GKSS 73/I/26

Computer programme for burn-up determination by means of fission products spectra

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Gesellschaft für Kernenergieverwertung
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9 Seiten, 1 Abbildung

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Summary

A computer programme package is presented which makes it possible to determinate burn-up of irradiated fuel rods from the fission products gamma spectra with a precision of 5 - 10 %. Needed for input data are the irradiation history, cross sections, fission yields, and isotopic data. Special attention will be paid to the calculation of selfabsorption in the fuel rod and to the error caused by insufficient knowledge of the irradiation history.

Zusammenfassung

Es wird ein Programmpaket vorgestellt, das es ermöglicht, aus den Spaltprodukt-Gammaspektren von bestrahlten Brennstäben den Abbrand mit einer Genauigkeit von 5 - 10 % zu bestimmen. Als Eingabedaten werden u.a. die Bestrahlungsgeschichte, Wirkungsquerschnitte, Spaltausbeuten und Isotopdaten benötigt. Besondere Beachtung wird der Berechnung der Selbstabsorption im Brennstab sowie der durch Ungenauigkeiten in der Bestrahlungsgeschichte hervorgerufenen Fehler der Abbrandresultate geschenkt.

Introduction

The method of burn-up measurements by γ-spectroscopy of fission products has been applied since about 10 years. Due to the development of gamma spectrometers with high efficiency and resolution, of fast electronic components, and magnetic tape data recorders, it can now be applied as a routine technique which enables detailed non-destructive investigations of the burn-up distribution of whole fuel rods. This, however, requires an extensive automation of the measuring device and a fast evaluation code. With this method an absolute exactitude of 5 - 10 % can be achieved.

Burn-up determination by fission product measurements

As commonly known the relation between the number of fissioned fuel atoms per sec and the number of produced fission product atoms per sec is given by the cumulative fission yield of the fission product in question. If the half life of a radioactive isotope is very large, compared with the duration of the irradiation, it is sufficient to measure the atomic density of the isotope and count it back to the moment of the end of the irradiation. The value obtained has to be divided by the fission yield in order to get immediately the total number of fission that accured. This number is equivalent to the burn-up.

If, however, the half life of the fission product is not large compared with the irradiation time, the decay of the isotope during the irradiation must be taken into account. In principle this correction could be exactly computed, provided that the neutron flux history at the place of the concerned fuel position is exactly known. Otherwise the relation between the number of fissions and the number of fission product atoms cannot be considered as unique. It has then to be proved to what degree inexactitudes at the given flux history affect the burn-up result. By doing this one obtains at the same time a criterion for the applicability of a fission product for the burn-up evaluation.

The programme system

Fig. 1. shows the flow chart of our programme system used for the evaluation of the γ-spectra. The programmes or subprogrammes stated in this figure will be explained in the following chapters.

1.) Calculation of the atomic densities from the γ-spectra

The γ-radiation of the fission products is recorded in an energy range from 0 - 2.5 mev by a 4096-channel-analyzer with magnetic tape output. This range contains virtually all γ-peaks of fission products suited for burn-up determination.

The counting rate of the spectrometer can be adjusted by the aid of a slit collimator so that a measuring time of about 10 minutes will be sufficient with regard to counting statistics.

The connection between the peak area I and the atomic density N of the fission product is as follows:

$$N = \frac{I}{\epsilon \cdot \beta \cdot s \cdot \lambda \cdot t}$$

with ϵ = efficiency of the detector

β = γ-emission probability per decay

s = self absorption in the fuel rod

λ = decay constant of the fission product

t = measuring time.

The determination of the net peak areas follows a standard method: the background levels on either side of a peak in the recorded γ-spectrum are determined; then the two points are connected by a straight line and the obtained background area is subtracted from the total pulse number between the boundary channels.

The calculation of the selfabsorption factors is a somewhat crucial problem, because several important γ-transitions have considerable selfabsorption in the fuel. In general,

it must be taken into account that by virtue of the flux depression in the fuel rods the fission product distribution itself shows a radial depression. The selfabsorption factors will, therefore, be calculated previously in a special code.

2.) Calculation of the corrections for the decay of fission products during the irradiation and for the decrease and breeding of fuel

The flux history underlying an irradiation is described by n pairs of values (ϕ_i, t_i), where t_i defines the time interval during that the flux has the constant value ϕ_i. It is assumed that the relative flux history expressed by the (ϕ_i, t_i) is valid for all fuel positions.

At the conversion of the fission product density into burn-up it has to be considered that besides the above cited decay correction the production of the fission products may have several sources, e. g. the fission of U-235 and of meanwhile breeded Pu-239. The two generating processes are running independently from each other, but they are attached to the same absolute flux.

This is formally expressed by:

$$N = N(235) + N(239)$$

$$N(235) = f_1 (...., x \cdot \phi_i, t_i, ...)$$

$$N(239) = f_2 (...., x \cdot \phi_i, t_i, ...)$$

with N = number of fission product atoms

$N(235)$ = fraction from fissions of U-235

$N(239)$ = fraction from fissions of Pu-239

$x \cdot \phi_i$ = correct flux in the i-th time interval

x is a factor common to all ϕ_i, which will be computed according to the procedure described below. With the aid of this x the fission densities FT from U-235 and Pu-239 can be calculated formally by:

$$FT(235) = f_3\ (\ldots.,\ x \cdot \phi_i,\ t_i,\ \ldots)$$

$$FT(239) = f_4\ (\ldots.,\ x \cdot \phi_i,\ t_i,\ \ldots)$$

In the programme the balance equations for the number of fissions, the fission product density, and the numer of remained fuel atoms for all n time intervals are calculated in a recursive way. Concerning the m-th interval the following relationships apply:

a) fissioning of U-235

$$FT(235)_m = FT(235)_{m-1} + \sigma_{f5} \cdot U_{m-1} \cdot x \cdot \phi_m \cdot t_m$$

$$N\ (235)_m = N\ (235)_{m-1} \cdot e^{-\lambda t_m} + \gamma_5 \cdot \sigma_{f5} \cdot U_{m-1} \cdot x \cdot \phi_m \cdot \frac{1}{\lambda} \cdot (1 - e^{-\lambda t_m})$$

$$U_m = U_o - FT(235)_m \cdot (1 + \alpha)$$

b) breeding and fissioning of Pu-239

$$FT(239)_m = FT(239)_{m-1} + \sigma_{fg} \cdot Pu_{m-1} \cdot x \cdot \phi_m \cdot t_m$$

$$N\ (239)_m = N\ (239)_{m-1} \cdot e^{-\lambda t_m} + \gamma_9 \cdot \sigma_{f9} \cdot Pu_{m-1} \cdot x \cdot \phi_m \cdot \frac{1}{\lambda} \cdot (1 - e^{-\lambda t_m})$$

$$Pu_m = Pu_{m-1} \cdot (1 - \sigma_{f9} \cdot x \cdot \phi_m \cdot t_m) + \sigma_{c8} \cdot U_{238,0} \cdot x\phi_m \cdot t_m$$

with

$FT(\ldots)_m$ = number of all fissions at the end of the m-th interval

$N\ (\ldots)_m$ = number of fission product atoms at the end of the m-th interval

U_m, Pu_m = number of U-235 resp. Pu-239 atoms at the end of the m-th interval

$U_{238,0}$ = number of U-238 atoms at the beginning of the irradiation

$\gamma_{5,\ 9}$ = fission yield of the fission product with reference to U-235 resp. Pu-239

$\sigma_{f5,9}\sigma_{c8}$ = effective cross sections for fission and capture

α = σ_{c5} / σ_{f5}

The computation of the flux factor x is achieved by an iteration procedure. Starting with x = 1 one obtains a first value for the fission product density. By comparing this with the measured value one obtains an improved x value which yields a new fission product density a. s. o., accordingly

$$x_k = x_{k-1} \cdot \frac{N\ \text{measured}}{N^{(k-1)}(235) + N^{(k-1)}(239)}$$

with k = number of iteration step

The procedure converges usually in less than 10 iteration steps.

3.) Error of the decay correction caused by insufficient knowledge of the irradiation history

The instrumentation equipment of a reactor yields in normal cases only the mean flux history for the whole core, e. g. from the power recorder. The individual history of a fuel section, however, can be affected by various causes, e. g. by long-term variations of absorber rods or by short-term variations of control rods.

The error caused by these deviations from the given flux is obviously independent from the measuring errors. So the amounts of the single errors are calculated in a separate code. The flux history is divided into groups of about one month's duration. To each group a factor is assigned which applies to all Φ_i of the group. These factors are statistical and normal distributed numbers scattering about the mean value 1 with a given scatter of e. g. 20 %. They are generated by calling a function subroutine which forms random numbers into a normal distribution. By repeating the procedure many times and letting the fission product density always fixed, one obtains a field of burn-up values the scattering of which defines the above metioned error, which will be called "variation error".

4.) Forming the average and the total error of a measurement

In order to eliminate systematic errors it should be tried to take as many γ-peaks as possible into account at the burn-up determination from fission product spectra. To be excluded from averaging are those burn-up values of which the statistical error of the peak area exceeds 20 % or where the variation error exceeds 5 %. To the remaining burn-up values the following weights g_i are assigned:

$$g_i = \frac{1}{(\Delta I_i)^2 + (\Delta K_i)^2 + (\Delta V_i)^2}$$

with ΔI_i = statistical error of the i-th peak

ΔK_i = sum of errors of the nuclear data of the i-th peak

ΔV_i = variation error of the fission product

Averaging of the burn-up values with reference to their weights is resulting in

$$\overline{FT} = \frac{\Sigma\, g_i \quad FT_i}{\Sigma\, g_i}$$

with $\overline{FT}$ = average burn-up

$\overline{FT}_i$ = burn-up value from peak i

The error of mean is then resulting in

$$\Delta\, \overline{FT} = \sqrt{\frac{\Sigma\, g_i\ (FT_i - \overline{FT})^2}{\Sigma\, g_i \cdot (N-1)}}$$

with N = number of evaluated peaks.

The results are issued in printed protocols and punched into cards which serve as input data for a plotter code for drawing burn-up profiles. The average computing time per spectrum is about 0.5 min.

The described programme system has been written for the evaluation of γ-measurements on fuel rods of light water reactors. It can be used without great difficulties for other fuel probes provided that, beyond the material data, the irradiation history, the cross sections, and the self absorption of the probe are available.

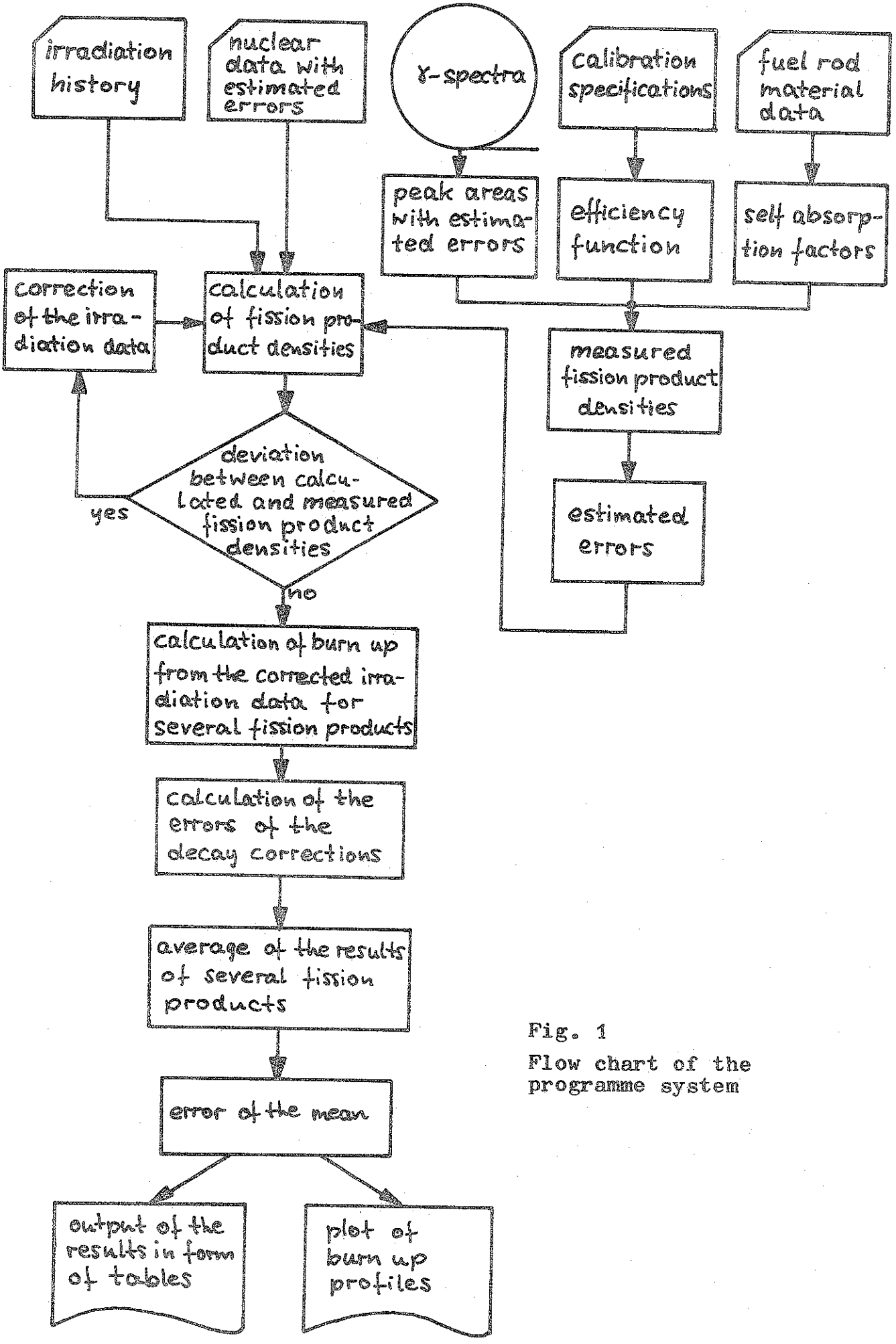

Fig. 1

Flow chart of the programme system