS/ML

.24% SAMPLE---CONTAMINATION OF ND BY NATURAL ND---MIXTURE .31

ISOTOPIC INVENTORY

ISOTOPE	BOL ATOMS	EOL MASS RATIO	EOL (A/O)	EOL ATOMS	FISSIONS BY ND148	NET PRODUCTION
PU238	.00000(.00000)	.0000 (.000)	.0000(.0000)	.0000 (.000)		.0000 (.000)
PU239	.81060(.00034)	1.0000	67.2943(.0226)	.5509-00(.262-02)	.1704-00(.363-02)	-.2597-00(.843-01)
PU240	.16449(.00084)	.4016-00(.160-03)	27.0274(.0116)	.2212-00(.106-02)		.5675-01(.843-01)
PU241	.02290(.00025)	.6427-01(.320-03)	4.9563(.0294)	.4057-01(.310-03)	.1100-01(.353-03)	.1707-01(.255-01)
PU242	.00201(.00005)	.1073-01(.110-03)	.7221(.0074)	.5911-02(.366-04)		.3900-02(.542-02)
TOTALS	1.00000			.8186-00(.104-01)	.1814-00(.387-02)	-.1814-00(.104-01)
AM243				.1102-03(.141-05)		.1102-03(.141-05)
CM244				.3754-05(.479-07)		.3754-05(.479-07)

RATIO OF NEUTRON CAPTURE TO FISSION FOR PU239= .5241(.0150)

-----QUANTITIES ENCLOSED BY PARENTHESES ARE ONE STANDARD DEVIATION ERRORS-----

ACKNOWLEDGEMENT

Many helpful discussions concerning isotopic dilution measurement methods and isotopic dilution analysis equations were held with W. Y. Matsumoto.

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APPENDIX A

CONSTANTS USED IN THE ISODIL PROGRAM

<u>ISODIL Name</u>	<u>Description</u>	<u>Value</u>	<u>Reference</u>
AVGDRO	Avogadro's number	$.60225 \times 10^{24}$	
UM(1)	Atomic weight of ^{234}U	234.0409	A1
UM(2)	Atomic weight of ^{235}U	235.0439	A1
UM(3)	Atomic weight of ^{236}U	236.0457	A1
UM(4)	Atomic weight of ^{238}U	238.0508	A1
PM(5)	Atomic weight of ^{238}Pu	238.0495	A1
PM(4)	Atomic weight of ^{239}Pu	239.0522	A1
PM(1)	Atomic weight of ^{240}Pu	240.0540	A1
PM(2)	Atomic weight of ^{241}Pu	241.0567	A1
PM(3)	Atomic weight of ^{242}Pu	242.0587	A1
C20	$^{142}\text{Nd}/^{150}\text{Nd}$ atom ratio for natural neodymium	4.824	A1
C	$^{148}\text{Nd}/^{150}\text{Nd}$ atom ratio for natural neodymium	1.0196	A1
T241	^{241}Pu half life	$14.5 \pm .4$	A2
ALF41	Capture to fission ratio for neutrons in ^{241}Pu	$.365 \pm .009$	A2
T137	^{137}Cs half life	$30.0 \pm .2$	A3
Y235	Thermal fission yield of ^{137}Cs from ^{235}U	$.0628 \pm .0003$	A4
Y239	Thermal fission yield of ^{137}Cs from ^{239}Pu	$.0674 \pm .0014$	A4
Y241	Thermal fission yield of ^{137}Cs from ^{241}Pu	$.0660 \pm .0017$	A4
Y238	Fast fission yield of ^{137}Cs from ^{238}U	$.062 \pm .004$	A5

<u>ISODIL Name</u>	<u>Description</u>	<u>Value</u>	<u>Reference</u>
Y235	Thermal fission yield of ^{148}Nd from ^{235}U	$.0169 \pm .0001$	A4
Y239	Thermal fission yield of ^{148}Nd from ^{239}Pu	$.0170 \pm .0003$	A4
Y241	Thermal fission yield of ^{148}Nd from ^{241}Pu	$.0189 \pm .0005$	A4
Y238	Fast fission yield of ^{148}Nd from ^{238}U	$.0193 \pm .0010$	A5
DEL35	^{238}U fissions per ^{235}U fission in the center rod of a PRTR cluster	$.06357 \pm .00636$	A6
DEL39	^{238}U fissions per ^{239}Pu fission in the center rod of a PRTR cluster	$.08263 \pm .00826$	A6
DEL41	^{238}U fissions per ^{241}Pu fission in the center rod of a PRTR cluster	$.08547 \pm .00855$	A6
DEL35	^{238}U fissions per ^{235}U fission in a middle ring rod for a PRTR Cluster	$.06087 \pm .00609$	A6
DEL39	^{238}U fissions per ^{239}Pu fission in a middle ring rod of a PRTR Cluster	$.07656 \pm .00766$	A6
DEL41	^{238}U fissions per ^{241}Pu fission in a middle ring rod of a PRTR Cluster	$.07995 \pm .00800$	A6
DEL35	^{238}U fissions per ^{235}U fission in an outer ring rod of a PRTR Cluster	$.04912 \pm .00491$	A6
DEL39	^{238}U fissions per ^{239}Pu fission in an outer ring rod of a PRTR Cluster	$.05634 \pm .00563$	A6
DEL41	^{238}U fissions per ^{241}Pu fission in an outer ring rod of a PRTR Cluster	$.06038 \pm .00604$	A6

<u>ISODIL Name</u>	<u>Description</u>	<u>Value</u>	<u>Reference</u>
MEV35	Energy release in ^{235}U fissions (MeV/fission)	201.7	A7
MEV39	Energy release in ^{239}Pu fissions (MeV/fission)	210.7	A7
MEV41	Energy release in ^{241}Pu fissions (MeV/fission)	213.8	A7
MEV38	Energy release in ^{238}U fissions (MeV/fission)	208.5	A7

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