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Improved and Consistent Determination of the Nuclear Inventory of Spent PWR Fuel on the Basis of Cell-Burnup Methods Using KORIGEN

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Abstract

For safe handling, processing and storage of spent nuclear fuel, a reliable experimentally validated method is needed for calculating the composition of the irradiated fuel, its radioactivity and heat release, and the emitted radiation. A cell-burnup method was developed for pressurized water reactors which makes it possible to determine the nuclear inventory, taking into account reactor-specific characteristics such as cell geometry, initial enrichment, and reactor control. Routine calculations can be performed using the burnup and decay program KORIGEN with reactor-specific sets of cross sections from the cell-burnup procedure--burnup-dependent for actinides on the basis of recent Karlsruhe data. The method is validated by numerous comparisons between calculations and measurements.

A description of the input along with examples is included for applications using KORIGEN. Improvements with respect to ORIGEN as regards procedure and data are described, and results of KORIGEN, ORIGEN and ORIGEN2 calculations are compared with one another.

Fuel and waste inventories are given for BIBLIS-type fuel of different burnups.

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1. Introduction

Energy generation in traditional reactors is based on the neutron-induced fission of heavy nuclei, which takes place in a controlled chain reaction, and the conversion of the kinetic energy of the nuclear fragments and the energy of the radiation released during the fission process into heat. Naturally occurring uranium enriched with uranium 235, which can be fissioned with thermal neutrons, is primarily used as the fuel. The fission products and transuranic nuclides which are formed during burnup of the nuclear fuel are for the most part radioactive; they emit alpha, beta, gamma or neutron radiation, whereby the decaying nuclei have very different lives and the radiation quite different energies. The nuclear decay heat formed by the retardation of this radiation continues to heat the fuel for a long period after its removal from the reactor.

Radioactivity, the release of heat and radiation, the resulting plutonium which can be used as a fuel in fast reactors and also in thermal reactors, and the residual uranium are all controlling factors with respect to the handling of unloaded, spent fuel elements during interim storage and transport, reprocessing, waste treatment, and fuel refabrication. At all points in the nuclear fuel cycle it is necessary to ensure the safety of the staff, the systems and the process flow against criticality, radiation and excessive heating, and the facitilities and procedures being created must be designed quantitatively in accordance with the amounts and compositions of the materials produced.

The inventory of irradiated fuel is determined by the fresh fuel and the course of irradiation in the reactor. In the irradiation history, the dominant integral characteristic of which is the burnup reached, the time-place history of the neutron spectrum typical for each reactor plays a decisive role. Therefore the determination of the inventory in discharged fuel requires in principle the use of multidimensional spectral burnup procedures for tracing inventory variations during irradiation.

The present report is limited to pressurized water reactors (PWR), in which moreover the compensation of the burnup-related excess reactivity

by boric acid, homogeneously distributed in the coolant, and the control rods, which during full-load operation are largely withdrawn, open up the possibility for a simplified theoretical treatment.

Our goal is the creating and validation of a spectral burnup method with which questions regarding the out-of-pile behavior of spent PWR fuel can be answered as simply as possible but with sufficient accuracy.

For this purpose the use of successfully tested spectral programs such as HAMMER and WIMS for determination of the burnup-dependent neutron spectrum in the reactor unit cell and the use of ORIGEN as the burnup module is considered. Since the nuclear data implemented in these programs is in part out-of-date, the goal was to use the most recent Karlsruhe data and also ORNL data which were generated at ORNL in conjunction with the further development of ORIGEN into ORIGEN2 by Droff et al. The report focuses on the determination of data records for pressurized water reactors of the BIBLIS type.

Validation of the procedure and the data by comparison of the calculations with appropriate measurements was very important to us. Post-irradiation analyses of fuels from the PWR Obrigheim (KWO) and other PWRs are used for this purpose.

Research on the fuel inventory of boiling water reactors and pressurized water reactors with recycled plutonium is being carried out at the present time and will be described in a later report.

2. General and Specific Aspects of Inventory Determination in PWRs

In determining theoretically the nuclear properties of spent nuclear fuels at the various stations in the fuel cycle, taking into account the complex interactions between the fuel and the neutron field in the reactor, the main thing that must be established is the fuel inventory-- the quantity of uranium and plutonium isotopes, transplutonic elements and fission products--during reactor operation until fuel unloading. The basic factors which must be considered are the following:

-- the composition, quantity and configuration (geometry) of the fresh fuel, the cladding, and the structural materials;

-- the mode of operation, whereby the normal procedure in PWRs with a cycle length of about one year is to unload at the end of the cycle 1/3 of the fuel elements after three cycles residence time in the reactor and to replace them with fresh fuel elements, with partial rearrangement of the fuel elements remaining in the reactor;

-- the compensation of the burnup-related excess reactivity in the PWR, involving mixing boric acid in with the coolant, and the effect of control rods.

Consideration of both the spatial and chronological aspects of in-pile calculations demands basically that spatially multidimensional and time-dependent calculation methods be used, which must also describe the neutron thermalization in the lattice of thermal reactors, including heterogeneity effects.

When we trace the fuel after unloading from the reactor, we find that the inventory variation is reduced to the radioactive nuclear transformations of the radionuclides formed in pile in the fuel and the structural materials (the neutron reactions with the fuel are negligible, since at this point only neutrons from spontaneous fission and (α,n)-reactions are present). For the purpose of determining these changes, the in-pile calculation must naturally treat the formation of the more than one thousand radionuclides important for the out-of-pile behavior of the fuel and the structural materials.

For in-pile calculations carried out for the purpose of tracing reactivity it is generally sufficient to consider the uranium and plutonium isotopes, the highly neutron-absorbent isotopes of the fission products xenon and samarium and one pseudo-nuclide representing the remaining fission products,as well as the moderator, structural and control rod materials, along with boron. A total of about 25 nuclides must be considered in investigations of this type. If one also desires to combine the available procedures for these in-pile calculations with out-of-pile methods for the determination of radioactivity, heat release, etc., then the nuclides considered for in-pile calculations must be supplemented by the large number of radionuclides mentioned above.

The smallest units which are generally unloaded from power reactors, stored on an interim basis, transported and finally broken down for the purpose of reprocessing are complete fuel assemblies. Therefore, the nuclide inventory of individual irradiated fuel assemblies must be determined primarily. In PWRs these fuel assemblies consist of fuel rods of identical initial enrichment which are arranged in a regular square lattice by means of spacers and other structural parts. BIBLIS fuel assemblies, for example, contain 236 rods in a 16 x 16 lattice with 20 positions for control rods [1]. The primary factors which determine the instantaneous change in the nuclide inventory in the fuel assembly are the neutron flux density, the neutron cross sections, which describe the interaction between the neutron field and the materials, and the instantaneous inventory itself. The neutron cross sections are a function of the time-dependent energy distribution of the neutrons, and neutron spectrum. These relationships will be treated in greater detail in the following subsections. At this point we will only point out the following: beginning with the respective inventory, one also needs to know--in order to trace its further developement over time--the associated neutron flux density or even the power density of the neutron spectrum (equivalent to the neutron flux density) for the purpose of determining the associated cross sections.

With regard to the history of the neutron flux density in the fuel assembly during its residence time in the reactor, the so-called "irradiation history," one must assume, based on preliminary calculations that generally have to be performed, a standard cyclical operation which is specific to the reactor under consideration: planned full-power days under nominal power and zero-load periods. Since for such precalculations or predeterminations the properties of irradiated, typical fuel assemblies are generally sought and since fuel assembly utilization plans for power reactors are generally designed with the goal of achieving the flattest possible power distribution--for reasons of fuel economy, among other things--, we can assume in these calculations fuel assemblies with a constant total power over time under full load. In the case of fuel assemblies which are at the edge of the core during a cycle and are loaded with less power, the target burnup for all fuel elements is achieved by inserting them in the core interior during subsequent cycles (shuffling) where the power loading is correspondingly greater. It is complex and time-consuming to determine the exact distribution of the total power of a fuel assembly to the individual rods and the neutron spectrum in the rods, since for that purpose one would have to consider in detail the effects of adjacent elements with possibly higher or lower burnup, of control rods which in normal operation may have been inserted in the upper zone of the fuel assemblies, of water-filled guide tubes for control rods, and of the axial reflector, etc. In the case of fuel assemblies at the edge of the core the effect of the radial reflector is an additional factor. In order to arrive at reliable statements about the inventory of entire fuel assemblies in PWRs using less time-consuming procedures, we will consider that for full-power operation, which represents the major part of irradiation, the following exists:

-- in the case of a flat radial power distribution over the reactor, the power changes only slightly from rod to rod;

-- the radial periodicity of the neutron spectrum in the regular fuel rod lattice is only mildly disturbed by the water-filled control rod guide tubes;

-- the neutron spectrum and the power density in the PWR are locally independent over wide ranges in the axial direction; only in the lower and upper regions of the fuel assemblies do spectral distortions and

a drop in the power density result due to neutron leakage, backscatter on the axial reflectors, and--in the upper zone--absorption in the rod cluster control rods, which in normal operation are largely withdrawn from the core.

Determination of the inventory of typical PWR fuel assemblies can therefore be done in good approximation with a one-dimensional, time-dependent cell program: disturbances in the spectrum are essentially limited to boundary regions and do not have much influence on the behavior of central quantities. The burnup portion of this time-dependent cell program must be capable of treating the large number of nuclides which must be taken into account in connections with issues involving the external fuel cycle. It is therefore advisable to use an extensive irradiation code as the burnup phase, for example the appropriately modified ORNL program ORIGEN [2] or the UK program FISPIN [3], whereby the cross sections and material data used as input for the burnup calculation must be used in a manner as consistent as possible with the cell calculation (cf. Chapter 3). Suitable cell programs are HAMMER [4] or WIMS [5], both of which have been successfully tested over a period of years. If based on a good treatment of neutron thermalization and resonance absorption, as well as nuclear data sufficiently accurate for these purposes, they permit a largely reliable determination of the neutron spectrum for uranium lattices. The effective cross sections for the large number of nuclides in the burnup and decay phases can then be determined on the basis of a modern core data library, e.g. KEDAK [6] or ENDF [7] (cf. Chap. 3). In this regard, a sufficiently good description of the heterogeneity of the lattice must be guaranteed. The cell-burnup calculation should be performed in time intervals of different lengths. These time intervals are determined such that the neutron spectrum only changes slightly in each interval and the effective cross sections in each interval can be kept constant.

The cell-burnup method begins with the fresh fuel and maximum boron concentration for compensation of the high excess reactivity at the beginning of the cycle and extends to the end of the third (last) cycle with 5-7 time intervals per cycle, reaching the target burnup.

The boron concentration is reduced during each cycle from its maximum value of 900 ppm (BIBLIS) at the beginning to about zero at the end of the cycle. In this way the major part of the variation of the neutron spectrum, which is caused by the unloading of spent fuel assemblies and the loading of fresh fuel assemblies, is taken into account.

This theoretically supported concept for the inventory determination of typical PWR fuel elements, a concept which can be realized using simple and available procedures, must be validated experimentally. The comparison of calculations based on this concept with measurements from post-irradiation analyses of spent PWR fuel assemblies in Chap. 5 will show that this is possible: the results are very satisfactory.

Before a detailed description of the one-dimensional cell-burnup procedure developed at KfK is given in Chapter 3, several physical and method-related aspects which play a role in the determination of the variation in the nuclide inventory will be discussed in the following subsections. The specific focus in these sections is on the determination of the effective neutron cross sections in the fuel of the reactor unit cell for the burnup calculation, taking into account local and chronological influences which take effect via the neutron spectrum.

2.1 Basic Formulation of the Physical Problem

Change in the nuclide inventory in a fission reactor takes place primarily by absorption of neutrons and--as a consequence--fission of the compound nuclei formed during that reaction and/or the emission of gamma quanta and neutrons. A further change takes place due to the decay of the unstable nuclides formed by neutron reactions and their radioactive daughter nuclei. If we use $N^i(\vec{r},t)$ [nuclides/cm^3] for the number of nuclides of type i per cm^3 at location $\vec{r}$ at time t, the designation $(E, \vec{r},t)$ [n/cm^2 sec MeV] for the differential neutron flux density at r, t and the neutron energy E, and the designation $\sigma_a^i(E)$ [cm^2] for the differential effective cross section of the nuclide i for absorption of neutrons having the

energy E, then the variation over time of the concentration of the nuclide i at location $\vec{r}$ is given by

$$(2.1) \qquad \frac{d}{dt}N^i(\vec{r},t) = -\int_0^\infty dE\sigma_a^i(E)\phi(E,\vec{r},t)\cdot N^i(\vec{r},t)$$

$$+ \text{ production rate - decay rate}$$

The production and decay rates noted only superficially here will be given more precisely below.

Using $\phi(\vec{r},t) = \int_0^\infty dE\phi(E,\vec{r},t)$ as the (total) neutron flux density and the effective microscopic absorption cross section as customarily defined

$$(2.2) \qquad \sigma_a^i(\vec{r},t) = \int_0^\infty dE\sigma_a^i(E)\phi(E,\vec{r},t)/\phi(\vec{r},t)$$

we can write the diff. equation (2.1) as

$$(2.3) \qquad \frac{d}{dt}N^i(\vec{r},t) = -\sigma_a^i(\vec{r},t)\cdot\phi(\vec{r},t)\cdot N^i(\vec{r},t) + \text{ additional terms}$$

Following the reasoning given above, the neutron spectrum in the fuel rod can now be determined within the framework of our problem from the one-dimensional neutron transport equation (in cylinder geometry) in the periodic fuel rod lattice, i.e. with a one-dimensional cell calculation with periodic boundary conditions. With respect to the cell calculation one must guarantee that effects from the spatial and energetic self-shielding--caused by the heterogeneous structure of the cell and the resonance character of the neutron cross sections, primarily of the heavy nuclides--are given sufficient consideration. The local variable remaining in the cylindricized reactor unit cell--consisting of the central fuel rod, the fuel rod cladding, the moderator and the outer cell edge--is the distance r from the fuel rod axis. If we assume, as is customary, first of all a zone division--zone $z = \{r \mid r_z < r \leq r_{z+1}\ ;\ z = 1,2..\}$--so that in each zone the neutron flux density and the neutron concentration are only slightly a function of location, and secondly if we divide the range of variation of neutron energy into energy groups $g = \{E \mid E_{g+1} < E \leq E_g, g = 1,2,....G\}$ then the cell calculation carried out in the multigroup pattern in zone division yields the neutron spectrum as a zone multigroup spectrum $\Phi_z^g(t)$.

This multigroup spectrum can be used to calculate the single-group cross section (2.2) determining the nuclide variation. If, as usual, we define the following, namely that

$$(2.4)\quad \sigma_{a,z}^{i,g}(t) = \int_{E_{g+1}}^{E_g} dE\sigma_a^i(E)\Phi_z(E,t) \Big/ \int_{E_{g+1}}^{E_g} dE\Phi_z(E,t)$$

as the microscopic effective absorption cross section in the group g - $\Phi_z(E,t)$ designates the neutron spectrum in zone z-, then we obtain in zone notation the required single-group cross section as

$$(2.5)\quad \sigma_{a,z}^{i}(t) = \sum_g \sigma_{a,z}^{i,g}(t)\cdot\Phi_z^g(t) \Big/ \sum_g \Phi_z^g(t)$$

The diff. equation (2.3) goes over to the following, if zones of nuclide and neutron flux densities are introduced which vary slightly locally:

$$(2.6)\quad \frac{d}{dt}N_z^i(t) = -\,\sigma_{a,z}^i(t)\cdot\Phi_z(t)\cdot N_z^i(t) + \text{additional terms.}$$

In the above equation

$$(2.7)\qquad N_z^i(t) = \frac{1}{V_z}\int_{V_z} dV\, N^i(\vec{r},t)$$

where V_z is the volume of zone z and

$$(2.8)\qquad \phi_z(t) = \frac{1}{V_z}\int_{V_z} dV\phi(\vec{r},t)$$

whereby $\phi_z(t)$ is linked to the power density in zone z, $P_z(t)$ [MW/cm^3], by

$$(2.9)\quad \phi_z(t) = 6.24\cdot 10^{18}\cdot P_z(t) \Big/ \sum_j E_j\cdot\sigma_{f,z}^j(t)\cdot N_z^j(t)\,.$$

E_j[MeV] and $\sigma_{f,z}^j(t)$ [cm^2] are the fission energy release and the effective fission cross section of the fission nuclide j.

Given known zone power density $P_z(t)$, Equation (2.9) permits determination of the neutron flux density $\phi_z(t)$ in zone z required for inventory determination. If we use the known rod power density $P_{rod}(t) = \sum_{z_{rod}} P_z(t)$ then we obtain

$$(2.10)\quad \Phi_z(t) = 6.24\cdot 10^{18}\cdot P_{rod}(t) / \sum_{z'_{rod}} \sum_j E_j \sigma^j_{f,z'}(t)\cdot N^j_{z'}(t)\Phi_{z'}(t)/\Phi_z(t)$$

As weights there appear here the ratios of the zone fluxes $\Phi_{z'}/\Phi_z$, which according to

$$(2.11)\quad \Phi_{z'}/\Phi_z = \sum_g \Phi^g_{z'} / \sum_g \Phi^g_z$$

can be determined from the multigroup spectra of the cell calculation.

2.2 Aspects Regarding the Time Behavior of the Effective Neutron Cross Sections

The multigroup cross sections $\sigma^{i,g}_{a,z}(t)$ defined in (2.4) are determined by the differential cross section $\sigma^i_a(E)$ and the behavior of the neutron spectrum $\Phi_z(E,t)$, which is specific for each zone and a function of the inventory at the moment, with the energy within group g. The variations of the spectrum over time within the range of the actinide resonances are caused by the buildup of plutonium and transplutonium isotopes during irradiation and by changes in fuel temperature. Since great temperature changes only occur during startup and shutdown of the reactor, i.e. during periods which are short compared with the total irradiation time, one can assume a constant temperature in normal inventory calculations. The relatively slow buildup of transuranic nuclides over the entire irradiation period is expressed in the resonance range by a decrease in neutrons with energies which correspond to the resonance energies of these nuclides. This depression in the neutron flux increases with the concentration of resonance nuclides and leads in the case of resonance-overlapping groups to a weight of peak values of the resonance cross sections which decreases with time. The microscopic effective group cross sections of the heavy nuclides therefore become smaller in their resonance range due to this effect of energetic resonance self-shielding with increasing burnup.

For the lighter and medium-weight nuclei the energetic self-shielding plays a secondary role in the formation of multigroup cross sections: their differential cross sections form a relatively flat curve in the

range of resonances of heavy nuclei, and their (broader) resonances in the upper keV range are largely replaced by normal fine-group divisions. The multigroup cross sections of these nuclei may therefore be assumed to be burnup-independent. The single-group cross sections defined in (2.5) which are required for the burnup calculation are formed by condensation of the multigroup cross sections with the time-dependent multigroup spectrum $\Phi_z^g(t)$. The multigroup spectrum varies over time primarily in accordance with the boron concentration in the moderator and in accordance with the concentrations of the fission product nuclides Xe-135 and Sm-149 which are highly absorbent in the thermal energy range, as well as the heavy nuclides U-235, Pu-239 and Pu-241, which moreover influence the thermal spectrum by absorption resonances at 0.3 eV.

2.3 Aspects Concerning the Solution of the Burnup Equations

The foregoing discussion concentrated on the changes in nuclide concentration resulting from neutron absorption. For consideration of all nuclear reactions in the reactor including radioactive decays and the formation of fission products, the simple equation (2.6) must be supplemented by the other loss and production terms specific for each nuclide, which themselves are functions of the nuclide concentrations being sought.

One obtains the well-known burnup equations in general form

$$\frac{d}{dt} N_z^i(t) = -\left[\sigma_{a,z}^i(t)\cdot\phi_z(t) + \lambda^i\right]\cdot N_z^i(t) + \sum_{\substack{j=1\\ j\neq i}}^{I}\left[f_{ij}\cdot\sigma_{a,z}^j(t)\cdot\phi_z(t) + l_{ij}\cdot\lambda^j\right]\cdot N_z^j(t) \qquad i = 1,2,3.....I \tag{2.12}$$

with the given initial values $N_z^i(o)$. I is the total number of nuclides considered, currently 1176 in the Karlsruhe Method. The first line on the right side of (2.12) is the loss rate of nuclide i by neutron absorption of this nuclide and by its readioactive decay with the decay constant

λ^i [sec^{-1}], the second line is the production rate of the nuclide i by neutron absorption in other nuclides and by its radioactive decays. The functions f_{ij} are the probabilities for the formation of nuclides i from n-absorption in nuclides j in accordance with the nuclear reactions to be considered (n,γ), (n,2n), (n,3n), (n,p), (n,α) and fission (n,f). For the formation of U-236 from U-235 by neutron capture, for example, $f_{6.5} = \sigma_c^5/\sigma_a^5$; for the formation of the fission product nuclide Ru-106 during fission of U-235, the following applies:

$$(2.13)\quad f_{106,5} = Y_{106,5} \cdot \sigma_f^5/\sigma_a^5,$$

where σ_c^j and σ_f^j as well as σ_a^j are calculated according to (2.4) and (2.5) with Φ-weighting, and the effective fission product yields y_{ij} with $\sigma_f \cdot \Phi$ weighting. The formation of the fission products can be treated equivalent to (2.13) by introducing effective, Φ-weighted $(Y\sigma_f)_{ij}$. The above example then reads

$$(2.14)\qquad f_{106,5} = (Y\sigma_f)_{106,5}/\sigma_a^5$$

For the Karlsruhe cell-burnup method, $\sigma_f\Phi$-weighted yields are currently being used.

The functions l_{ij} are the probabilities for the formation of nuclides i during decay of the nuclides j with the decay constant λ^j in accordance with splitting in β^-, β^+ and α decays and, for metastable nuclei (e.g. Am-242m) into γ-transitions to the basic state. The burnup equations (2.12) form a linear system of differential first-order equations in time with some coefficients which are variable over time having the form $\sigma_{x,z}^i(t) \cdot \phi_z(t)$. The method customarily used for numerical solution of this system consists in dividing the irradiation time, in which $\phi_z(t)>0$, into intervals (t_k, t_{k+1}) such that in each interval $\sigma_{x,z}^i(t)$ and $\phi_z(t)$ are slightly variable functions of t and $\sigma_{x,z}^i(t_k)$ can be replaced by its initial value $\sigma_{x,z}^i(t_k)$. The time variation of ϕ in the time interval k is then logically taken into account by the fact that $\phi_z(t)$ in this interval is already replaced in the burnup equations by its mean value over time

$$(2.15)\quad \overline{\phi_z^k} = \frac{1}{\Delta t_k} \int_{t_k}^{t_{k+1}} dt\ \phi(t)\ ,\Delta t_k = t_{k+1} - t_k$$

By means of this procedure a good description of the nuclide buildup by neutron absorption is obtained, for the exact solution of the equation valid for limitation to absorption, $\frac{d}{dt}N(t) = -\sigma_a \cdot \phi(t) \cdot N(t)$, is $N(t) = N(o) \cdot \exp(-\sigma_a \cdot \bar{\phi} \cdot t)$, and with regard to neutron flux density this expression explicitly depends only on the latter's mean value over time $\bar{\phi}$.

The burnup equations (2.12) in this way go over into a system of diff. equations with constant coefficients for each time interval and each zone, giving appropriate consideration to the time behavior of the neutron flux density; for numerical solution of these equations, even with regard to the large number of nuclides which are finally important in out-of-pile tests, the ORNL program ORIGEN [2] can be used zone-wise beginning with the burnup module.

The mean value over time $\overline{\phi_z^k}$ of the neutron flux density in the time interval k and in zone z is obtained in satisfactory approximation from Eq. (2.9) given constant zone power and a fission cross section which is only slightly variable over time by using a Taylor development of $1/\sum_j E_j \cdot \sigma_{f,z}^j \cdot N_z^j(t)$ at the respective interval beginning until the 2nd order in t, whereby the time derivations of $N^j(t)$ which appear are expressed by the burnup equation.

The simplification used in ORIGEN involving a uniform energy release of 200 MeV per fission for all fission nuclides has proved to be inadequate for many cases, since for PWR the concentrations of important plutonium and transplutonium isotopes are calculated too high [8]. In the Karlsruhe cell-burnup method, therefore, a nuclide-related fission energy release including the heat which is generated by retarding gamma radiation from (n,γ) processes is used--cf. also Section 4.1.2.

3. Effective Neutron Cross Sections for Actinides with KfK Data and Methods

From the statements in Chapter 2 it follows that one-dimensional cell-burnup procedures are suitable for answering questions regarding the external fuel cycle for PWR. A corresponding program was developed at KfK:

-- for calculation of the neutron spectrum, the cell program HAMMER [4] was selected for historical reasons;

-- for the burnup calculation, a suitably modified version of KORIGEN was used, since it can take into account a large number of nuclides;

-- the cell-burnup method was completed by linking HAMMER and the modified KORIGEN version to the program system HAMKOR [9].

HAMMER has its own nuclear library with data which are based largely on ENDF/B-II [10]. So that more recent nuclear data for a large number of nuclides could be used in the burnup calculation and nonetheless a major change in the HAMMER program could be avoided, the following solution was followed. From the HAMMER cell calculation only the neutron spectrum is used. This is largely reliable due to the good treatment of the thermalization of the neutrons in the moderator. By accessing the Karlsruhe nuclear data library KEDAK [6], effective multigroup cross sections are calculated--outside the framework of the HAMMER program. After condensation with the HAMMER neutron spectrum to one energy group, these are then made available for the burnup calculation. In the following subsection, 3.1, the procedure will be described in greater detail.

The effective cross sections determined for a specific reactor during a HAMKOR calculation are stored permanently in a library, the cross section file KORIDATA. The stand-alone program KORIGEN has the capacity to use these cross sections, which vary with burnup for the actinides. Thus simple inventory calculations for the specific reactor are possible with KORIGEN which are consistent with the corresponding cell inventory calculations.

So that consistent inventory calculations with KORIGEN can be made possible for a great number of reactor, fuel element and fuel types, a comprehensive library with corresponding cross section sets is desirable. In this sense the program system HAMKOR functions as an instrument for the generation of these cross section sets.

3.1 Generation of Reactor-Specific, Burnup-Dependent Cross Sections for Actinides

3.1.1 The HAMKOR Program System

The HAMKOR calculation (schematicized sequence, Fig. 3.1) is accomplished in given time intervals, beginning with a cell calculation for t=0: HAMMER first calculates the neutron spectrum for the fresh fuel (in 83 energy groups) in accordance with the geometric data for the unit cell and the initial nuclide densities including the boron content of the moderator.

For t = 0 HAMKOR then calculates the effective cross sections for n-capture and nuclear fission. Calculation of the effective cross sections is carried out in a 83-group structure (in accordance with the HAMMER group structure). The thermal energy range (up to 0.625 eV), in particular the range around 0.3 eV where U-235, Pu-239 and Pu-241 have thermal resonance, is will resolved with 30 energy groups so that no additional resonance treatment is necessary. In the epithermal range (E > 0.625 eV), on the other hand, the resonances are no longer resolved in the given group structure, so that an additional resonance treatment is necessary in order to consider the resonance absorption which sometimes varies greatly with burnup.

The resonance treatment is carried out on the basis of f-factor formalism according to Bondarenko (σ_0 concept) [11], whereby the effective group cross section σ_x^g can be represented, assuming a constant total collision density in the group g, as the product of the infinitely diluted cross section $\sigma_{x,\infty}^g$--i.e. the energetic mean value of $\sigma_x(E)$ in group g--and the self-shielding factor $f_x^g(\sigma_0,T)$. Energetic resonance self-shielding

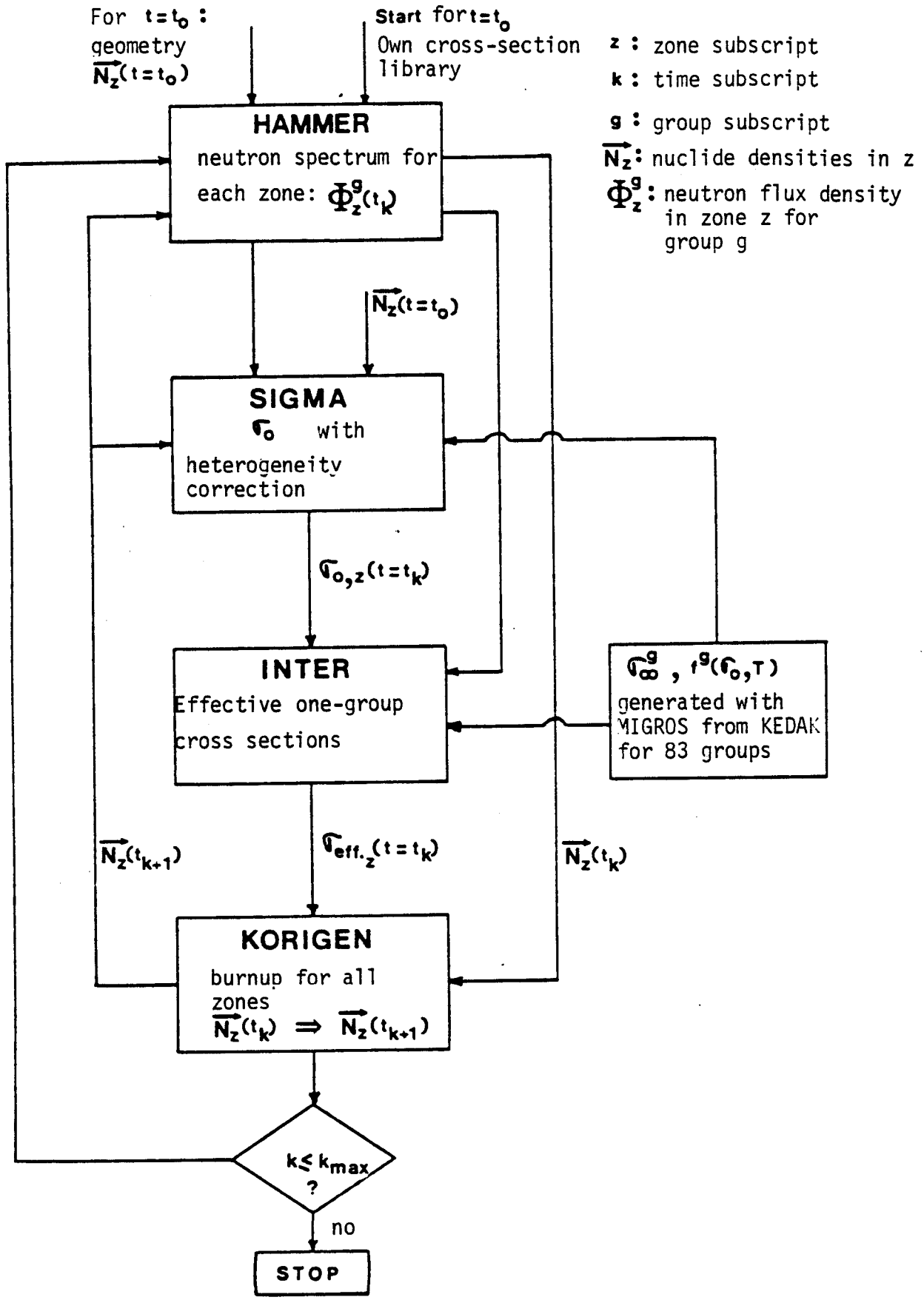


Figure 3.1. Schematic representation of the program HAMKOR.

is here described by the f-factor which depends on the nuclide mixture by way of the background cross section σ_0.

The σ_0-concept was used for treatment of resonance absorption because of its simplicity, the experience acquired with it at KfK, and the availability of necessary programs and program parts (especially programs for processing KEDAK data), although it is not absolutely suitable for thermal reactors since at low neutron energies the condition of constancy of total reaction rate in each energy group is not satisfied under certain conditions. This applies to the range of 1-eV resonance of the absorption cross section of Pu-240, which is therefore subjected to special treatment (see the end of this section.

F-factor formalism permits determination of the quantities necessary for calculation of the effective group cross sections--$\sigma^g_{x,\infty}$ and $f^g_x(\sigma_0,T$--independent of the cell burnup calculation. For this purpose the KfK program for generation of microscopic group constants, MIGROS3 [12], is used, which calculates the infinitely diluted group cross sections and the f-factors (for various σ_0 and T values) from the KEDAK data and stores them in a file. This file serves as the nuclear data base for all HAMKOR calculations. It is therefore not necessary to recalculate the group constants $\sigma^g_{x,\infty}$ and $f^g_x(\sigma_0,T)$ in each case from the KEDAK data during the HAMKOR calculation.

HAMKOR first calculates the background cross section for the actual nuclide mixture and the prevailing operating temperature. For consideration of the resonance absorption, which is reduced in heterogeneous configurations in comparison with the homogeneous mixture, σ_0 is subjected to a newly derived heterogeneity correction [13]. The lattice effect, finally, is taken into account by a Dancoff correction using the method described by Sauer [14].

The corresponding f-factor for each nuclide and each reaction is now sought in the HAMKOR library or interpolated for the actual background cross section σ_0. By multiplication with the infinitely diluted group cross section, the effective group cross section is finally obtained. Using the neutron

spectrum calculated by HAMMER (for t = 0), the effective 83-group cross sections are condensed to four and finally one group. The one-group cross sections calculated in this way are transferred to KORIGEN for inventory calculation and stored in a library for later inventory calculations. KORIGEN calculates on the basis of these cross sections the new nuclide densities for the first time interval which is selected as ca. 5 days in order to take into account establishment of Xe equilibrium. The actual nuclide densities at the end of the first time interval, including the given boron concentration of the moderator, are transferred to HAMMER for recalculation of the spectrum--at the beginning of the 2nd time interval. There follows a new calculation of the effective cross sections and, based on these, an inventory calculation using KORIGEN for the 2nd time interval. The nuclide densities at the end of the 2nd time interval are again transferred to HAMMER for calculation of the neutron spectrum--at the beginning of the 3rd time interval--and so forth. This procedure is continued until the end of the 3rd irradiation cycle. Each cycle is divided into ca. 5 time intervals; for the type BIBLIS, for example (3 cycles at 333 d full load and 40 d zero load) the time intervals are 5, 45, 83, 100 and 100 [sic] d.

The effective one-group cross sections generated with this time-dependent method include the neutron spectrum which varies over time and also the energetic resonance self-shielding which generally increases with increasing burnup. These effects are treated in very refined fashion in an 83-group structure; the inventory calculation is carried out in a single energy group. However, this does not represent an approximation, since during buildup and depletion of the nuclides in the reactor only total reaction rates and, accordingly, cross sections averaged over the entire neutron spectrum enter into the burnup equations.

The total cross section set is stored in the cross-section library KORIDATA and is therefore available for simpler but consistent inventory calculations with the stand-alone version of KORIGEN.

As already explained above, it is necessary for treatment of resonance absorption due to the formalism used to resolve well the 1 eV resonance

of Pu-240. Since the group structure, however, is given by the cell program used--HAMMER--(here only one group lies above the 1 eV resonance), a special treatment for the capture cross section of Pu-240 is carried out. For this purpose a one-time WIMS calculation is performed: the cell program WIMS resolves the 1-eV resonance of Pu-240 with 12 energy groups very easily. The behavior of the effective capture cross section of Pu-240 as a function of the burnup is now adjusted to the behavior calculated using WIMS, assuming the KEDAK value for the fresh reactor (cf. Fig. 3.2). This procedure guarantees, on the one hand, that the most recent Karlsruhe nuclear data from KEDAK can also be used for Pu-240 and, on the other hand, that the resonance absorption of Pu-240 will be treated well even at high burnups. The resulting capture cross section of Pu-240 is supported by the comparison with experimental post-irradiation analyses (Chap. 3).

The time-dependent cell method HAMKOR described above was developed as the first step towards a comprehensive program system for the fuel cycle of thermal reactors, in order to obtain the possibility, in a suitable time frame, of performing consisten inventory calculations on the basis of up-to-date nuclear data. The time-dependent cell method, which was to be newly developed in conjunction with the fuel cycle program system, is integrated modularly into the Karlsruhe program system KAPROS [15]. The module for the cell calculation will be able to have direct access to the KEDAK data and calculate the effective cross sections, both for the spectral calculation and for transfer to the burnup module, using a suitable formalism. The burnup module will be created as before on the basis of KORIGEN. Until this program system is completed, the HAMKOR system can continue to be used as an instrument for providing effective cross sections. The comparison with experimental post-irradiation analyses in Chap. 5 will show that consistent inventory calculations based on the cross sections calculated with HAMKOR are suitable for determining the nuclear inventory of spent PWR fuels reliably.

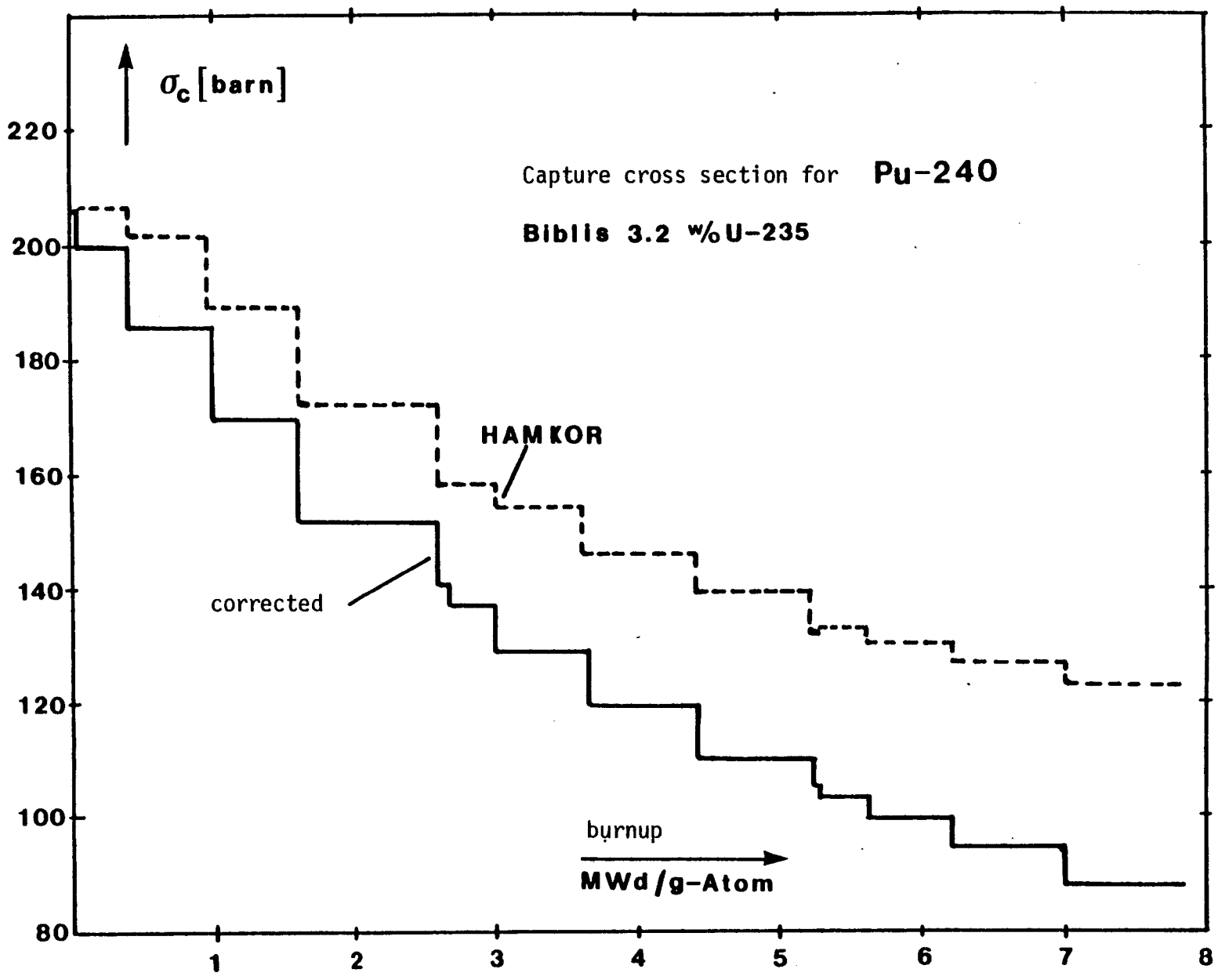


Figure 3.2. Burnup dependence of the capture cross section of Pu-240.

3.1.2 Burnup-Dependent Cross Sections for Biblis-Type Pressurized Water Reactors

BIBLIS B was selected as the typical representative of large German pressurized reactors and also agrees in the essential data with the subsequent plant, Grafenrheinfelden. The important reactor data are given in Tab. 3.1. Uranium fuel with an initial enrichment of 3.2, 3.4 and 3.6 w/o was analyzed. The discharge burnups on which the analysis is based are 33, 36 and 40 GWd/t U. Irradiation occurred in 3 cycles of 333 d each with zero load periods of 40 d. For boric-acid control, a boron concentration in the moderator varying from 900 ppm to 0 per cycle was used.

Burnun-dependent cross sections were generated for the actinides U-235, U-236, U-238, Np-237, Pu-238-242, Am-241, Am-243 and Cm-244 on the basis of KEDAK data. The capture cross section of Pu-240 was corrected in each case in accordance with the method described in Sec. 3.1.1. These cross section sets, along with data records for the PWR Trino Vercellese and for KWO [Obrigheim Nuclear Power Plant], which were created in connection with post-irradiation analyses, are implemented in the KORIGEN cross section library, KORIDATA. The data records of the KORIDATA library are tabulated in Appendix C. The designation for the data records is given in Table 3.2. The cross sections of important actinides for the BIBLIS case, 3.2 w/o initial U-235, burnup 33 GWd/tU, are represented graphically in Appendix C.2 and are discussed in the following section in comparison with the corresponding ORNL cross sections.

Unique, reactor-specific cross sections are being developed for fission products. At the present time an ORNL data record is used for fission products in inventory calculations: constant one-group cross sections for a mean burnup state over time [16].

Table 3.1. BIBLIS B reactor data [1]

Thermal power	3733 MW
Fuel	UO_2
U-235 initial enrichment	3.2 w/o
Mean power density of reactor	36.7 MW/t
Cell:	
Fuel rod diameter	10.75 mm
Cladding wall thickness (Material: Zircaloy)	.725 mm
Volume ratio of moderator to fuel	2.06
Mean Temperatures [°C]:	
Fuel	833
Coolant	310
Cladding	344

Table 3.2. KfK data records in the KORIDATA library

Type	Initial U-235 Enrichment [w/o]	Burnup [GWd]	Designation
BIBLIS B	3.2	33	33P3U-32-KFK-BIB
"	3.4	36	36P3U-34-KFK-BIB
"	3.6	40	40P3U-36-KFK-BIB
KWO *	3.13	30	30P3U-31-KFK-KWO
"	3.0	38	38P3U-30-KFK-KWO
"	2.83	38	38P3U-28-KFK-KWO
Trino	3.13	30	30P4U-31-KFK-TRI

* KWO = Kernkraftwerk Obrigheim [Obrigheim Nuclear Power Plant]

Table 3.3. Cell data for the ORNL reference PWR [16]

Fuel rod diameter	8.348 mm
Cladding wall thickness	.583 mm
Moderator/fuel volume ratio	1.88

3.2 Comparison of KfK and ORNL Cross Sections and Procedures for Generating Them

For comparison of the actinide cross sections generated using KfK methods and data, corresponding effective one-group cross sections from Oak Ridge are available, which are also implemented in the KORIDATA file and can be used alternately by KORIGEN. The cross sections generated at ORNL differ both in the differential nuclear data on which they are based (ENDF/B-IV) and in the method used to generate them and the reactor model of the KfK cross sections.

3.2.1 The ORNL Model for the Generation of Effective Actinide Cross Sections

The ORNL model [16] is based on a large PWR, approximately comparable to the BIBLIS type, with a thermal power of 3800 MW, which is run in 3 cycles of 293.3 d full load and 71.7 d zero load each. The final burnup , as in the BIBLIS type, is 33 GWd/tU with 3.2 w/o enriched uranium. The cell dimensions are somewhat smaller than in the BIBLIS type (cf. Table 3.1 and 3.3).

The ORNL model places great value on consideration of the effect of differently enriched zones at first loading, rearrangement, and reloading of the reactor. One third of the total inventory is unloaded per cycle and reloaded with shuffling of the fuel. The reloading fuel has an initial enrichment of 3.2%, the first loading 2.1%, 2.6% and 3.1% for each 1/3 of the total load.

The fuel is assigned a mean burnup state of 5 GWd/tU during its first cycle in the reactor, 16 GWd/tU during its second cycle, and 27 GWd/tU during its third cycle. Effective 84-group cross sections are generated with given nuclide concentrations for these burnup states, with which cross sections the cell spectrum is calculated in 84 groups. The 84-group spectrum is used to condense the 84-group cross sections to 5 groups. These effective 5-group cross sections are viewed as representative with respect to the mean energetic resonance self-shielding per cycle. This

means that the energetic resonance self-shielding during each cycle is kep constant (cf. in this connection the KfK method, Sec. 3.2.2 and 3.1).

With the effective 5-group cross sections a diffusion calculation is now carried out, beginning with the fresh reactor (t = 0), for a quasi-zero-dimensional model--three 0.001 cm thin plates with 3.1, 2.6 and 3.1% enrichment--which determines a spectrum in 5 groups that is used to condense the 5-group cross sections to one group. The resulting cross sections are considered to be the effective one-group cross sections at time t=0 which were sought. Due to the condensation spectrum used, they contain effects from the different enrichment zones.

For each enrichment zone there follows now a burnup calculation which provides the nuclide concentrations at the end of the first time interval t_1. Thus a spectrum calculation in 5 groups is again performed, which is followed by a condensation of the representative 5-group cross sections to one group. A burnup calculation for the next time interval follows, and so forth. This procedure is continued until the end of the first cycle with more or less equal time increments and then--taking into account fuel reshuffling, unloading and reloading--continued to the end of the tenth cycle. At the end of this tenth cycle the composition of the reload fuel unloaded from the reactor with a burnup of 33 GWd/tU (initial enrichment 3.2 w/o U-235) is practically steady-state (equilibrium cycle). The effective one-group cross sections of the actinides obtained in this equilibrium cycle are considered to be the cross sections which are to be used in the improved ORIGEN code for batch burnup calculations in equilibrium operation.

The variable boric acid concentration in the moderator for reactor control--decreasing from about 900 ppm boron at the beginning of the cycle to 0 at the cycle end--which is customary in PWRs and which sometimes affects the burnup-dependent actinide cross sections rather severely, is not considered by the ORNL model; for simplicity's sake a constant boron concentration of 550 ppm is assumed [16].

3.2.2 Comparison of the KfK and ORNL Methods for Generating Effective Actinide Cross Sections

The KfK and ORNL methods have different emphases:

KfK Method: consistent treatment of energetic resonance self-shielding and of variation of the spectrum over time due to control with boric acid and the buildup of fission products and transuranic elements

ORNL Method: consideration of the discontinuous effect of fuel zones with different burnup on the spectrum during loading, reshuffling and unloading.

The two methods are compared with one another in detail in the following.

Energetic Resonance Self-Shielding:

-- is consistently carried along in the KfK method; this is made possible by coupling the inventory calculation with the cell calculation: the actual nuclide densities are transferred to the cell program at the beginning of each time interval, whereby the correct self-shielding is calculated for each time increment.

-- In the ORNL method only one cell calculation with given and therefore inconsistent nuclide densities is performed for each cycle, i.e. a mean value is calculated for the energetic resonance self-shielding which is viewed as valid for the entire cycle.

Spectrum Variation Due to Control Using Boric Acid:

-- In the KfK method the boric acid concentration of the moderator is varied per cycle from ca. 900 ppm to 0; this sometimes has a strong effect on the effective cross sections, since the spectrum is relatively hard at the beginning of a new cycle due to the high boron concentration and becomes softer in accordance with the decreasing boron concentration. This effect is superimposed on the hardening of the spectrum from the buildup of transuranic elements and fission products.

-- This spectrum variation is only considered in the ORNL method as an average for programming reasons: constant boron concentration of 500 ppm in each cycle.

Consideration of the Zones of Different Burnup Resulting from Discontinuous Loading, Reshuffling and Unloading:

-- is considered by the KfK method only indirectly by way of the variable boron concentration of the moderator: the per-cycle boron concentration, which is at first high and then decreases, reflects the variation over time of the excess reactivity of the entire reactor: at the beginning of a cycle the high excess reactivity, caused by the partial reloading of the reactor with fresh fuel elements, is compensated for by increased mixing in of boric acid in the moderator; during the cycle the boric acid concentration is reduced in accordance with the decrease in excess reactivity caused by progressive burnup.

-- is simulated by the ORNL method on the basis of the small-plate model (cf. Sec. 3.2.1) rather extensively (coupled diffusion burnup calculation in 5 energy groups until the tenth cycle). It appears questionable whether this simple plate model adequately describes the interaction of fuel elements of different burnup.

In summary one can conclude that the KfK method has clear advantages over the ORNL method: the effects which substantially affect the neutron spectrum and enter into the effective one-group cross sections and also sometimes strongly influence their behavior with increasing burnup (energetic resonance self-shielding, control using boric acid, hardening of the spectrum due to fission products and transuranic elements) are treated consistently by the KfK method. The ORNL method treats these effects in part inconsistently and more approximately and in a simple model focuses on effects resulting from zones of different burnup, which results in costly computer procedure.

3.2.3 Comparison of the KfK Cross Sections of the BIBLIS Type (3.2 w/o U-235) with the ORNL Cross Sections

Decisive for the behavior of the effective cross sections as a function of burnup is the history of the microscopic cross sections with energy and the neutron spectrum which is variable with burnup. The cross sections of the most important actinides analyzed can be broken down into three groups:

I) Actinides with cross sections in the thermal energy range ($E \leq 0.5$ eV) which have resonance ("thermal resonance"): U-235, Pu-239, Pu-241, Am-241 (cf. Table 3.4 and Fig. 3.3). The cross sections of these nuclides have high 2200 m/s values (cf. Table 3.4 and Fig. 3.3), so that for thermal reactors the effective one-group cross section is determined by the curve of the microscopic cross section in the thermal energy range.

II) Actinides with microscopic cross sections which have the first resonance in the epithermal energy range ($E \gtrsim 0.5$ eV), which resonance determines the one-group cross section for thermal reactors: U-236, U-238, Pu-242, Cm-244. The thermal cross section (2200 m/s value) is very small; the resonance energy is several eV (Tab. 3.4 and Fig. 3.3).

III) Special Cases:

a) Pu-240 and Am-243, the microscopic capture cross sections of which have resonances at 1 eV which determine the effective one-group cross section for thermal reactors.

b) Pu-238, the capture cross section of which has a high 2200 m/sec value, while the first resonance is at ca. 3 eV (Tab. 3.4): the effective one-group cross section is determined by the 1/v fraction.

In the following cases I, II and III are discussed in detail.

I) Actinides with Thermal Resonance

The effective one-group cross sections of these nuclides are distinguished by a sharply increasing resonance self-shielding with progressive burnup and by great sensitivity to spectral changes resulting from boric acid

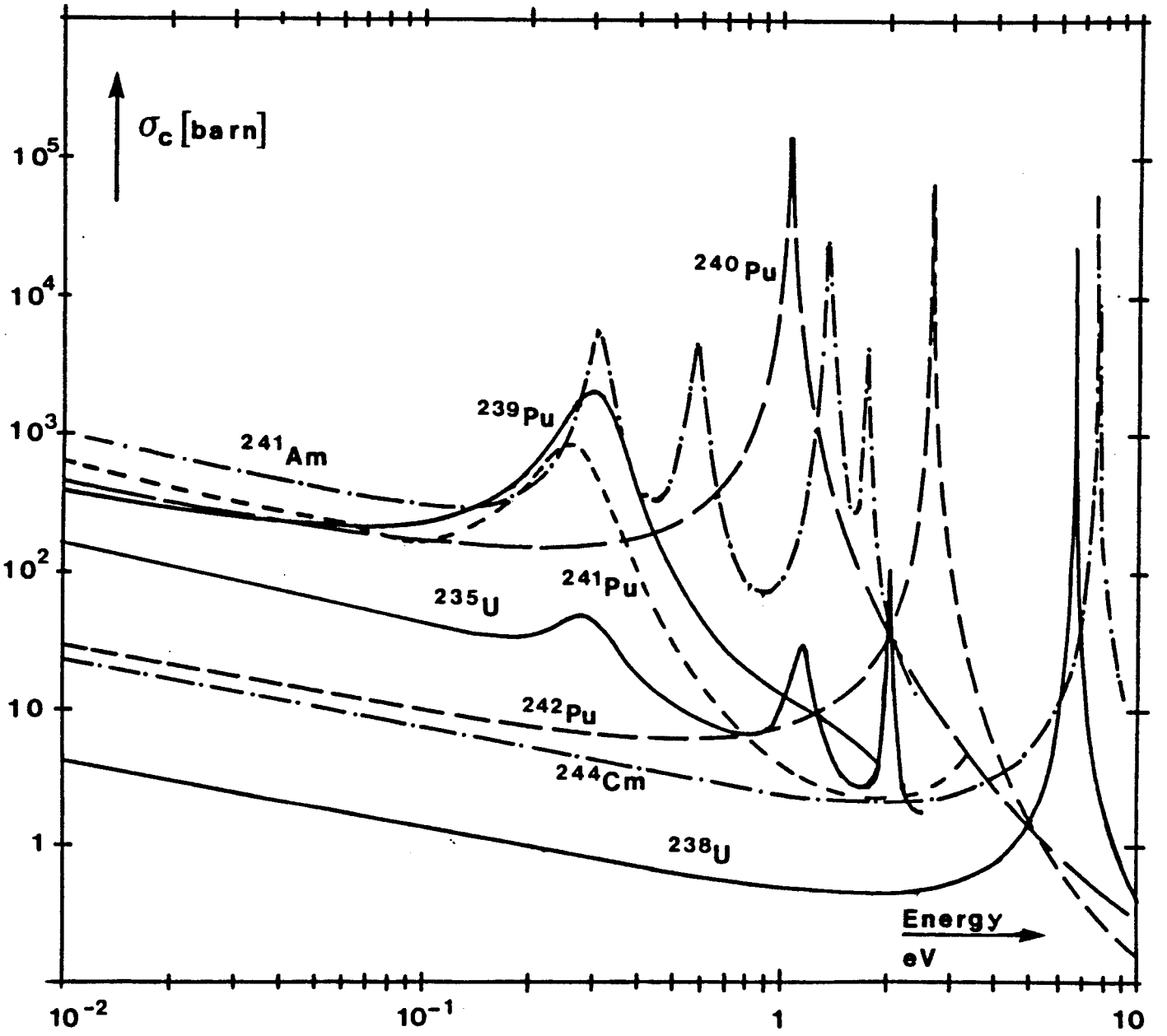


Figure 3.3. Capture cross sections for the most important actinides in the energy range up to 10 eV.

Nuclide	Category	Resonance energies [eV] 1st resonance	2nd resonance	2200 m/s-cross section [b] capture	fission	Resonance integral* [b] capture	fission
U235	I	0.273	1.14	99.4	588	168	238
236	II	5.45	29.7	5.2	-	346	-
238	II	4.393	6.67	2.7	-	279	-
Np237	I	0.489	1.32	169.1	-	642	
Pu238	III	2.885	9.975	542	17	162	31
239	I	0.297	7.82	271	745	194	280
240	III	1.058	20.46	290	-	7977	-
241	I	0.26	4.276	365	1012	193	582
242	II	2.65	14.6	19.0	-	1124	-
Am241	I	0.308	0.577	610	3.2	1452	-
243	III	0.420	0.983 1.356	79.3	-	1847	-
Cm244	I	7.668	16.79	14	1.0	637	-

$$* \int_{0.5eV}^{15MeV} \sigma(E) \frac{dE}{E}$$

Table 3.4. Resonance energies, resonance integrals and 2200 m/sec cross sections for the actinides analyzed.

control and buildup of fission products and transuranic elements. The general trend of these one-group cross sections can be characterized as follows for the KfK method.

Beginning with the greatest value at time t = 0 (fresh fuel), the cross section drops during the first cycle--more or less sharply depending on the nuclide--due to the sharply increasing hardening of the spectrum resulting from the buildup of transuranic elements and fission products. In the first cycle this effect is so strong that the decreasing boron concentration, which would result in a progressively softer spectrum, is not noticeable in the spectrum. At the beginning of the second cycle the spectrum becomes even harder due to the boron concentration, which is now high: the cross section drops discontinuously. During the second cycle the spectrum becomes softer in accordance with the decreasing boron concentration. This effect competes with the resonance shielding which increases depending on the nuclide (in the case of U-235 it decreases): the cross section can rise (σ_f,σ_c of Pu-241) but can also remain more or less constant, i.e. the opposing effects are more or less balanced (σ_f,σ_c of Pu-241). At the beginning of the third cycle the cross section again drops discontinuously, and then rises again during the cycle as the spectrum becomes softer (in this case the "boron effect" outweighs the increasing self-shielding reaching saturation). Common to the one-group cross sections of the actinides with thermal resonance is also the strong dependence on initial U-235 enrichment. This is explained by the fact that the thermal neutron absorption in U-235 is increased with a greater initial concentration of U-235 and that the neutron spectrum therefore becomes harder.

The qualitative progression of the corresponding ORNL cross sections is determined by the method used: jumps at the beginning of the cycle due to the altered energetic resonance self-shielding, smaller variations within the cycles due to the altered fuel composition. Generally the changes in cross sections within the cycles are so small that they could be disregarded. Characteristic is the rise in the cross section of U-235 at each cycle beginning: the decreasing shielding in U-235 is not covered up by the boron effect in this case. The decrease in the U-235 cross

section within the cycles is a logical result of the method used: the increasing hardening of the spectrum (fission products, transuranic elements) brings about a decrease in the effective U-235 cross section since the boron concentration and the resonance self-shielding are kept constant. With consistent treatment of the resonance shelf-shielding and control with boric acid, an increase in the cross section within the cycles is possible.

II) Actinides with the First Resonance in the Epithermal Range

The characteristic of the effective cross sections of these nuclides is their insensitivity to changes in the thermal spectrum (e.g. hardening or "boron effect"). The cross section is more or less constant over the entire burnup range (σ_c of U-238, Cm-244). Weak changes in the cross section are caused by the varying resonance self-shielding (σ_c of Pu-242). However, this is very slight, since the resonances lie in the 1/E range of the spectrum and are loaded with correspondingly low weight.

III) Special Cases

a) Pu-240 and Am-243

The capture cross section of Pu-240 has a highly pronounced resonance at 1 eV which determines the effective one-group cross section almost exclusively. This resonance is not noticeably affected by changes in the thermal spectrum. Since this resonance lies on the lower limit of the 1/E spectrum, however, and thus is loaded with great weight, and is furthermore very pronounced, a sharply increasing resonance self-shielding is evident sich increasing Pu-240 concentration: with KfK σ_c (Pu-240) drops from an initial level of ca. 200 b to 90 b with a burnup of 33 GWd/tU.

The strong self-shielding is also expressed in the ORNL method: σ_c (Pu-240) makes a sharp jump at the beginning of the cycle. Within a cycle it hardly changes at all, since the energetic resonance self-shielding is kept constant within the cycle in accordance with the ORNL model.

The effective capture cross section of Am-243 is determined by the resonance at 1.356 eV; the two resonances below 1 eV are not important. The behavior of the one-group cross section is therefore similar to that for σ_c (Pu-240): σ_c drops slightly with increasing burnup due to the slightly increasing resonance self-shielding. The effect is considerably less than in the case of Pu-240 due to the smaller Am-243 concentration and the less pronounced resonance.

b) The capture cross section of Pu-238 represents an exception, since its 2200 m/s value is relatively high but the first resonance does not occur until ca. 3 eV (Tab. 3.4). Accordingly, a slight dependence on the varying boron concentration is evident in the effective one-group cross section.

Quantitatively, it is difficult to compare effective KfK and ORNL cross sections of Category I since besides the different cell dimensions other reactor parameters which have a great effect on these cross sections--such as uranium density, for example--differ. The KfK cross sections for U and Pu isotopes (with the exception of U-236 and Np-237) are based on KEDAK-3 (1979) [17], and the ORNL cross sections on ENDF/B-IV (1974). The KfK cross sections of the Am isotopes and Cm-244 are based on the most recent Karlsruhe evaluation (1981). The effective capture cross sections of Am-243 and Cm-244, which are hardly sensitive to different reactor parameters, differ greatly between KfK and ORNL; the KfK value for σ_c (Am-243) is ca. 30% above the corresponding ORNL value, and that for σ_c(Cm-244) ca. 50% below the corresponding ORNL value. Both cross sections are determining factors for the Cm-244 concentration. With the KfK values better agreement with experimental post-irradiation analyses can be achieved for Cm-244 (Chap. 5). More recent effective cross section sets for 4.1% enriched uranium fuel obtained by Croff et al. [18], which are based on ENDF/B-V, contain values for the capture cross sections of Am-243 and Cm-244 which are consistent with the KfK values.

4. Modifications in the Stand-Alone KORIGEN Code as Compared with Bell's ORIGEN

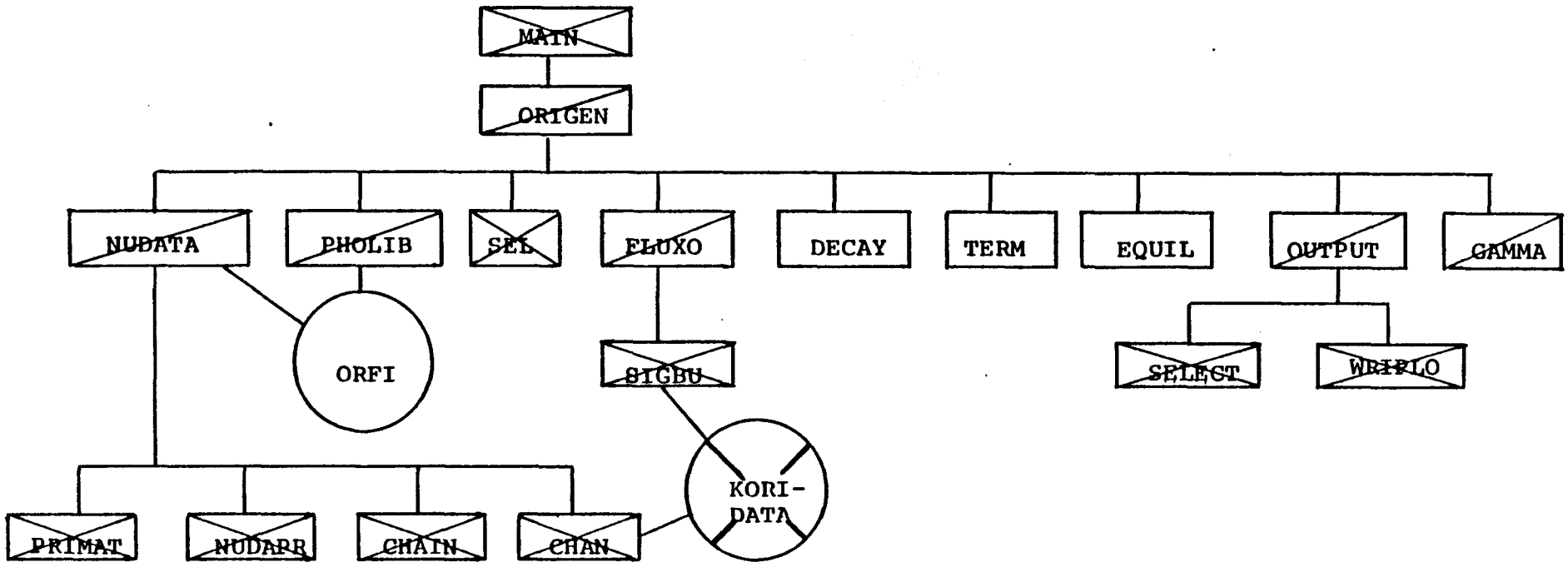
These modifications concern

-- the use of reactor-physical effective cross sections for important nuclides, burnup-dependent for actinides,
-- nuclide-specific fission energy release,
-- n-emission,
-- check of the nuclear data used,
-- input and input verification,
-- representation and evaluation of results.

For this purpose the control program MAIN and the subroutines SIGBU, CHAN, SEL, SELECT, PRIMAT, NUDAPR, CHAIN, INVNUC, WRIPLO and DESCRI were recreated and the system service routines present at KfK--CONVX, CONVY and CONVZ--were installed (their function is described in the following sections), and the ORIGEN program parts ORIGEN, NUDATA, FLUXO, OUTPUT, PHOLIB and GAMMA were modified. Fig. 4.1 shows the major program parts of KORIGEN in a block diagram.

4.1 Use of Data

A portion of the nuclear data is still included in the total of six ORFI files in the form introduced by Bell [2]. Until the switch to Karlsruhe data is completed, therefore, the formalism associated with the structure of these data records, particularly the outdated concept of spectral indices--but with reduced effect (Appendix A.2)--and the relation to the thermal neutron flux are retained. Reactor-specific effective cross sections for important nuclides, burnup-dependent for actinides, are stored in the file KORIDATA--see Section 3.1.2 and Appendix C.



unchanged program parts from ORIGEN

modified program parts from ORIGEN

new program parts

base data: in old structure, but more nuclides, in part new data, modified sequence of nuclides

new data with 1-group cross sections: for a mean burnup for 7 LWR spectra [16,18], burnup-dependent for 14 LWR spectra (this report and [16, 18]--cf. also App. B.1).

Figure 4.1. KORIGEN block diagram (important routines)

4.1.1 Provision of Effective Cross Sections

The coefficient matrix of the burnup equations--cf. Section 2.3, Eq. (2.12)--, referred to briefly in the following as transition matrix, as is customary, contains the microscopic zone-dependent reaction rates $\sigma_{x,z}(t)\cdot\phi_z(t)$ or, in Bell's notation, $\sigma_{x,OR,z}(t)\cdot\phi_{th,z}(t)$ where

$$\sigma_{x,OR,z}(t) \equiv \sigma_{x,z}(t)\cdot\phi_z(t)/\phi_{th,z}(t) \tag{4.1}$$

For simplification of notation, the zone and time dependence $_z(t)$ is not included in the following. In KORIGEN, before the beginning of the irradiation calculation, the $\sigma^i_{x,OR}$ cross sections are first determined in the subroutine NUDATA from the data of the ORFI files (currently 1176 nuclides) using the spectral parameters THERM, RES and FAST--cf. Appendix A.1. Depending on the calculation of such an $\sigma^i_{x,OR}$ value, an inquiry is made by the subroutine CHAN in the KORIDATA file specified by the input (card B: LPU, card U1: ALLWQ) whether a reactor-specific one-group cross section σ^i_x determined for a mean burnup exists. Any σ^i_x value found is multiplied by ϕ/ϕ_{th} from Eq. (A-9) and replaces the $\sigma^i_{x,OR}$ previously calculated. This overwriting of cross section values can be performed alternately or in addition by input of one-group values by way of cards U2--cf. Appendix B.1: input description. At the end of the run through NUDATA there then exist a transitional matrix {A} and σ_z and σ_f cross sections (TOCAP, FISS), which for the important nuclides and neutron reactions represented in the KORIDATA file are based on reactor-specific one-group cross sections or--in the case of exclusive or additional card input--contain the card values. In the KORIDATA file at the present time the one-group cross sections calculated by Croff et al. [16] for a mean burnup state are implemented.

The introduction of burnup-dependent cross sections for actinides is done at the beginning of each irradiation interval (input: card G) using the subroutine SIGBU, which is called up by FLUXO, and is controlled via input of the indicator KSB on card C and by card T1. SIGBU modifies the cross sections $\sigma_{c,OR}$, $\sigma^{ex}_{c,OR}$ (n-capture for excited state), $\sigma_{a,OR}$ and $\sigma_{f,OR}$ in accordance with the burnup B reached in the preceding irradiation

intervals. When SIGBU (B=0) is first called up the burnups B_i [MWd/gatSM]* which are to be increased for actinides in the KORIDATA file are read with B_1=0 stored one-group cross sections $\sigma_{c,i}$, $\sigma_{c,i}^{ex}$ and $\sigma_{f,i}$ and processed for i > 1 to form conversion factors

$$(4.2) \qquad S_{x,i} = \sigma_{x,i} / \sigma_{x,i-1} \qquad , \; i > 1$$

For B=0, $\sigma_{x,OR}$ in the transitional matrix and in $\sigma_{a,OR}$(TOCAP) and $\sigma_{f,OR}$ (FISS) is replaced by $\sigma_{x,1} \cdot \phi/\phi_{th}$. For B > 0 at the beginning of additional irradiation intervals, the index l is first of all determined such that

$$(4.3) \qquad \begin{array}{ll} B_l \leq B < B_{l+1}, & \text{falls } B < B^{max} \equiv Max\{B_i\}, \\ B_l = B^{max} \quad , & \text{falls } B \geq B^{max} \, , \end{array}$$

and subsequently the cross section assigned to burnup B and kept constant in the following irradiation interval is determined as $\sigma_{x,OR} = \sigma_{x,l} \cdot \phi/\phi_{th}$. The latter is accomplished by multiplying the $\sigma_{x,OR}$ value used in the preceding irradiation interval by the product of the conversion factors $S_{x,i}$ for the B_i values in each interval.

For optimal consideration of the burnup dependence of the actinie cross sections it is recommended that the irradiation intervals (t_{k-1}, t_k) be selected such that

$$(4.4) \quad gat_{SM} \cdot B_l \leq \sum_{k=1}^{l-1} P_k \cdot \Delta t_k < gat_{SM} \cdot B_{l+1}, \quad l = 2,3,4....$$

Here $\Delta t_k = t_k - t_{k-1}$ [d], where $t_o = 0$, the length of the k-th irradiation interval, P_k [MW/tSM] is the associated power density, and gat_{SM} the number of gram atoms of the actinide nuclides per ton initial heavy material (= 4203.4 for U-235 or U-238 fuel with 3.2 w/o U-235). The burnup support points B_l of the cross section sets implemented in KORIDATA are contained in the KORIDATA printout (see Appendix C.1).

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* Megawatt days per gram atom of heavy material.

The introduction of burnup-dependent actinite cross sections can be achieved by means of the file KORIDATA and/or cards (T2 to T6). If data are entered by means of cards for one nuclide and one cross section type, for which values already exist in the KORIDATA file, then the card values are used in the calculation. The cross sections from the KORIDATA file can therefore be temporarily modified and supplemented for an actual calculation. Regarding the contents of the present state of the KORIDATA files see Section 3.1.2, Tab. 3.2.

4.1.2 Nuclide-Specific Fission Energy Release

The energy released in the reactor per fission of a fission nuclide and useful as heat is composed primarily--for details see reference [19], for example--of the kinetic energy of the fission products and the neutrons (prompt and delayed), the energy of the prompt gamma radiation, the energy of the delayed beta and gamma radiation, and the energy of the gamma radiation emitted after neutron capture in the fuel and the cladding and structural materials:

$$
\begin{aligned}
E_T &= E_{kin}(SP)+E_{kin}(n)+E(\gamma\text{-prompt}) \\
&\quad + E(\beta,\gamma\text{-delayed})+E_c \qquad (4.5) \\
&= E_F+E_c
\end{aligned}
$$

E_F combines the first components, which result only directly from fission. E_c is primarily the energy of the gamma quanta released per fission after capture of fission neutrons, chiefly (n,γ)-reactions in U-238 (other components, such as those stemming from alpha decay of the transuranic nuclides Pu-238, Cm-242, etc., play a secondary role in this connection) and is a function of the number of secondary neutrons available for this reaction and the reactor materials. With $\bar{\nu}$ representing the nuclide-specific yield of secondary neutrons per fission, $\bar{\nu}$-1 in the critical reactor gives the number of excess neutrons which are either captured in the reactor materials or--to a lesser extent--lost by leakage. If we designate as Q_γ the gamma energy released in the reactor under investigation

on a spatial and chronological average per excess neutron--for BIBLIS normal fuel (3.2% initial U-235 enrichment, 33 GWd/tU mean unloading burnup) K. Baumung [20], taking into account leakage, obtains a value of $Q_\gamma \approx 5.5$ MeV--then

$$E_c \approx (\bar{\nu}-1)\cdot Q_\gamma \tag{4.6}$$

We should note that E_F and E_c are a function of the energy E_n of the neutrons inducing fission--E_c primarily by way of $\bar{\nu}$. According to [19]--see that report for details--$dE_F/dE_n = -0.71$ and $\frac{d}{dE}[(\bar{\nu}-1)\cdot Q_\gamma] = 0.75$ for $E_n < 2.6$ MeV for U-235 fission, for example. These opposing dependences result in the relatively weak total increase of 0.04 MeV per fission given an increase in the neutron energy of 1 MeV. This value is representative for the other fission nuclides as well. This means that for LWR it is sufficient to have a set of E_F and $\bar{\nu}$ values--for Q_γ a redetermination for other PWRs and also for BWR is underway--; for fast reactors the nuclide-specific E_F and $\bar{\nu}$ values and also Q_γ must be determined consistently.

Table 4.1 lists the values for E_F and $\bar{\nu}$-1 used in KORIGEN for PWR calculations and the value E_T which results from the former when $Q_\gamma = 5.5$ MeV. The ORIGEN2 values [21] given for E_T for the purpose of comparison are only slightly greater by 0.2% for the important fissile material isotopes U-235, Pu-239 and Pu-241. The differences are probably due to the value Q_γ which was determined for another reactor and to the analytical representation of E_T as a function of the atomic number Z and the mass number A which was used in approximation in ORIGEN2. A comparison calculation showed that the inventory of irradiated BIBLIS normal fuel calculated with the ORIGEN2 E_T values does not differ for all practical purposes from the inventory determined using KORIGEN E_T values.

Table 4.1. Nuclide-specific fission energy releases in KORIGEN and ORIGEN2

Nuclide	E_F [MeV]	$\bar{\nu}-1$ [e]	E_T [MeV] KORIGEN	E_T [MeV] ORIGEN2	Δ (%)
Th232	183.1 [a]	0.9	188.1	193.4	2.9
U 233	191.8 [b]	1.498	200.0	201.0	0.5
U 234	190.3 [c]	1.37	197.8	201.3	1.8
U 235	193.6 [b]	1.423	201.4	201.7	0.15
U 236	192.8 [c]	1.451	200.8	202.1	0.6
U 238	192.6 [b]	1.34	200.0	202.8	1.4
Np237	195.1 [c]	1.67	204.3	206.1	0.9
Pu238	197.8 [c]	1.889	208.2	210.2	1.0
Pu239	199.7 [b]	1.885	210.1	210.6	0.2
Pu240	195.0 [d]	1.880	205.3	211.0	2.8
Pu241	200.4 [b]	1.924	211.0	211.3	0.14
Pu242	200.6 [c]	1.580	210.9	211.7	0.4
Am241	202.3 [c]	2.110	213.9	215.2	0.5
Am243	202.1 [c]	1.80	212.0	215.9	2.0
Cm244	206.6 [c]	2.24	218.9	220.2	0.6

[a] Unik 1971 /19/

[b] Trapp 1978 /22/

[c] ENDF/B-V

[d] ENDF/B-IV

[e] KEDAK4 /6/

$$\Delta = \frac{\text{ORIGEN2} - \text{KORIGEN}}{\text{KORIGEN}}$$

4.1.3 Improved Calculation of the Neutron Emission of Irradiated Fuel

For reasons of criticality and safety, the issue of the neutron emission from irradiated fuel, especially in reprocessing, is important. The neutrons emitted from irradiated fuel stem on the one hand from the spontaneous fission of the transactinide nuclei and on the other hand from (α,n)-reactions, primarily with the oxygen (O17, O18) of the fuel oxide.

Neutrons from Spontaneous Fission

For calculation of the neutrons released during spontaneous fission (per time and mass unit) we need not only the half-life compared with spontaneous fission but also the neutron yield per spontaneous fission $\bar{\nu}_{sf}$. KORIGEN explicitly uses nuclide-specific values which are based in part on measurements and in part on calculations [21, 23]. These values are implemented for all spontaneously splitting nuclei in a data statement in the subroutine GAMMA.

ORIGEN used the following semiempirical equation dependent only on the mass number A for determination of $\bar{\nu}_{sf}$:

$$\bar{\nu}_{sf} = 2.84 + (A-244) \cdot 0.1225 \tag{4.6}$$

As a comparison between the nuclide-specific $\bar{\nu}_{sf}$ values and those calculated using (4.6) shows (Fig. 4.2), the differences are slight. The main producers of neutrons from spontaneous fission are primarily the Cm isotopes 242 and 244, with longer cooling times (> 50 yrs) Cm-246.

(α,n)-Reactions

The helium nuclei emitted during α-decay of the actinides are absorbed preferentially by the light nuclei due to the low atomic number. In oxide fuel most of the (α,n) reactions take place with the oxygen (primarily O18) of the oxide. For calculation of the neutron yield expected per α-decay of an actinide nucleus, ORIGEN used the following semi-empirical interpolation formula, which depends solely on the energy of the α particles:

$$n(E_\alpha) = 10^{-10}\, E_\alpha^{3.65}\ n/\alpha\text{-decay}, \qquad E_\alpha \text{ in MeV} \tag{4.7}$$

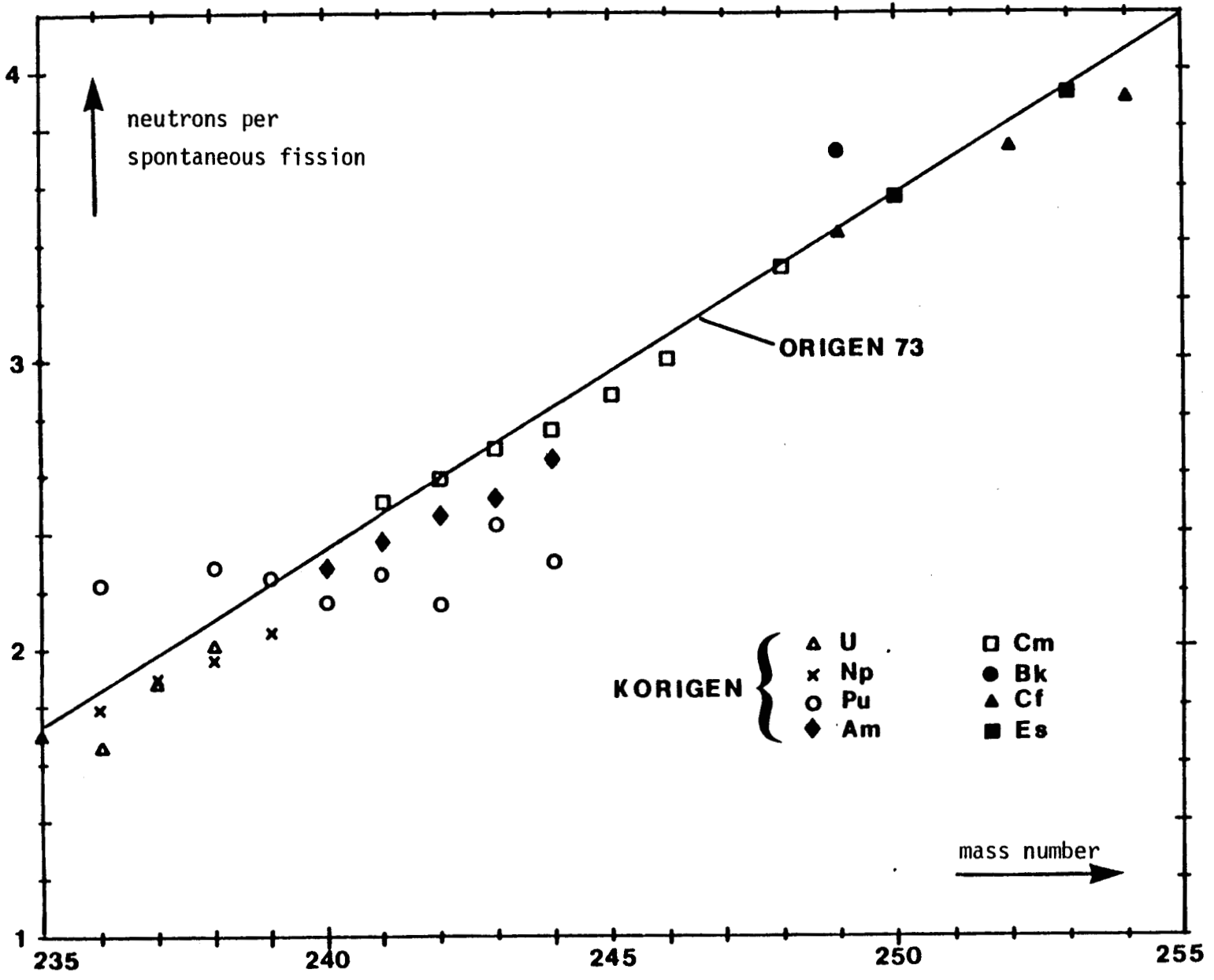


Figure 4.2. Neutron yield in spontaneous fission.

In KORIGEN we use explicitly experimental values of neutron yield (per mass and time unit) for the most important producers of (α,n) neutrons --Pu-238, Pu-240, Pu-242, Am-241, Cm-242, Cm-244 [21, 24]. For less important producers of (α,n) neutrons we use the following interpolation formula [21] based on the experimental values of Pu-239 and Cm-242; it gives the neutron yield per α-decay:

$$(4.8) \qquad n(E_\alpha) = 2.152 \cdot 10^{-18}\, E_\alpha^{14.01} \ \text{n/}\alpha\text{-decay} \quad , \ E_\alpha \text{ in MeV}$$

Calculation of the (α,n)-neutrons is performed in KORIGEN such that $n(E_\alpha)$ is first calculated using Eq. (4.8) (in subroutine NUDATA). In the subroutine GAMMA the neutron yield for those nuclei for which experimental values are available--they are in a DATA statement in subroutine GAMMA--is replaced by the corresponding experimental value.

In order to use the formulas (4.7) and (4.8), one needs the energy of the emitted α-particles. In ORIGEN the Q-value of the respective nuclide was inserted erroneously. KORIGEN contains in the ORFI4-(actinide)-file the mean $\bar{E}_\alpha$ value according to Lederer [25] for all α-emitters among the actinides--Tab. 4.2. This value is used in Eq. (4.8).

Figures 4.3 shows a comparison of the equations (4.7) and (4.8) for n(E) with the explicit values:

-- the relation (4.8) used in KORIGEN shows good agreement with the experimental values in the relevant range 5 MeV $\leqslant E_\alpha \leqslant$ 6.2 MeV;

-- in the range 5 to ca. 5.7 MeV the formulas (4.7) and (4.8) yield more or less the same results;

-- for $5.7 < E_\alpha \leqslant 6.2$ MeV, the equation (4.7) used in ORIGEN clearly underestimates the measured values (Cm-242, Cm-244);

-- for $E_\alpha > 6.2$ MeV the interpolation formula (4.8) yields unrealistically high values due to the large exponent of E_α.

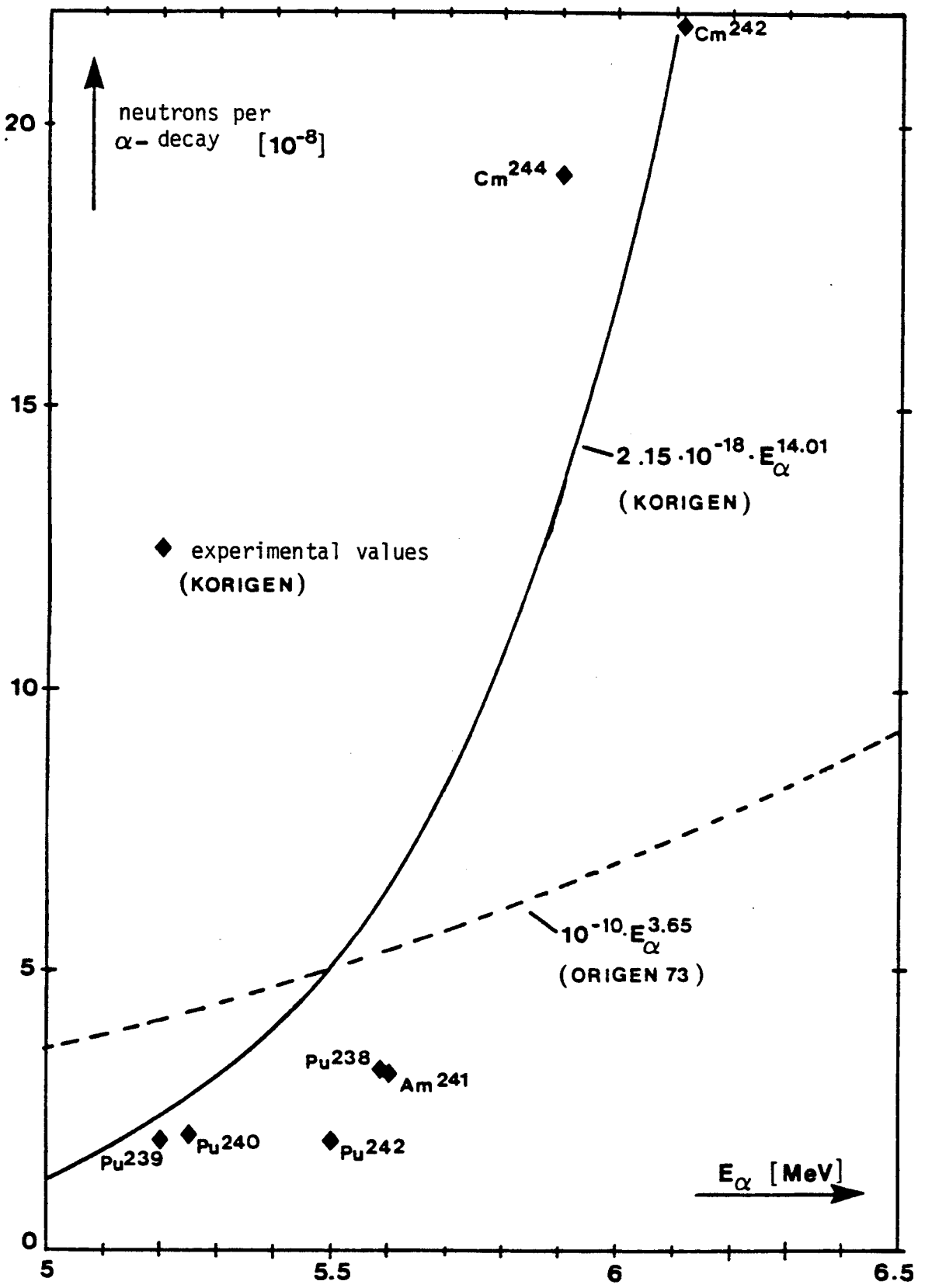


Figure 4.3. Neutron yield from (α,n)-reactions in oxygen.

Table 4.2. Mean α-energy of the most important α-emitters among the actinides [25].

Nuclide	E_α [MeV]	Nuclide	E_α [MeV]
Pu238	5.593	Am241	5.638
Pu239	5.244	Cm242	6.216
Pu240	5.256	Cm244	5.902
Pu242	4.983		

4.1.4 Specific Data Corrections

Cm-242: The Q value obtained from [26] (0.196 MeV) was corrected according to Lederer et al. [25]: Q = 6.216 MeV.

Pu-236: The handling of fresh FBR fuel elements which are fabricated with reprocessed LWR Pu is made difficult by the hard gamma radiation of TL208 (2.6 MeV). TL208 is a decay member of Pu-236 which is contained in reprocessed plutonium. A reliable calculation of the formation of Pu-236 in thermal reactors is therefore necessary. A detailed analysis of the buildup of Pu-236 in PWR fuel yielded corrections of Np-236 and Np-237 data [27].

Pu-236 is formed primarily by β^--decay from Np-236, which is itself formed from (n,2n)-reactions in Np-237. Np-236 occurs in two states, a short-lived state ($T_{1/2}$ = 22.5 yrs) and a long-lived state ($T_{1/2} = 1.29 \cdot 10^6$ yrs [28], see also Tab. 4.3).

Only the short-lived isomer of Np-236--i.e. in KORIGEN the ground state--has significance for the buildup of Pu-236. The percentage of (n,2n)-reactions in Np-237, which lead to the ground state of Np-236, is 72.5% [27]. The (n,2n)-cross section of Np-237 averaged over the fission spectrum (Cranberg-Watt) is 3.35 mb according to KEDAK-3 (ORIGEN: 1.3 mb, i.e. the value calculated by Pearlstein in 1965). The corresponding (n,2n) cross section of Np-237 averaged over the reactor spectrum of the

PWR Trino Vercellese and validated with post-irradiation analyses is 1.259 mb [27]. This value contains all corrections which are necessary on the basis of the formalism used in both KORIGEN and ORIGEN for description of the (n,2n)-reactions (cf. Appendix A.1). The KORIGEN library contains the value derived from this value and averaged over the fission spectrum of 4.26 mb, which, when the representative spectral parameters THERM = 0.701, RES = 0.304 and FAST = 2.01 (cf. Appendix A) are used--this will generally be the case for the user--yields the value averaged over the reactor spectrum of 1.259 mb and which is consistent with the KEDAK value of 3.35 mb mentioned above.

Table 4.3. Nuclear data of Np-236.

	$T_{1/2}$	Decay Fraction [%] β^-	E_c	Reference
Np236 m	$1.29 \cdot 10^6$ a*	9	91	/28/
Np236 g	22.5 h	50	50	/28,25/

Pb-210: Share of α-decays in the total decays FA = $1.7 \cdot 10^{-8}$

Bi-210: FA = $1.3 \cdot 10^{-6}$

Pu-241: FA = $2.50 \cdot 10^{-5}$

Cm-250: FA = 0, share of spontaneous fission in the total decays, FSF = 1, Q = 200 MeV

Tritium: The fission yields of tritium, formed from ternary fission, are very uncertain. On the basis of a study of the literature [30] the values listed in Tab.e 4.4 were assumed for the H^3 fission yields in thermal reactors. The fission yields originally contained in ORIGEN were ca. 30% higher for the most important fission nuclei in thermal reactors, U-235 and Pu-239 (Table 4.4).

* According to the 5th edition of the KfK nuclide map (Dec. 81) $T_{1/2}$ = $1.15 \cdot 10^5$ yrs. This value will be implemented in the future in the KORIGEN file.

Table 4.4. Tritium yield [%] in thermal reactors.

Fission Nucleus	ORIGEN	KORIGEN
U235	$1.30 \cdot 10^{-2}$	$0.92 \cdot 10^{-2}$
U238	$2.0 \cdot 10^{-2}$	$2.3 \cdot 10^{-2}$
Pu239	$2.50 \cdot 10^{-2}$	$1.64 \cdot 10^{-2}$

4.1.5 Introduction of Limit Values for the Maximum Permissible Annual Intake as a Measure of Radiotoxicity

In order to determine the radiotoxicity of radioactive nuclides, Bell's ORIGEN used the so-called "Radionuclide Concentration Guide" (RCG) values which gave the maximum permissible concentration of nuclides (in Ci/m^3) in air (inhalation) or water (ingestion). The "Hazard Index" expressed in the ORIGEN table gave the volume of air or water which was required to dilute the nuclide to the RCG value. The RCG values were obtained from the US Code of Federal Regulations [31].

In KORIGEN there was a departure from this concept; the concept of the limits of the maximum permissible annual intake of activity, on which the Second Radiation Protection Ordinance of the Federal Republic of Germany of October 1976 is based, is now being used. The maximum permissible annual intake values for inhalation and ingestion given in the Radiation Protection Ordinance represent the maximum permissible intake of activity per year which applies to an individual. The radiotoxicity index expressed in the KORIGEN tables, which is calculated from the activity and the maximum permissible annual intake value, therefore gives the factor by which the maximum permissible activity intake per year for an individual is exceeded.

The maximum permissible annual intake values of the Radiation Protection Ordinance are limits derived according to "International Commission on Radiological Protection (ICRP) Publication No. 2 (1950)" which refer to

the 30 mrem concept valid in the Federal Republic of German. The maximum permissible annual intake values for the nuclides not listed in the Radiation Protection Ordinance were determined by H. Haug [32] by conversion of older RCG values or by interpolation--he also compiled the maximum permissible annual intake values from the Radiation Protection Ordinance.

4.2 Data Control

4.2.1 Expanded Printout of the Transition Matrix

The transition matrix is first printed out before the irradiation calculation in subroutine NUDATA by specifying option IR > 0 on card B (cf. App. B.1) and contains, to the extent used, the constant one-group cross sections from the KORIDATA file. If burnup-dependent actinide cross sections are used, the matrix at time step IR is printed out again with correspondingly modified cross sections in subroutine FLUXO. Therefore, control of the burnup-dependent cross sections is also possible.

4.2.2 Printouts of Decay and Production Chains

By selecting the option IPNUDA > 0 on card A (cf. App. B.1), a comfortable output of the transition and the production matrices is generated in which the nuclides are listed with alphanumeric names and the reactions are also designated alphanumerically.

The transition matrix printed out in the subroutine PRIMAT (for excerpt see Fig. 4.4) contains for each case the sequence number of the respective nuclide, its alphanumeric name, the decay constant, the absorption and fission cross section, and all nuclides--by alphanumeric name and number--from which it is formed, with information on the respective reactions and the value of the matrix element.

The production matrix (for excerpt see Fig. 4.5) contains all nuclides with the respective reactions and the matrix element that can be built

up from the source nuclide. This offers the possibility of finding out very quickly in what way a nuclide is built up and which other nuclides can develop from it. Furthermore, using the matrix elements one can estimate how probable a given production path is.

4.2.3 Printouts of Reactor-Specific Effective Neutron Cross Sections

For consistent inventory calculations, the neutron cross sections associated with the fuel and reactor under investigation must be entered as a function of burnup for actinides and burnup-averaged for 267 additional nuclides which are important as regards radiotoxicity, thermal release, etc. (including the same for actinides). In controlling the input of these data by means of cards U1 and T1, a printout on the unit FT09F001 (DD card file!) of the cross sections used in an actual case can be obtained by way of KTIW = 0 and/or ITIW = 0 on card U1 or card T1.

The printed data contain the contents of the KORIDATA library for the fuel investigated and in addition provide information at KTIW = 0 on the burnup-averaged cross sections stored in KORIDATA for other fuels. The printout is performed in the subroutines CHAN and SIGBU.

TRANSITION MATRIX (ACTINIDES AND THEIR DAUGHTERS)

PRODUCING NUCLIDE AND REACTION WITH DECAY CONSTANT RESP. CROSS SECTION

I	NUCLIDE	DIS(I)	TOCAP(I)	FISS(I)	I	NUCLIDE	REAC.	A(I,J)	I	NUCLIDE	REAC.	A(I,J)	I	NUCLIDE	REAC.	A(I,J)
307	U 232	3.05E-10	2.91E+02	1.51E+02	303	PA232	NEG.	6.12E-06	323	PU236	A	7.70E-09	308	U 233	N,2N	6.63E-03
					309	U 234	N,3N	8.84E-05								
308	U 233	1.39E-13	4.75E+02	4.23E+02	304	PA233	NEG.	2.97E-07	307	U 232	N,G	1.40E+02	309	U 234	N,2N	1.45E-02
					310	U 235	N,3N	9.45E-05								
309	U 234	8.98E-14	1.34E+02	3.06E+00	305	PA234M	NEG.	9.86E-03	306	PA234	NEG.	2.87E-05	324	PU238	A	2.50E-10
					308	U 233	N,G	5.15E+01	310	U 235	N,2N	3.02E-02	311	U 236	N,3N	1.45E-04
310	U 235	3.12E-17	3.89E+02	3.17E+02	325	PU239	A	9.13E-13	309	U 234	N,G	1.31E+02	311	U 236	N,2N	1.31E-02
311	U 236	9.38E-16	5.26E+01	1.34E+00	316	NP236M	POS.	1.55E-14	317	NP236	POS.	4.45E-06	326	PU240	A	3.36E-12
					310	U 235	N,G	7.14E+01	313	U 238	N,3N	3.01E-04				
312	U 237	1.19E-06	3.02E+02	1.58E+00	327	PU241	A	3.81E-14	311	U 236	N,G	5.12E+01	313	U 238	N,2N	3.02E-02
313	U 238	4.92E-18	6.84E+00	6.80E-01	328	PU242	A	5.68E-14	312	U 237	N,G	3.00E+02				
314	U 239	4.91E-04	3.58E+01	1.05E+01	313	U 238	N,G	6.13E+00								
315	U 240	1.37E-05	4.20E+00	4.95E-01	330	PU244	A	2.66E-16	314	U 239	N,G	2.53E+01				
316	NP236M	1.70E-14	3.56E+00	3.56E+00	318	NP237	N,2N*	2.52E-03								
317	NP236	8.56E-06	3.56E+00	3.56E+00	318	NP237	N,2N	6.63E-03								
318	NP237	1.03E-14	2.22E+02	3.56E+00	312	U 237	NEG.	1.19E-06	332	AM241	A	5.08E-11				
319	NP238	3.79E-06	1.33E+03	1.21E+03	333	AM242M	A	7.22E-13	318	NP237	N,G	2.18E+02				
320	NP239	3.41E-06	1.68E+02	0.0	314	U 239	NEG.	4.91E-04	335	AM243	A	2.98E-12	319	NP238	N,G	1.22E+02
321	NP240M	1.56E-03	0.0	0.0	315	U 240	NEG.*	1.37E-05								
322	NP240	1.78E-04	0.0	0.0	320	NP239	N,G	1.68E+02								
323	PU236	7.70E-09	2.79E+02	1.40E+02	316	NP236M	NEG.	1.53E-15	317	NP236	NEG.	4.11E-06	324	PU238	N,3N	6.03E-05
324	PU238	2.50E-10	2.53E+02	1.67E+01	319	NP238	NEG.	3.79E-06	338	CM242	A	4.92E-08	325	PU239	N,2N	1.00E-02
					326	PU240	N,3N	7.04E-05								
325	PU239	9.13E-13	1.12E+03	7.20E+02	320	NP239	NEG.	3.41E-06	339	CM243	A	7.69E-10	324	PU238	N,G	2.36E+02
					326	PU240	N,2N	7.03E-03	327	PU241	N,3N	1.59E-04				
326	PU240	3.36E-12	7.11E+02	3.97E+00	321	NP240M	NEG.	1.56E-03	322	NP240	NEG.	1.78E-04	340	CM244	A	1.21E-09
					325	PU239	N,G	3.98E+02	327	PU241	N,2N	1.59E-02	328	PU242	N,3N	1.49E-04
327	PU241	1.53E-09	1.07E+03	8.02E+02	341	CM245	A	2.58E-12	326	PU240	N,G	7.07E+02	328	PU242	N,2N	1.49E-02
328	PU242	5.68E-14	2.03E+02	3.11E+00	334	AM242	POS.	2.08E-06	342	CM246	A	4.64E-12	327	PU241	N,G	2.63E+02
329	PU243	3.89E-05	3.18E+02	1.42E+02	343	CM247	A	1.41E-15	328	PU242	N,G	2.00E+02				
330	PU244	2.66E-16	1.12E+00	0.0	344	CM248	A	5.94E-14	329	PU243	N,G	1.76E+02				
331	PU245	1.82E-05	1.94E+02	0.0	330	PU244	N,G	1.12E+00								
332	AM241	5.08E-11	6.83E+02	8.97E+00	327	PU241	NEG.	1.53E-09	333	AM242M	N,2N	3.72E-02	334	AM242	N,2N	3.72E-02
333	AM242M	1.44E-10	6.29E+03	5.23E+03	332	AM241	N,G*	8.36E+01								
334	AM242	1.20E-05	1.29E+03	1.16E+03	333	AM242M	ISOM.	1.44E-10	332	AM241	N,G	5.91E+02				
335	AM243	2.98E-12	2.61E+02	2.43E+00	329	PU243	NEG.	3.89E-05	333	AM242M	N,G	1.05E+03	334	AM242	N,G	1.28E+02
336	AM244	1.91E-05	1.61E+03	1.61E+03	335	AM243	N,G	2.58E+02								
337	AM245	9.30E-05	0.0	0.0	331	PU245	NEG.	1.82E-05								
338	CM242	4.92E-08	3.87E+01	1.50E+00	334	AM242	NEG.	9.94E-06	339	CM243	N,2N	1.00E-02				
339	CM243	7.71E-10	7.31E+02	6.63E+02	338	CM242	N,G	3.72E+01								
340	CM244	1.21E-09	2.92E+01	5.80E+00	336	AM244	NEG.	1.91E-05	339	CM243	N,G	6.74E+01				
341	CM245	2.58E-12	2.42E+03	1.17E+03	337	AM245	NEG.	9.30E-05	349	CF249	A	6.27E-11	340	CM244	N,G	2.34E+01
342	CM246	4.64E-12	1.23E+01	3.33E+00	350	CF250	A	1.68E-09	341	CM245	N,G	1.25E+03				
343	CM247	1.41E-15	1.54E+02	6.36E+01	351	CF251	A	2.45E-11	342	CM246	N,G	8.97E+00				
344	CM248	6.48E-14	3.12E+01	3.87E+00	352	CF252	A	8.07E-09	343	CM247	N,G	9.04E+01				
345	CM249	1.80E-04	3.70E+01	3.50E+01	353	CF253	A	1.35E-09	344	CM248	N,G	2.73E+01				
346	CM250	1.26E-12	1.40E+00	0.0	354	CF254	A	3.98E-10	345	CM249	N,G	1.96E+00				
347	BK249	2.51E-08	1.19E+03	0.0	345	CM249	NEG.	1.80E-04	355	ES253	A	3.92E-07				
348	BK250	5.98E-05	2.35E+03	2.10E+03	346	CM250	NEG.	1.14E-12	347	BK249	N,G	1.19E+03				
349	CF249	6.27E-11	2.66E+03	1.18E+03	347	BK249	NEG.	2.51E-08								
350	CF250	1.68E-09	1.01E+03	0.0	348	BK250	NEG.	5.98E-05	349	CF249	N,G	1.47E+03				
351	CF251	2.45E-11	7.06E+03	2.76E+03	350	CF250	N,G	1.01E+03								
352	CF252	8.33E-09	6.42E+01	2.95E+01	351	CF251	N,G	4.30E+03								
353	CF253	4.50E-07	1.39E+03	7.68E+02	352	CF252	N,G	3.47E+01								
354	CF254	1.33E-07	6.25E+01	0.0	353	CF253	N,G	6.25E+02								

Figure 4.4. Excerpt from the transition matrix.

PRODUCED NUCLIDES FOR ACTINIDES AND THEIR DAUGHTERS

PRODUCED NUCLIDES AND REACTIONS WITH DECAY CONSTANTS RESP. CROSS SECTIONS

I	NUCLIDE	DIS(I)	TOCAP(I)	FISS(I)	I	NUCLIDE	REAC.	COEFF.	I	NUCLIDE	REAC.	COEFF.	I	NUCLIDE	REAC.	COEFF.
316	NP236M	1.70E-14	3.56E+00	3.56E+00	311	U 236	POS.	1.55E-14	323	PU236	NEG.	1.53E-15				
317	NP236	8.56E-06	3.56E+00	3.56E+00	311	U 236	POS.	4.45E-06	323	PU236	NEG.	4.11E-06				
318	NP237	1.03E-14	2.22E+02	3.56E+00	304	PA233	A	1.03E-14	6	HE 4	A	1.03E-14	319	NP238	N,G	2.18E+02
					317	NP236	N,2N	6.63E-03	316	NP236M	N,2N*	2.52E-03	0	NP235	N,3N	1.91E-05
319	NP238	3.79E-06	1.33E+03	1.21E+03	324	PU238	NEG.	3.79E-06	320	NP239	N,G	1.22E+02				
320	NP239	3.41E-06	1.68E+02	0.0	325	PU239	NEG.	3.41E-06	322	NP240	N,G	1.68E+02				
321	NP240M	1.56E-03	0.0	0.0	326	PU240	NEG.	1.56E-03								
322	NP240	1.78E-04	0.0	0.0	326	PU240	NEG.	1.78E-04								
323	PU236	7.70E-09	2.79E+02	1.40E+02	307	U 232	A	7.70E-09	6	HE 4	A	7.70E-09	0	PU237	N,G	1.39E+02
324	PU238	2.50E-10	2.53E+02	1.67E+01	309	U 234	A	2.50E-10	6	HE 4	A	2.50E-10	325	PU239	N,G	2.36E+02
					0	PU237	N,2N	5.23E-03	323	PU236	N,3N	6.03E-05				
325	PU239	9.13E-13	1.12E+03	7.20E+02	310	U 235	A	9.13E-13	6	HE 4	A	9.13E-13	326	PU240	N,G	3.98E+02
					324	PU238	N,2N	1.00E-02	0	PU237	N,3N	1.17E-04				
326	PU240	3.36E-12	7.11E+02	3.97E+00	311	U 236	A	3.36E-12	6	HE 4	A	3.36E-12	327	PU241	N,G	7.07E+02
					325	PU239	N,2N	7.03E-03	324	PU238	N,3N	7.04E-05				
327	PU241	1.53E-09	1.07E+03	8.02E+02	312	U 237	A	3.81E-14	6	HE 4	A	3.81E-14	332	AM241	NEG.	1.53E-09
					328	PU242	N,G	2.63E+02	326	PU240	N,2N	1.59E-02	325	PU239	N,3N	1.59E-04
328	PU242	5.68E-14	2.03E+02	3.11E+00	313	U 238	A	5.68E-14	6	HE 4	A	5.68E-14	329	PU243	N,G	2.00E+02
					327	PU241	N,2N	1.49E-02	326	PU240	N,3N	1.49E-04				
329	PU243	3.89E-05	3.18E+02	1.42E+02	335	AM243	NEG.	3.89E-05	330	PU244	N,G	1.76E+02				
330	PU244	2.66E-16	1.12E+00	0.0	315	U 240	A	2.66E-16	6	HE 4	A	2.66E-16	331	PU245	N,G	1.12E+00
331	PU245	1.82E-05	1.94E+02	0.0	337	AM245	NEG.	1.82E-05	0	PU246	N,G	1.94E+02				
332	AM241	5.08E-11	6.83E+02	8.97E+00	318	NP237	A	5.08E-11	6	HE 4	A	5.08E-11	334	AM242	N,G	5.91E+02
					333	AM242M	N,G*	8.36E+01	0	AM240	N,2N	1.01E-03	0	AM239	N,3N	1.01E-05
333	AM242M	1.44E-10	6.29E+03	5.23E+03	334	AM242	ISOM.	1.44E-10	319	NP238	A	7.22E-13	0	HE 3	A	7.22E-13
					335	AM243	N,G	1.05E+03	332	AM241	N,2N	3.72E-02	0	AM240	N,3N	3.82E-04
334	AM242	1.20E-05	1.29E+03	1.16E+03	328	PU242	POS.	2.08E-06	338	CM242	NEG.	9.94E-06	335	AM243	N,G	1.28E+02
					332	AM241	N,2N	3.72E-02	0	AM240	N,3N	3.82E-04				
335	AM243	2.98E-12	2.61E+02	2.43E+00	320	NP239	A	2.98E-12	6	HE 4	A	2.98E-12	336	AM244	N,G	2.58E+02
336	AM244	1.91E-05	1.61E+03	1.61E+03	340	CM244	NEG.	1.91E-05								
337	AM245	9.30E-05	0.0	0.0	341	CM245	NEG.	9.30E-05								
338	CM242	4.92E-08	3.87E+01	1.50E+00	324	PU238	A	4.92E-08	6	HE 4	A	4.92E-08	339	CM243	N,G	3.72E+01
339	CM243	7.71E-10	7.31E+02	6.63E+02	325	PU239	A	7.69E-10	6	HE 4	A	7.69E-10	0	BK243	NEG.	1.54E-12
					340	CM244	N,G	6.74E+01	338	CM242	N,2N	1.00E-02	0	CM241	N,3N	1.01E-04
340	CM244	1.21E-09	2.92E+01	5.80E+00	326	PU240	A	1.21E-09	6	HE 4	A	1.21E-09	341	CM245	N,G	2.34E+01
341	CM245	2.58E-12	2.42E+03	1.17E+03	327	PU241	A	2.58E-12	6	HE 4	A	2.58E-12	342	CM246	N,G	1.25E+03
342	CM246	4.64E-12	1.23E+01	3.33E+00	328	PU242	A	4.64E-12	6	HE 4	A	4.64E-12	343	CM247	N,G	8.97E+00
343	CM247	1.41E-15	1.54E+02	6.36E+01	329	PU243	A	1.41E-15	6	HE 4	A	1.41E-15	344	CM248	N,G	9.04E+01
344	CM248	6.48E-14	3.12E+01	3.87E+00	330	PU244	A	5.94E-14	6	HE 4	A	5.94E-14	345	CM249	N,G	2.73E+01
345	CM249	1.80E-04	3.70E+01	3.50E+01	347	BK249	NEG.	1.80E-04	346	CM250	N,G	1.96E+00				
346	CM250	1.26E-12	1.40E+00	0.0	348	BK250	NEG.	1.14E-12	0	CM251	N,G	1.40E+00				
347	BK249	2.51E-08	1.19E+03	0.0	349	CF249	NEG.	2.51E-08	348	BK250	N,G	1.19E+03				
348	BK250	5.98E-05	2.35E+03	2.10E+03	350	CF250	NEG.	5.98E-05	0	BK251	N,G	2.45E+02				
349	CF249	6.27E-11	2.66E+03	1.18E+03	341	CM245	A	6.27E-11	6	HE 4	A	6.27E-11	350	CF250	N,G	1.47E+03
350	CF250	1.68E-09	1.01E+03	0.0	342	CM246	A	1.68E-09	6	HE 4	A	1.68E-09	0	ES250	NEG.	3.86E-13
					351	CF251	N,G	1.01E+03								
351	CF251	2.45E-11	7.06E+03	2.76E+03	343	CM247	A	2.45E-11	6	HE 4	A	2.45E-11	352	CF252	N,G	4.30E+03
352	CF252	8.33E-09	6.42E+01	2.95E+01	344	CM248	A	8.07E-09	6	HE 4	A	8.07E-09	353	CF253	N,G	3.47E+01
353	CF253	4.50E-07	1.39E+03	7.68E+02	345	CM249	A	1.35E-09	6	HE 4	A	1.35E-09	355	ES253	NEG.	4.49E-07
					354	CF254	N,G	6.25E+02								
354	CF254	1.33E-07	6.25E+01	0.0	346	CM250	A	3.98E-10	6	HE 4	A	3.98E-10	0	ES254	NEG.	1.33E-11
					0	CF255	N,G	6.25E+01								
355	ES253	3.92E-07	1.37E+02	0.0	347	BK249	A	3.92E-07	6	HE 4	A	3.92E-07	0	ES254	N,G	4.21E+01
					0	ES254M	N,G*	9.52E+01								

Figure 4.5. Excerpt from the production matrix.

4.3 Input

The newly added options -- use of reactor-specific cross sections, improved representation of results, etc.--require expanded program control via the input. In order to make it easier for former ORIGEN users to use KORIGEN, the expansion was undertaken from the standpoint of compatibility of ORIGEN and KORIGEN inputs. For this reason the input continues to be formatted. Indicators for the control of new program parts must be input in card columns that were previously not used. Some new [cards], e.g. the U and T cards, which are required for processing the cross section files, must be additionally inserted.

The input description which is printed out by the program in the event that card A is omitted, i.e. for example in the case of non-existing input, is given in Appendix B.1 (subroutine DESCRI). In Appendix B.2 there follow a number of input examples, which covers a large number of the program options.

4.3.1 Input Data Log

The input data log (Input Reproduction) provides a printed record of the input as it is accepted by the program and processed--with the exception of differences due to different format usage in the case of read and write statements. In this way one can check to see whether the intended input agrees with that actually processed; errors which occur in the case of linked input due to failure to follow the prescribed format can be detected. Because the reading of the input, its printout in the data log and the execution of the associated part of the calculation follow in sequence, the cause of an erroneous interruption of the calculation --for example, set indicators and cards found by the program--can be localized more or less. An example of an input data log is given in Appendix B.3.

4.3.2 Input of Elements for the Cladding and Structural Materials in g/Basis

The input of natural elements of the cladding and structural materials, e.g. Fe, Ni and Zr, was performed in ORIGEN in gram atoms relative to the selected basis. Because the data on these elements generally exist in grams, conversion using the atomic weights was necessary for this purpose. In KORIGEN the direct input of these elements in g/basis is possible using the indicator NEXT = 5 on Card K. For this purpose the atomic weights ${}_ZM$ (Z = atomic number) of the natural elements, scaled to ${}_6M^{12}$, initialized in field ATW in the subroutine BLOCKDATA. In the case of the artificial elements (Pu, Am, etc.)--not used in this connection--, DUMMY values are given. KORIGEN calculates the number of the gram-atoms/basis of the stable isotopes having the mass numbers A, i.e. ${}_ZN^A$, according to the formula

$$ {}_ZN^A = {}_ZG/{}_ZM \cdot {}_ZP^A $$

with ${}_ZG$ in g/basis for the element with the atomic number Z and with ${}_ZP^A$ as the natural isotope frequency (number of isotopes with the atomic and mass numbers Z and A relative to the total number of nuclides with the atomic number Z in the material).

It should be noted that in calculating back to g/basis--only for the purpose of presenting results--the whole-number mass numbers are used in KORIGEN and ORIGEN for isotopes and elements for the sake of simplicity:

$$ (4.10) \quad {}_ZG^A_{OR} = {}_ZN^A \cdot A, \qquad {}_ZG_{OR} = \sum_{A(Z)} {}_ZN^A \cdot A $$

The resulting imprecision, which is accepted by way of compromise, is determined by the nuclear mass defect per nucleon and is a maximum of 0.1% for $A \geqslant 10$. With the increasing tendency toward small mass numbers it reaches 0.8% for hydrogen and is therefore negligible on the whole in comparison with uncertainties in the initial concentrations.

4.4 Tables of Results

The large number of nuclides considered in the program yields a quantity of information upon printout of all results which is virtually impossible to handle and often not relevant. An advantage with regard to a selective output of results was the option already implemented in ORIGEN of outputting only certain quantities, e.g. the radioactivity and the thermal output, and at the user's option, for example only for fission products. On the other hand, there is frequently an interest in quantities which were not calculated previously but the determination of which is easily possible from the results internally present, for example the alpha activity and the neutron multiplication factor k_∞. In the following sections the possibilities implemented in KORIGEN for further result selection and also for the output of additional results are described.

4.4.1 Reduction of the Main Tables to Elements or Isotopes of Interest

In studies on the behavior of irradiated fuel, it is often the case that only the uranium and plutonium isotopes, some Am and Cm isotopes, and the ten fission product nuclides which are most important from the standpoint of radioactivity, to cite some examples, are of interest. With the corresponding selection of the indicator JTO on Card C--see input description--it is possible for the elements or nuclides desired in the output to be input on Card S2; see in this respect example 7 in Appendix B.2. Up to 50 elements and single nuclides can be given. If Pu and Am-241 are given, for example, then Pu and Am appear in the element tables and all Pu isotopes and Am-241 in the nuclide tables. The selection is made using the subroutine SELECT which is called up by the subroutine OUTPUT. The input card S2 is read in by way of ENTRY SEL in SELECT.

4.4.2 Cut-Off for all Main Tables

The use of a cut-off in the output of results, by which the printout of the radioactivity, nuclide concentration, etc. is suppressed at a given point in time for all time steps if values fall below a given cut-off value, was previously limited to the SUMMARY tables. Since many nuclides

due to their low formation rate or short life, particularly fission products, then appear in the important nuclide tables with practically negligible values (usually .0), it is advisable to also make a cut-off possible for these tables. In KORIGEN the cut-off values read in on Card I and used in appropriate allocation for the main nuclide tables, if NCU is set = 1 on Card S2. The main element tables are excepted from the cut-off due to their comparatively short length. Selection of the nuclides is carried out in the subprogram SELECT.

We should note especially the useful option by which the cut-off can be applied to main nuclide tables without selection of elements and/or nuclides. Cards S2 consist for this purpose only of one blank card and one additional card with 1 in column 5.

4.4.3 Totals for All Nuclides

In addition to the totals within the individual nuclide groups (light elements, actinides, fission products), which were already calculated in ORIGEN, KORIGEN calculates the totals over the nuclide groups, i.e. the sum over all nuclides, for each table selected by input (concentration in g or g-A/basis, activity, heat, radiotoxicity index for inhalation and ingestion, alpha activity, gamma heat). Of decisive importance for the printout of the totals is the selection of a table of a single nuclide group, i.e. if, for example, only the printout of the activity of the actinides is desired, then the total activity of all nuclides is printed out in addition.

The totals are only calculated for post-irradiation times; the corresponding table appears in compact form with the heading "Light Elements + Actinides + Fission Products" after printout of the tables of the individual nuclide groups.

4.4.4 Separate Tables for the Alpha Activity

KORIGEN has the option of calculating the alpha activity separately. This makes it possible to classify total radioactivity according to alpha or beta activity. The table in which the alpha activity is printed out--only for the actinides--is controlled by means of the NTO field (cf. Appendix B.1, Input Description), whereby NTO > 1 is selected, unlike the previous usage. This means that a complete changeover of the NTO field was avoided, so that compatibility with earlier KORIGEN inputs--and with ORIGEN--was preserved. However, the consequence of this is not only programming that is not very pretty but also a situation in which the table of alpha activity does not directly follow the table of total activity but is not printed out until the end of each nuclide group.

4.4.5 Relative Cut-Off for Neutron Emission

For the purpose of reducing the tables for neutron emission to the most important neutron producers, a relative cut-off was introduced internally in the program. For the spontaneous fission nuclides, 10^{-4}% was assumed for the first post-irradiation time interval. The (α,n) producers appear in the table only if the neutron emission in the first post-irradiation time interval does not fall below 10^{-3}% of the total rate and at the same time 1% of the total rate in the last time interval.

4.4.6 Output of Concentrations during Irradiation in Grams and/or Gram Atoms

The output of the nuclide concentrations during irradiation was expanded to the practical unit gram/basis. The possibility therefore exists of obtaining interim results during irradiation in grams and/or gram atoms. Output of activity, radiation, etc., during irradiation does not occur.

4.4.7 Variable Output Format

The output format of some quantities which vary over a wide range (e.g. decay times, matrix elements) and must nonetheless be fit into a narrowly limited format was variably designed using the newly created subroutine CONVZ [33]. This means that either the F format or the E format is used, depending on the magnitude of the respective variables, whereby in the latter the E as well as unnecessary digits in the exponent are suppressed. This guarantees that each number is printed out with a maximum number of valid digits.

4.5 Further Changes and Modifications

4.5.1 Modified Calculation of Gamma Power

KORIGEN calculated the gamma power P_γ from the photon spectra:

$$P_\gamma = \sum_{i=1}^{N} E_i \cdot I_i \tag{4.11}$$

Here I_i is the absolute gamma intensity (photon/decay) of the respective nuclide in group i, and E_i is the mean energy of group i (the spectra of the actinides are divided into 18 groups, that of the fission products and light elements into 12 groups).

4.5.2 Neutron Multiplication Factor k_∞

For the purpose of tracing the criticality variation of the fuel under investigation, KORIGEN calculates the multiplication factor

$$k_\infty = \sum_i \bar{\nu}_i \sigma_f^i N_i / \sum_i \sigma_a^i N_i \tag{4.12}$$

Summation is done over all nuclides including those of the structural materials and the fission products formed. k_∞ is printed out during irradiation at the beginning of each time interval. If no output is desired during irradiation (JTO=1; cf. App. B.2, Input Description), k_∞ is printed out at the beginning of the first and last time interval of the irradiation phases.

4.5.3 Change in the Nomenclature of Some Nuclides

The alphanumeric nuclide names were changed to the KEDAK convention. For nuclides with a one-letter designation (U, I, H, etc.) this meant that a blank was inserted after the first letter, e.g. U235 became U_235.

4.6 Evaluation of Results

4.6.1 Permanent Storage of the Main Tables

All main tables (concentrations, activity, thermal power, radiotoxicity) can be permanently stored on a storage medium specified by the user. This takes place in subroutine WRIPLO by use of the option KPL=1/2 (cf. Appendix B.2). All tables which are selected for the print output are also stored: by nuclide, by element, by group, totals. The data are stored unformatted and are primarily suitable for further graphic processing with the Karlsruhe plotting program PLOTEASY [34].

One data record with a header line--for later identification--is stored per output table as follows:

1st record	MAX,NAME,KENN1,KENN2
2nd record	(X(I),I=1,MAX)
3rd record	(Y(I),I=1,MAX)

Here X(I) and Y(I) are the data pairs for the respective irradiation or decay phase, MAX the number of data pairs, NAME an alphanumeric designation (REAL*8) of the data record (Tab. 4.5), KENN1 a numeric designation for the respective table (Tab. 4.6), and KENN2 a numeric designation of the respective time interval.

KENN1 consists of two digits: the first digit designates the element or nuclide group, the second the type of stored ordinate (concentration, activity, etc.).

Table 4.5. Alphanumeric designation of the stored data records

Data Record	Name
Single nuclide	Nuclide name, e.g. 'U_235___'
Single element	Element name, e.g. 'PU_____'
Total over one nuclide group	'TOTAL___'
Total over one element group	'TOTALS__'
Total of the concentration in gram atoms over all nuclides	'GRATOMS_'
Total of the concentration in grams over all nuclides	'GRAMS___'
Total of the total activity over all nuclides	'CURIES__'
Total of the thermal power over all nuclides	'WATTS___'
Total of the gamma power over all nuclides	'GAM-WATT'
Total of the radiotoxicity index for ingestion over all nuclides	'HAZARD-W'
Total of the radiotoxicity index for inhalation over all nuclides	'HAZARD-A'
Total of alpha activity over all nuclides	'ALPHA-CI'

KENN2 is simply counted from 1 for each completed time interval for which printed output is produced. The total control information, i.e. MAX, NAME, KENN1 and KENN2, is printed out via the Fortran unit 3 and can therefore be used for later identification of a data record.

For example, KENN1=23 means that NPLOT=2, JT=3, i.e. that the activity of the actinides was stored by nuclide.

Table 4.6. Meaning of the numeric identification code KENN1.

Data Record	JT	Data Record	NPLOT
Concentration in gram atoms	1	Light elements	
Concentration in grams	2	- by nuclide	1
Activity	3	- by element	4
Thermal power	4	Actinides	
Gamma power	5	- by nuclide	2
Radiotoxicity for ingestion	6	- by element	5
Radiotoxicity for inhalation	7	Fission products	
Alpha activity	8	- by nuclide	3
		- by element	6
		Totals over all nuclides	JT

5. Comparison of Results of Consistent Inventory Calculations with Experimental Post-Irradiation Analyses for PWRs

Experimental post-irradiation analyses can be used for checking inventory calculations. For PWRs there are experimental studies of dissolved fuel from whole or half fuel elements--batch analyses--and of dissolved single pellets--pellet analyses.

The reliability of the comparison of the present measurements with results of the completed inventory calculations becomes clear from the following: in the PWR compensation of excess reactivity is achieved by means of boric acid, which is distributed homogeneously in the coolant. Brief changes in reactivity, e.g. for the purpose of increasing power, are accomplished in modern power reactors (KWU) by the operation of control rods, which during normal operation are largely withdrawn from the reactor core [35]. Thus the power distribution in a PWR fuel assembly is flat in the axial direction and periodic in a radial direction in accordance with the fuel rod lattice, except for brief control procedures and with the exception of the upper and lower edge zones--and the lateral edge zones if the fuel assembly is at the edge of the core. Disturbances in the radial periodicity from fuel rod to fuel rod due to water-filled guide tubes for control rods should be slight, so that on the whole one can use a one-dimensional cell-inventory calculation with periodic boundary conditions in an undisturbed unit cell representative for the fuel assembly under investigation for the purpose of determining the fuel assembly inventory.

An important aspect is consideration of spectral influences: the hardening of the spectrum due to the buildup of transuranic elements and fission products, the variable boric acid concentration of the coolant, and the energetic self-shielding which generally increases with progressive burnup. This is treated consistently in the cell calculation (cf. Chapter 3). The local power density, calculated in the case of batch analyses from the experimentally determined burnup and the irradiation history of the respective fuel element, is also taken into account.

For each batch with associated burnup, a cell-inventory calculation is performed, the result of which is compared with the measurement.

For the purpose of comparison with pellet analyses, cell-inventory calculations are also carried out for selected support points in the burnup. Between these burnup support points interpolation is carried out continuously. In this way one obtains a curve of nuclide concentration as a function of burnup which applies to pellets which are typical of the fuel assembly containing these pellets, i.e. pellets from the central axial zone of fuel rods which are located neither at the edge of the fuel assembly nor directly next to guide tubes for control rods in the fuel assembly.

In the case of pellets which come from the lower or upper axial zone of a fuel rod, from edge rods or fuel rods which are adjacent to guide tubes for control rods, it is possible for weak spectral differences to be noticeable. When plotted over burnup this leads to a slight systematic fanning out of measured values for most pellets, so that one obtains a band of measuring points instead of a measurement curve. This systematic fanning-out phenomenon in pellet analyses is superimposed on the experimental variation or scatter. An accurate comparison of measured and calculated concentrations for individual pellets would only be possible using a multi-dimensional reactor and cell calculation.

5.1 Comparison with Experimental Analyses of Dissolved Fuel Assemblies from the PWR Obrigheim (KWO)

An important position in checking the consistent inventory calculations by comparison with experimental results was assumed by the measurements which were performed in conjunction with the "Isotopic Correlation Experiment (ICE)" using irradiated fuel from the PWR Obrigheim in 1977-78 [36]. The measurements were undertaken independently by the laboratories of the European Institute for Transuranic Elements (TUI), the Institute for Radiochemistry (IRCh), the Karlsruhe Reprocessing Plant (WAK)--all of which are on the grounds of the Karlsruhe Nuclear Research Center--

and the International Atomic Energy Agency (IAEA) in Vienna. The KWO [Obrigheim Power Plant] fuel assemblies 168, 170, 171, 172 and 176 were analyzed, which had been irradiated during the 3rd, 4th and 6th KWO cycle between September 1971 and June 1975. The fuel assemblies were split lengthwise before dissolution, so that two batches per assembly were available for analysis. All laboratories participated in the determination of the concentration of U and Pu isotopes. From the variation in measured values from the various laboratories it is possible to deduce the measurement inaccuracy. The concentrations of Am and Cm, the isotope ratios of the fission products Kr, Xe, Nd and Cs and burnup were determined solely by TUI. Cell and inventory calculations for the KWO batches were performed using the method described in Chapter 3. The KWO cell data and the irradiation history of the fuel assemblies analyzed are listed in Tables 5.1 and 5.2. The dependence of the Pu-240 capture cross section on burnup was corrected using the method presented in Chap. 3. The KWO cross sections for 30.3 GWd/t burnup are compiled in the appendix. This set of cross sections is available for inventory calculations using KORIGEN in the KORIDATA library under the designation "33P3U_31_KFK_KWO."

Table 5.1. KWO reactor data [40]

Fuel	UO_2
Initial enrichment	3.13 wt% U-235
Thermal power	907.5 MW
Fuel density (U)	8.55 g/cm^3
Coolant density	0.789 g/cm^3
Cell:	
Fuel rod diameter	0.93 cm
Cladding wall thickness (Zircaloy)	0.7 mm
Moderator-fuel volume ratio	1.99
Mean temperature [°C]	
Fuel	755
Cladding	332
Coolant	299

Table 5.2. Power history, 3rd-6th KWO cycle [41]

Time [d]	Operating State	KWO-Cycle	Time [d]	Operating State	KWO-Cycle
5.8	VL[1]		126.9	VL	
1.0	NL[2]		9.2	NL	
79.6	VL		84.7	VL	
2.0	NL		3.5	NL	4
33.5	VL		23	VL	
41.5	NL	3	3.0	NL	
131.5	VL		60	VL	
5.8	NL		13.8	87% VL	
36.9	VL		380	NL[3]	5
28.0	NL		143.7	VL	
			3.0	NL	6
			100.	VL	

[1]VL = full load [2]NL = zero load [3]fuel assemblies 380d out-of-pile

For checking individual batches, the time-dependent specific power of the respective fuel elements is required. This is obtained from the experimental burnup values and the irradiation history (Tab. 5.2) in accordance with

$$P \cdot \sum_i p_i \, \Delta t_i = B$$

where B is the experimental burnup, Δt_i the irradiation interval, p_i the ratio of the reactor power run in t_i to the maximum reactor power (p_i and Δt_i should be derived from Table 5.2), and P the specific power of the respective fuel assembly which is to be calculated. Summation should be performed over all irradiation intervals Δt_i. The specific power in Δt_i is obtained as p_i P (generally p_i = 100%).

The burnup values, determined radiochemically, involve an inaccuracy of ±(3-5)%. This can be deduced from experiments in which the burnup has been measured using several methods and/or by several laboratories [38]

(cf. also Sec. 5.2, 5.3). In ICE, where only one single burnup measurement is available per batch, one can also deduce the uncertainty in burnup by comparisons with the operator data for the burnup of a whole fuel assembly. Measurements of neutron emission in irradiated KWO fuel assemblies and the use of the experimental Cm-244-burnup correlation [37] showed that the operator data are very accurate (Tab. 5.3). If one compares the burnup values based on TUI measurements for the fuel assemblies analyzed in ICE with the operator data (Tab. 5.4), then one finds deviations ranging from -4.4 to +3.4%.

Table 5.3. Comparison of the burnup values of some KWO fuel elements according to operator data and based on measurement of neutron emission (after [37])

Fuel Ass.	Burnup [GWd/tU] Operator	Measurement [37]
169	24.5	25.2 ± 0.5
222	30.6	31.3 ± 0.9
166	24.2	24.3 ± 0.5
243	27.4	27.4 ± 0.6
237	31.3	31.3 ± 0.8
224	30.6	30.5 ± 0.8

5.1.1 Comparison between Calculation and Measurement for U and Pu Isotopes

The results of the consistent inventory calculations are compared with the corresponding measurements of TUI, WAK, IRCh and IAEA for the U and Pu isotopes in Figures 5.1 to 5.7 and will be discussed in the following.

U-235: the measuring points vary by up to ca. ±2% uniformly around the calculated curve.

U-236: is underestimated by ca. (1-2)% by the calculation. The experimeantal spread is ca. ± 1%.

Table 5.4. Comparison of the burnup values [41] measured radiochemically by TUI with the operator data [41] for the KWO fuel elements analyzed in conjunction with ICE.

Fuel As.	Batch	Burnup [GWd/tU] TUI	Operator	Deviation* [%]
168	86 87	28.4 30.3 }29.35	30.018	- 2.23
171	88 89	28.4 29.08 }28.74	30.052	- 4.4
176	90 91	29.58 27.98 }28.78	29.647	- 2.9
172	92 93	26.53 29.27 }27.90	26.979	+ 3.4
170	94 95	25.93 28.08 }27.0	27.764	- 2.7

* (TUI-operator)/operator

Pu-238: the experimental spread is very great: ca. ± 15%. The Pu-238 concentration is comparatively difficult to measure and must be corrected to the fuel discharge time based on the alpha decay of Cm-242. Since Cm-242 was only measured by TUI in ICE, only the TUI values for Pu-238 were corrected accordingly. The calculated Pu-238 concentrations approach the TUI measured values within ca. 10%, whereas the measured values of the other laboratories are ca. 20-25% above the calculated values.

Pu-239: the measuring points scatter uniformly around the calculated curve on the average of ca. ±3%. For the two main fissile materials in the LWR fuel, U-235 and Pu-239, there is therefore very good agreement with the existing measurements.

Pu-240-242: are somewhat underestimated in each case by the calculation. The deviations--ca. 5% for Pu-240, ca. 6% for Pu-241 and ca. 9% for Pu-242--are, however, not substantially beyond the spread of the measured values: ± 3% for Pu-240, ± 4% for Pu-241 and 242.

5.1.2 Comparison between Calculation and Measurement for Am and Cm Isotopes

Radiochemical measurements of the concentrations of Am and Cm isotopes involve great uncertainty. This makes comparison between calculation and measurement difficult. In conjunction with ICE, measurements of Am and Cm were carried out solely by TUI. Therefore there is no possibility for comparison with measurements of other laboratories, by means of which the measurement inaccuracy could be quantified. The experimental uncertainty, however, can be estimated by reference to other experiments (Sec. 5.2, 5.3) and by analysis of the measurement methods.

Am-241 is determined by alpha spectrometry. The alpha spectrum of Am-241, however, is almost completely covered by that of Pu-238. The measurement is therefore performed such that the alpha spectrum of Pu-238 and Am-241 are recorded before Pu separation, and that of Pu-238 only after Pu separation. The alpha spectrum of Am-241 is determined finally by the difference between these two spectra, i.e. by the difference between two numbers which are virtually the same size and involve errors. Furthermore, the Am-241 concentration is calculated back to the discharge time (the measurement takes place ca. 3 yrs after unloading). Since Am-241 is formed for the most part (ca. 75%) out of pile from Pu-241, the relative error increases four-fold upon back calculation in accordance with the concentration ratios! I.e. with a measurement error of ± 25% in Am-241--a realistic value--an error of about ±100% can be expected at unloading.

Calculation and measurement accordingly deviate up to 50-60% from one another. In the case of measurements of TRINO2 pellets (Sec. 5.2) the differences, on the other hand, are only ca. 5% maximum: in this case the researchers did not calculate back.

An independent support for the calculation is obtained from the comparison of the resonance integral or 2200 m/s cross sections with experimental values (Tab. 5.5): in particular, very good agreement is apparent for the neutron capture of Am-241. A sensitivity analysis moreover showed that the high experimental values for the concentration of Am-241 cannot be achieved even with an imperceptible absorption cross section of Am-241. On the basis of the good agreement of the Pu isotopes, calculation of the Am-241 concentration should therefore be reliable.

Measurements of the Am-241 concentration of individual KWO pellets published in the interim, which were performed for the first time using isotope diluting analysis [49], agree very well with the calculated Am-241 concentrations, while the alpha-spectrometric measured values in part differ a great deal (cf. Sec. 5.2).

Table 5.5. Comparison of integral KEDAK data with experimental values.

	Resonance Integral[a] for n-capture [b]		2200 m/s cross sections [b] for n-capture	
	KEDAK	Experiment	KEDAK	Experiment
Am241	1549[b]	1538 ± 135[b] /44/	610	832 ± 20[c] /44/ 853 ± 68[c] /45/
Am243	1847	2210 ± 150 /45/	79.3	73 ± 6 /47/
Am244	637	650 ± 50 /46/	14.4	14 ± 4 /46/

a) $\int_{E_c}^{15MeV} \sigma(E) \frac{dE}{E}$, $E_c = 0.5eV$ b) $E_c = 0.369eV$

c) These are effective values which are consistent with the KEDAK value [48].

Am-243 is determined by mass spectrometry by way of the ratio Am-243/Am-241. The direct error is the Am-241 determination therefore also enters into the determination of Am-241 [sic: probably error for "Am-243"--Transl. Note]. The calculation yields ca. 30% higher concentrations than the measurement within the measurement uncertainty.

Cm-242 exists in very small concentrations (ca. 100 mg/t U) due to its short half life ($T_{1/2}$=163 d), and the measuring inaccuracy is accordingly great: sometimes more than 100%, as can be determined from comparative measurements (Sec. 5.2, 5.3). For the KWO batches the calculation yields concentrations of Cm-242 which are ca. 30% smaller than the measurement.

Cm-244 can be determined comparatively easily by alpha spectrometry due to its pronounced alpha spectrum, its sufficiently long half life ($T_{1/2}$ = 18 yrs) and its adequate concentration. As recent comparative measurements of KWO pellets show (cf. Sec. 5.2)--for the first time isotope dilution analysis was used to determine the Cm-244 concentration--, the measuring inaccuracy must nonetheless be assumed to be ca. ± (20-30%) (cf. Sec. 5.2). Furthermore, Cm-244 is highly sensitive to uncertainties in burnup, since the Cm-244 concentration increases more or less with the 4th power of the burnup [37]. The differences between calculated and measured Cm-244 concentrations are within the measuring inaccuracy at 20-25%.

5.1.3 Comparison between Calculation and Measurement for Several Isotope Ratios of Fission Products

For isotopes of the fission products Kr, Xe, Nd and Cs, atomic ratios were measured by TUI in conjunction with ICE. This offers the possibility of also checking the Karlsruhe cell-burnup method in the range of the fission products using post-irradiation analyses. Unlike the situation with the actinides, only ORNL cross sections have be used to date for the fission products. This is generally sufficient, as shown by the comparison between calculation and measurement in the case of the KWO example (Fig. 5.8, 5.13-5.20). For Cs-134/Cs-137, for example, calculation and measurement agree to within about 3%. KfK cross sections for fission products are currently being developed.

5.2 Comparison with Experimental Analyses of Pellets from the PWR Obrigheim

For KWO fuel there are also pellet measurements available which were performed jointly by TUI and the "Joint Research Center at Ispra" (JRC) [38]. The analyzed pellets are from fuel rods of the KWO fuel assembly nos. 124 and 210. Fuel assembly no. 124, which was irradiated in the 2nd, 4th and 5th KWO cycle, had an initial enrichment of 3.0 wt% U-235, and fuel assembly no. 210--irradiated in the 3rd to 5th KWO cycle--was initially enriched with 2.83 wt % U-235. Becuase of the different initial enrichment, the different irradiation history and the different U densities (cf. Tab. 5.6), cell-inventory calculations were carried out for each fuel assembly type, and cross section sets were generated according to the standard method (Chap. 3).

Table 5.6. Input data for KWO fuel assemblies no. 124 and no. 210 which differ from Table 7.1 [38]

	Fuel Ass. 124	Fuel Ass. 210
Initial enrichment [wt % U-235]	3.0	2.83
Uranium density [g/cm^3]	9.17	8.95

Table 5.7. Irradiation history of KWO fuel assemblies 124 and 210 (after [38]).

Time /d/	KWO Cycle	FA 124	FA 210
258	2	VL	NL
48		NL	NL
295	3	NL	VL
27		NL	NL
283	4	VL	VL
23		NL	NL
229	5	VL	VL

The calculated actinide concentrations are plotted in Fig. 5.21 through 5.31 over burnup. The measured values from the pellet analyses, which in the best case are spread around the calculated curves, are also shown.

For the U isotopes 235 and 236 the same picture is evident as in ICE: U-235 agrees very well with the measurements. U-236 is slightly underestimated. The measured values vary more sharply than in the batch measurements, as expected.

In the case of Pu-238 the calculated curves are well enclosed by the highly scattered measured values. From comparison measurements of the two laboratories one can deduce a measuring inaccuracy of up to ca. 25%. For Pu-239 the agreement is not as good as in ICE: for FA no. 124 the calculation yields a slight overestimation, and for FA no. 210 a somewhat greater overestimation (cf. Fig. 5.24). The measured values do vary greatly. For the Pu isotopes 240-242 there is very good agreement: the calculated curve runs in each case through the family of measuring points.

The comparison between calculated Am-241 concentrations and those measured with alpha spectrometry (Fig. 5.29) demonstrates essentially the very great inaccuracy of the alpha-spectrometric measurements. Recent measurements performed with more exact isotope dilution analysis [49] convirm the calculated Am-241 concentrations (cf. Fig. 5.29): the deviation between calculation and measurement is a maximum of 16%, whereas the alpha-spectrometric measured values are up to a factor of 3 greater than the calculated values. For Am-243 there are only very few measured values, which however are close to the calculated curves.

Cm-242, like Am-241, involves great experimental uncertainty, as one can see from the great spread of measured values (Fig. 5.31). Comparison measurements of the same pellet differ by factors of 2 to 3!

Even recent measurements [50]--also alpha spectrometric--show differences of up to 90%. The calculated curves of Cm-242 concentrations run through the lower measuring points.

Cm-244 is calculated very well; the calculated curves lie within the family of measuring points (Fig. 5.30).

Recent measurements [50] of single KWO pellets--the much more precise method of isotope dilution analysis was used for Cm-244 for the first time--show differences from the older alpha-spectrometric measurements which are as much as ± 40%. The recent measurements (Fig. 5.32) are almost exclusively such that agreement with the calculated values is better. The burnup was also redetermined in conjunction with this measurement; the differences from the older values are between -10 and +10%, which in part yields distinct shifts in the Cm-244 concentration values in the concentration/burnup diagram (Fig. 5.32).

In summary, very good agreement between calculated and measured concentrations of U and Pu isotopes and of Cm-244 can be noted for the example involving KWO pellets. As expected in pellet measurements, the measured values vary greatly, however. For Am-241 and Cm-242 the great experimental inaccuracy of alpha-spectrometric measurement is very clearly shown.

5.3 Comparison with Experimental Analyses of Irradiated Pellets from the PWR Trino Vercellese

Pellet measurements were also performed using irradiated fuel from the PWR Trino Vercellese [39].

The pellets analyzed were taken from fuel rods of the Trino fuel assembly no. 509-069. This fuel assembly (cf. Tab. 5.8) was irradiated between October 1964 and July 1971 over four cycles in the Trino reactor. As a particular feature, FA no. 509-069 has an interim out-of-pile residence time of 3 years before renewed irradiation (the Trino reactor was shut down during this period).

In Figures 5.34 to 5.41 the consistently calculated actinide concentrations are plotted in comparison with the measured values. U-235 is also calculated very well in the case of the Trino pellet. U-236 is slightly

underestimated. The Pu isotopes 238-242, i.e. also Pu_{tot}, are underestimated here--to different degrees.

Table 5.8. Cell data of the Trino FA no. 509-069.

Thermal power	825 MW
FA No. 509-069:	
Fuel density (U)	9.32 g/cm^3
Coolant density	0.765 g/cm^3
Fuel rod diameter	0.902 cm
Cladding wall thickness	0.383 mm
Moderator/fuel volume ratio	1.79

Table 5.9. Irradiation history of the Trino FA No. 509-069

Time [d]	Trino Cycles	State
226	1	VL
86		NL
263	2	VL
51		NL
292	3	VL
1117		NL
416	4	VL

VL = full load
NL = zero load

Am-241 was not calculated back to discharge by the experimenters in the present case, so that the relative uncertainties in the experimental Am-241 concentrations stem from the determination of the alpha spectra of Pu-238 + Am-241 and Pu-238 and the subsequent subtraction (cf. Sec. 5.1.2). Calculation and measurement agree relatively well with one another in this case, with several exceptions (Fig. 5.40). The

measurements of Am-243 are very informative: in the case of several measured data the corresponding Am-241 measured values were lacking (Am-243 is determined by mass spectrometry via Am-243/Am-241). In these cases a mean Am-241 concentration of $246 \cdot 10^{-6}$ IMA (cf. [39]) was used as a basis by the researchers. For these measured points the Am-243 values are very close to one another (cf. Fig. 5.40), whereas the other values vary greatly, apparently due to the less accurate Am-241 measurements. For Am-243 there are comparison measurements in the case of two pellets; the differences are 2 and 16%. The former difference, however, only relates to the mass spectrometric measurement of Am-241/Am-243.

Cm-242 and Cm-244 are underestimated before calculation on the average of 20 to 30%. For both isotopes there are comparison measurements in the case of five pellets. The differences are small here for Cm-242--up to ca. 5%, but greater for Cm-244--up to ca. 10%.

5.4 Supplementary Comparisons with Experimentally Analyzed PWR Batches

5.4.1 SENA Reactor

The French PWR SENA (Société d'Energie Nucléaire Franco-Belge des Ardennes) is comparable to KWO in power, but has distinctly smaller cell dimensions (Tab. 5.10). For comparison calculations, measurements of the 2nd reprocessing campaign involving spent SENA fuel assemblies are available [42].

These measurements included 12 batches which were initially enriched to 3.358 wt % U-235. The final burnup was between 18.6 and 20.6 GWd/tU. The concentrations of U and Pu isotopes were measured.

The relative differences between calculated and measured concentrations are listed in Table 5.11. As with KWO and Trino, there is very good agreement for U-235 and Pu-239, but also for Pu-240. Pu-241 is slightly underestimated, and Pu-242 somewhat more greatly underestimated. Pu-238 is greatly underestimated at ca. 30%, but the measured values were not corrected based on the alpha decay of Cm-242.

Table 5.10. SENA reactor data

Fuel	UO_2
Initial enrichment	3.358 wt % U-235
Thermal power	905 MW
Cell:	
Fuel rod diameter	0.902 cm
Cladding wall thickness	0.38 mm
Moderator/fuel volume ratio	1.61

Table 5.11. Relative deviations [%] between calculated and measured concentrations of U and Pu isotopes for SENA batches

Batch Nr.	Burnup [GWd/tU]	U235	U236	Pu238	Pu239	Pu240	Pu241	Pu242
100	18.6	-1.0	-3.8	-37	+1.1	+.05	-2.7	-13.9
200	20.3	+.06	-5.6	-38	+2.1	-1.3	-5.3	-16.3
300	19.4	-1.4	-3.4	-33	+1.1	+0.2	-2.4	-12.5
400	20.2	-0.3	-4.5	-35	+1.9	-1.4	-4.4	-17
500	19.4	-1.7	-4.5	-30	+1.0	+0.9	-3.0	-12
600	20.3	-0.6	-3.0	-33	+1.5	-0.8	-4.3	-15
700	19.4	-0.6	-2.2	-32	+1.2	-0.5	-3.2	-14
800	20.1	-0.1	-3.7	-33	+1.8	-1.6	-4.4	-16
900	20.3	-0.3	-4.5	-34	+1.9	-1.0	-4.4	-17
1000	20.6	-0.5	-11	-35	+1.6	-0.1	-4.3	-14
1100	20.3	+0.2	-4.2	-33	+1.9	-0.9	-4.6	-16
1200	19.7	+0.2	-9.2	-28	+0.9	+0.3	-2.2	-13

5.4.2 Robinson Reactor

The American PWR H.B. Robinson is a reactor of medium power with a thermal output of 2300 MW. The cell dimensions are similar to those of the French SENA reactor (Tab. 5.12). Measurements of isotope frequencies of the actinides Am, Pu, Am and Cm are available [43]. The experimental analyses were performed ca. 2 years after removal from the reactor using two dissolved fuel rods from the Robinson fuel assembly B05 having a burnup of 31364 MWd/t. The initial enrichment was 2.561 wt % U-235.

The deviations between calculated and measured isotope frequencies are listed in Table 5.13. For the U and Pu isotopes, the accustomed good agreement is obtained.

For the isotope frequencies of Am-241, 243 and Cm-244, very good agreement is also achieved. This may lie be due, on the one hand, to the fact that the measured values were not calculated back to the time of unloading from the reactor (e.g. Am-241) and, on the other hand, to the fact that the measured quantities are relative quantities (e.g. Cm-244 fraction of Cm ca. 90%, Cm-242 is for the most part decayed).

Table 5.12. H. B. Robinson reactor data [43]

Fuel	UO_2
Initial enrichment	2.561 wt % U-235
Thermal power	2300 MW
Cell:	
Fuel rod diameter	0.9294 cm
Cladding wall thickness	0.617 mm
Moderator/fuel volume ratio	1.62

Table 5.13. Comparison between calculated and measured isotope frequencies of irradiated fuel rods from the Robinson Reactor.

	Measurement [%][43]	Calculation [%]	Difference [%]
U235/U	0.653	0.652	-
U236/U	0.347	0.336	-3.2
U238/U	98.99	99.0	-
Pu238/Pu	1.56	1.41	-9.7
Pu239/Pu	55.79	56.8	1.9
Pu240/Pu	24.91	23.24	-6.7
Pu241/Pu	12.29	12.90	5.0
Pu242/Pu	5.45	5.61	3.5
Am241/Am	62.4	61.5	-1.4
Am243/Am	37.29	38.8	3.9
Cm242/Cm	3.28	2.49	-24
Cm244/Cm	90.66	89.4	-1.4

5.5 Summary Evaluation for PWR Fuel

Extensive checking of experimental post-irradiation analyses (Sec. 5.1-5.4) shows the ability of the KfK methods and data to reliably calculate PWR inventories. This was demonstrated for different initial enrichment levels (2.6-3.4 wt % U-235) and for different lattices (V_M/V_B = 1.61-2.0). Absolute concentrations of U and Pu isotopes can be calculated with satisfactory accuracy. For U-235 the agreement between measurement and calculation is very good. Calculations of the concentrations of the transplutonium isotopes Am-241, Am-243 and Cm-244 are more reliable than the common alpha-spectrometric measurements, as shown by the good agreement obtained with recent precision measurements using isotope dilution analysis. For important fission product ratios good agreement between calculation and measurement is noted. From the extensive comparisons of calculations with experiments one can conclude that the KfK methods and data for consistent inventory calculations are suitable for reliably calculating inventories

for PWR, even for PWR for which no comparisons with experiments have yet been performed but for which consistent cross section sets have been generated: e.g. BIBLIS fuels with 3.2-3.6 wt % initial U-235 with high final burnups of 33-40 GWd/tU aimed at better uranium utilization.

5.6 Comparison with Measurements of Pu Fuel from KWO

When recycled plutonium is used in a LWR ("thermal recycling"), the neutron spectrum in the fuel becomes comparatively hard because the thermal absorption cross sections of the fissile Pu isotopes are distinctly higher than those of the U isotopes. The effective cross sections of U and Pu fuel differ greatly, so that it is absolutely essential for inventory calculations for Pu fuel elements that consistent effective cross sections be used. A corresponding data record for the ORNL reference reactor (cf. Chap. 3) is available in KORIGEN--the creation of a KfK data record for Pu fuel is planned.

The ORNL data record for Pu fuel--2.91% Pu_{fiss}, 0.65% U-235--was used to check irradiated Pu fuel from KWO--3.2% Pu_{fiss} in U_{nat}. This is justified for initial comparison calculations because the lattices of KWO and the ORNL reference reactor are very similar. The different initial enrichment is of secondary importance in this case; what is important is consideration of the hard spectrum in the Pu fuel (for example, the absorption cross sections of U-235, Pu-239 and Pu-241 for Pu fuel are ca. 50% smaller than for U fuel).

The comparison of the calculated concentrations of the Pu isotopes 238, 240, 241, 242 (Fig. 5.42) and of Am-243 and Cm-244 (Fig. 5.43) relative to Pu-239 with the measurements shows good agreement for the Pu isotopes 240-242 over the entire burnup range. For Pu-238, Am-243 (in this case there is only one measuring point) and Cm-244 as well, the differences are not too great. The measured values were made available to us by KWU [Kraftwerk Union] [54].

5.7 Comparison of the Calculated Decay Heat from the Thermal Fission of U-235 with Experiments

For comparison with the calculated decay heat of U-235, the revised ANS 5.1 standard of 1979 [51] was used along with recent measurements [52, 53]. The revised ANS standard is based on several experiments which were supplemented by calculations.

The decay power of U-235 for a short irradiation pulse ($\phi_{th}=10^{15}$ n/(cm^2·s), Δt = 0.1 s) and the decay heat for a quasi-infinite irradiation period (in the case of the KORIGEN calculation T = 4 yrs >> cooling time) were calculated and compared, in each case in the time range of 1 s - 10^4 s after interruption of the chain reaction.

In the case of the pulse, by means of which the amount of short-lived fission products can be estimated, one obtains after ca. 20 s cooling time good agreement between the calculation and the ANS standard as well as the recent measurement of Dickens [53] (Fig. 5.44). In the time range 1 s < t < 20 s one obtains an underestimation of the ANS or Dickens values of max. 20%.

In the case of quasi-infinite irradiation, one finds after ca. 20 s cooling time very good agreement between KORIGEN calculation, ANS standard, and recent measurements (Fig. 5.45). In the time range < 20 s the ANS values are underestimated by the calculation by a maximum of ca. 6%. The recent values of Dickens et al. [53], on the other hand, also agree in this time range with the KORIGEN calculations.

Figures 5.1 to 5.20: Graphic comparison of the actinide and fission product concentrations from KWO batches measured by ICE with results of consist KORIGEN calculations.
Figures 5.21 to 5.32: Graphic comparison of measured actinide concentrations from KWO pellets with results of consistent KORIGEN calculations.
Figures 5.33 to 5.41: Graphic comparison of measured actinide concentrations from TRINO2 pellets with results of consistent KORIGEN calculations.

Translators Note: in the following figures (Abb. = Fig.), terms requiring translation are keyed as follows:

A burnup
B measurement
C open symbols: precision measurement (isotope dilution analysis)
D Precision measurement (isotope dilution analysis)
E 3 yrs after unloading from the reactor
F KWO Pu fuel rods

The abbreviations TUI, WAK, IAEA, IRCh and KWO are explained on pages 65-66.

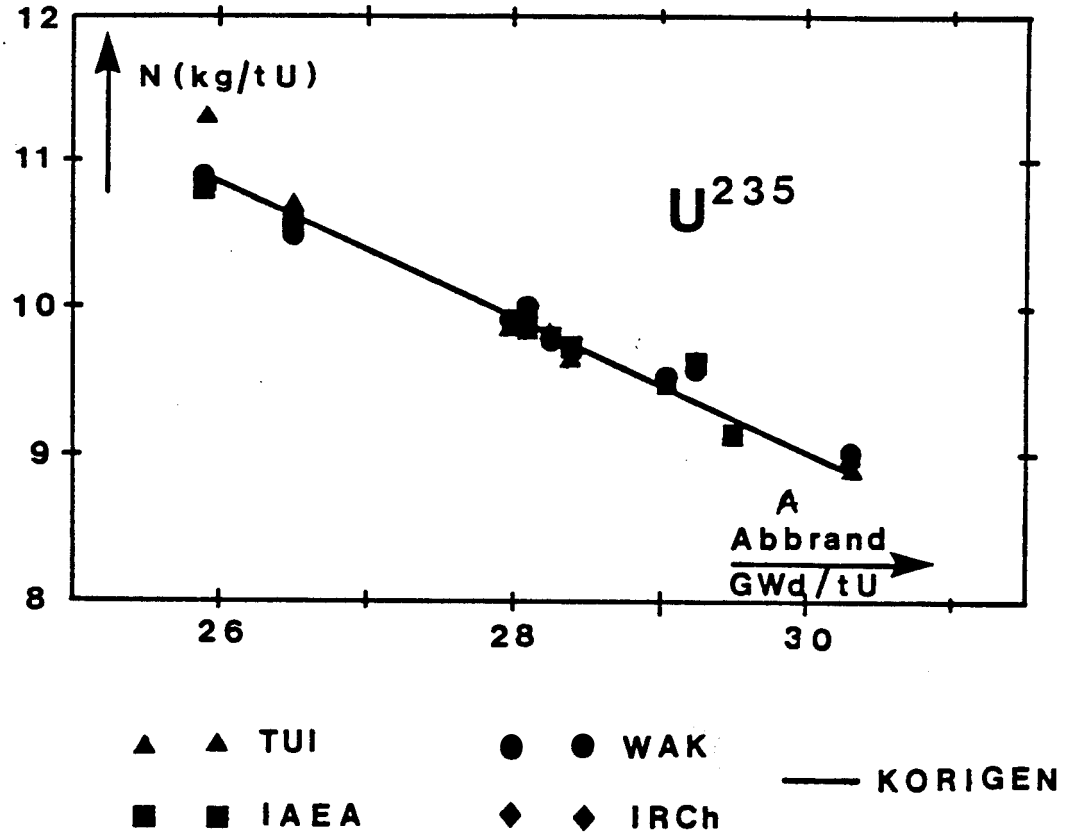


Abb.5.1

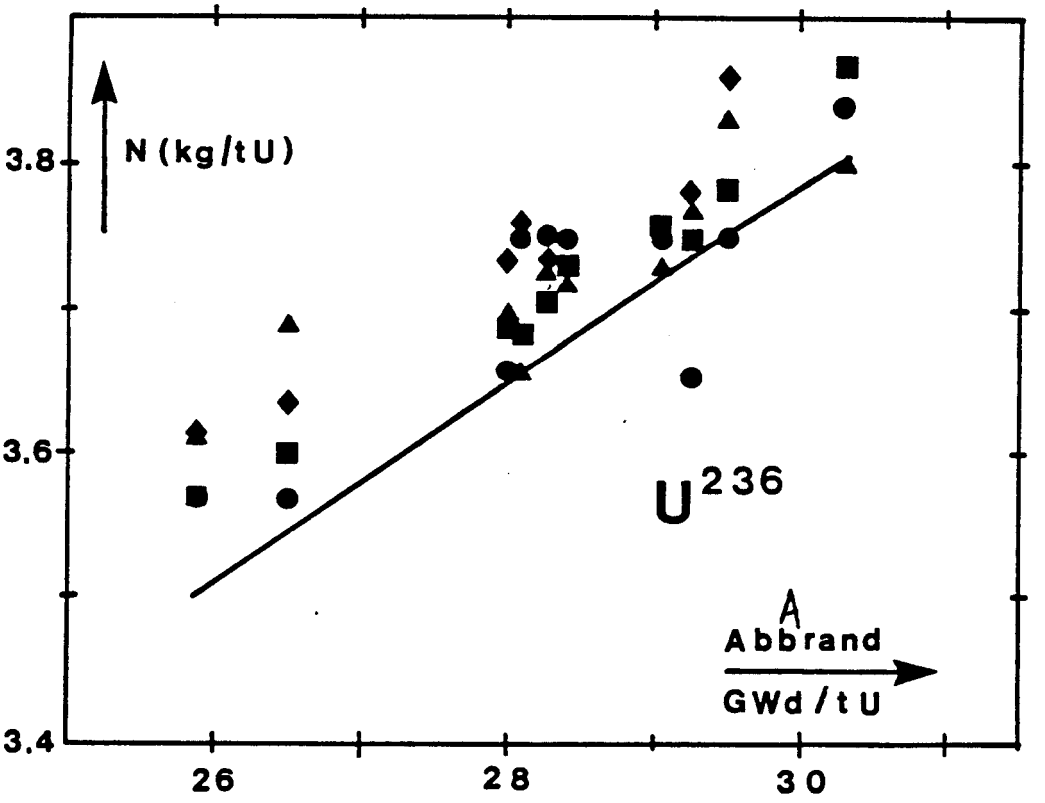
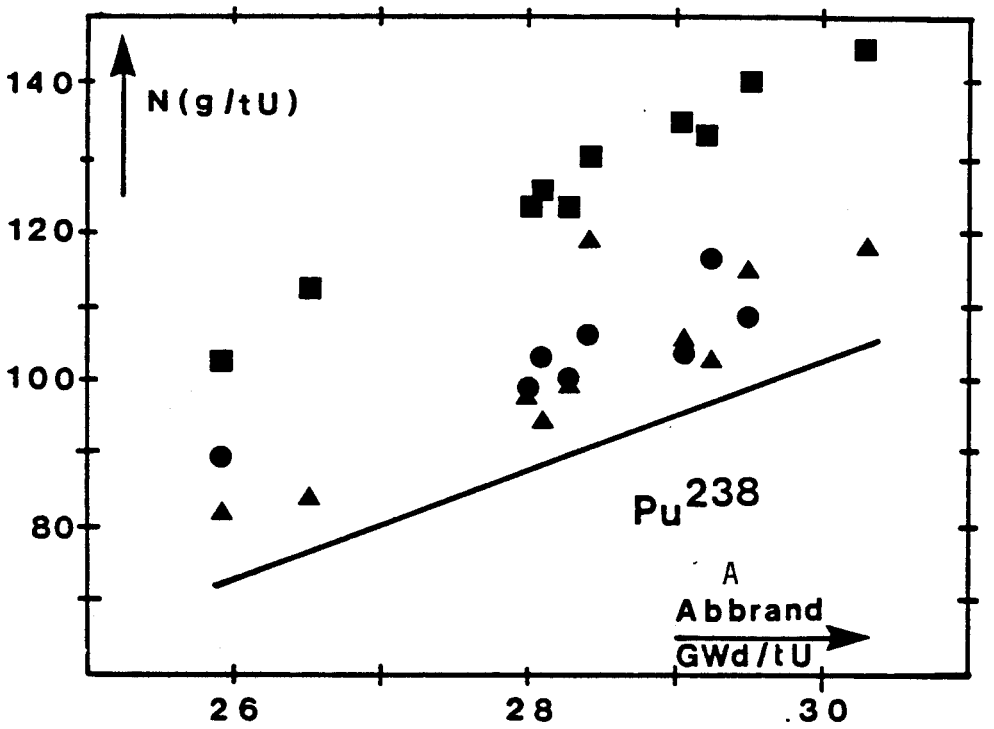


Abb.5.2

KWO-Batches (ICE)



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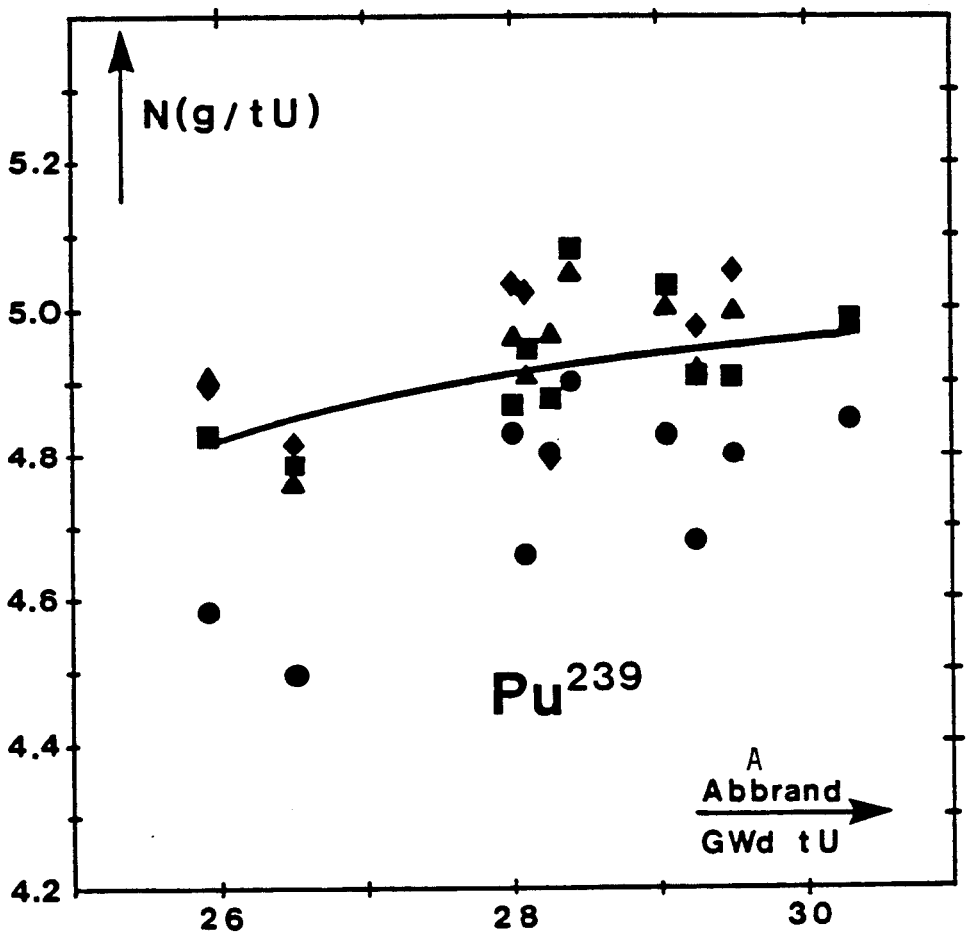


Abb.5.4

▲ ▲ TUI ● ● WAK
■ ■ IAEA ◆ ◆ IRCh —— KORIGEN

KWO-Batches (ICE)

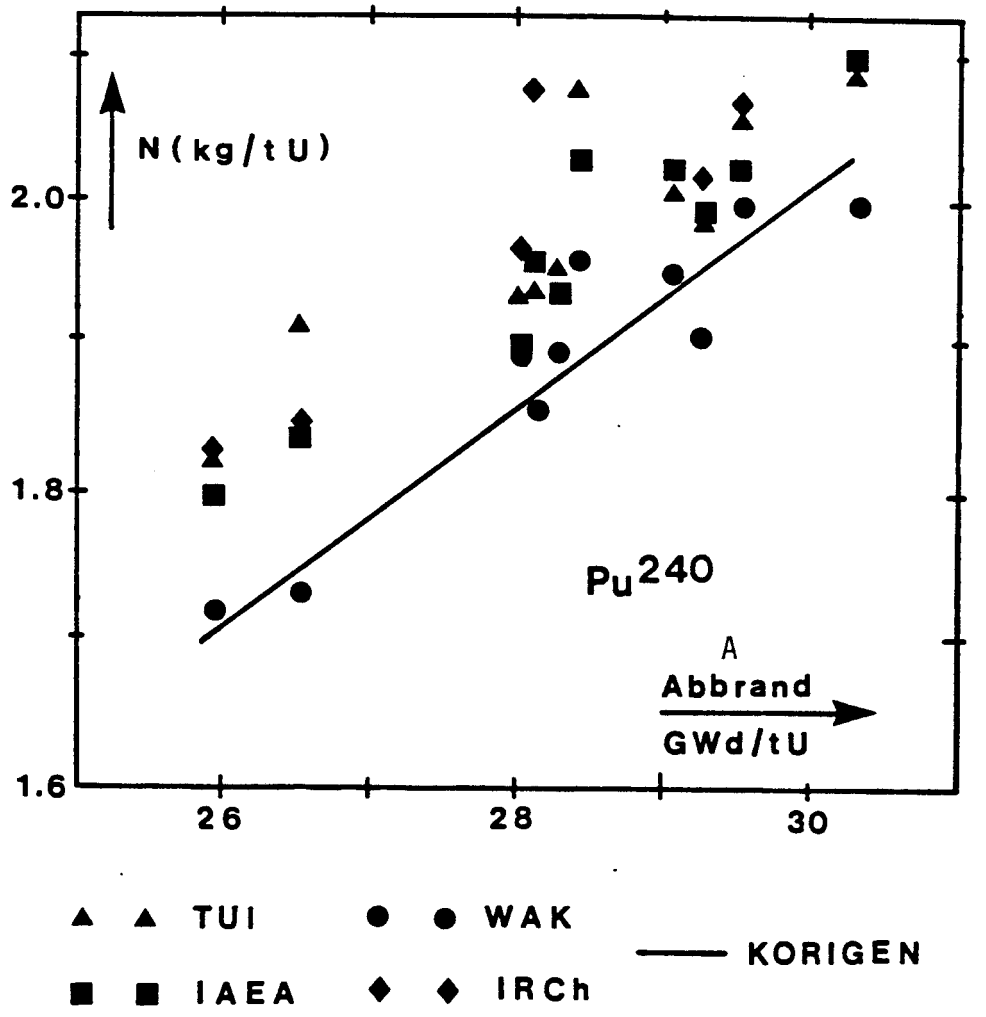


Abb.5.5

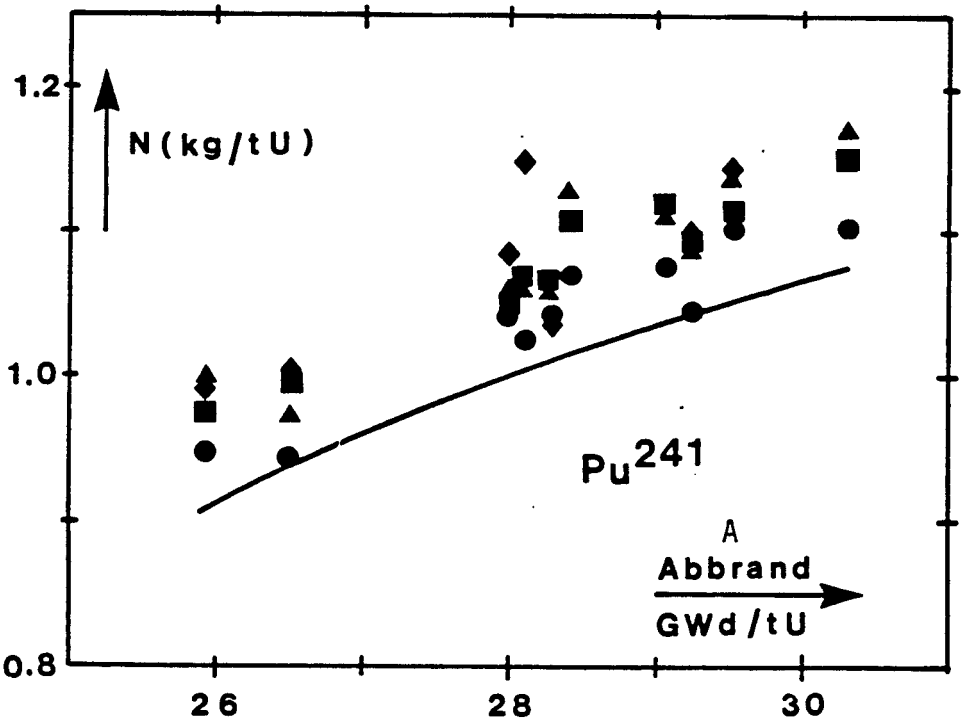
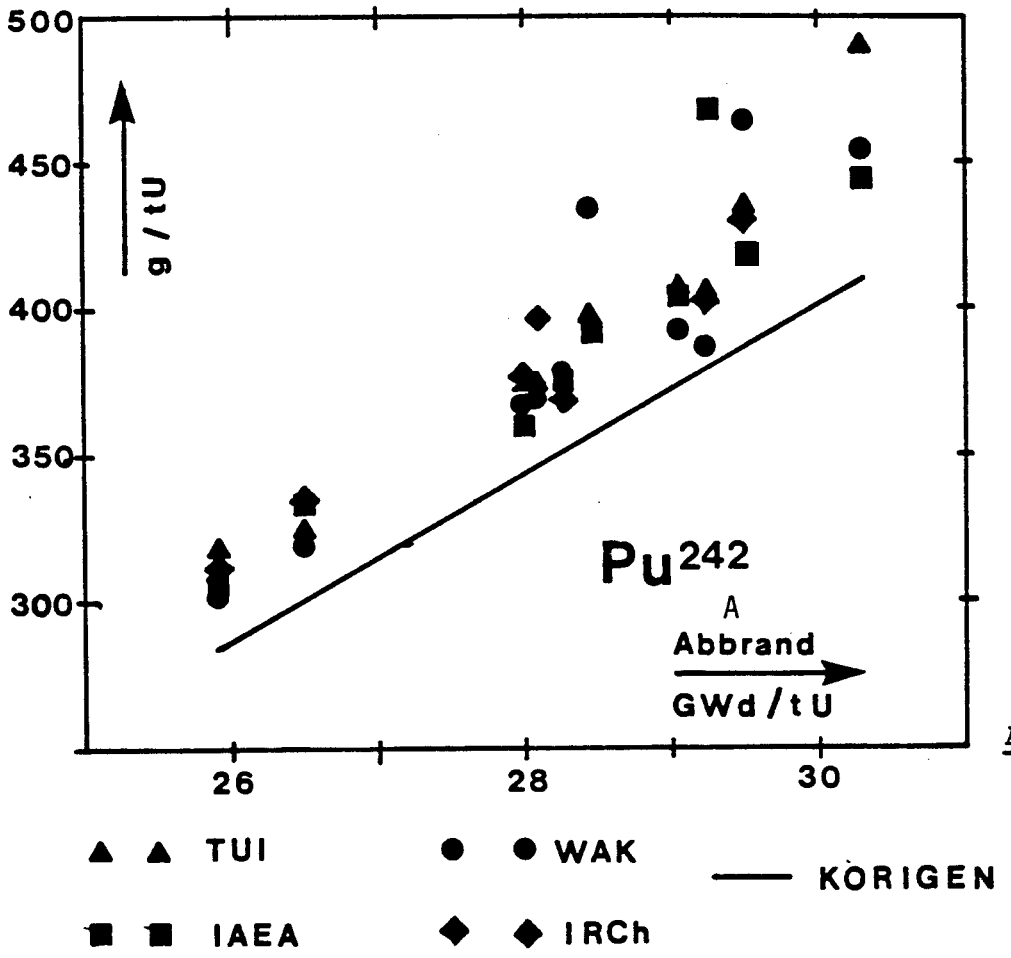


Abb.5.6

KWO-Batches (ICE)



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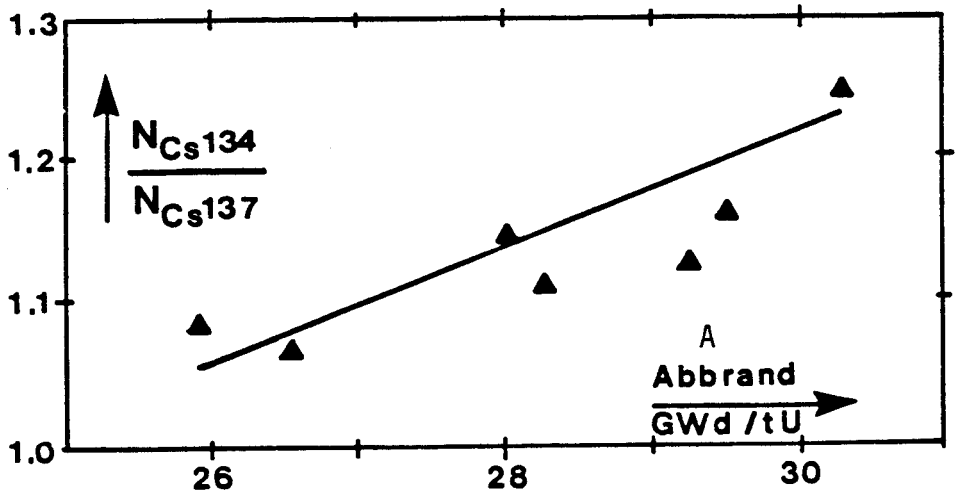


Abb.5.8

KWO-Batches (ICE)

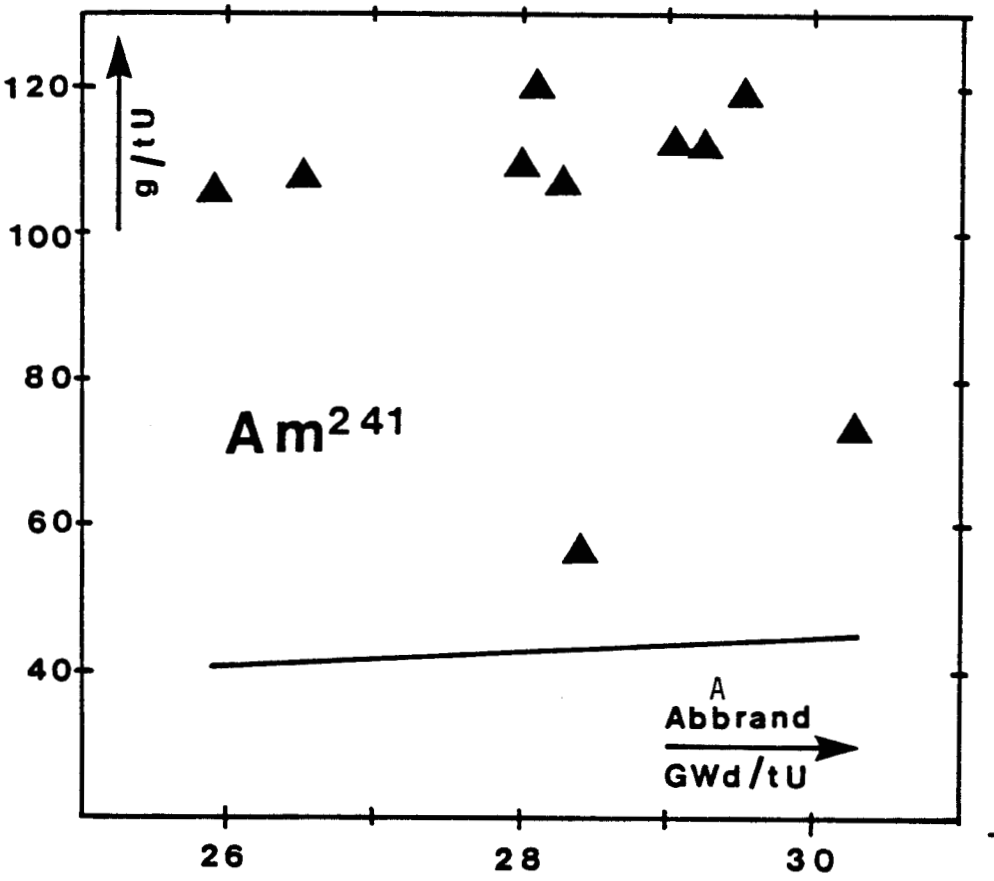


Abb.5.9

B

▲ ▲ Messung TUI —— KORIGEN

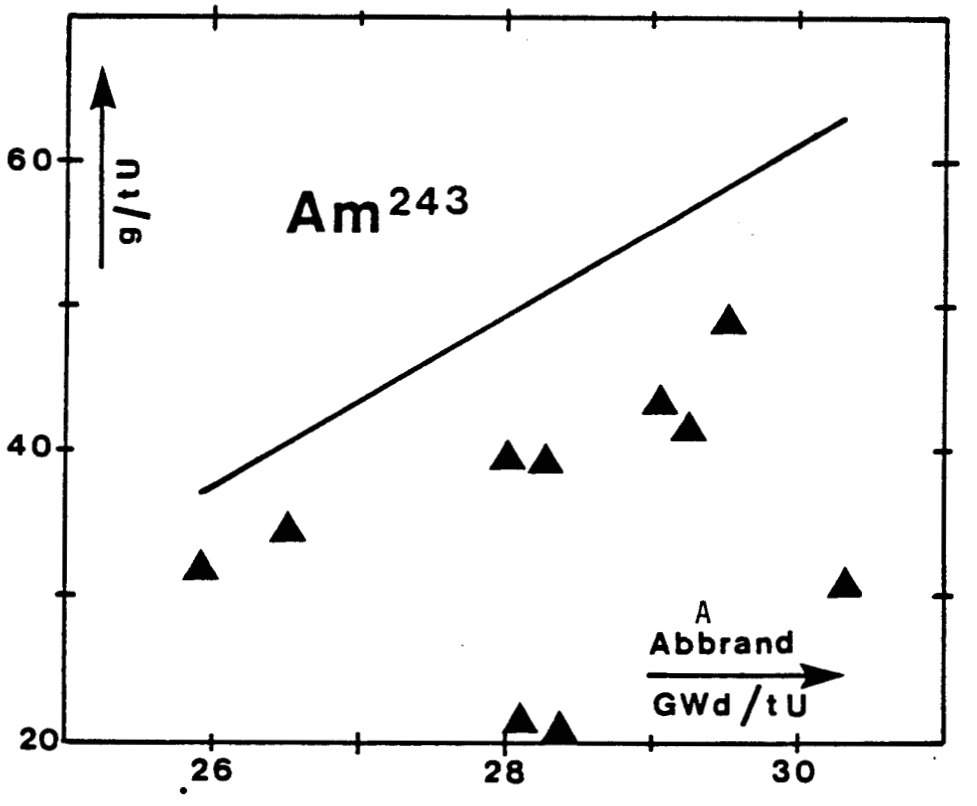


Abb.5.10

KWO-Batches (ICE)

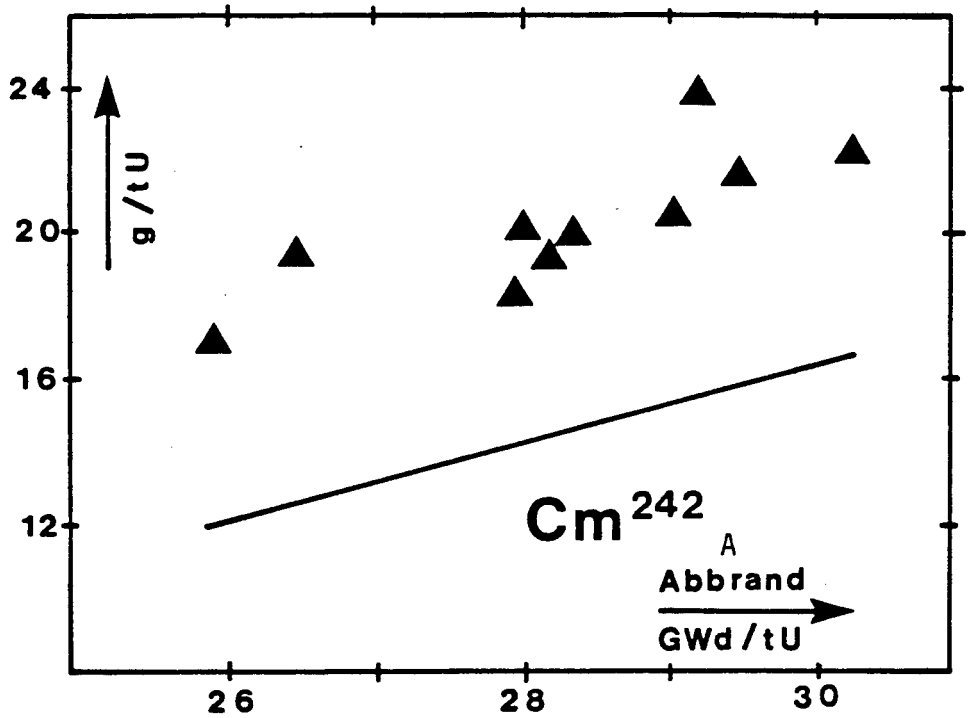


Abb.5.11

B

▲ ▲ Messung TUI —— KORIGEN

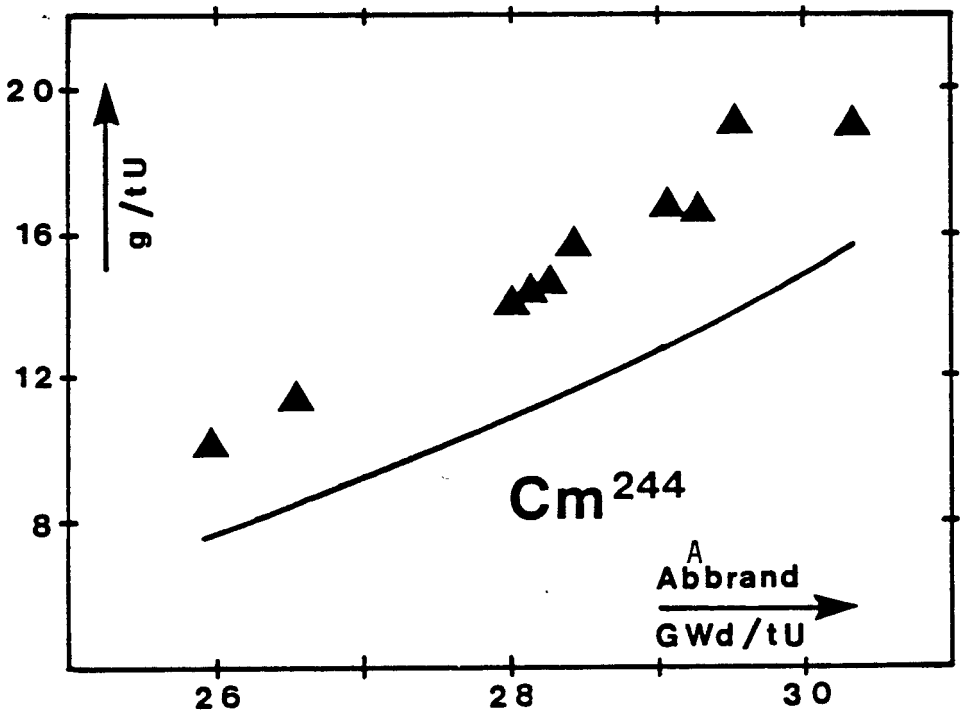
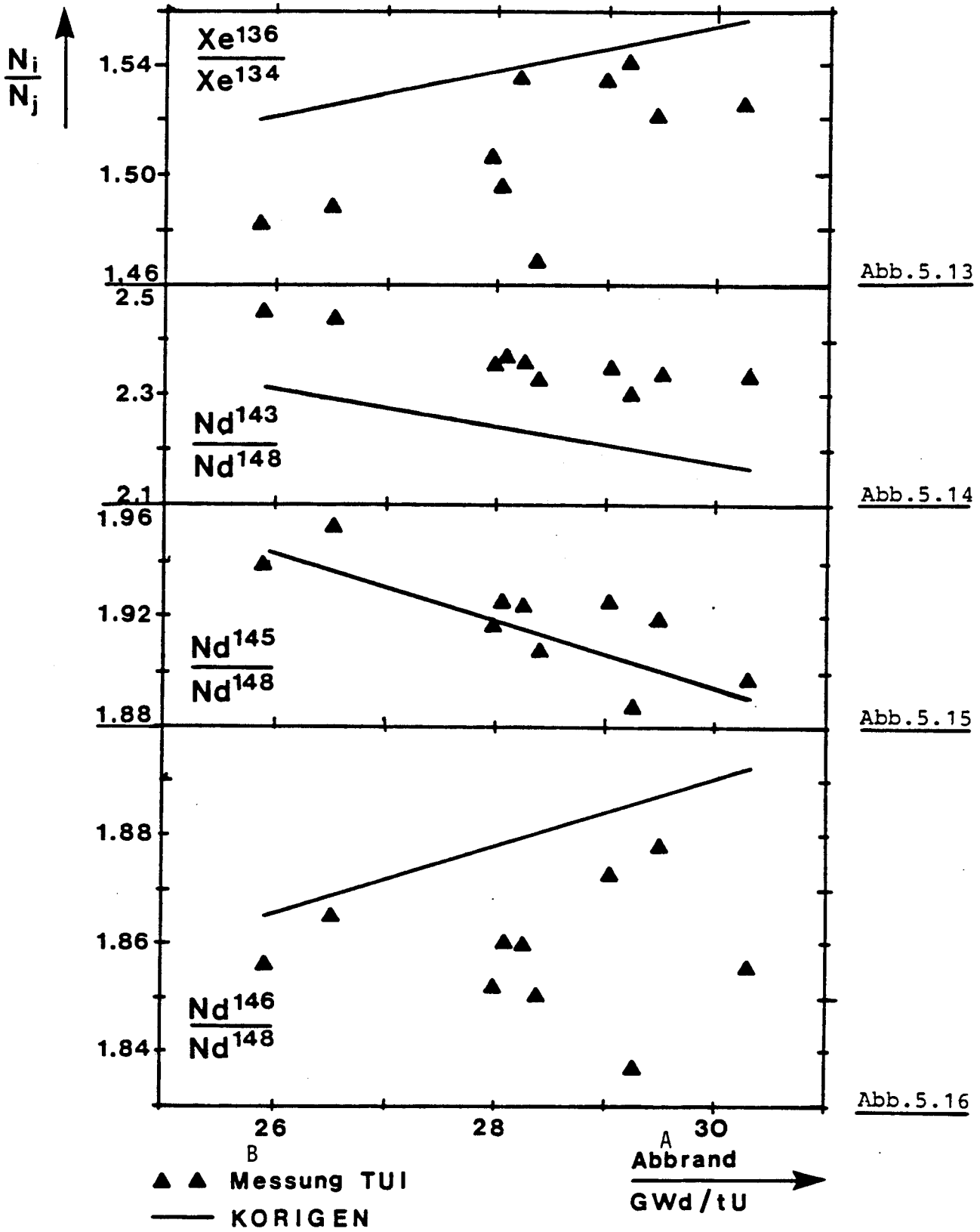
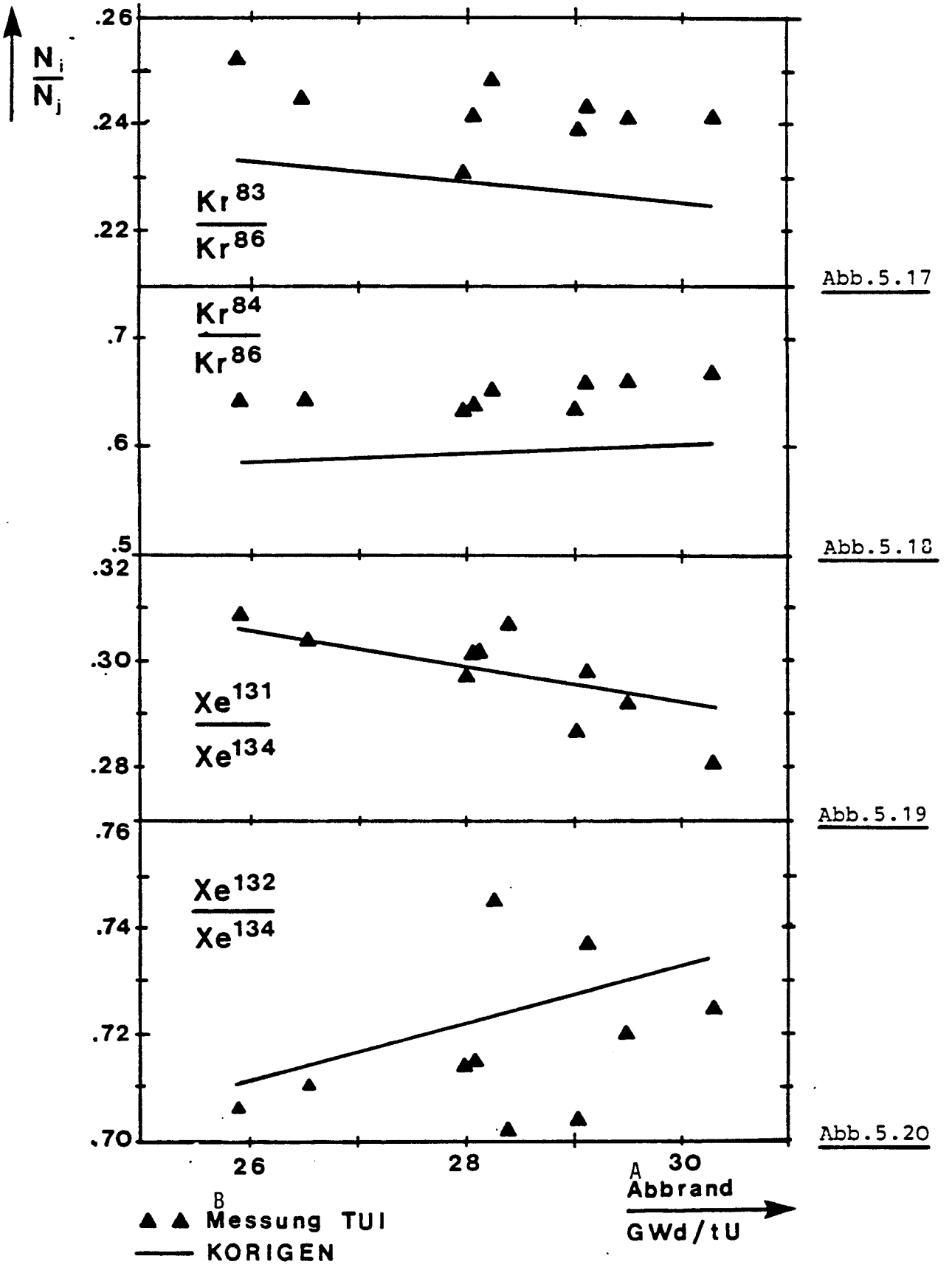


Abb.5.12

KWO-Batches (ICE)


Ni/Nj
Xe136/Xe134
1.54
1.50
1.46
Abb.5.13
2.5
2.3
2.1
Nd143/Nd148
Abb.5.14
1.96
1.92
1.88
Nd145/Nd148
Abb.5.15
1.88
1.86
1.84
Nd146/Nd148
Abb.5.16
26
28
30
B
A
Abbrand
GWd/tU
Messung TUI
KORIGEN


$\frac{N_i}{N_j}$
.26
.24
.22
$\frac{Kr^{83}}{Kr^{86}}$
Abb.5.17
$\frac{Kr^{84}}{Kr^{86}}$
.7
.6
.5
Abb.5.18
.32
.30
.28
$\frac{Xe^{131}}{Xe^{134}}$
Abb.5.19
.76
.74
.72
.70
$\frac{Xe^{132}}{Xe^{134}}$
Abb.5.20
26
28
30
A Abbrand
GWd/tU
B Messung TUI
KORIGEN

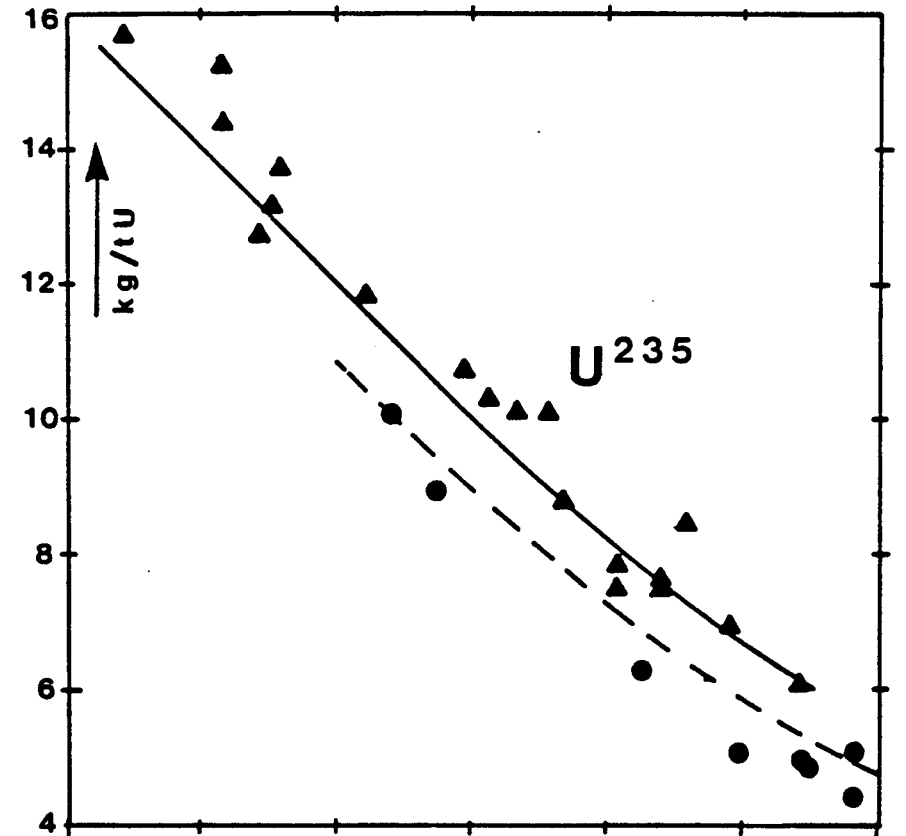


Abb.5.21

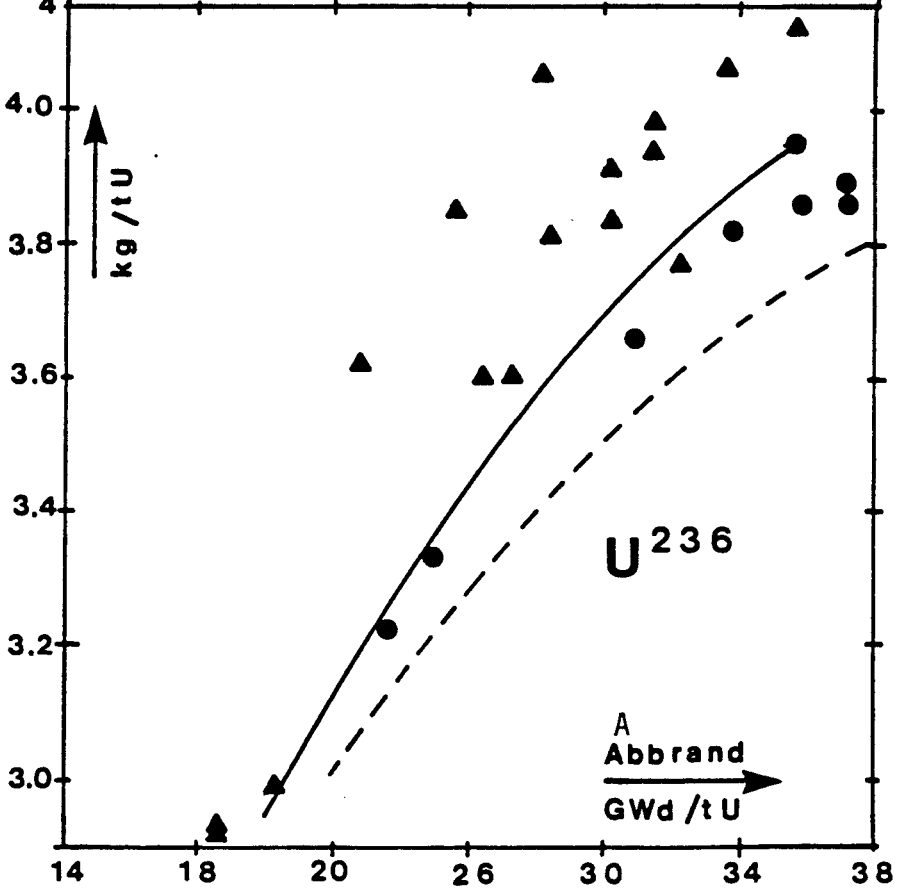


Abb.5.22

▲ ▲ KWO 124 } B Messung — KWO 124 } KORIGEN
● ● KWO 210 } Messung - - - KWO 210 } KORIGEN

KWO-Pellets

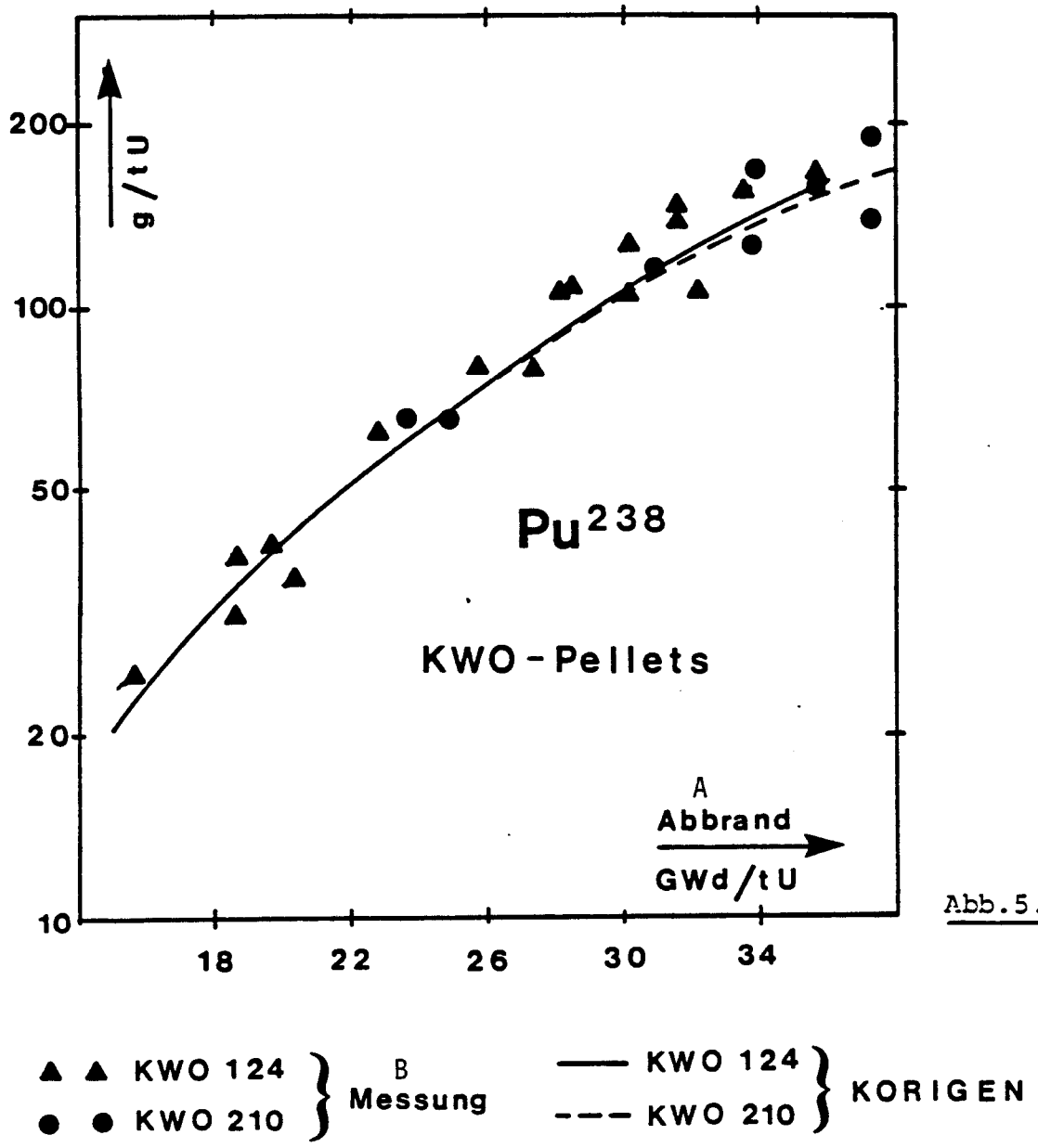

200
100
50
20
10
g/tU
Pu238
KWO-Pellets
A
Abbrand
GWd/tU
18
22
26
30
34
KWO 124
KWO 210
B
Messung
KWO 124
KWO 210
KORIGEN

Abb.5.23

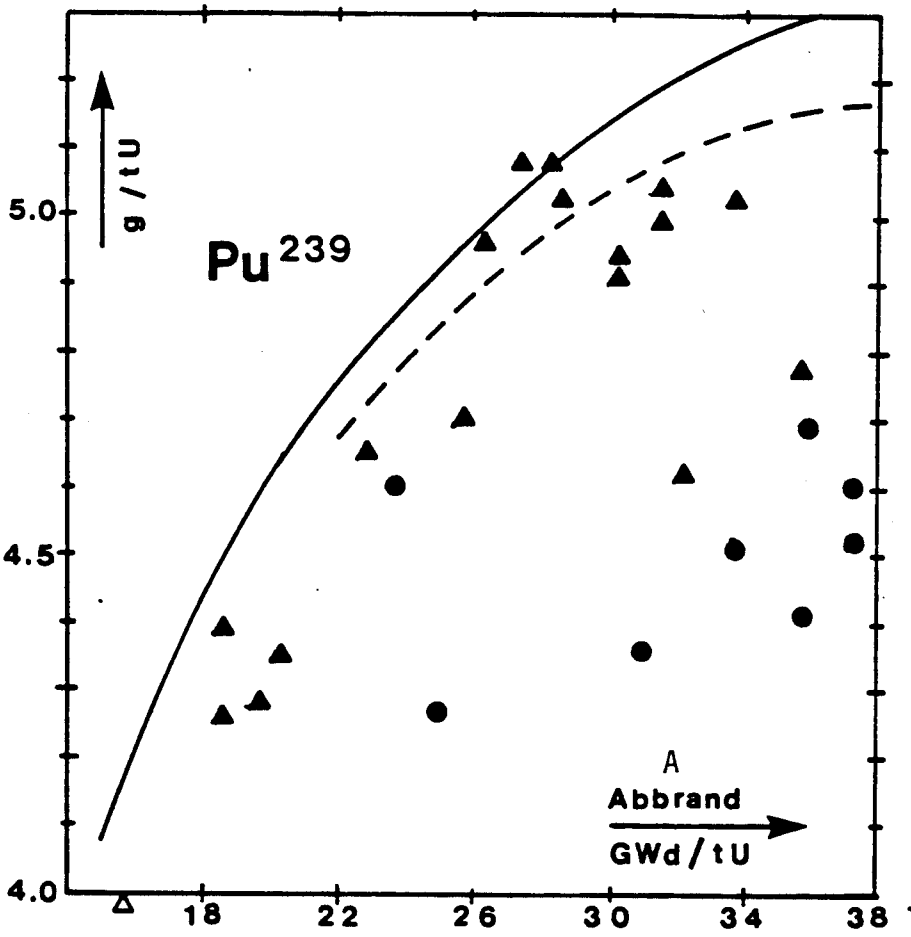


Abb.5.24

KWO 124 } B Messung — KWO 124 } KORIGEN
KWO 210 } Messung - - - KWO 210 } KORIGEN

KWO-Pellets

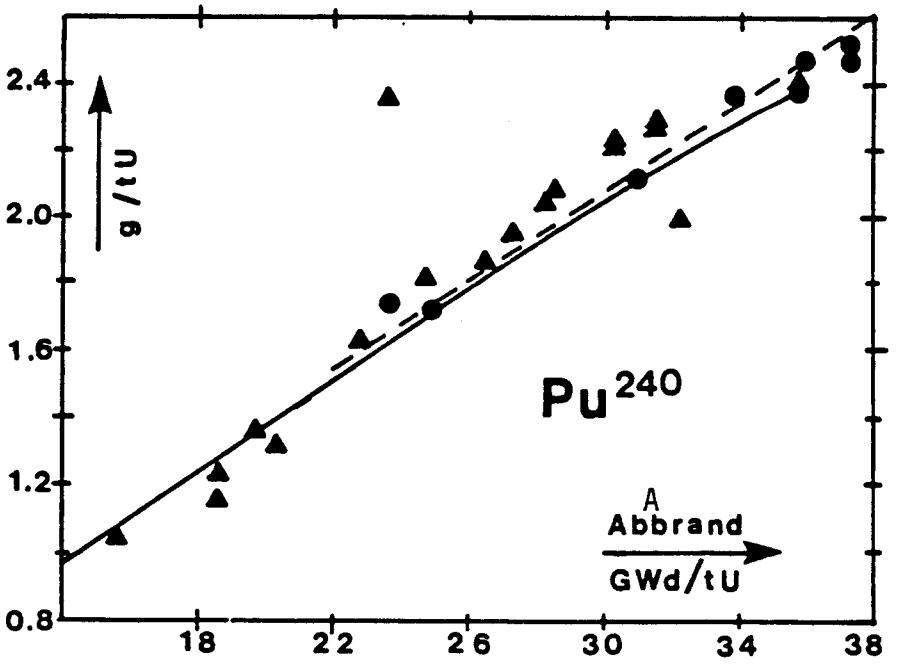


Abb.5.25

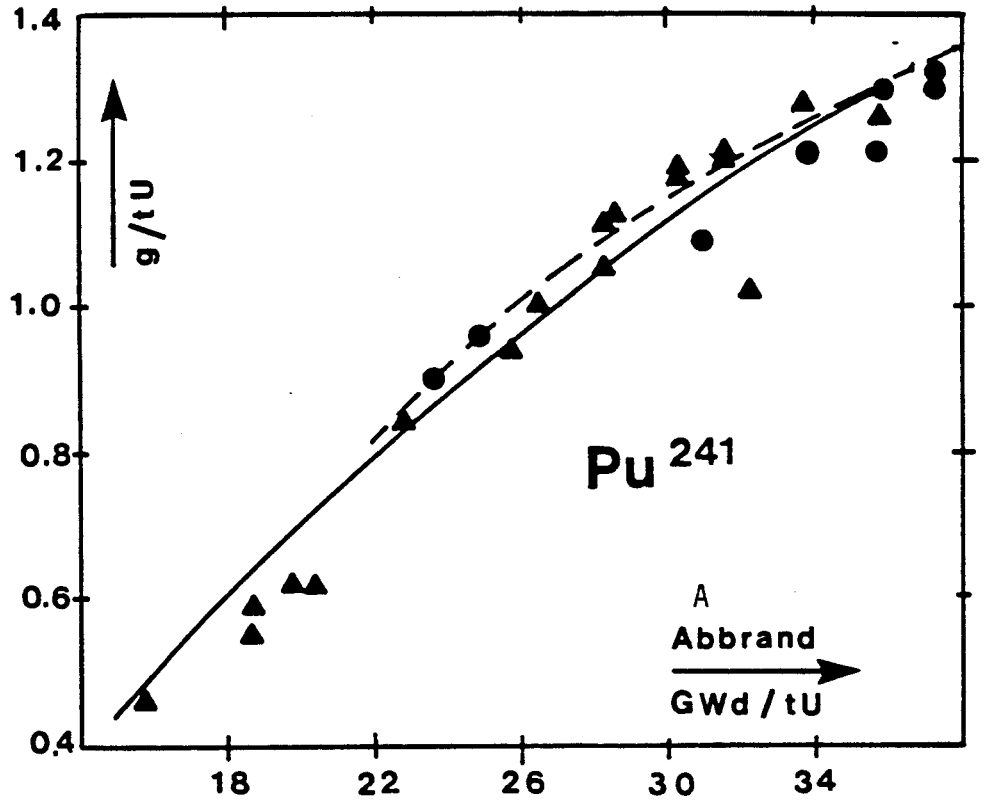


Abb.5.26

▲ ▲ KWO 124 } B Messung — KWO 124 } KORIGEN
● ● KWO 210 } B Messung ---- KWO 210 } KORIGEN

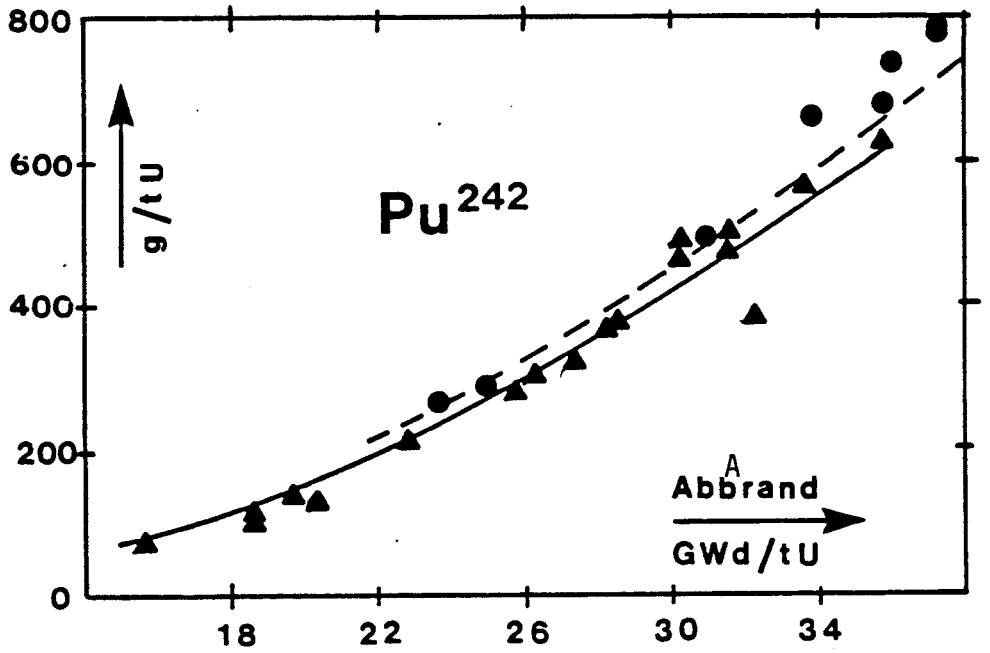


Abb.5.27

KWO-Pellets

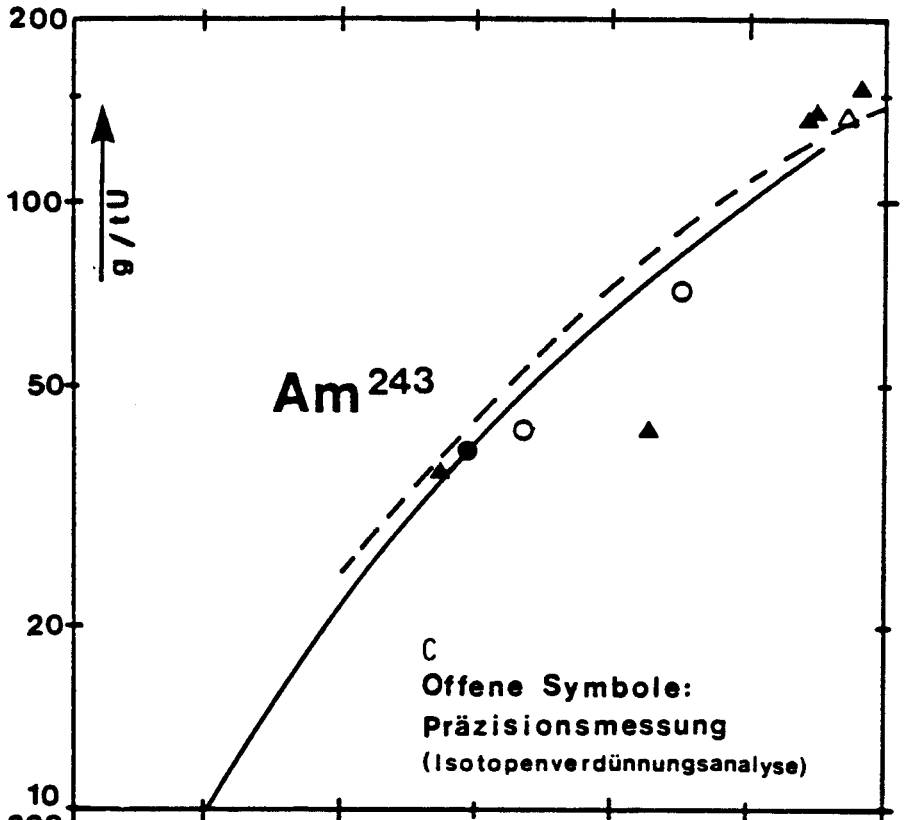


Abb.5.28

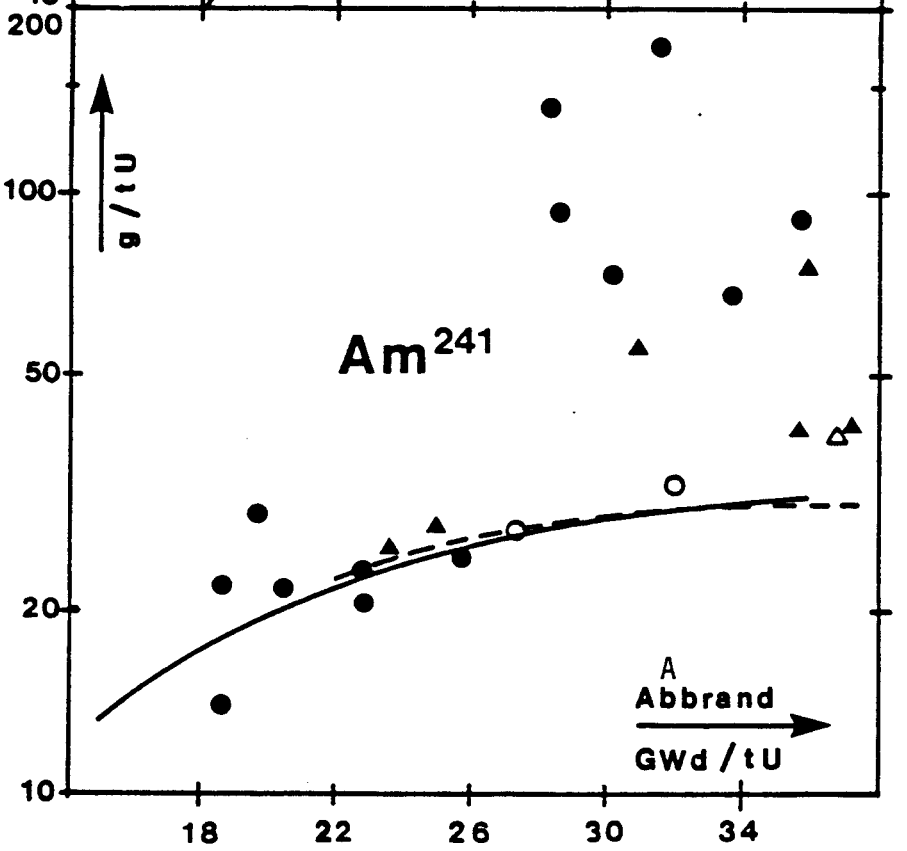


Abb.5.29

▲ ▲ KWO 210 } B Messung
● ● KWO 124 }

— — — KWO 210 } KORIGEN
——— KWO 124 }

KWO-Pellets

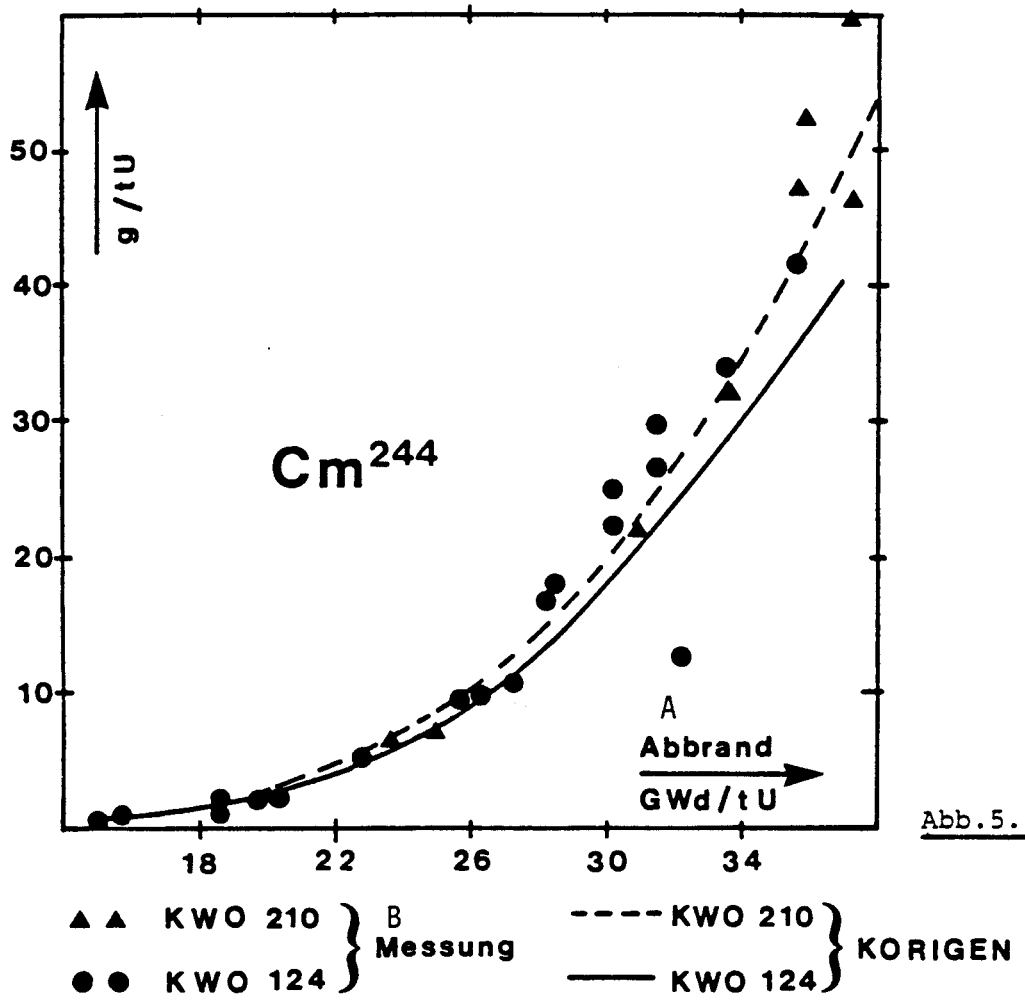


Abb.5.30

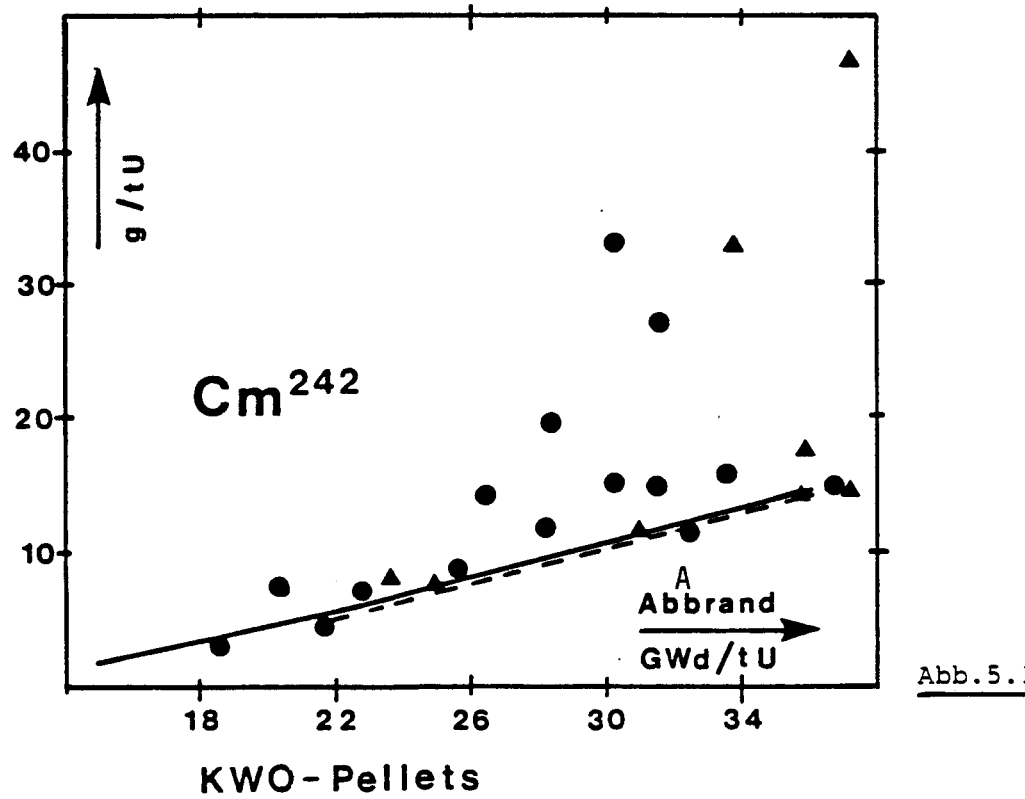


Abb.5.31

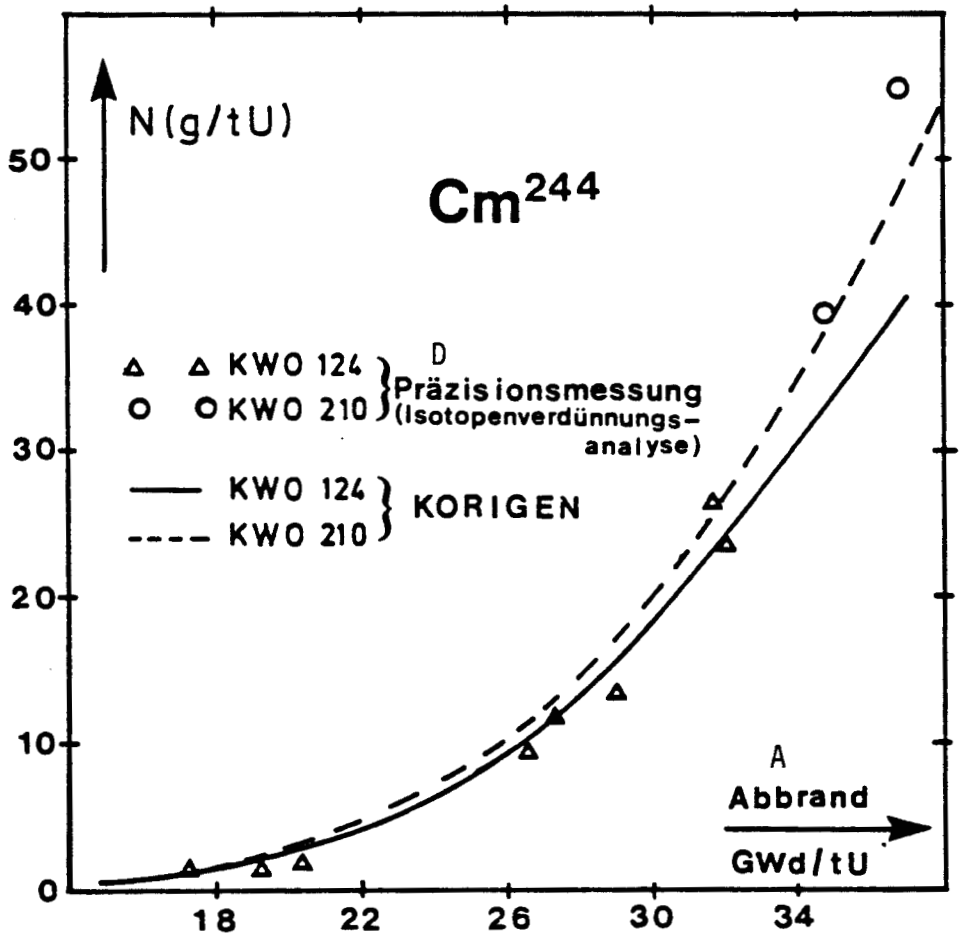
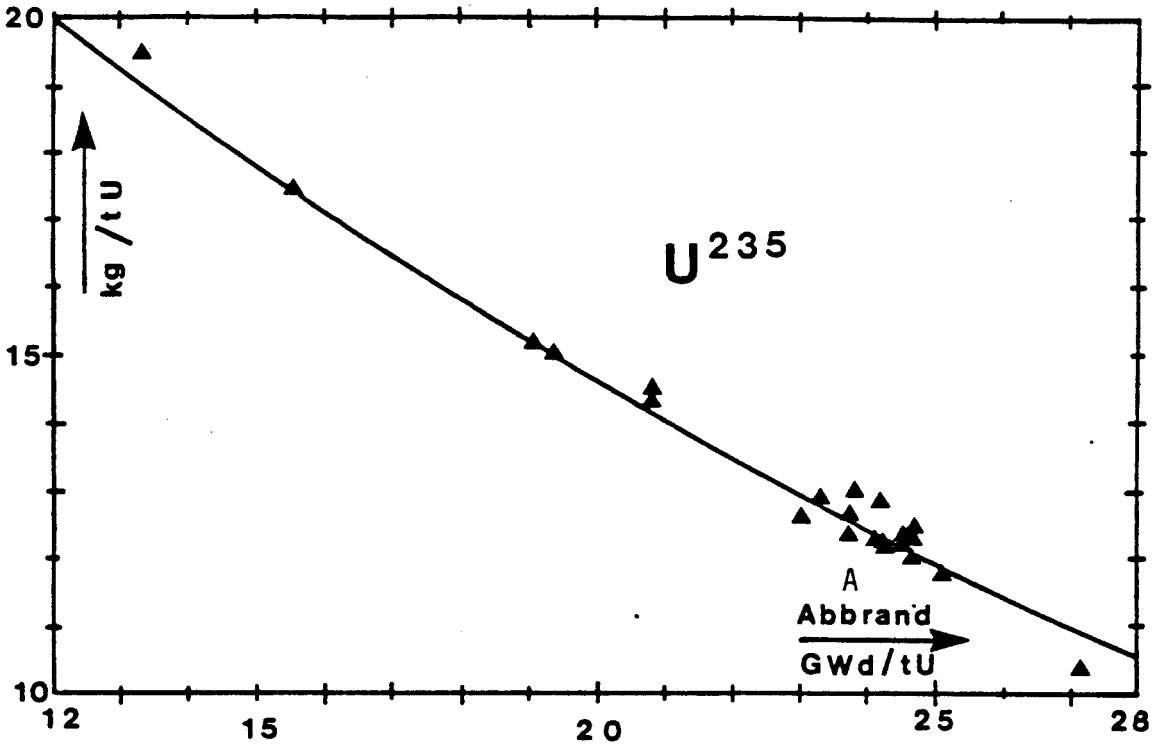

N(g/tU)
Cm244
50
40
30
20
10
0
KWO 124
KWO 210
D
Präzisionsmessung
(Isotopenverdünnungs-
analyse)
KWO 124
KWO 210
KORIGEN
A
Abbrand
GWd/tU
18
22
26
30
34

Abb.5.32



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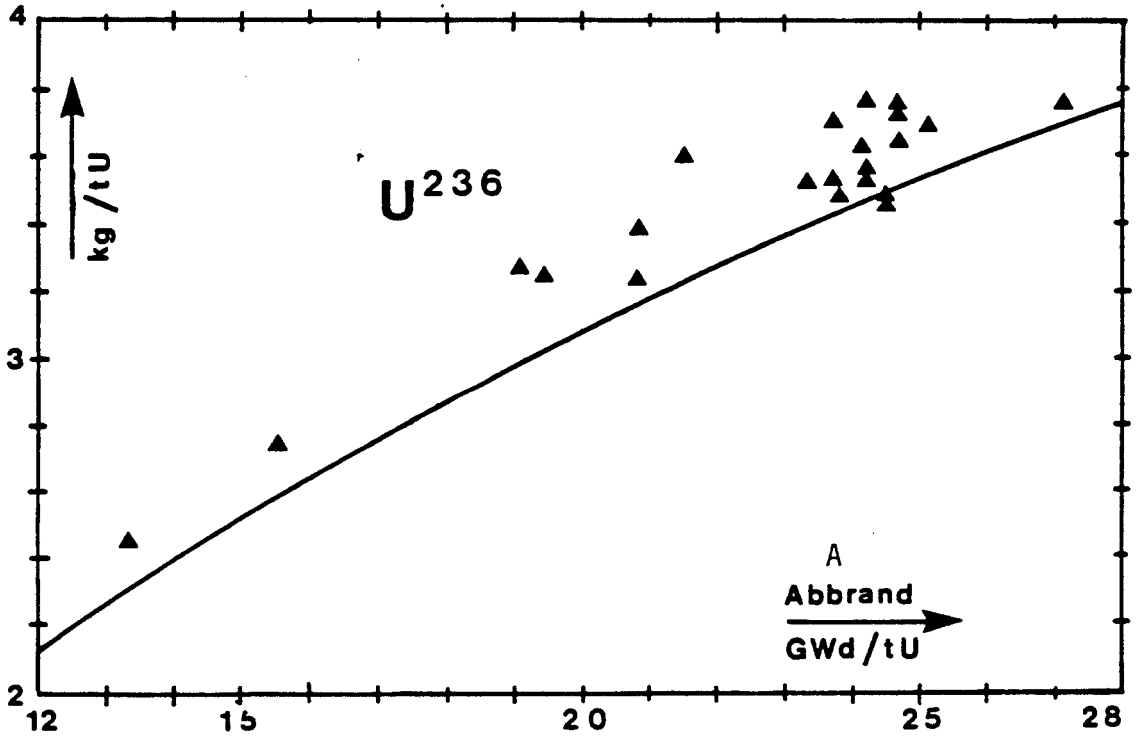
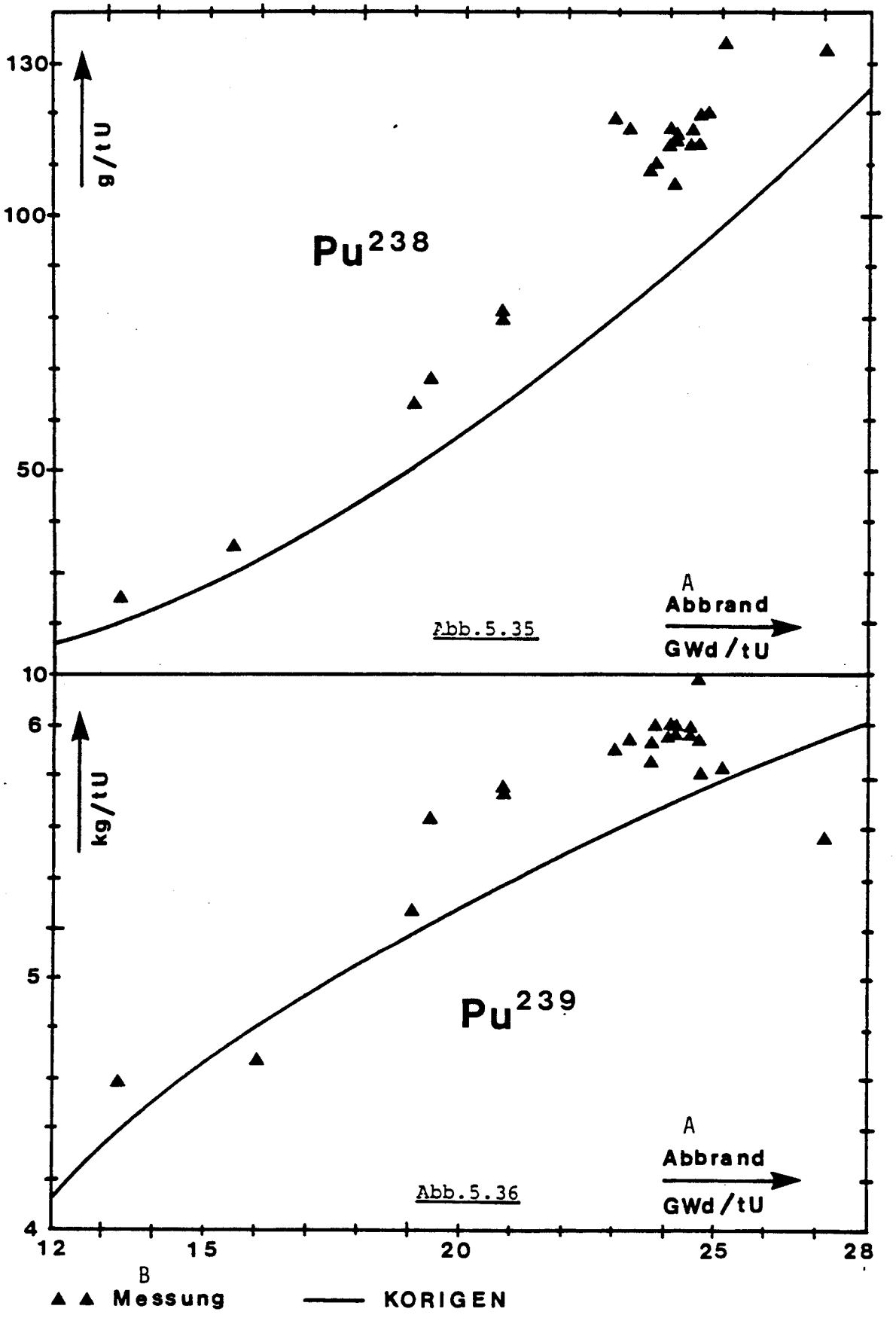
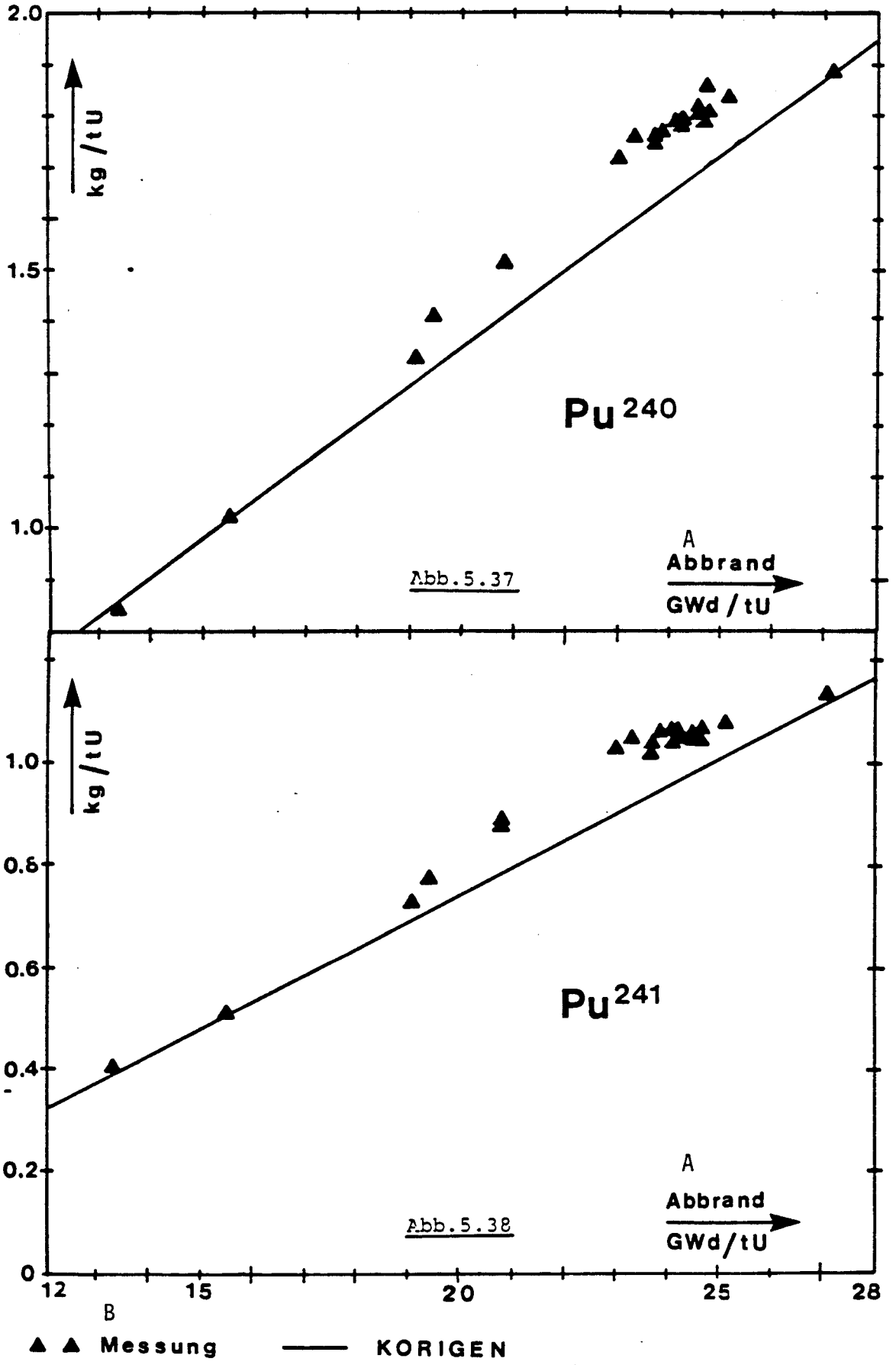


B
▲ ▲ Messung —— KORIGEN

Abb.5.34 TRINO2- Pellets



TRINO2-Pellets



TRINO2-Pellets

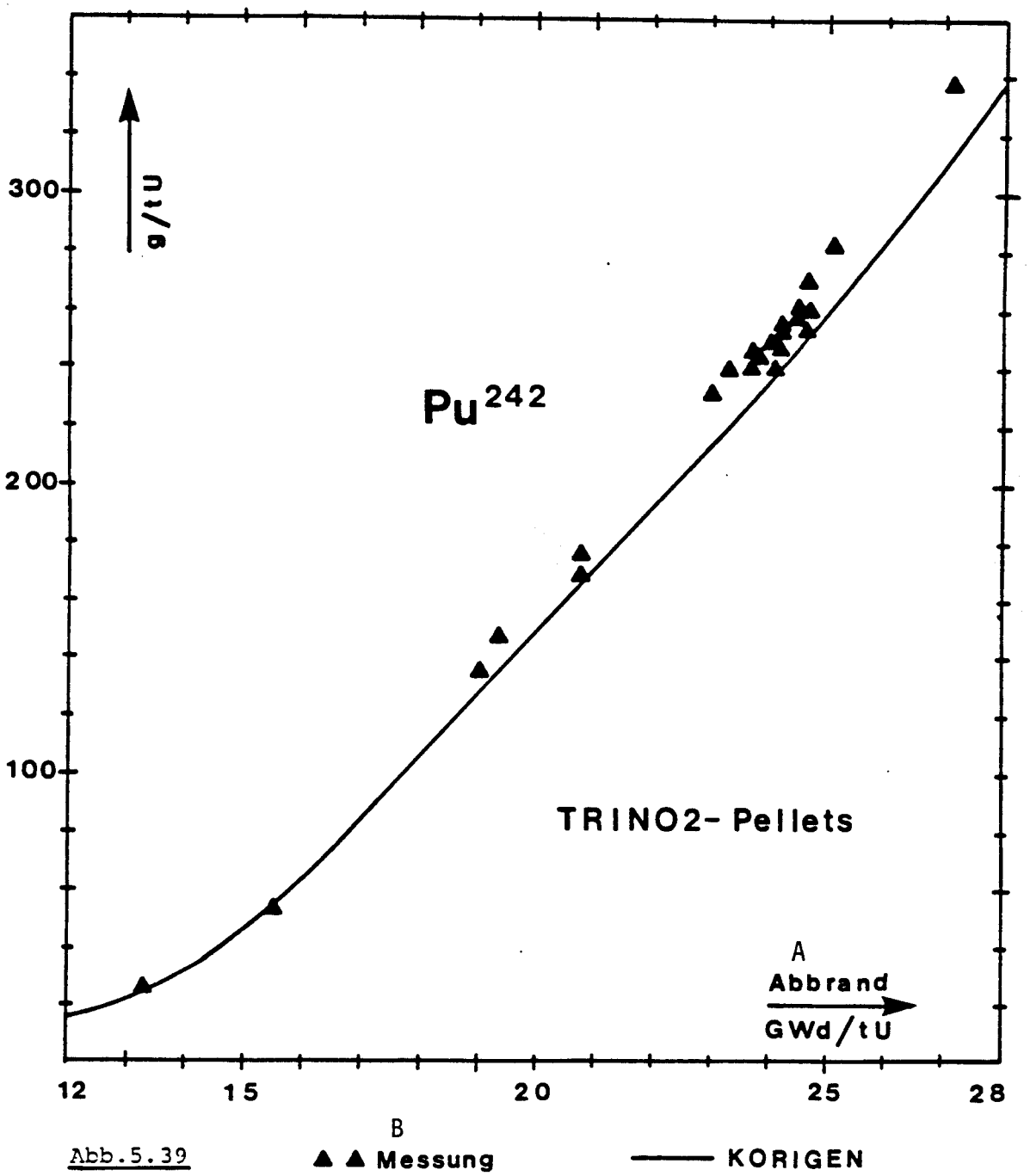

g/tU
300
200
100
Pu242
TRINO2- Pellets
A
Abbrand
GWd/tU
12
15
20
25
28
B
▲ ▲ Messung
KORIGEN

Abb.5.39

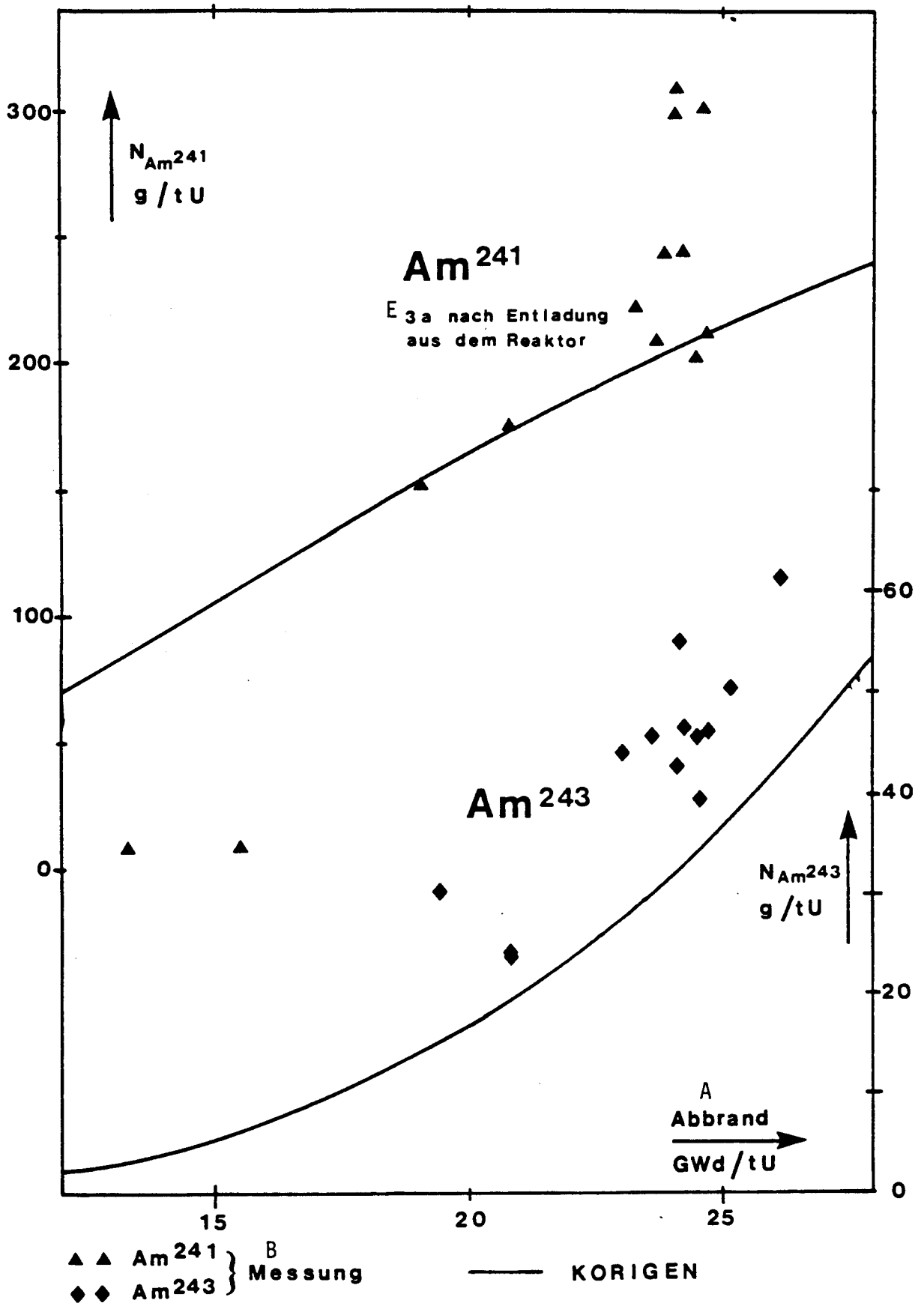
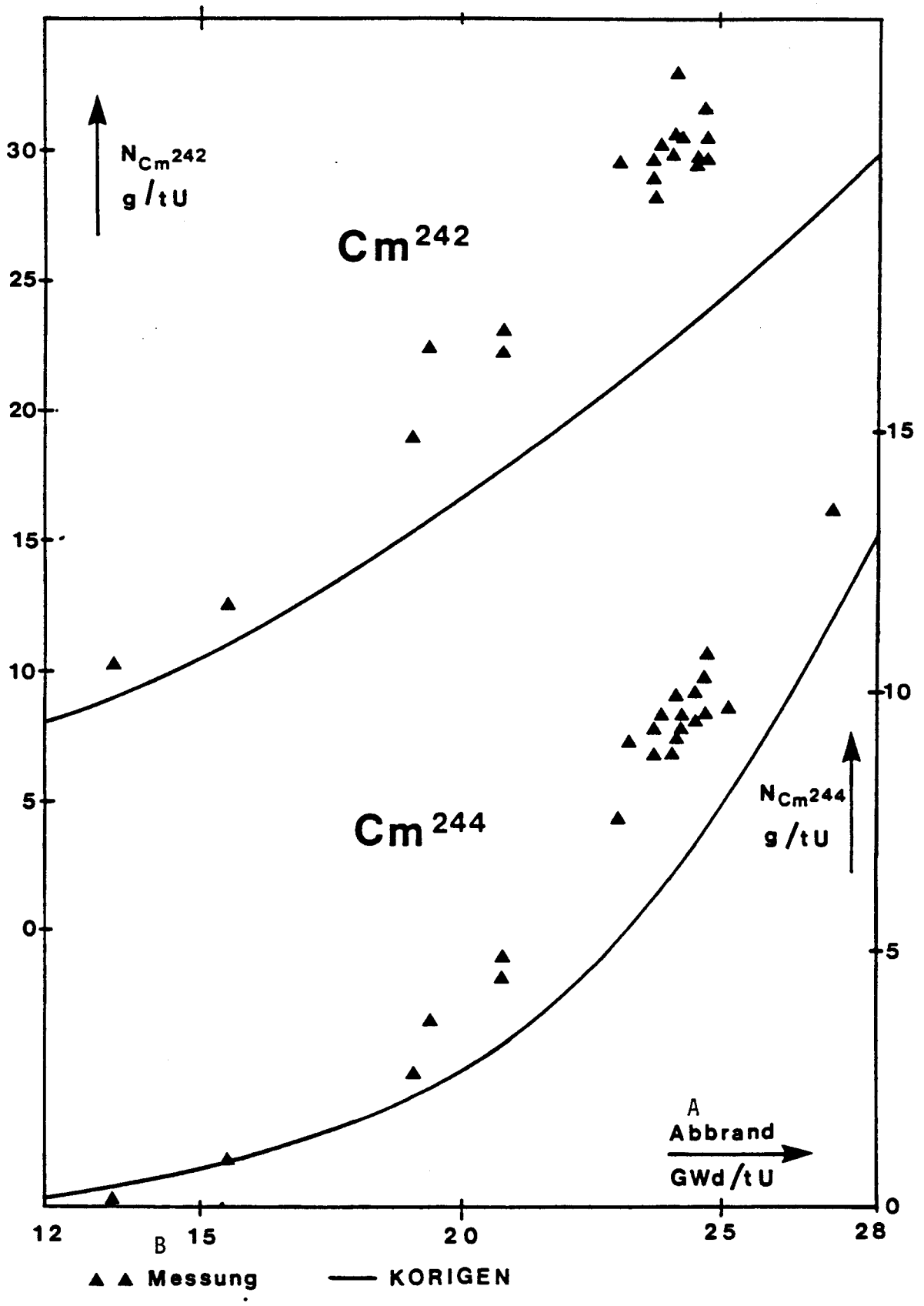


Abb.5.40 TRINO2-Pellets



TRINO2-Pellets

Abb.5.41

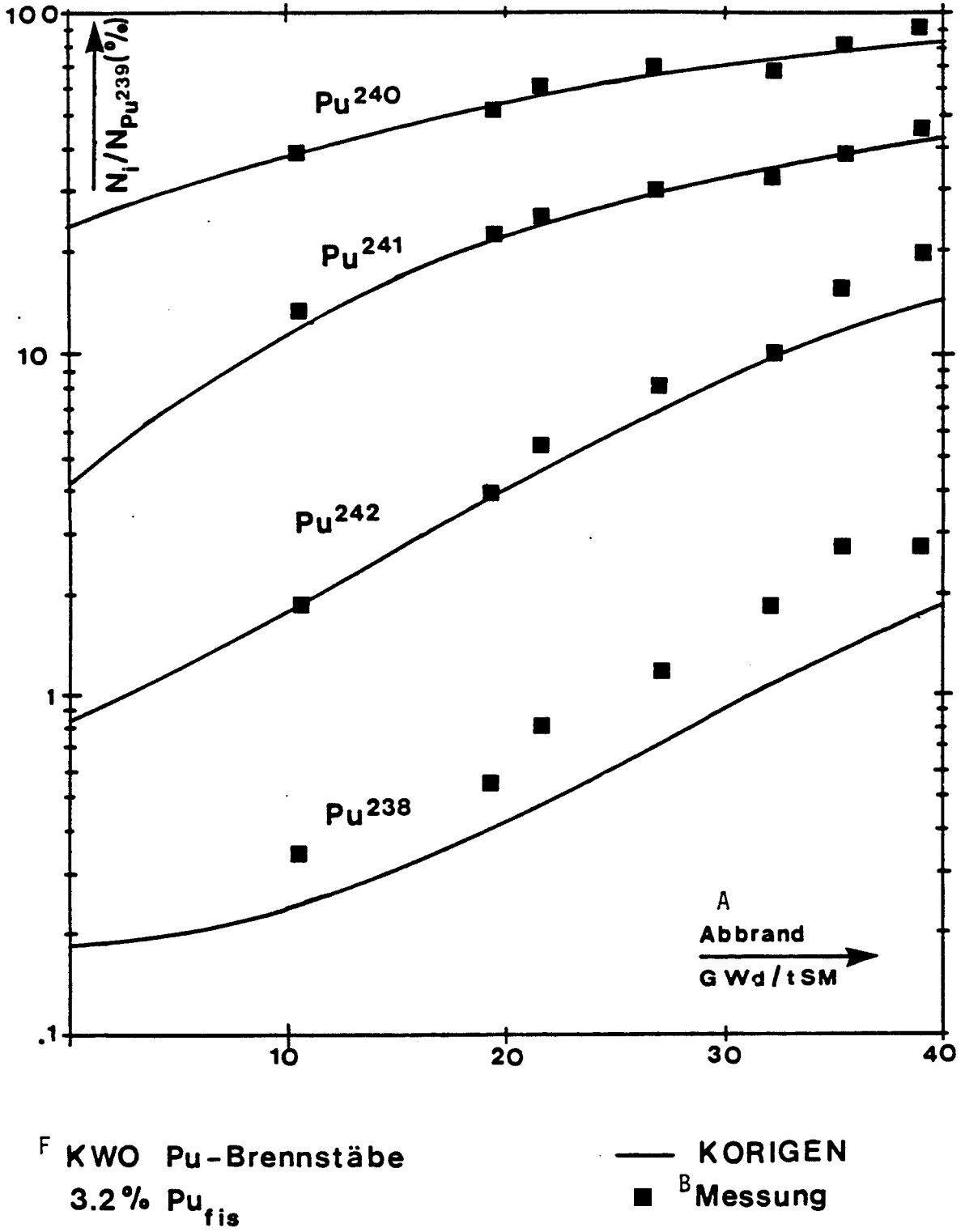


Figure 5.42. Comparison between the calculated and the measured concentrations of Pu isotopes from KWO Pu fuel, scaled to Pu-239.

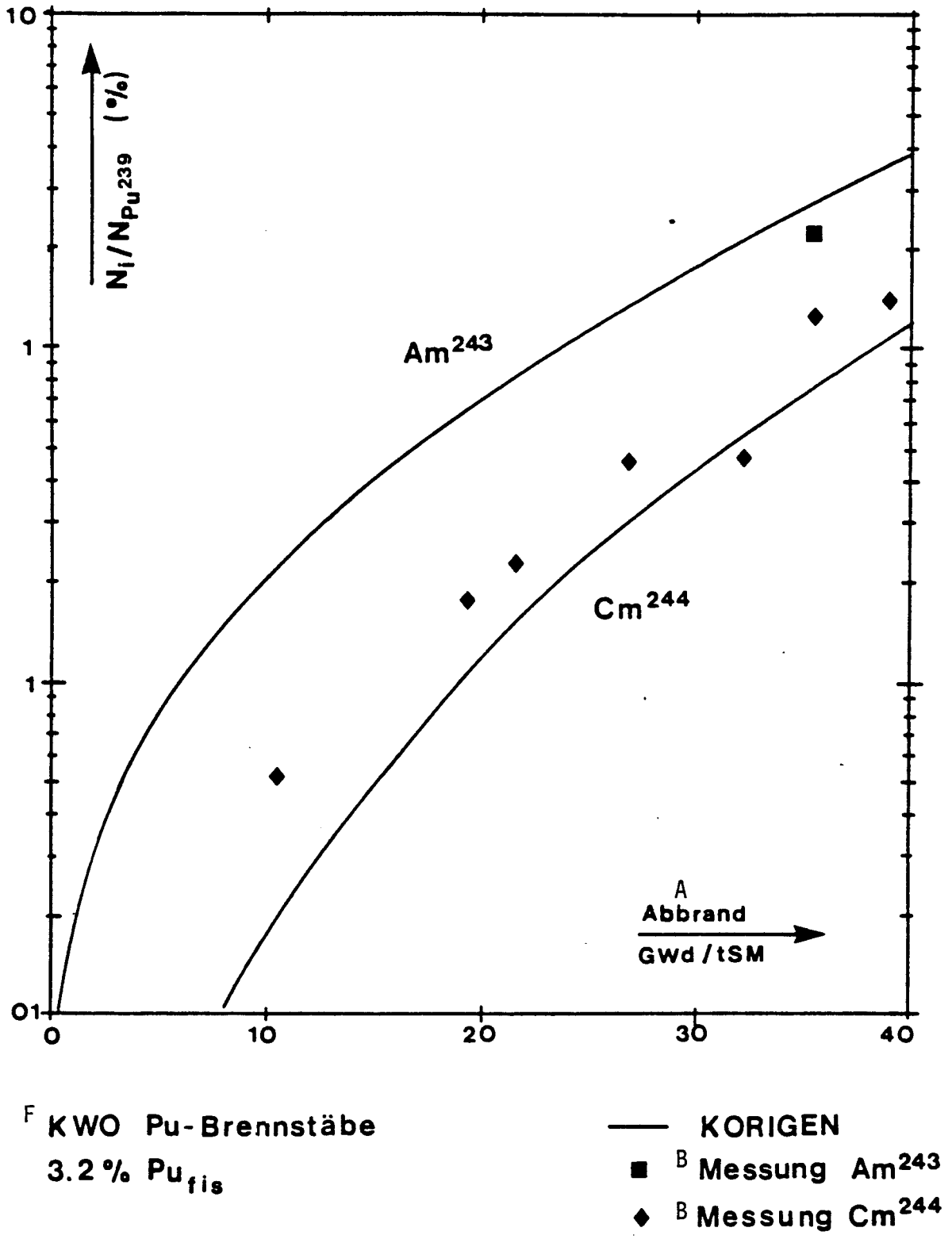


Figure 5.43. Comparison between the calculated and measured isotope ratios Am-243/Pu-239 and Cm-244/Pu-239 from KWO Pu fuel.

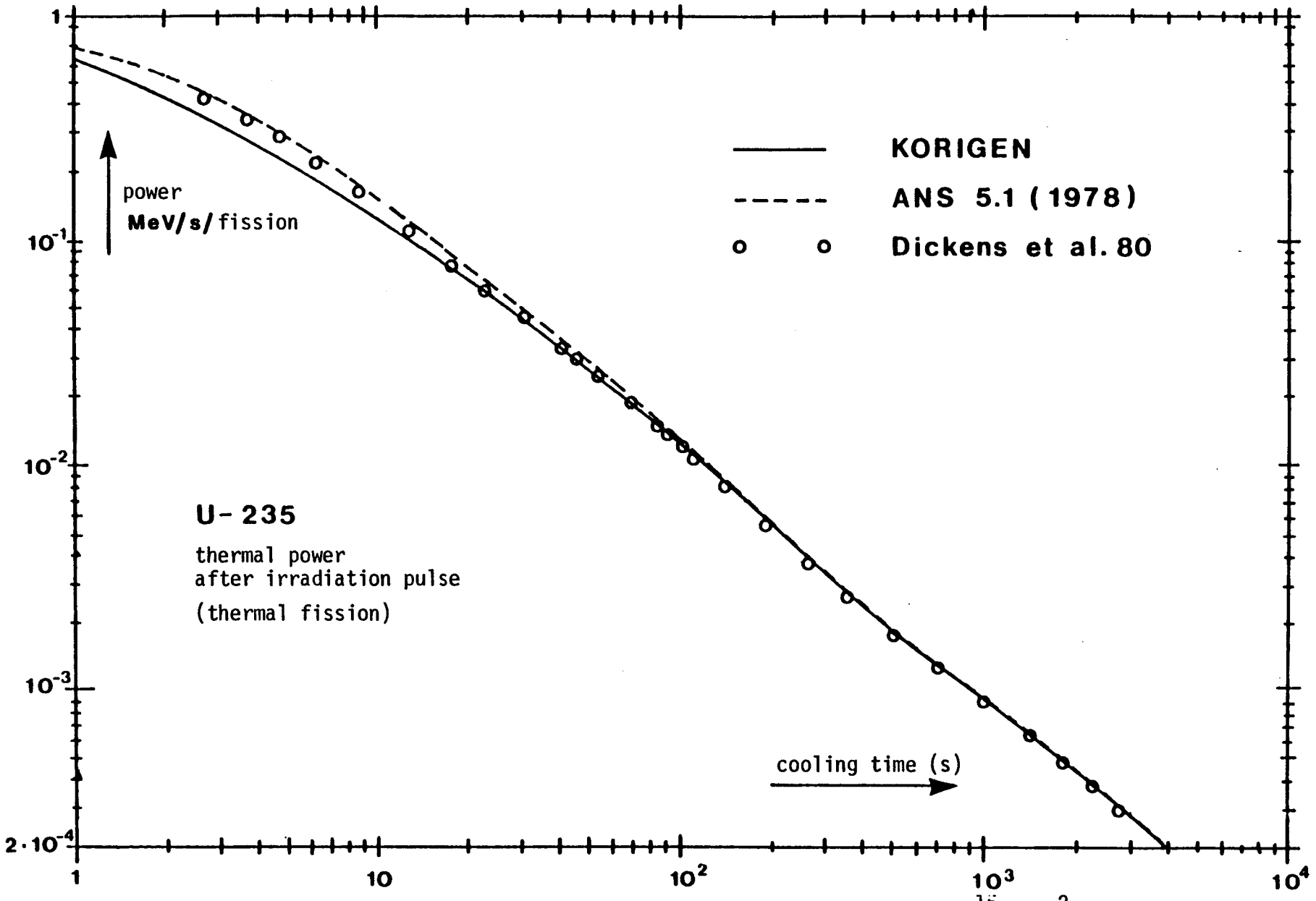


Figure 5.44. Decay power of U-235 after a short irradiation pulse ($\phi_{th} = 10^{15}$ n/cm^{2}s, $\Delta t = 0.1$ s).

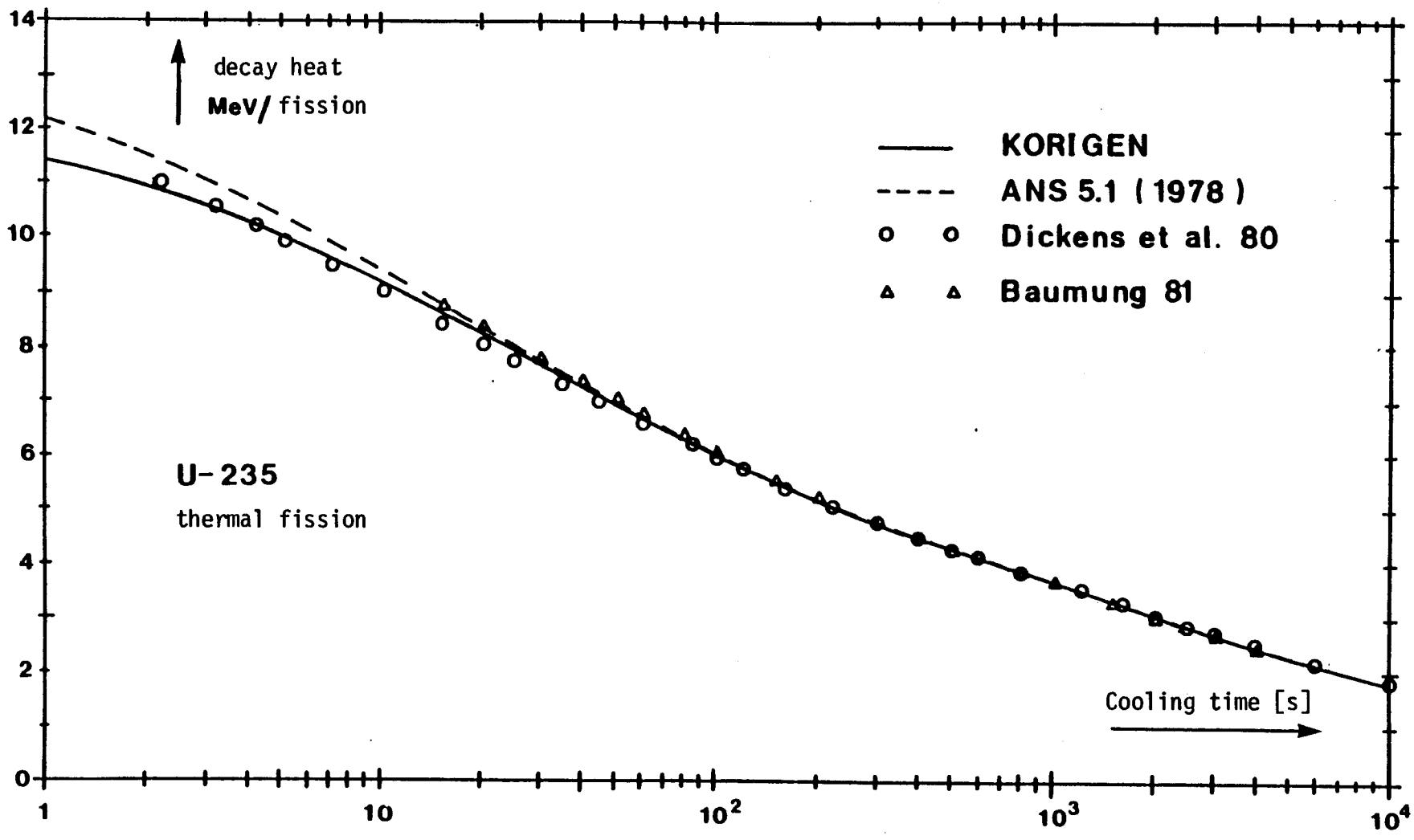


Figure 5.45. Decay heat resulting from thermal fission of U-235.

6. Comparison of KORIGEN, ORIGEN and ORIGEN2 for a Typical PWR Fuel

For the purpose of indicating inadequacies in the older ORIGEN program with reference to an important example, a comparison between the results of inventory calculations with KORIGEN and ORIGEN is carried out for a typical PWR fuel (3.2 wt% initially enriched uranium, burnup 33 GWd/tU). In addition, results of inventory calculations using ORIGEN2 are utilized which are taken from a report of Croff et al. [55] and were also determined for uranium fuel with the initial enrichment 3.2 wt% and the final burnup 33 GWd/tU. The ORIGEN2 results relate to the ORNL reference reactor given in [16], the lattice of which does not differ too greatly from the BIBLIS lattice (cf. Chap. 3).

The KORIGEN results relate to the BIBLIS reactor, i.e. they were obtained using the data record for BIBLIS presented in Chap. 3, and to the above fuel type. The ORIGEN calculation was performed using the original ORIGEN data and the spectral parameters THERM = 0.632, RES = 0.333 and FAST = 2.0 for the fuel specified above. Due to these different data records and the different methods--particularly the use of burnup-dependent actinide cross sections and nuclide-specific fission energy release in KORIGEN and ORIGEN2--some distinct differences in the results of the inventory calculations are apparent. In the following some results are compared in detail.

6.1 Actinide Inventory

Comparison between KORIGEN and ORIGEN. The most important fission nuclides in LWR, U-235 and Pu-239, are underestimated by ORIGEN: U-235 by ca. 9%, Pu-239 by ca. 4% (cf. the good agreement between KORIGEN and experiment in Chap. 5). This is caused on the one hand by the high absorption cross sections of these nuclides in ORIGEN, and on the other hand by the excessively low value for fission energy release in ORIGEN (200 MeV/fission for all fission nuclides), so that an increased number of fissions is necessary to reach the given burnup. The higher Pu isotopes 241 and 242, which tend to be underestimated with KORIGEN compared with the experiment, are also calculated lower by ORIGEN than by KORIGEN (Pu-241: -13%, Pu-242: -25%).

The Pu vector therefore differs essentially in the range of the higher Pu isotopes (cf. Tab. 6.1). In the case of the transplutonium isotopes it is striking that Cm-244 is calculated by ORIGEN ca. 40% higher than by KORIGEN, which results in a clear overestimation of the experiments. This is caused primarily by the capture cross section of Am-243, which is ca. 55% greater than the KORIGEN value.

Comparison between KORIGEN and ORIGEN2. The essential differences in this case involve U-235, Pu-239 and Cm-244 (cf. Tab. 6.1). ORIGEN2 calculates both U-235 and Pu-239 lower than KORIGEN (by ca. 3.5% and 8%, respectively) due to the greater absorption cross sections. Cm-244 is calculated ca. 30% lower by ORIGEN2 than by KORIGEN. This is caused primarily by the excessively small capture cross section of Am-243 in the ORIGEN file [16] (based on ENDF/B-IV), but also by the excessively large (by a factor of 2 compared with KORIGEN) capture cross section of Cm-244 (likewise from ENDF/B-IV) (cf. Chap. 3.2.3). The capture cross sections of Am-241, 243 and Cm-244 in KORIGEN are based on a new KfK evaluation [6]. The Pu vector differs clearly in the case of Pu-239 and Pu-240 (cf. Tab. 6.1), but hardly at all in the case of the higher isotopes. This is caused primarily by the difference in the calculated concentration of Pu-239 of ca. 8%.

Table 6.1. Comparison of the actinide inventory [g/tU] or irradiated uranium fuel (initial enrichment 3.2 w/o U-235, burnup 33 GWd/tU) upon unloading from the reactor, calculated with KORIGEN, ORIGEN and ORIGEN 2.

	KORIGEN	ORIGEN	ORIGEN2 /55/
U235	8.25+3	7.51+3	7.96+3
236	4.05+3	4.43+3	3.96+3
238	9.43+5	9.43+5	9.44+5
Np237	426.	469.	432.
Pu238	132.	163.	128.
239	5.39+3	5.19+3	4.94+3
240	2.23+3	2.19+3	2.31+3
241	1.22+3	1.06+3	1.22+3
242	486.	369.	460.
Am241	37.0	24.0	29.8
Am243	83.7	100.	85.4
Cm242	12.4	10.7	11.7
Cm244	23.9	33	18.8
Pu_{fiss}	6.61+3	6.25+3	6.16+3
Pu_{tot}	9.45+3	8.97+3	9.06+3
Pu-Vektor			
Pu38:Pu39:Pu40 :Pu41:Pu42	1.4:57.0:23.6: 12.9:5.1	1.8:57.9: 24.4:11.8:4.1	1.4:54.5:25.5: 13.5:5.1

Table 6.2. Comparison of the activity [curie/tU] of important fission products 1 year after unloading of the fuel (initial enrichment 3.2 w/o U-235, burnup 33 GWd/tU) from the reactor, calculated with KORIGEN, ORIGEN and ORIGEN2.

	KORIGEN	ORIGEN	ORIGEN2 /55/
H3	470	676	517
Kr85	8.82+3	1.05+4	8.87+3
Sr90	7.30+4	7.50+4	7.14+4
Y90	7.30+4	7.51+4	7.14+4
Zr95	2.81+4	3.04+4	3.15+4
Ru106	2.51+5	2.86+5	2.93+5
Rh106	2.51+5	2.86+5	2.93+5
Cs134	1.12+5	1.80+5	1.12+5
Cs137	1.04+5	1.06+5	1.02+5
Ba137m	9.85+4	9.87+4	9.64+4
Ce144	4.50+5	4.67+5	5.17+5
Pr144	4.50+5	4.67+5	5.17+5
Pm147	1.23+5	8.30+4	1.06+5
Eu154	1.13+4	6.85+3	9.63+3

6.2 Fission Products

ORIGEN overestimates the fission products generally, because due to the fission energy release of 200 MeV/fission for all fission nuclei used in ORIGEN (the nuclide-specific values are all higher) a higher number of fissions is necessary in order to reach a given burnup.

Significant differences in the calculated fission product concentrations are found in the case of tritium and Cs-134. The tritium inventory, due to increased fission yields (cf. Sec. 4.1.4), is calculated ca. 40% higher by ORIGEN than by KORIGEN (Tab. 6.2); ORIGEN2 calculates the tritium inventory only slightly higher than KORIGEN (Tab. 6.2). The concentration

of Cs-134 is calculated ca. 50% higher by ORIGEN than by KORIGEN and ORIGEN2 (Tab. 6.2). This is the result of a Cs-133 capture cross section in ORIGEN increased by a factor of 2. The Cs-134 concentration calculated by KORIGEN is confirmed by experimental post-irradiation analyses (cf. Sec. 5.1.3).

6.3 Activity and Thermal Power

The total activity of the calculated fuel is calculated more or less equally by KORIGEN, ORIGEN and ORIGEN2 (Tab. 6.2). The greatest difference is evident in the case of the actinides: ORIGEN calculates their activity (1 yr after discharge) ca. 12% smaller than KORIGEN and ORIGEN2. The reason for this is the Pu-241 concentration, which is calculated too low by ORIGEN (cf. Sec. 6.1).

The total thermal power is likewise calculated equally by KORIGEN, ORIGEN and ORIGEN2. Only ORIGEN2 calculates too small an amount for the thermal output of the actinides 1 yr after discharge (Tab. 6.3). The reason for this is an incorrect Q value for Cm-242 (cf. Sec. 4.1.4), since Cm-242 in this time range provides the greatest contribution to the thermal output of the actinides. The calculated gamma fractions of the thermal power differ by a maximum of 17%, whereby ORIGEN calculates the highest and ORIGEN2 the lowest value (Tab. 6.3).

Table 6.3. Comparison of the totals of activity, thermal power and gamma power 1 year after discharge of the irradiated fuel (initial enrichment 3.2 w/o U-235, burnup 33 GWd/tU) from the reactor, calculated with KORIGEN, ORIGEN and ORIGEN2.

		KORIGEN	ORIGEN	ORIGEN2/55/
activity [Ci/tU]	actinides	1.34+5	1.17+5	1.33+5
	fission products	2.15+6	2.29+6	2.36+6
	total	2.38+6	2.49+6	2.51+6
thermal power ($\alpha+\beta+\gamma$) [W/tU]	actinides	511	504	198
	fission products	8.90+3	1.08+4	9.88+3
	total	9.88+3	1.18+4	1.02+4
gamma power [W/tU]	actinides	0.22	0.19	0.27
	fission products	2.98+3	3.52+3	3.20+3
	total	3.45+3	3.90+3	3.32+3

Table 6.4. Comparison of the neutron emission rates calculated with KORIGEN, ORIGEN and ORIGEN2 at different cooling times.

		discharge	1 yr after discharge	5 yrs after discharge
Total neutron emission rate [n/(sec tU)]	KORIGEN	8.93+8	4.12+8	2.50+8
	ORIGEN2	7.92+8	3.4+8	1.91+8
	ORIGEN	7.10+8	4.48+8	3.30+8
Fraction of neutrons from spontaneous fission [%]	KORIGEN	61	79	93.5
	ORIGEN2	59	76	93
	ORIGEN	83.5	92	96
Fraction of neutrons from (α,n)-reactions [%]	KORIGEN	39	21	6.5
	ORIGEN2	41	24	7
	ORIGEN	16.5	8	4

6.4 Neutron Emission

The calculated neutron emission rates differ rather greatly in some cases. KORIGEN calculates the highest neutron emission at discharge (Tab. 6.4). On the other hand, ORIGEN calculates the highest n-emission after discharge. This is explained by the different origin of the emitted neutrons. The greatest percentage of emitted neutrons comes from spontaneous fission. The neutron yield per spontaneous fission hardly differs in KORIGEN, ORIGEN2 and ORIGEN (cf. Sec. 4.1.6). Because Cm-244 is by far the greatest producer of neutrons from spontaneous fission in the time range considered (up to ca. 10 years after discharge) and because the concentration of Cm-244 is overestimated by ORIGEN (Sec. 6.1), ORIGEN calculates the highest n-emission rate based on spontaneous fissions. ORIGEN2, which underestimates the Cm-244 concentration, calculates the lowest emission rate from spontaneous fissions, on the other hand.

Up to ca. 2 years after discharge, most (α,n) reactions with the oxygen of the fuel oxide are induced by α-particles from the decay of Cm-242. The percentage of neutrons from (α,n) reactions therefore drops in accordance with the half life of Cm-242 (Tab. 6.4). Three years after discharge the share of (α,n) neutrons in the total number of emitted neutrons has already dropped below 10%. One should keep in mind that ORIGEN greatly underestimates the (α,n) neutron emission rate, whereas KORIGEN and ORIGEN2 hardly differ at all. The reason for this lies in the neutron yield (per α-decay) of Cm-242 (and Cm-244) used in KORIGEN and ORIGEN2. The KORIGEN and/or ORIGEN2 values seem more reliable, however, because they are based on measurements, whereas the ORIGEN values are based on a semi-empirical interpolation formula (Sec. 4.1.6) which was created not for (α,n)-reactions in oxygen but for (α,n) reactions in beryllium.

Therefore we can state that the total neutron emission rate for spent LWR fuel (for cooling times > 1 year) is overestimated by ORIGEN (ca. 30%) and underestimated by ORIGEN2 (ca. 25%). The fraction of (α,n)-neutrons, which is important in the case of short cooling periods (up to a maximum of 3 yrs), is greatly underestimated by ORIGEN (factor of 3 at discharge).

Measurements of neutron emission, which were performed using spent KWO fuel assemblies with uranium and also MOX fuel, agree to within 15-30% with the KORIGEN calculations [56].

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Appendices

A. The Spectral Parameters THERM, RES, FAST

In Bell's ORIGEN the spectral parameters THERM, RES and FAST, which are to be determined for the reactor under consideration from spectral calculations, play a central role, since by using them the one-group cross sections required in the burnup calculation can be calculated from the 2200 m/s cross sections, the resonance integrals and the cross sections averaged over the fission spectrum. Several approximations are contained in this simple procedure which are dispensed with if one determines the one-group cross section from a multigroup calculation, as happens for KORIGEN for the important actinide, fission product and structural material cross sections. However, because even in KORIGEN at the present time we must still resort to the original Bell data for a large number of nuclides of secondary importance and in these cases the spectral parameters must be used, we will discuss in the following the definition and the significance of the spectral parameters in KORIGEN calculations, as well as the approximations made using this method.

A.1 Definition of the Spectral Parameters According to Bell and Discussion of the Approximation Assumptions

The one-group cross section (of reaction x on nuclide i) still required in the burnup calculation with KORIGEN as based on ORIGEN, which must be multiplied by the thermal neutron flux density ϕ_{th} in order to obtain the total reaction rate, is given by:

$$\text{(A-1)} \qquad \sigma_{x,OR} = \int_0^\infty \sigma_x^i(E) \cdot \phi(E)\,dE/\phi_{th}$$

with the thermal flux density

$$\phi_{th} = \int_0^{0.5eV} \phi(E)\,dE$$

Bell divides the energy zone of neutrons in the reactor into a thermal, a resonance and a fast zone:

$$\sigma^{i}_{x,OR} = \left[\int_{0}^{0.5ev} \sigma^{i}_{x}(E)\phi(E)dE + \int_{0.5eV}^{1MeV} \sigma^{i}_{x}(E)\phi(E)dE + \int_{1MeV}^{10MeV} \sigma^{i}_{x}(E)\phi(E)dE) \right] / \phi_{th}$$

(A-2)

$$= \sigma^{i}_{x,th} + \sigma^{i}_{x,res} + \sigma^{i}_{x,fast}$$

i) Thermal Zone

The approximation assumptions made in thiszone are as follows:

a) The absorption cross section follows a 1/v curve:

$$\sigma(v) = \sigma_{2200} \cdot \frac{v_{th}}{v}, \quad v_{th} = 2200 \text{ m/s}, \ \sigma_{2200} = \sigma(v_{th})$$

or

(A-3) $$\sigma(E) = \sigma_{2200} \cdot \sqrt{\frac{E_{th}}{E}} = \sigma_{2200} \cdot \sqrt{\frac{T_{th}}{T}}$$

b) The neutron spectrum in the Thermal Zone can be described by a Maxwell-Boltzmann distribution:

(A-4) $$\phi(E)\ dE = C \cdot \frac{E}{kT} e^{-\frac{E}{kT}} \cdot \frac{dE}{kT}$$

T is the absolute temperature, k the Boltzmann constant, and C a normalization constant.

For the thermal cross section σ_{th} averaged over this Maxwell spectrum we then obtain

$$\sigma_{th} = \sigma_{2200} \cdot \sqrt{\frac{\pi}{4} \cdot \frac{T_{th}}{T}}, \quad T_{th} = 293.16 \text{ K}$$

(A-5) $$= \sigma_{2200} \cdot \text{THERM}$$

$\text{THERM} = \sqrt{\frac{\pi}{4} \cdot T_{th}/T}$ is therefore the factor by which, for example, the 2200 m/s absorption cross section is to be multiplied in order to obtain the absorption cross section averaged over the thermal spectrum, if this spectrum can be represented by a Maxwell-Boltzmann distribution and the absorption cross section shows a 1/v curve.

Assumption (a) is not satisfied in the case of several actinides, in particular. The curve of these cross sections, which differs from 1/v, can be corrected, however, by using temperature-dependent Westcott factors which are applied to σ_{th}.

Assumption (b) is decisive, since the thermal spectrum in a LWR changes with progressive burnup. This applies both to the form (deviation from the Maxwell-Boltzmann distribution) and to the location of the spectrum. This means that the cross section σ_{th} averated over the thermal spectrum varies with progressive burnup sometimes rather sharply (particularly with the actinides). This cannot be taken into account in this method. Bell therefore used fitted values for σ_{th} for the most important actinides.

ii) Resonance Zone

Here the approximation assumption is made that the neutron spectrum follows a 1/E curve in the resonance zone (0.5 eV - 1 MeV):

$$\sigma_{res} = \frac{\int_{0.5eV}^{1MeV} \sigma(E) \frac{dE}{E}}{\int_{0.5eV}^{1MeV} \frac{dE}{E}} \cdot \frac{\phi_{res}}{\phi_{th}}, \quad \phi_{res} = \int_{0.5eV}^{1MeV} \phi(E)\,dE$$

$$= \int_{0.5eV}^{1MeV} \sigma(E) \frac{dE}{E} \cdot \frac{\phi_{res}}{14.509 \cdot \phi_{th}} = RI \cdot RES \qquad \text{(A-6)}$$

The cross section σ_{res} averaged over the spectrum in the resonance zone is therefore calculated in this approximation from the resonance integral RI and the spectral parameter RES. RES is given by the ratio of resonance

flux and thermal flux divided by the lethargy width of the resonance zone.

The assumption of a 1/E spectrum is valid only outside the resonances. Nuclides with pronounced resonances in the eV range exhibit increasing shielding of these resonances with progressive burnup. The cross section calculated according to (A-6) therefore leads to an overestimation of the actual effective cross section. Therefore Bell used effective resonance integrals for resonance nuclides such as U-235, U-238, and Pu-239, which integrals correspond to a mean burnup state and were adjusted for the most part. A correct description of the variable resonance self-shielding with progressive burnup is only possible using a time-dependent multi-group method.

iii) Fast Zone

In this case the following approximations are assumed:

a) The neutron spectrum in the fast energy zone (E > 1 MeV) has the same energy dependence as the fission spectrum $\chi(E)$.

b) The analyzed reaction has a threshold at E ⩾ 1 MeV

$$\sigma_{fast} = c \cdot \int_{1\ MeV}^{10\ MeV} \sigma(E)\ \chi(E)dE/\phi_{th}$$

$$= \frac{\int_{1\ MeV}^{10\ MeV} \sigma(E)\chi(E)dE}{\int_{1\ MeV}^{10\ MeV} \chi(E)dE} \cdot \frac{c \cdot \int_{1\ MeV}^{10\ MeV} \chi(E)dE}{\phi_{th}}$$

$$= \frac{\int_{0}^{10MeV} \sigma(E)\chi(E)dE}{\int_{0}^{10\ MeV} \chi(E)dE} \cdot \frac{\int_{0}^{10\ MeV} \chi(E)dE}{\int_{1\ MeV}^{10\ MeV} \chi(E)dE} \cdot \frac{\phi_{fast}}{\phi_{th}}$$

$$= \langle\phi\rangle_\chi \cdot 1.45 \cdot \frac{\phi_{fast}}{\phi_{th}} \text{ (for a Cranberg fission spectrum)}$$

(A-7)

$$= \langle\sigma\rangle_\chi \cdot \mathbf{FAST}$$

The cross section averaged over the fast neutron spectrum is calculated in this approximation from the cross section averaged over the fission spectrum and the spectral parameter FAST, which is given by the ratio of the fast to the thermal flux multiplied by the ratio of the total fission spectrum to the fraction of the fission spectrum above 1 MeV.

Assumption (a) is only valid above ca. 3-5 MeV, because in the range up to ca. 5 MeV degradations occur due to inelastic scattering on U-238. In the case of reactions with thresholds above ca. 5 MeV (particularly n,2n-reactions) this leads to an underestimation of σ_{fast}. In the case of reactions the threshold of which is just above 1 MeV this can lead to an overestimation of σ_{fast}.

For reactions which have a threshold below 1 MeV, one likewise obtains an overestimation of σ_{fast} which is considerable under some conditions. One should keep in mind that one only obtains an estimate in the fast zone using this method. Reactions to which this applies ((n,2n), (n,p), etc.) generally do not play a significant role, so that the accuracy is generally sufficient. In the case of specialized issues, for example in studies of the Pu-236 concentration in irradiated fuel [27], such reactions can certainly be significant, however. In the case of Pu-236 the production path leads via the reaction Np-237 (n,2n) Np-236 and the subsequent β^--decay to Pu-236. For reliable calculation of the Pu-236 production therefore a correct spectrum-averaged (n,2n)-cross section is a prerequisite. In such cases the corresponding corrections must be applied in Eq. (A-7).

A.2 Significance of the Spectral Parameters for KORIGEN Calculations

Calculation of the effective neutron cross sections used in KORIGEN, σ_{OR}--cf. (A-1)--is performed according to the following formula for important actinides, fission products and light elements:

$$\sigma_{OR} = \sigma \cdot \phi/\phi_{th} \tag{A-8}$$

where the one-group cross sections σ--burnup-dependent for the actinides--are taken from the KORIDATA file--cf. Sec. 4.1.1. Since according to (A-6) RES = $\phi_{res}/(14.509 \times \phi_{th})$ and according to (A-7) FAST = $1.45 \cdot \phi_{fast}/\phi_{th}$, the ratio required in (A-8) of the total and the thermal neutron flux density can be determined from the spectral parameters RES and FAST because $\phi = \phi_{th} + \phi_{res} + \phi_{fast}$:

$$(\phi/\phi_{th})_{Sp} = 1 + 14.509 \cdot \mathrm{RES} + \mathrm{FAST}/1.45 \tag{A-9}$$

The subscript Sp characterizes in the following the flux ratio calculated from the spectral parameters.

Determination of the cross sections of the less important nuclides which are required for the KORIGEN calculation is done in accordance with the previous section A.1 using

$$\sigma_{OR} = \sigma_{2200} \cdot \mathrm{THERM} + \mathrm{RI} \cdot \mathrm{RES} + \langle\phi\rangle_X \cdot \mathrm{FAST} \tag{A-10}$$

For most inventory calculations with KORIGEN the discharge inventory must be determined from the known initial inventory and the <u>power density</u> of the fuel during irradiation. The reaction rates of the important neutron reactions for which one-group cross sections exist are independent of the ratio ϕ/ϕ_{th} determined from the spectral parameters.

The microscopic reaction rate namely is calculated according to the following:

$$\sigma_{OR} \cdot \phi_{th} = \sigma \cdot (\phi/\phi_{th})_{Sp} \cdot \frac{P}{1.602 \cdot 10^{-19}} / \sum_j E_j \sigma_f^j (\phi/\phi_{th})_{Sp} \cdot N_j \tag{A-11}$$

$$= \frac{\sigma \cdot P}{1.602 \cdot 10^{-19}} / \sum_j E_j \sigma_f^j N_j$$

and is a function only of the one-group cross sections σ, σ_f^j from consistent spectral calculations, but not of $(\phi/\phi_{th})_{Sp}$. With regard to these reactions, the spectral parameters in KORIGEN calculations are therefore without effect. The demand for consistency of spectral parameters is obtained first with respect to the reaction rates of the less important nuclides for which the Bell formalism is still being used at the present time due to a lack of consistently calculated one-group cross sections. For these nuclides it is, however, generally sufficient to use representative spectral parameters for the reactor analyzed, e.g. for PWR those given by Croff et al. [16]--THERM = .701, RES = .304, FAST = 2.01--with which the calculations described in Section 5 are performed for the purpose of comparison with measurements.

Then

(A-12) $$\sigma_{OR} \cdot \phi_{th} \simeq \sigma \cdot \phi$$

We should point out the dependence of the thermal neutron flux density (determined by the program as a byproduct) on the spectral parameters, which is calculated according to

(A-13) $$\phi_{th} = \frac{P}{1.602 \cdot 10^{-19}} \Big/ \sum_j E_j \sigma_f^j \cdot \left(\frac{\phi}{\phi_{th}}\right)_{Sp} \cdot N_j$$

The total neutron flux density consistently calculated in the case of reactor-specific one-group fission cross sections σ_f^j is obtained from the above as

(A-14) $$\phi = \phi_{th} \cdot \left(\frac{\phi}{\phi_{th}}\right)_{Sp}$$

If the thermal neutron flux density ϕ_{th} is given instead of the power density P, then for the important nuclides with known one-group cross sections

(A-15) $$\sigma_{OR} \cdot \phi_{th} = \sigma \cdot \left(\frac{\phi}{\phi_{th}}\right)_{Sp} \cdot \phi_{th} = \sigma \cdot \phi$$

is trivially independent of $(\phi/\phi_{th})_{Sp}$ if the inputted ϕ_{th} is determined

from the actual total neutron flux density such that (A-14) is satisfied. For the other nuclides Eq. (A-12) is again valid if representative spectral indices are used.

B. Input for KORIGEN Calculations

B.1 Input Description

```
*******************
*                 *
*  K O R I G E N  *
*                 *
*******************
```

U. FISCHER, H. W. WIESE

KERNFORSCHUNGSZENTRUM KARLSRUHE
INSTITUT FUER NEUTRONENPHYSIK UND REAKTORTECHNIK

THIS IS A SHORT GUIDE TO THE USE OF KORIGEN, THE KARLSRUHER VERSION OF THE OAK RIDGE ISOTOPE GENERATION AND DEPLETION CODE ORIGEN.
IT WILL BE PRINTED IF THE CODE FAILS IN READING A CARD 'A',E.G. IN THE CASE OF ABSENCE OF INPUT CARDS.

SINCE KORIGEN IS CURRENTLY IMPROVED, IT IS RECOMMENDED TO RUN A KORIGEN JOB WITHOUT INPUT CARDS TO GET LAST INFORMATIONS.

USER'S COMMENTS WILL BE VERY MUCH APPRECIATED; PLEASE CALL TEL. 07247/82 -2452 OR -2414

DATE OF LAST CHANGE : 15. 10. 82

THE MAIN FEATURES OF KORIGEN IN COMPARISON TO THE ORIGINAL VERSION OF M. J. BELL /1/ ARE :

1. TO USE BURN-UP DEPENDENT REACTOR-SPECIFIC ONE-GROUP CROSS SECTIONS FROM KFK AND ORNL FOR ACTINIDE NUCLEI.
2. TO USE UPDATED BURNUP AVERAGED ONE-GROUP CROSS SECTIONS.
3. THE SPECTRAL INDICES CAN BE CHANGED DURING IRRADIATION.
4. ACCOUNT IS TAKEN FOR THE NUCLIDE DEPENDENT FISSION ENERGY RELEASE AND THE ENERGY RELEASE RESULTING FROM (N,GAMMA)-REACTIONS.
5. THE GAMMA POWER IS CALCULATED DIRECTLY FROM THE GAMMA SPECTRA.
6. IMPROVED CALCULATION OF THE NEURON EMISSION RESULTING FROM (ALPHA,N)-REACTIONS AND SPONTANEOUS FISSION.
7. THE INFINITE MULTIPLICATION FACTOR IS CALCULATED DURING IRRADIATON.
8. DECAY DATA, Q-VALUES, FISSION YIELDS ETC. ARE UPDATED FOR NEARLY 1400 NUCLIDES.
9. OUTPUT OPTIONS ARE INCREASED RESP. IMPROVED, E. G. SELECTION OF TABLES, NUCLIDES OR ELEMENTS OF INTEREST, CUT-OFF FOR ALL TABLES, TOTALS OVER ALL NUCLIDES,OUTPUT TABLES FOR THE ALPHA-ACTIVITY ETC.
10. INTRODUCTION OF ANNUAL MAXIMUM PERMISSIBLE RADIO-TOXICITY VALUES, VALID FOR THE F.R.G.
11. THE INPUT IS REPRODUCED FOR CHECK PURPOSES.
12. THE UPDATED INPUT DESCRIPTION CAN BE PRINTED.
13. RESULTS CAN BE STORED FOR SUBSEQUENT USE E.G. FOR PLOTS.

THE ORIGINAL ORIGEN CODE WAS IMPLEMENTED AT KFK BY C. BROEDERS, WHO ALSO PERFORMED FIRST IMPROVEMENTS. E. STEIN, R. HEGER, AND E. WIEGNER CONTRIBUTED ESSENTIALLY TO THE FURTHER DEVELOPMENT OF THE CODE. M. MARZO AND S. SHI WERE STRONGLY ENGAGED IN GENERATING AND TESTING KFK DATA.

DESCRIPTION OF THE FORMATTED INPUT OF KORIGEN

THIS DESCRIPTION IS LARGELY IDENTICAL TO THE ORIGINAL DESCRIPTION OF BELL /1/. INPUT PARAMETERS WHICH HAVE BEEN NEWLY INTRODUCED OR THE MEANING OF WHICH HAS BEEN EXTENDED ARE UNDERLINED BY *****. SOME OF THESE WERE ALREADY CONTAINED IN BELL'S VERSION BUT WERE NOT CONTAINED IN HIS DESCRIPTION.

CARD A FORMAT(18A4,I3,I3,I2)

TITLE A 72-CHARACTER ALPHAMERIC TITLE WHICH WILL BE PRINTED AS THE HEADING ON THE PRINTOUT OF THE NUCLEAR DATA LIBRARY

NLIBE AN INTEGER THAT IDENTIFIES THE LIBRARY TO BE USED
NLIBE = 1/2/3/4 FOR HTGR/LWR/LMFBR/MSBR

IPNUDA AN INDICATOR FOR THE PRINTING OF THE TRANSITION AND
****** PRODUCTION MATRIX WITH ALPHA-NUMERIC CHARACTERISTICS FOR REACTION TYPES AND NUCLIDES AT THE BEGINNING OF THE IRRADIATION.
IPNUDA= 0/1 : THE MATRIX IS NOT PRINTED/IS PRINTED ON UNIT FT45F001 (A DD-CARD FOR THIS UNIT IS NEEDED)

IPET AN INDICATOR FOR PRINTING THE PROCESSED NUCLEAR DATA
**** AND RADIOTOXICITY VALUES IN AN OVERVIEW TABLE
IPET = 0/1 : TABLE IS PRINTED/IS NOT PRINTED

CARD B FORMAT(4F10.5,7I2)

THERM RATIO OF THE NEUTRON REACTION RATE FOR A 1/V-ABSORBER WITH A POPULATION OF NEUTRONS HAVING A MAXWELL-BOLTZMANN DISTRIBUTION OF ENERGIES AT ABSOLUTE TEMPERATURE, T, TO THE REACTION RATE WITH 2200-M/SEC NEUTRONS.

RES RATIO OF THE RESONANCE FLUX PER UNIT LETHARGY TO THE THERMAL NEUTRON FLUX.

FAST RATIO OF FLUX ABOVE ONE MEV TO THE FRACTION OF THE FISSION SPECTRUM ABOVE 1 MEV, DIVIDED BY THE THERMAL NEUTRON FLUX.

ERR A TRUNCATION ERROR LIMIT, 1.E-25 RECOMMENDED

NMO, NDAY, NYR MONTH, DAY AND YEAR WHEN CASE IS RUN, TO HELP IN IDENTIFYING OUTPUT.

MPCTAB A DUMMY VARIABLE.

INPT AN INPUT OPTION FOR THE ACCESS OF THE NUCLEAR DATA.
INPT = 0 : THE NUCLEAR DATA LIBRARY (DATA SETS ORFI3-8) IS READ FROM FORTRAN UNIT 7 (THIS IS THE STANDARD CASE).
INPT = 1 : THE NUCLEAR DATA LIBRARY IS INPUT ON CARDS.

IR AN OUTPUT OPTION. IR = 0/1 : THE TRANSITION MATRIX
** IS NOT PRINTED/THE TRANSITION MATRIX IS PRINTED AS BEING CONSTRUCTED FROM THE DATA LIBRARY. IF IT IS MODIFIED ACCORDING TO THE BURN-UP DEPENDENCE OF THE ACTINIDE CROSS SECTIONS, IT WILL BE PRINTED AT TIME STEP IR, TOO. THIS PRINTOUT OF DECAY CONSTANTS, CROSS SECTIONS, MATRIX ELEMENTS IS A TOOL FOR CONTROLLING THE INPUT OF NUCLEAR DATA TO THE CODE.

```
  LPU      AN OPTION FOR SECONDARY CONSTANT CROSS SECTION INPUT
  ***      LPU = 0 : NO SECONDARY CONSTANT CROSS SECTION INPUT
           LPU > 0 : LPU GIVES THE NUMBER OF ACTINIDE NUCLIDES
           FOR WHICH THE NUCLEAR DATA LIBRARY SHALL BE MODIFIED
           INDEPENDENT OF BURN-UP; LPU < 31.
           LPU =-1 : UPDATED BURN-UP AVERAGED ONE GROUP CROSS
           SECTIONS FOR DIFFERENT TYPES OF REACTORS WILL BE
           READ FROM UNIT ISB AND/OR FROM CARDS.
           RECOMMENDATION : LPU = -1
           --------------
IF LPU > 0, A CARD OF TYPE W1 AND THEN LPU CARDS OF TYPE W2
WILL BE EXPECTED FOLLOWING CARD B. IF LPU = 0  CARD C WILL
FOLLOW. IF LPU =-1, A CARD OF TYPE U1 AND THEN CARDS U2 ARE
EXPECTED TO FOLLOW CARD B.

CARD W1        FORMAT(I6,10I7)

  (NUCL(I),I=1,LPU)
  ****     LPU INTEGER VALUES IDENTIFYING THE ACTINIDE NUCLIDES
           FOR WHICH THE NUCLEAR DATA WILL BE MODIFIED IN THE
           SUBSEQUENT CALCULATION, E.G. 922380 942390 952421
           - SEE CARD K. THIS CARD MAY BE CONTINUED.

CARD W2        FORMAT(7X,2F9.2,F5.3,4F9.2,F4.3,F9.2,I1)

  SIGNG,RING,FNG1,SIGF,RIF,SIGFF,SIGN2N,FN2N1,SIGN3N,IT
  ****     FOR DEFINITIONS OF THE VARIABLES SEE REF. /1/, P.27,
           WE HAVE IT = NLIBE - SEE CARD A.

CARD U1        FORMAT(6A8,2A8,1X,I5,5X,I5)

  WQTIT    A 48-CHARACTER ALPHAMERIC TITLE FOR IDENTIFYING
  ****     INPUT
  (ALLWQ(I),I=1,2)
  ****     A 16-CHARACTER ALPHAMERIC NAME FOR IDENTIFYING THE
           REACTOR SPECTRUM DEPENDENT BURN-UP AVERAGED CROSS
           SECTIONS ON UNIT ISB (DSN=INR909.KORIDATA)
           AVAILABLE FROM /2,3/ ON THE KORIDATA LIBRARY :
           33P3U___ORNL_78_,33P3RU__ORNL_78_,33P3RM__ORNL_78_,
           27B4U___ORNL_78_,27B4RU__ORNL_78_,27B4RM__ORNL_78_,
           33P3U___KFK_80__,50P5U___ORNL_80_.
           WITH THE FOLLOWING MEANING :
           FIRST TWO FIGURES : BURN-UP IN GWD/THM
           FOLLOWING LETTER  : P FOR PWR,B FOR BWR
           FOLLOWING FIGURE  : NUMBER OF CYCLES
           LAST TWO LETTERS  : U/RU FOR URANIUM FUEL IN A NORMAL
                               LWR/IN A RECYCLE LWR
                               RM FOR MOX-FUEL IN A RECYCLE LWR
           AT THE PRESENT THE BURN-UP AVERAGED CROSS SECTIONS
           33P3U___KFK_80__ AND 33P3U___ORNL_78_ ARE IDENTICAL.
           NOTE : IF SUBSEQUENT BURNUP DEPENDENT ACTINIDE
                  SECTIONS FROM KFK ARE USED (SEE THE CARD T1
                  BELOW),ON CARD U1 THE DATA SET 33P3U___KFK_80
                  SHOULD BE TAKEN.
```

```
  ISB      FORTRAN UNIT FROM WHICH THE ONE-GROUP CROSS SECTIONS
  ***      WILL BE READ - SEE CARD T1.
  KTIW     AN INDICATOR FOR PRINTING CROSS SECTIONS FROM UNIT
  ****     ISB > 5. KTIW = 0/1 : PRINTED / NOT PRINTED.

IF ISB > 5 AND IF NO MODIFICATION OF THE KORIDATA VALUES
OR NO ADDING OF FURTHER ONE-GROUP CROSS SECTIONS IS WANTED, A
BLANK CARD AND THEN CARD C IS EXPECTED TO FOLLOW CARD U1.
IF ISB > 5 AND IF MODIFYING AND/OR ADDITIONAL DATA WILL BE GI-
VEN, CARDS OF TYPE U2 AND THEN CARD C IS EXPECTED FOLLOWING
CARD U1.
IF ISB = 5, CARDS U2 AND THEN CARD C WILL FOLLOW CARD U1.
CARD U2        FORMAT(I6,2X,A8,8E8.2)

  NUNUM    NUCLIDE IDENTIFIER - SEE CARD K.
  ****
  SIGTYP   TYPE OF CROSS SECTION TO BE READ,
  ****     = SIGNA___ : (N,ALPHA)
           = SIGNP___ : (N,PROTON)
           = SCAPT___ : (N,GAMMA) TO GROUND STATE NUCLIDE
           = SFISS___ : (N,FISSION)
           = SCAPT1__ : (N,GAMMA) TO EXCITED STATE NUCLIDE
  (SIG(I),I=1,8)
  ****     EFFECTIVE ONE-GROUP CROSS SECTIONS IN BARNS.
           IF ON CARD U1 ALLWQ=33P3U___ORNL_78_,ONLY SIG(1) HAS
           TO BE PUNCHED,IF ALLWQ=33P3RU__ORNL_78_,ONLY SIG(2)
           HAS TO BE PUNCHED, ETC.
IMPORTANT NOTE : 1. THE CODE WILL FIRST CALCULATE CROSS
SECTIONS FROM UNIT 7 ( OR FROM CARDS IF INPT= 1,CARD A) I.E.
ONE-GROUP CROSS SECTIONS CALCULATED WITH THERMAL CROSS SECTIONS,
RESONANCE INTEGRALS AND FAST CROSS SECTIONS FROM OLD ORIGEN
(IF NOT MODIFIED BY CARD INPUT) BUT WITH UPDATED SPECTRAL
INDICES FROM CARD B.
2.IF LPU = -1 (AS RECOMMENDED),THE CODE WILL TAKE THE CROSS
SECTIONS FROM UNIT ISB > 5, MULTIPLY THESE BY TOTAL/THERMAL
FLUX RATIO  1.+14.509*RES+FAST/1.45  AND REPLACE THE VALUES
COMPUTED BEFORE. FOR NUCLIDES FOUND ON CARDS U2 THE SIG(I)
VALUES WILL FURNISH THE CROSS SECTIONS FINALLY BEING USED.
THE CARDS U2 HAVE TO BE IN THE SAME ORDER AS THE NUCLIDES IN
THE LIBRARY ON UNIT 7. FURTHERMORE, IF FOR ONE NUCLIDE SEVERAL
TYPES OF CROSS SECTIONS WILL BE READ, THE CARDS U2 SHOULD BE
GIVEN IN THE ORDER OF THE SIGTYP VALUES AS LISTED ABOVE.
THE LAST CARD U2 MUST BE A BLANK CARD.

CARD C         FORMAT(13I5)

  MMN      NUMBER OF TIME INTERVALS DURING IRRADIATION PERIOD,
           MMN < 11.
  MOUT     TOTAL NUMBER OF INTERVALS FOR IRRADIATION AND POST-
           IRRADIATION  PERIODS, MOUT < 11.
  NOBLND   NUMBER OF MATERIALS TO BE BLENDED, ENTER 0 OR 1 IF
           NO BLENDING IS TO BE DONE, NOBLND < 11 ( SEE COMMENT
           ON NOBLND BELOW )
  INDEX    AN INPUT INDICATOR. INDEX = 0/1 : POWER(M)/FLUX(M)
           WILL BE READ ON A SUBSEQUENT CARD. IF MNN = 0, INDEX
           IS NOT USED BY THE CODE IN THIS STEP.
```

NTABLE AN OUTPUT INDICATOR. IF NTABLE = 0, ALL ISOTOPES AND ALL TIMES WILL BE GIVEN IN THE OUTPUT.
IF NTABLE = 1, ONLY SUMMARIES OF MOST IMPORTANT ISOTOPES WILL BE GIVEN.

MSTAR AN OUTPUT INDICATOR. WHEN SUMMARIZING THE MOST IMPORTANT ISOTOPES, THE CODE ELIMINATES THE ISOTOPES WHOSE VALUES ARE BELOW SOME THRESHOLD IN TIME PERIOD M = MSTAR (SEE CARD TYPE I).

NGO AN INDICATOR WHICH TELLS THE CODE WHETHER THE PRESENT CALCULATION WILL BE CONTINUED IN A SUBSEQUENT SET OF TIMES, OR WHETHER A NEW CALCULATION IS TO BE DONE.
NGO < 0 : THIS INDICATES A NEW CALCULATION WITH NEW INITIAL CONDITIONS USING THE SAME NUCLEAR DATA. A CARD OF TYPE C WILL BE EXPECTED FOLLOWING CARD K OR N.IN CASE OF A BLANK CARD INSTEAD OF CARD C THE PROGRAM WILL STOP EXECUTION.
NGO = 0 : THIS INDICATES A NEW CALCULATION WITH A NEW SET OF NUCLEAR DATA. A CARD OF TYPE A WILL BE EXPECTED FOLLOWING CARD K OR N.
NGO > 0 : THIS INDICATES THAT THE PRESENT CALCULATION WILL BE CONTINUED. A CARD OF TYPE O FOLLOWING CARD K OR N WILL BE EXPECTED.

MPROS AN INDICATOR FOR CONTINUOUS CHEMICAL PROCESSING OPTION.
MPROS = NUMBER OF GROUPS OF CHEMICAL ELEMENTS PROCESSED. MPROS = 0 FOR NO CHEMICAL PROCESSING.

MFEED CONTINUOUS FEED OPTION FOR FLUID FUEL REACTOR.
MFEED = 0 FOR NO CONTINUOUS FEED; MFEED > 0 FOR CONTINUOUS FEED.

JTO AN OUTPUT INDICATOR TO SELECT TABLES AND NUCLIDES
*** OR ELEMENTS FOR PRINTING - SEE FOLLOWING TABLE.

KPL AN INDICATOR TO CONTROL STORAGE OF OUTPUT TABLES
*** FOR SUBSEQUENT USE E.G. FOR PLOTTING ETC.
KPL = 0/1/2 : NO STORAGE/STORAGE ON UNIT 20 AND PRINTING SOME CONTROL INFORMATION ON UNIT 3/ STORAGE ON UNIT 20 AND PRINTING OF COMPLETE CONTROL INFORMATION ON UNIT 3.
NOTE : JOB CONTROL CARDS HAVE TO BE SPECIFIED FOR UNIT 3 AND 20.

KSB AN INDICATOR FOR SECONDARY BURN-UP DEPENDENT CROSS
*** SECTION INPUT.
KSB = 1 : FOR ACTINIDES THE PREVIOUSLY CALCULATED RESP. INPUT BURNUP-INDEPENT CROSS SECTIONS WILL BE REPLACED BY BURNUP DEPENDENT CROSS-SECTIONS READ FROM CARDS OR FROM UNIT ISB - SEE CARD T1.
KSB = 0 : NO BURN-UP DEPENDENT CROSS SECTIONS WILL BE READ.
RECOMMENDATION : KSB = 1

NGF AN INDICATOR FOR THE CHANGE OF SPECTRAL INDICES.
*** TOGETHER WITH NGO = 0 ON THIS CARD A VALUE OF E.G. NGF = 6 TELLS THE CODE THAT AT THE BEGINNING OF THE NEXT SEQUENCE OF TIME STEPS THE CROSS SECTIONS WILL BE RECALCULATED WITH A NEW SET OF SPECTRAL INDICES AND THAT THE FINAL NUCLIDE VECTOR OF TIME STEP 6 WILL BE USED AS THE INITIAL VECTOR FOR THE SUBSEQUENT CALCULATION. FOLLOWING CARD K OR N CARDS OF TYPE A, B, (U1, U2) AND O WILL BE EXPECTED WITH MSUB = NGF = 6 IN OUR CASE ON CARD O.
NGF = 0 : NO CHANGE OF SPECTRAL INDICES.(THIS IS THE STANDARD CASE

COMMENT FOR NOBLND > 1 :

1. NOBLND MUST BE GIVEN FOR IRRADIATION AND DECAY PERIODS BEFORE BLENDING;IN CASE OF CONTINUED IRRADIATION (MORE THAN 10 IRRADIATION PERIODS IN TOTAL) NOBLND HAS TO BE GIVEN FOR THE LAST CONTINUATION ONLY.
2. AFTER IRRADIATION FORMALLY AT LEAST ONE DECAY STEP FOR THE NON-BLENDED FUEL IS NEEDED
3. THE IRRADIATED FUELS WILL BE BLENDED AT DISCHARGE.THE VALUE OF MSUB.GE.0 FOR THE COMMON DECAY PERIODS IS IRRELEVANT.FOR THE FIRST COMMON DECAY PERIODS TMO=0. IS RECOMMENDED.

CONTROL OF PRINTING TABLES AND NUCLIDES/ELEMENTS
WITH THE OUTPUT INDICATOR JTO

JTO	OUTPUT TABLES?	SELECTED TABLES ?	SELECTED NUCL. OR ELEMENTS ?	GAMMA TABLES ?
0	YES	NO	NO	YES
1	NO	NO	NO	NO
2	YES	YES	NO	YES
3	YES	YES	YES	YES
4	YES	NO	YES	YES
5	YES	NO	NO	NO
6	YES	YES	NO	NO
7	YES	YES	YES	NO
8	YES	NO	YES	NO
9	NO	NO	NO	YES
-1	PRINTING AS IN PREVIOUS STEP			

IF JTO = 2/3/6/7 CARDS S1, IF JTO = 4/8 CARD S2,
IF JTO = 0/1/5/9/-1 CARD D IS EXPECTED FOLLOWING CARD C OR O.

```
CARDS S1        FORMAT(3X,7(7X,3I1))

(NTO(I),I=1,63) THE NTO VALUES CONTROL THE PRINTING OF OUTPUT
                TABLES, THEY ARE PUNCHED ON THREE CARDS :
            1. CARD : TABLES FOR LIGHT ELEMENTS
            2. CARD : TABLES FOR ACTINIDES (HEAVY NUCLIDES)
            3. CARD : TABLES FOR FISSION PRODUCTS.
          ON EACH OF THESE THREE CARDS 7 SETS OF 3 INTEGER
          VALUES 0 OR 1 WILL BE PUNCHED :
            1. SET : GRAMATOM TABLE
            2. SET : GRAM TABLE
            3. SET : CURIE TABLE
            4. SET : POWER TABLE
            5. SET : GAMMA POWER TABLE
            6. SET : RADIOTOXICITY TABLE FOR INHALATION
            7. SET : RADIOTOXICITY TABLE FOR INGESTION
          IF IN THESE SETS THE
            1. VALUE = 1 : A NUCLIDE TABLE
            2. VALUE = 1 : AN ELEMENT TABLE
            3. VALUE = 1 : SUMMARY TABLES FOR NUCLIDES AND
                           ELEMENTS
          WILL BE PRINTED. PUNCHING OF ZEROS OR BLANKS WILL
          SUPPRESS THE PRINTING OF THE CORRESPONDING TABLES.
SPECIAL OPTION : IF NTO = 2 ON THE 3.SET (CURIE TABLE) ON THE
                 2. S2-CARD (ACTINIDES) IS CHOSEN,THE ALPHA-
                 ACTIVITY OF THE ACTINIDE NUCLEI WILL BE PRINTED
                 ADDITIONALLY
NOTE : PRINTING OF SUMMARY TABLES WILL BE CONTROLLED BY THE
       CUTOFF VALUES ON CARD I.
       FROM AN IRRADIATION PERIOD ONLY GRAMATOM AND GRAM TAB-
       LES WILL BE PRINTED. THE VALUES IN SETS 3.-7. THEN WILL
       BE IGNORED.

IF  JTO = 3/7  CARDS S2, IF  JTO = 2/6  CARD D IS EXPECTED
FOLLOWING CARDS S1.

CARDS S2        FORMAT(I5/8(4X,A2,I3,A1)/I5)

  NSEL    NUMBER OF ELEMENTS AND NUCLIDES TO WHICH PRINTED
  ****    AND STORED  OUTPUT WILL BE REDUCED
  ((ELE(N),MWT(N),STA(N)),N=1,NSEL)
  ****    ELE(N) = NAME OF ELEMENT IN KFK-GRUBA NOMENCLATURE,
          E.G.,C_ FOR CARBON, U_ FOR URANIUM, PU FOR PLUTONIUM
          MWT(N) = ATOMIC MASS NUMBER
          STA(N) = _/M FOR GROUND STATE/ISOMERIC STATE;
          THIS CARD MAY BE CONTINUED.
  NCU     AN INDICATOR FOR EXTENDED USE OF THE CUTOFF VALUES
  ***     GIVEN ON CARD I.
          NCU = 0/1 : CUTOFF VALUES ARE APPLIED TO SUMMARY
          TABLES ONLY/ ARE APPLIED TO ALL TABLES.

  IMPORTANT : IF THE MWT AND STA VALUES ARE NOT PUNCHED ,
  OUTPUT WILL BE PRINTED FOR ALL ISOTOPES OF ELEMERNT ELE(N)
  AND , OF COURSE , FOR THIS ELEMENT.
  SPECIAL OPTION : IF ONLY THE FIRST AND THE LAST OF THE CARDS
  S2 ARE PUNCHED WITH NSEL=0 AND NCU=1 , THE CUTOFF ACCORDING
  TO CARD I WILL BE APPLIED TO ALL TABLES ALL NUCLIDES AND ALL
  ELEMENTS : THIS IS A COMFORTABLE WAY TO REDUCE THE OUTPUT
  TO THE MOST IMPORTANT NUCLIDES RESP. ELEMENTS AND HENCE IS
  RECOMMENDED FOR USE IN GENERAL.
```

CARD D FORMAT(20A4)

TITLE A TITLE FOR THE CALCULATION CONTAINING UP TO 80 ALPHAMERIC CHARACTERS. IT WILL BE PRINTED AS THE HEADING OF THE OUTPUT TABLES.

IF MMN > 0 CARDS OF TYPE E OR F WILL BE EXPECTED.

IF INDEX = 0 CARD E AND CARD G, IF INDEX = 1 CARD F AND CARD G WILL FOLLOW.

CARD E FORMAT(10E8.2)

(POWER(M),M=1,MMN)
SPECIFIC THERMAL POWER OF FUEL IN IRRADIATION PERIOD M (MW/UNIT OF FUEL). THERE MAY BE PERIODS OF ZERO POWER; HOWEVER, THERE MAY NOT BE TWO CONSECUTIVE ZERO-POWER INTERVALS, AND THE FINAL IRRADIATION PERIOD MAY NOT HAVE ZERO POWER.

CARD F FORMAT(10E8.2)

(FLUX(M),M=1,MMN)
THERMAL NEUTRON FLUX IN IRRADIATION PERIOD M (NEUTRONS/CM**2 * SEC), OR THE TOTAL NEUTRON FLUX FOR A FAST REACTOR. THERE MAY BE PERIODS WHEN THE FLUX IS ZERO DURING THE IRRADIATION PERIOD; HOWEVER, THERE MAY NOT BE TWO CONSECUTIVE PERIODS OF ZERO FLUX, AND THE FINAL IRRADIATION PERIOD MAY NOT HAVE ZERO FLUX.

CARD G FORMAT(10E8.2)

(T(M),M=1,MOUT)
ELAPSED TIME SINCE THE BEGINNING OF THE CALCULATION (MEASURED IN TERMS OF TUNIT).
IF 0.LE.MMN.LT.MOUT THE TIME AFTER DISCHARGE REFERS TO DISCHARGE

CARD H FORMAT(10A4,F7.0,A3)

BASIS A 40-CHARACTER ALPHAMERIC TITLE THAT IS THE UNIT OF FUEL ON WHICH THE CALCULATION IS BASED AND THAT WILL BE PRINTED OUT AS THE BASIS FOR THE CALCULATION E.G. 'METRIC TON OF FUEL CHARGED TO REACTOR'.

TCONST A FACTOR TO CONVERT THE INPUT VALUES OF T(M) INTO SECONDS E.G., TCONST = 3.155E7 IF VALUES OF T(M) ARE INPUT IN TERMS OF YEARS.

TUNIT AN ALPHAMERIC DESIGNATION FOR THE INPUT UNITS OF T(M), E.G., ' D ' FOR DAYS.

NOTE : THE BASIS IS ARBITRARY AND MAY BE CHOSEN BY THE USER, BUT IT HAS TO BE VERIFIED THAT THE INPUT GRAMATOMS AND POWERS REFER TO THIS SAME BASIS.

```
CARD I          FORMAT(8E10.3)

  (CUTOFF(MS),MS=1,7)
          THRESHOLD VALUES FOR OUTPUT (SUMMARY) TABLES.
          ANY ISOTOPE WHOSE VALUE IN THE TIME PERIOD MSTAR IS
          LESS THAN THE CORRESPONDING CUTOFF VALUE WILL BE
          OMITTED FROM THE (SUMMARY) TABLE.
          ( SEE ALSO CARDS S2 : SPECIAL OPTIONS )
            MS = 1 : GRAMATOM TABLE
                 2 : GRAM TABLE
                 3 : CURIE TABLE
                 4 : POWER TABLE
                 5 : GAMMA POWER TABLE
                 6 : RADIOTOXICITY TABLE FOR INHALATION
                 7 : RADIOTOXICITY TABLE FOR INGESTION

A VALUE OF .001 IS RECOMMENDED FOR TABLES 1 TO 5, AND A VALUE
OF 1. IS RECOMMENDED FOR TABLES 6 AND 7.

IF NOBLND > 1  CARD J IS EXPECTED TO FOLLOW CARD I.

CARD J          FORMAT(8E10.3)

  (FACT(N),N=1,NOBLND)
          VALUES OF ATOMIC FRACTIONS OF THE MATERIALS IN A
          CALCULATION FOR A BLENDED FUEL.
          IN CASE OF CONTINUED IRRADIATION CARD J IS EXPECTED
          TO FOLLOW CARD O OF THE LAST CONTINUATION (SEE BELOW)
          CARD J MUST BE GIVEN ONLY FOR THE FIRST OF THE FUELS
          TO BE BLENDED.
CARD K          FORMAT(5(I6,E9.2),I5)

  ((INUCL(I),XCOMP(I)),I=1,5),NEXT
          INUCL : NUCLIDE IDENTIFIER FOR AN ISOTOPE IN FRESH
          FUEL  = ATOMIC NUMBER*10000 + ATOMIC WEIGHT*10 + IS,
          WHERE IS = 0/1 FOR GROUND STATE/ EXCITED STATE.
          XCOMP : CONCENTRATION OF NUCLIDE INUCL IN FRESH FUEL
          NEXT : INDICATOR GIVING THE TYPE OF THE 5 ISOTOPES
          ON THE CARD. THUS ALL FIVE MUST BE OF ONE TYPE.


             NEXT = 1 : ISOTOPES OF CLADDING AND STRUCTURAL
                        MATERIALS IN GRAMATOMS
                    2 : HEAVY MATERIAL ISOTOPES IN GRAMATOMS
                    3 : FISSION PRODUCT ISOTOPES IN GRAMATOMS
                    4 : ELEMENTS OF CLADDING AND STRUCTURAL
                        MATERIALS IN GRAMATOMS.
                    5 : ELEMENTS OF CLADDING AND STRUCTURAL
                        MATERIALS IN GRAMS

          ALL CARDS OF THIS TYPE SHOULD BE FOLLOWED BY A SING-
          LE BLANK CARD. WHEN THE PROGRAM ENCOUNTERS THE BLANK
          CARD, IT CONTINUES TO THE NEXT PART OF THE INPUT. IF
          IT ENCOUNTERS A BLANK INUCL FIELD, IT SKIPS THE REST
          OF THE INFORMATION ON THAT CARD AND READS THE NEXT
          CARD K.
```

THE FOLLOWING CARDS L AND M ARE READ ONLY IF MPROS > 0

CARD L FORMAT(8(E10.2,I2))

((PRATE(M),NOPROS(M)),M=1,MPROS)
PRATE(M) IS A FIRST ORDER REMOVAL CONSTANT FOR CHEMICAL PROCESSING BY PROCESSING STREAM M, 1/SEC, NOPROS(M) IS THE NUMBER OF ELEMENTS REMOVED BY STREAM M.

CARD M FORMAT(20I4)

((NZPROS(M,N),N=1,NOPROS(M)),M=1,MPROS)
NZPROS(M,N) IS THE ATOMIC NUMBER OF ELEMENT N IN PROCESSING STREAM M.

THE FOLLOWING CARD N IS READ ONLY IF MFEED > 0.

CARD N FORMAT(5(I6,E9.2),I5)

((INUCL(I),XCOMP(I)),I=1,5),NEXT
XCOMP(I) IS THE CONTINUOUS FEED RATE OF ISOTOPE INUCL(I) IN GRAMATOMS PER UNIT OF FUEL. THE UNIT OF FUEL IS GIVEN BY THE VARIABLE BASIS. INUCL AND NEXT FOLLOW THE SAME CONVENTIONS AS DESCRIBED FOR CARDS OF TYPE K.

IF AT THE BEGINNING OF THE IRRADIATION CALCULATION KSB = 1 ON CARD C, CARD T1 IS EXPECTED FOLLOWING CARD K OR CARD M OR CARD N.NOTE THAT THE LAST CARD K OR N SHOULD BE A BLANK CARD.

CARD T1 FORMAT(6A8,2A8,1X,3I5)

BUTIT A 48-CHARACTER ALPHAMERIC TITLE FOR IDENTIFYING
**** INPUT DATA ON UNIT ISB.
(ACTWQ(I),I=1,2)
**** A 16-CHARACTER ALPHAMERIC NAME FOR IDENTIFYING THE REACTOR SPECTRUM AND BURN-UP DEPENDENT ACTINIDE CROSS SECTION LIBRARY.
ON THE KORIDATA LIBRARY (DSN=INR909.KORIDATA) THE FOLLOWING TYPES OF BURNUP DEPENDENT ACTINIDE CROSS-SECTION SETS ARE AVAILABLE :

1. ORNL ACTINIDE CROSS-SECTIONS FOR US-PWRS AND -BWRS /2,3/ :
33P3U___ORNL_78_,33P3RU__ORNL_78_,33P3RM__ORNL_78_,
27B4U___ORNL_78_,27B4RU__ORNL_78_,27B4RM__ORNL_78_,
50P5U___ORNL_80_.
FOR EXPLANATIONS SEE CARD U2.

2. KFK ACTINIDE CROSS-SECTIONS FOR EUROPEAN PWRS :
30P3U_31_KFK_KWO,38P3U_30_KFK_KWO,38P3U_28_KFK_KWO,
33P3U_32_KFK_BIB,36P3U_34_KFK_BIB,40P3U_36_KFK_BIB,
30P4U_31_KFK_TRI
HERE ADDITIONALLY THE INITIAL ENRICHMENT IS CHARACTERIZED BY THE LAST TWO FIGURES OF THE FIRST 8-BYTE WORD E.G. 31 = 3.1% U235.THE RACTOR TYPES ARE : PWR OBRIGHEIM (KWO), PWR BIBLIS (BIB), PWR TRINO VERCELLESE (TRI).ALL THESE KFK CROSS-SECTIONS ARE PROCESSED FROM THE LATEST KEDAK FILE.DETAILS CAN BE SEEN IN CHAPTER 3 OF THIS REPORT.

```
          IF ON CARD T1 BURNUP DEPENDENT ACTINIDE CROSS-
          SECTIONS FROM KFK ARE USED, FOR  BURNUP -INDE-
          PENDENT CROSS-SECTIONS OF NON-ACTINIDE NUCLEI THE
          DATA SET 33P3U___KFK_80_ ON CARD U1 SHOULD BE TAKEN
  ISB     FORTRAN UNIT FROM WHICH THE BURN-UP DEPENDENT ACTI-
  ***     NIDE CROSS SECTIONS WILL BE READ - SEE CARD U1.
  IFR     AN INDICATOR FOR READING BURN-UP DEPENDENT
  ***     TOTAL FLUX/THERMAL FLUX RATIOS.
          IFR = 1/0 : RATIOS ARE READ FROM UNIT 5 (CARDS) /
          FROM UNIT ISB; IFR = 0 AND ISB = 5 IS NOT ALLOWED.
  ITIW    AN INDICATOR FOR PRINTING BURN-UP DEPENDENT ACTI-
  ****    NIDE CROSS SECTIONS FOR ISB > 5.
          ITIW = 0/1 : PRINTED/NOT PRINTED.

IF  ISB > 5  AND IF NO MODIFICATION OF THE KORIDATA VALUES
OR NO ADDING OF FURTHER ACTINIDE DATA IS WANTED A BLANK CARD
AND THEN CARDS L, M, N OR O ARE EXPECTED TO FOLLOW CARD T1.
IF  ISB > 5  AND IF MODIFYING AND/OR ADDITIONAL ACTINIDE DATA
WILL BE GIVEN CARDS T5 AND T6 ARE EXPECTED FOLLOWING CARD T1.
THESE MODIFYING OR ADDITIONAL DATA SHOULD CORRESPOND TO THE
LIBRARY DATA IN NUMBER AND BURN-UP MESH.
IF  ISB = 5  CARDS T2, T3, T4, T5, T6 WILL FOLLOW CARD T1.

CARD T2        FORMAT(I6)

  IBUT    NUMBER OF BURN-UP MESHPOINTS FOR BURN-UP DEPENDENT
  ****    ACTINIDE CROSS SECTIONS, IBUT < 101.
CARD T3        FORMAT(10E8.2)

  (BUT(I),I=1,IBUT)
  ****    BURN-UP MESHPOINTS IN MWD/GRAMATOM HEAVY MATERIAL,
          INCREASING AND STARTING WITH .0

CARD T4        FORMAT(10E8.2)

  (FLURAT(I),I=1,IBUT)
  ****    TOTAL FLUX/THERMAL FLUX RATIOS CORRESPONDING TO THE
          BUT(I) VALUES. IF THE FIRST FLURAT VALUE IS FOUND TO
          BE ZERO, ALL FLURAT VALUES WILL BE SET CONSTANT AND
          EQUAL TO  1.+14.509*RES+FAST/1.45 . IF RES = 1.,
          I.E. FOR A FAST REACTOR, ALL FLURAT VALUES WILL BE
          SET EQUAL TO 1. THE KORIDATA LIBRARY CONTAINS
          FLURAT VALUES EQUAL TO ZERO FOR ALL I.
CARD T5        FORMAT(I6,4X,A8)

  NUNUM   NUCLIDE IDENTIFIER - SEE CARD K
  ****
  SIGTYP  TYPE OF CROSS SECTION TO BE READ
  ****    SIGTYP = SCAPT___/SCAPT1__/SFISS___ FOR CAPTURE
          CROSS SECTION TO GROUND STATE NUCLIDE/ CAPTURE CROSS
          SECTION TO EXCITED STATE NUCLIDE/ FISSION CROSS SEC-
          TION.
```

CARD T6 FORMAT(10E8.2)

(SIG(I),I=1,IBUT)
**** EFFECTIVE ONE-GROUP CROSS SECTIONS OF TYPE 'SIG-TYP' IN BARNS CORRESPONDING TO THE BUT(I) VALUES.
NOTE : SIG(I)*TOTAL FLUX = REACTION RATE. IN ORDER TO GET ORIGEN CROSS SECTIONS THE SIG VALUES WILL BE MULTIPLIED BY THE FLURAT VALUES.
THIS CARD AS WELL AS CARDS T3 AND T4 MAY BE CONTINUED IF IBUT > 10.

WE HAVE TWO CARDS FOR EACH ADDITIONAL NUCLIDE AND CROSS SECTION. THESE SETS OF TWO CARDS MAY BE IN ANY ORDER, BUT THE LIGHTEST MUST NOT BE LIGHTER THAN THE FIRST NUCLIDE IN THE LIBRARY ON UNIT ISB, I.E. 922340.
THE LAST CARD T5 HAS TO BE A BLANK CARD.

THE PROGRAM WILL NOW CALCULATE THE ISOTOPIC COMPOSITION FOR ALL MOUT TIME PERIODS AND WRITE OUTPUT. AFTER WRITING OUTPUT, THE PROGRAM WILL EITHER BE READY TO START A NEW PROBLEM (IF NGO< 1) OR CONTINUE THE PRESENT ONE (NGO > 0). IN THE LATTER CASE, THE INPUT FOR THE CONTINUATION OF THE CALCULATION HAS THE FORM :
CARD O FORMAT(13I5)

MMN,MOUT,NOBLND,INDEX,MSUB,MSTAR,NGO,MPROS,MFEED,
JTO,KPL,KSB,NGF

MSUB : THE TIME PERIOD IN THE LAST CALCULATION CONSIDERED AS THE START OF THE NEW CALCULATION; ALSO USED TO INDICATE THAT BATCH CHEMICAL PROCESSING OCCURS IF THE VALUE IS NEGATIVE.
MFEED : AN INPUT INDICATOR FOR CONTINUOUS FEED
MFEED = 0 : NO FEED
> 0 : CONTINUOUS FEED AT THE SAME RATE AS IN THE PREVIOUS CALCULATION.

ALL OTHER VARIABLES HAVE THEIR FORMER MEANING; KSB IS ONLY USED AT THE BEGINNING OF IRRADIATION .

THIS CARD IS FOLLOWED BY CARDS OF TYPE D, E, F, G AND H OR H' (SEE BELOW) TO COMPLETE THE INPUT FOR THE CONTINUED CALCULATION. THIS PROCEDURE MAY BE REPEATED AS DESIRED.
IF NOBLND>1 ON CARD O (IN THE CASE OF CONTINUED IRRADIATION OF THE FUEL TO BE BLENDED),CARD J HAS TO BE INSERTED BEFORE CARD D FOR THE LAST CONTINUED IRRADIATION PERIODS.
THE CALCULATION WILL STOP WHEN A BLANK CARD IS READ INSTEAD OF A CARD OF TYPE C.

WHEN CONTINUING A CALCULATION THAT WAS STARTED IN A PREVIOUS SET OF TIME PERIODS, A CARD OF TYPE H MAY BE MODIFIED TO INCLUDE ONE ADDITIONAL PIECE OF INFORMATION :

CARD H' FORMAT(10A4,F7.0,A3,F10.3)

BASIS,TCONST,TUNIT AS BEFORE ON CARD H

TMO THE TIME MEASURED IN TERMS OF TUNIT TO WHICH THE TIMES READ ON CARD G ARE REFERENCED. THUS IT IS POSSIBLE TO CALCULATE THE POSTIRRADIATION PROPERTIES OF A FUEL, SAY, FOR TEN YEARS AFTER DISCHARGE IN TEN TIME PERIODS, AND THEN TO CALCULATE THE PROPERTIES FOR TEN SUBSEQUENT TIME PERIODS BY SETTING MSUB = 10 AND TMO = 10.0 YEARS. THE ARRAY 'TIME' WILL HAVE THE VALUES 11 THROUGH 20, AND THE CODE WILL CALCULATE THE PROPERTIES FOR THE ELAPSED TIME SINCE THE TENTH YEAR. POSTIRRADIATION PROPERTIES ARE CALCULATED WITH RESPECT TO THE TIME OF DISCHARGE IF TMO IS NOT GIVEN A VALUE.

A NEGATIVE VALUE FOR THE VARIABLE MSUB INDICATES THAT BATCH CHEMICAL PROCESSING IS ASSUMED TO OCCUR AT TIME T(-MSUB), AND THAT DATA GIVING PROCESSING INFORMATION ARE REQUIRED TO BE READ. WHEN MSUB HAS A NEGATIVE VALUE, CARDS OF TYPE P ARE EXPECTED TO FOLLOW CARD H'.

CARD P FORMAT(I6,4X,E10.3)

LEMENT THE ATOMIC NUMBER OF A CERTAIN CHEMICAL ELEMENT TO BE REMOVED

FREPRO THE FRACTION OF THE MATERIAL THAT REMAINS AFTER PROCESSING. THE ARRAY FREPRO IS INITIALLY SET EQUAL TO 1.0; THUS, IF NO DATA ARE READ FOR AN ELEMENT, PROCESSING DOES NOT AFFECT ITS CONCENTRATION.

CARDS OF TYPE P ARE EXPECTED TO BE READ UNTIL A BLANK CARD IS ENCOUNTERED.

****

TO RUN A KORIGEN JOB ON THE KFK IBM/370 COMPUTER JCL CARDS AS IN THE FOLLOWING EXAMPLE MAY BE USED :

```
//INR238KO JOB (0238,106,P5H1M),WIESE
// EXEC FHG,LIB=NUSYS,NAME=KORIGEN
//G.FT03F001 DD SYSOUT=A,DCB=*.FT06F001
//G.FT07F001 DD DSN=INR909.ORFI(M3),DISP=SHR
//G.FT07F002 DD DSN=INR909.ORFI(M4),DISP=SHR
//G.FT07F003 DD DSN=INR909.ORFI(M5),DISP=SHR
//G.FT07F004 DD DSN=INR909.ORFI(M6),DISP=SHR
//G.FT07F005 DD DSN=INR909.ORFI(M7),DISP=SHR
//G.FT07F006 DD DSN=INR909.ORFI(M8),DISP=SHR
//G.FT30F001 DD DSN=INR909.KORIDATA,DISP=SHR
//G.SYSIN DD *
      (INPUT DATA)
//
```

IN THE ABOVE EXAMPLE ISB = 30 IS ASSUMED.

IF RESULTS ARE WANTED TO BE STORED, A DD-CARD FOR UNIT 20 HAS TO BE INSERTED.

IF A TABLE OF THE DECAY CHAINS IS DESIRED THE DD-CARD

```
//G.FT45F001 DD SYSOUT=A,DCB=*.FT06F001
```

MUST BE INCLUDED. SEE CARD A.

FOR PRINTING THE KORIDATA LIBRARY (DSN=INR909.KORIDATA) A DD-CARD

```
//G.FT09F001 DD SYSOUT=A,DCB=*.FT06F001
```

IS NEEDED - SEE CARDS U1 AND T1.

****

REFERENCES :

/1/ M.J. BELL : ORIGEN - THE OAK RIDGE ISOTOPE GENERATION AND DEPLETION CODE ORNL-4628 MAY 1973

/2/ A.G. CROFF ,M.A. BJERKE,G.W. MORRISON,L.M.PETRIE : REVISED URANIUM-PLUTONIUM CYCLE PWR AND BWR MODELS FOR THE ORIGEN COMPUTER CODE ORNL/TM-6051 SEPTEMBER 1978

/3/ A.G. CROFF , M.A. BJERKE : ALTERNATIVE FUEL CYCLE PWR MODELS FOR THE ORIGEN COMPUTER CODE ORNL/TM-7005 FEBRUARY 1980

B.2 Input Examples

As a supplement to the input description, seven input examples are compiled in the following. In them the most important input options are realized. By referring to these examples and to the description, the user should find it easier to assemble an input for an actual application.

For the purpose of orientation, the special characteristics of the examples included will be given first with references to associated indicators and variables. The examples themselves will follow. For Example 1, which represents the input for the output given in Appendix D.1, we have included the "Input Reproduction" generated by the program. It is used for input control and contains the assignment of the input values to the input cards A, B, etc.

Errors in input can be found more easily by means of the Input Reproduction.

Input Example 1: PWR (CPU time: 34 sec)*

Irradiation:
- with continuation (more than 10 intervals, MSUB > 0)
- gram input for light elements (NEXT)
- burnup-averaged (LPU, 33P3U___KfK_80__) and burnup-dependent (KSB, 33P3U_32_KfK_BIB) cross sections from KORIDATA

Fuel Decay:
- selected tables (JTO,NTO) with cutoff (NSEL,NCU)

Waste Decay:
- separation of U and Pu at beginning (reprocessing WA, MSUB<0)
- selected tables as in preceding interval (JTO = -1)
- times after discharge (TMO>0)

---

* Calculated with a load module on the KfK computer system, an IBM 3033 370-168 with a REGION of 1 MB without application of OVERLAY.

Input Example 2: PWR

Irradiation:

- with continuation (more than 10 intervals, MSUB>0)
- selected tables (JTO, NTO)
- gram input for light elements (NEXT)
- times continuously referred to start of irradiation (TMO>0)
- cross sections from KORIDATA (LPU, KSB, file 50P5U)
- high burnup

Fuel Decay:

- selected tables (JTO, NTO) with cutoff (NSEL, NCU)

Input Example 3: PWR with U and MOX Fuel (Pu Recycling)
(CPU time: 74 sec)

U Fuel Irradiation:
- with continuation (more than 10 intervals, MSUB>0)
- cross sections from KORIDATA (LPU, KSB, file 33P3PU)
- gram input for light elements (NEXT)
- blending (NOBLEND = 2 in continuation of irradiation)
- blending fractions in continuation of irradiation (FACT)

U Fuel Decay:

- selected tables (JTO, NTO) with cutoff (NSEL,NCU)

MOX Fuel Irradiation:

- cross sections from KORIDATA (LPU, KSB, File 33P3RM)
- gram input for light elements (NEXT)

MOX Fuel Decay:

- selected tables (JTO,NTO) with cutoff (NSEL,NCU)

Mixing and Decay of Mixed Fuel:

- selected tables with cutoff

Separation of U and Pu (WA [reprocessing]) and Decay of Waste from the Mixed Fuel:

- selected tables
- times after discharge (TMO>0)

Input Example 4: FBR (CPU time: 20 sec)

Overwriting of basis data for 6 actinides in basis data format (LPU = 6)

Irradiation:

- selected tables with cutoff
- fuel basis: 1.688 t HM

Input Example 5: PWR (CPU time: 47 sec)

Irradiation:

- cross sections from KORIDATA, but not burnup-dependent (LPU, file 50P5U)
- continuation of irradiation with altered spectral parameters (NGF)
- selected tables without cutoff
- times continuously referred to start of irradiation (TMO>0)

Fuel Decay:

- combined with the last part of irradiation (0 < MMN < MOUT)
- decay times relate to the time of discharge

Input Example 6: PWR (CPU time: 19 sec)

Irradiation:

- no additional data from KORIDATA, only basis data
- with continuation (MSUB>0)
- times continuously referred to start of irradiation (TMO>0)
- selected tables without cutoff

Fuel Decay:
Separation U and Pu (WA) and decay of the waste:

Input Example 7: PWR (CPU time: 14 sec)

Only irradiation:

- cross sections from KORIDATA, (LPU, KSB, file 33P3U)
- capture cross section Cs-133 overwritten (burnup-averaged)
- capture cross section Pu-240 overwritten (burnup-dependent)
- selected elements and nuclides

Key to German Terms Used in Following Input Examples (see notation in left margin):

*	U fuel in recycling reactor
**	1 TSM [ton heavy metal] reactor charge
***	decay of U fuel
****	MOX fuel in recycling reactor
+	decay of MOX fuel
++	PWR with Pu recycling, decay of fuel until reprocessing
+++	PWR with Pu recycling, decay of waste after reprocessing (7 yrs)
++++	1 TSM [ton heavy metal] times after discharge
#	PWR 3.2% U-235, partial irradiation, 1 cycle

```
*******  EINGABEBEISPIEL 1 **********  [Input Example 1]
REFERENCE REACTORS : BIBLIS 3.2% U235 33GWD/THM  10/1982                    2  0 1
   0.701     0.304     2.01       1.E-25 6 880 0 0 0-1
BIBLIS 3.2% U235   33GWD/THM                    33P3U   ORNL 78     27          0

    9    9    0    0    0    9    1    0    0    1    0    1
IRRADIATION  BIBLIS  3.2% U-235 33 GWD/THM
33.033  33.033  33.033  33.033    0.    33.033  33.033  33.033  33.033
  20.    111.    222.    333.    373.    393.    484.    595.    706.
1 THM CHARGED TO REACTOR                  86400. D
     1.E-4     1.E-3     1.E-03    1.E-03     1.E-3     1.E 6     1.E 6
05    0.935    06     110.7   08    136033.  13     1185.8  14    1079.5        5
15    48.8     16     32.9    22    846.9    24     23390.  25    2096.         5
26    75494.   27     119.8   28    25626.8  29     9.7     40    288600.       5
41    508.7    42     504.8   50    4269.    73     408.8                       5
92234 1.038    92235 136.17   92238 4066.39                                      2

BIBLIS 3.2% U235 33GWD/THM                    33P3U 32 KFK BIB    27          0

    5    5    0    0    9    5    1    0    0   -1    0    1    0
IRRADIATION  BIBLIS  3.2% U-235 33 GWD/THM
0.      33.033  33.033  33.033  33.033
  746.    766.    857.    968.   1079.
1 THM CHARGED TO REACTOR                  86400. D    706.
    0    4    0    0    5    1    1    0    0    3    0
                     11         1         1         1
                     11         2         1         1
                     1          1         1         1
    0
    1
FUEL DECAY  BIBLIS-TYPE 3.2% U235
   1.      2.      5.      7.
1 THM CHARGED TO REACTOR                  3.155E7 Y
    0   10    0    0   -4    1   -1    0    0   -1    0
WASTE DECAY  BIBLIS-TYPE 3.2% U235  REPROCESSING  7 Y  AFTER DISCHARGE
   10.     20.     50.    100.    200.    1000.      1.E4    1.E5    1.E6    1.E7
1 THM CHARGED TO REACTOR                  3.155E7 Y   7.
    92            .01
    94            .01
```

```
***** EINGABEBEISPIEL 2 *****  [Input Example 2]
BASIC NUCLEAR DATA LIBRARY FOR LWR                                          2  0 1
      .525      .197     2.441     1.E-25 7 380 1 0 0-1
AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-7005  50P5U   ORNL 80     27          0

   10   10    0    0    0   10    1    0    0    6    0    1    0
                     1
                    11

IRRADIATION PERIOD 1  821.5D  30GWD/THM  FUEL 3.5% U 235
   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52
     10.     90.    170.    250.    430.    510.    590.    670.    750.   821.5
1 MT OF HEAVY MATERIAL CHARGED TO REACT. 86400. D
    1.E-10    1.E-10    1.E-10    1.E-10    1.E-10         .1         .1
 6    113.59    7     43.123    8     880.94   13     1290.8   14    1104.4        5
15    47.151   16     23.579   22     808.89   24     23154.   25    2092.3        5
26    75271.   27     128.68   28     24556.   29     20.259   41    975.48        5
42    490.18                                                                       5
92234 1.141    92235 148.94   92238 4053.5                                         2

BURNUP DEPEND. ACTIN. X-SECT. FROM ORNL/TM-7005 50P5U   ORNL 80     27    0    0

    6    6    0    0   10    6    1    0    0   -1    0    1    0
IRRADIATION PERIOD 2  328.6D  42GWD/THM
   36.52   36.52   36.52   36.52   36.52   36.52
   871.5   921.5   971.5  1041.5  1091.5  1150.1
1 MT OF HEAVY MATERIAL CHARGED TO REACT. 86400. D      821.5
    0   10    0    0    6    1   -1    0    0    3    0    0    0
                     11        11        11                  1
                     11        11        11
                     11        11        11
    0
    1
FUEL DECAY 10Y FROM DISCHARGE
    180.    240.    365.   557.5    730.   912.5   1095.   1825.   2555.   3650.
1 MT OF HEAVY MATERIAL CHARGED TO REACT. 86400. D
```

```
      ***** EINGABEBEISPIEL 3 ***** [Input Example 3]
      BASIC NUCLEAR DATA FOR LWR                                                  2    1
      0.592      0.366      2.39          1.0E-2410 780 0 0 0-1
      AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-6051  33P3RU  ORNL 78     27

         10   10    0    0    0   10    1    0    0    1    0    1
 *    U-BRENNSTOFF IM REZYKLIER-REAKTOR  3.2% U 235
        40.909  40.909  40.909  40.909  40.909     0.0  40.909  40.909  40.909  40.909
       10.     73.     147.     220.     293.    365.    375.    438.    512.    585.
 **'   1 TSM  REAKTORBELADUNG                      86400. D
          1.E-07    1.E-06    1.E-10    1.E-10    1.E-10    1.E+05    1.E+03
       3    1.        5    .2        6    10.       7    10.       8    1.34E+05    5
       9    3.       11    10.      12    2.       13    21.      14    16.         5
      15    2.       19    10.      20    10.      22    10.      23    11.         5
      24    3.       25    1.       26    14.      27    10.      28    2.          5
      29    .5       30    10.      42    1.       50    1.       74    2.          5
      922340  1.239  922350 136.17  922380 4066.01                                  2

      BURNUP DEPEND. ACTIN. X-SECT. FROM ORNL/TM-6051 33P3RU  ORNL 78     27

         7    7    2    0   10    1    1    0    0   -1    0    1
      0.7       0.3
 *    U-BRENNSTOFF IM REZYKLIER-REAKTOR  3.2% U 235
        40.909     0.0  40.909  40.909  40.909  40.909  40.909
       73.     145.     155.     218.     292.     365.     439.
       1 TSM  REAKTORBELADUNG                      86400. D
          0    6    2    0    7    1    0    0    0    7
                            11        11        11
                            11        11        11
                            11        11        11

         1
 *** ZERFALL DES U-BRENNSTOFFS
      10.     50.     100.     150.     200.     365.
 **    1 TSM  REAKTORBELADUNG                      86400. D
      BASIC NUCLEAR DATA FOR LWR                                                  2    1
       0.509      0.73      5.49          1.0E-2410 780 0 0 0-1
       AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-6051  33P3RM  ORNL 78     27

         10   10    0    0    0   10    1    0    0    1    0    1
****  MOX-BRENNSTOFF IM REZYKLIER-REAKTOR 2.91%PU-FISS  .68% U 235
        40.909  40.909  40.909  40.909  40.909     0.0  40.909  40.909  40.909  40.909
       10.     73.     147.     220.     293.    365.    375.    438.    512.    585.
 **    1 TSM  REAKTORBELADUNG                      86400. D
          1.E-07    1.E-06    1.E-10    1.E-10    1.E-10    1.E+05    1.E+03
       3    1.        5    .2        6    10.       7    10.       8    1.34E+05    5
       9    3.       11    10.      12    2.       13    21.      14    16.         5
      15    2.       19    10.      20    10.      22    10.      23    11.         5
      24    3.       25    1.       26    14.      27    10.      28    2.          5
      29    .5       30    10.      42    1.       50    1.       74    2.          5
      922340 0.2278  922350 28.9064 9223803982.03  942380 4.256   94239 93.753       2
      942400 48.221  942410 27.593  942420 15.65                                     2

      BURNUP DEPEND. ACTIN. X-SECT. FROM ORNL/TM-6051 33P3RM  ORNL 78     27

         7    7    2    0   10    1    1    0    0   -1    0    1
****  MOX-BRENNSTOFF IM REZYKLIER-REAKTOR 2.91%PU-FISS  .68% U 235
        40.909     0.0  40.909  40.909  40.909  40.909  40.909
       73.     145.     155.     218.     292.     365.     439.
 **    1 TSM  REAKTORBELADUNG                      86400. D
          0    6    2    0    7    1    1    0    0    7
                            11        11        11
                            11        11        11
                            11        11        11

         1
 +    ZERFALL DES MOX-BRENNSTOFFS
      10.     50.     100.     150.     200.     365.
 **    1 TSM  REAKTORBELADUNG                      86400. D
          0    6    0    0    7    1    1    0    0    3
                            11        11        11
                            11        11        11
                            11        11        11

         1
 ++         DWR MIT PU-REZYKLIERUNG  ZERFALL DES BRENNSTOFFS BIS ZUR WA
      1.      2.      3.      5.      6.      7.
 **    1 TSM  REAKTORBELADUNG                      3.155E7 Y
          0   10    0    0   -6    1   -1    0    0    3
                            11        11        11
                            11        11        11
                            11        11        11

         1
 +++        DWR MIT PU-REZYKLIERUNG ZERFALL DES ABFALLS NACH DER WA ( 7 Y )
      10.     20.     50.     100.     200.     1000.   10000.  100000. 1.0 E+06 1.0E+07
++++  1 TSM  ZEITEN NACH ENTLADUNG                 3.155E7 Y         7.
          92    0.01
          94    0.01
```

```
***** EINGABEBEISPIEL 4 *****  [Input Example 4]
 LMFBR NUCLEAR DATA  (OLD ORIGEN )                                          3  0 1
        1.        1.        1.    1.E-25 6 280 1 0 0 6
922350 922380 942390 942400 942410 942420
        4.40E-01        0.0   1.70E 00                  1.12E-03.0   3.52E-063
        2.50E-01        0.0   5.20E-02                  1.53E-03.0   1.12E-053
        3.90E-01        0.0   1.80E 00                  2.00E-04.0   4.35E-063
        3.40E-01        0.0   4.20E-01                  5.10E-04.0   2.62E-063
        4.80E-01        0.0   2.40E 00                  2.10E-03.0   5.90E-063
        4.80E-01        0.0   3.10E-01                  5.60E-04.0   5.50E-063
   10   10    0    0    0   10   -1    0    0    7    0    0    0

                    11
                    11
    0
    1
SNEAK-9B  BURN-UP CALCULATION ZONE 1  IRRADIATION 10Y
  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6  5.58-6
      1.      2.      3.      4.      5.      6.      7.      8.      9.     10.
1.688THM CHARGED TO ZONE 1              3.155E7 Y
     1.-10     1.-10     1.-10     1.-10     1.-10       .1        .1
92235 99.57    92238 5870.    94239 1018.    94240 92.08    94241 8.548        2
94242 2.975                                                                     2
```

```
***** EINGABEBEISPIEL 5 *****  [Input Example 5]
BASIC NUCLEAR DATA FOR LWR                                                  2    1
     .478      .320      2.12    1.E-24 71680 1 0 0-1
AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-7005  50P5U   ORNL 80     27         1

   10   10    0    0    0   10    0    0    0    6    0    0   10

                    1

1ST IRRADIATION PERIOD 821.5D  3.5% U 235
   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52   36.52
     10.     100.    190.    280.    370.    460.    550.    640.    730.   821.5
1MT OF HM CHARGED TO REACTOR             86400. D
                1.E-6
92234 1.141    92235 148.94   92238 4053.5                                      2

BASIC NUCLEAR DATA FOR LWR                                                  2    1
     .4788      .358      2.80    1.E-24       1 0 0-1
AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-7005  50P5U   ORNL 80     27         1

    4    4    0    0   10    4    0    0    0   -1    0    0    4
2ND IRRADIATION PERIOD  219.1D
   36.52   36.52   36.52   36.52
   871.5   921.5   971.5  1040.6
1MT OF HM CHARGED TO REACTOR             86400. D      821.5
BASIC NUCLEAR DATA FOR LWR                                                  2    1
     .4775      .447      3.12    1.E-24       1 0 0-1
AVERAGED CONSTANT X-SECTIONS FROM ORNL/TM-7005  50P5U   ORNL 80     27         1

    2    7    0    0    4    2   -1    0    0    6

                    1
                    1
3RD IRRADIATION PERIOD  109.5 D  AND SUBSEQUENT DECAY
   36.52   36.52
  1090.6  1150.1    200.    400.    600.    800.   1000.
1MT OF HM CHARGED TO REACTOR             86400. D     1040.6
```

```
***** EINGABEBEISPIEL 6 *****  [Input Example 6]
BASIC NUCLEAR DATA FOR LWR                                                  2    1
      .701      .304      2.01     1.E-24
    3    3    0    1    0    1    1    0    0    2

          111

IRRADIATION 1
  5.E+13  5.E+13  5.E+13
    100.    200.    300.
1 MT HM                                  86400. D
      1.-7
92234 1.24     92235 136.     92238 4070.                                    2

    2    2    0    1    3    1    1    0    0   -1
IRRADIATION 2
  5.E+13  5.E+13
    400.    500.
1 MT HM                                  86400. D       300.
    0    3    0    1    2    1    1    0    0   -1
FUEL DECAY
      1.      2.      3.
1MT HM                                  3.155E7 Y
    0    4    0    1   -3    1   -1    0    0   -1
WASTE DECAY
    100.    200.    300.    400.
1MT HM                                  3.155E7 Y         3.
    92            .01
    94            .01
```

```
***** EINGABEBEISPIEL 7 *****  [Input Example 7]
BASIC NUCLEAR DATA LIBRARY FOR LWR                                          2    1
      .701      .304      2.01     1.E-24          -1
AVERAGED CONSTANT X-SECTIONS FROM KFK-3014      33P3U   KFK 80      27        0
551330  SCAPT                                                     1.2E+01

    5    5    0    0    0    5   -1    0    0    7    0    1

                   1
                   1
    7
    U          PU        AM        CM        CS133     CS134     CS137
    1
# DWR  3.2% U 235  TEILBESTRAHLUNG 1 ZYKLUS
    37.5    37.5    37.5    37.5    37.5
     10.     73.    147.    220.   293.3
** 1T SM REAKTORBELADUNG                   86400. D
                 1.-10
92234 1.239    92235 136.2    92238 4066.                                   2

BURNUP DEPENDENT X-SECTIONS FROM KFK-3014       33P3U 32 KFK BIB    27    0    0
942400    SCAPT
175.2   176.0   170.4   160.7   151.2   143.0   143.0   135.9   131.5   127.3
123.9   121.0   121.0   118.0   116.5   115.0   113.8   112.7
```

```
*****************************************************
*                                                   *
*  K O R I G E N   I N P U T   R E P R O D U C T I O N  *
*                                                   *
*****************************************************

CARD A REFERENCE REACTORS : BIBLIS 3.2% U235 33GWD/THM  10/1982                2  0 1
CARD B    0.70100   0.30400   2.01000  1.00E-25 6 880 0 0 0-1
CARD U1BIBLIS 3.2% U235   33GWD/THM                    33P3U   ORNL 78     27         0
CARD U2     0
-------------------------------------------------------------------------------
CARD C     9    9    0    0    0    9    1    0    0    1    0    1    0    0
CARD D IRRADIATION  BIBLIS  3.2% U-235 33 GWD/THM
CARD E 3.30E+013.30E+013.30E+013.30E+010.0     3.30E+013.30E+013.30E+013.30E+01
CARD G 2.00E+011.11E+022.22E+023.33E+023.73E+023.93E+024.84E+025.95E+027.06E+02
CARD H 1 THM CHARGED TO REACTOR                8.640E+04 D
CARD I  1.000E-04 1.000E-03 1.000E-03 1.000E-03 1.000E-03 1.000E+06 1.000E+06 1.000E-03
CARD K  50000 9.35E-01 60000 1.11E+02 80000 1.36E+05130000 1.19E+03140000 1.08E+03    5
CARD K 150000 4.88E+01160000 3.29E+01220000 8.47E+02240000 2.34E+04250000 2.10E+03    5
CARD K 260000 7.55E+04270000 1.20E+02280000 2.56E+04290000 9.70E+00400000 2.89E+05    5
CARD K 410000 5.09E+02420000 5.05E+02500000 4.27E+03730000 4.09E+02     0 0.0         5
CARD K 922340 1.04E+00922350 1.36E+02922380 4.07E+03     0 0.0          0 0.0         2
CARD K      0 0.0          0 0.0          0 0.0          0 0.0          0 0.0         0
CARD T1BIBLIS 3.2% U235 33GWD/THM                    33P3U 32 KFK BIB    27    0    0
CARD T5     0
-------------------------------------------------------------------------------
CARD O     5    5    0    0    9    5    1    0    0   -1    0    1    0    0
CARD D IRRADIATION  BIBLIS  3.2% U-235 33 GWD/THM
CARD E 0.0     3.30E+013.30E+013.30E+013.30E+01
CARD G 7.46E+027.66E+028.57E+029.68E+021.08E+03
CARD H'1 THM CHARGED TO REACTOR                8.640E+04 D    706.000
```

```
CARD O      0    4    0    0    5    1    1    0    0    3    0    0    0    0

CARD S1           000       110       100       100       100       000       000
                  000       110       200       100       100       000       000
                  000       100       100       100       100       000       000

CARD S2     0
            1

CARD D FUEL DECAY  BIBLIS-TYPE 3.2% U235

CARD G 1.00E+002.00E+005.00E+007.00E+00

CARD H'1 THM CHARGED TO REACTOR                3.155E+07 Y      0.0

CARD O      0   10    0    0   -4    1   -1    0    0   -1    0    0    0    0

CARD D WASTE DECAY  BIBLIS-TYPE 3.2% U235  REPROCESSING  7 Y  AFTER DISCHARGE

CARD G 1.00E+012.00E+015.00E+011.00E+022.00E+021.00E+031.00E+041.00E+051.00E+061.00E+07

CARD H'1 THM CHARGED TO REACTOR                3.155E+07 Y      7.000

CARD P      92      1.000E-02

CARD P      94      1.000E-02

CARD P       0      0.0

CARD C      0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

C. The Data File KORIDATA

C.1 Contents of the KORIDATA Data File

10010	SCAPT	3.47E-02	2.89E-02	1.61E-02	3.42E-02	3.09E-02	2.31E-02	3.47E-02	2.81E-02
50100	SIGNA	4.02E+02	3.35E+02	1.86E+02	3.64E+02	3.28E+02	2.36E+02	4.02E+02	3.25E+02
50110	SCAPT	5.35E-04	4.48E-04	2.56E-04	4.87E-04	4.41E-04	3.21E-04	5.35E-04	4.36E-04
60120	SCAPT	3.11E-04	2.74E-04	1.44E-04	3.21E-04	2.89E-04	2.07E-04	3.11E-04	2.64E-04
70140	SIGNP	1.81E-01	1.62E-01	9.34E-02	1.85E-01	1.69E-01	1.26E-01	1.81E-01	1.56E-01
80160	SIGNA	2.72E-03	2.75E-03	3.13E-03	2.04E-03	2.06E-03	2.25E-03	2.72E-03	2.69E-03
110230	SCAPT	5.28E-02	4.73E-02	2.70E-02	5.47E-02	4.98E-02	3.71E-02	5.28E-02	4.56E-02
240520	SCAPT	3.35E-01	2.81E-01	1.62E-01	3.06E-01	2.77E-01	2.03E-01	3.35E-01	1.85E-01
250550	SCAPT	1.52E+00	1.39E+00	9.04E-01	1.57E+00	1.46E+00	1.15E+00	1.52E+00	1.35E+00
260560	SCAPT	2.68E-01	2.29E-01	1.31E-01	2.73E-01	2.48E-01	1.88E-01	2.68E-01	2.23E-01
280580	SCAPT	4.87E-01	4.07E-01	2.31E-01	4.43E-01	4.01E-01	2.91E-01	4.87E-01	3.60E-01
380880	SCAPT	1.24E-03	1.20E-03	1.05E-03	1.23E-03	1.21E-03	1.11E-03	1.24E-03	6.92E-03
380890	SCAPT	5.26E-02	5.03E-02	3.53E-02	5.28E-02	5.31E-02	4.39E-02	5.26E-02	5.53E-02
380900	SCAPT	8.74E-02	8.15E-02	4.72E-02	8.68E-02	8.71E-02	6.58E-02	8.74E-02	8.51E-02
390900	SCAPT	4.33E-01	4.15E-01	2.89E-01	4.36E-01	4.40E-01	3.63E-01	4.33E-01	4.11E-01
390910	SCAPT	1.65E-01	1.57E-01	1.06E-01	1.66E-01	1.67E-01	1.36E-01	1.65E-01	1.60E-01
400900	SCAPT	2.51E-02	2.47E-02	2.23E-02	2.54E-02	2.55E-02	2.40E-02	2.51E-02	3.02E-02
400910	SCAPT	2.75E-01	2.77E-01	2.52E-01	2.81E-01	2.88E-01	2.73E-01	2.75E-01	2.80E-01
400920	SCAPT	5.43E-02	5.36E-02	4.62E-02	5.56E-02	5.62E-02	5.18E-02	5.43E-02	5.87E-02
400930	SCAPT	1.03E+00	1.07E+00	1.03E+00	1.06E+00	1.11E+00	1.09E+00	1.03E+00	1.06E+00
400940	SCAPT	1.94E-02	1.94E-02	1.85E-02	2.01E-02	2.03E-02	1.98E-02	1.94E-02	2.51E-02
400950	SCAPT	2.32E-01	2.37E-01	2.33E-01	2.40E-01	2.46E-01	2.45E-01	2.32E-01	2.42E-01
410931	SCAPT	4.32E-01	4.23E-01	4.03E-01	4.32E-01	4.28E-01	4.17E-01	4.32E-01	4.23E-01
410930	SCAPT	4.10E-01	4.17E-01	3.96E-01	4.23E-01	4.35E-01	4.24E-01	4.10E-01	4.14E-01
410950	SCAPT	8.51E-01	8.75E-01	8.72E-01	8.79E-01	9.06E-01	9.08E-01	8.51E-01	8.77E-01
420950	SCAPT	4.22E+00	4.36E+00	3.91E+00	4.30E+00	4.53E+00	4.29E+00	4.22E+00	4.30E+00
420960	SCAPT	6.87E-01	7.12E-01	7.13E-01	7.10E-01	7.35E-01	7.41E-01	6.87E-01	7.12E-01
420970	SCAPT	6.93E-01	7.05E-01	6.55E-01	7.14E-01	7.35E-01	7.08E-01	6.93E-01	7.05E-01
420980	SCAPT	2.37E-01	2.46E-01	2.56E-01	2.47E-01	2.54E-01	2.62E-01	2.37E-01	2.51E-01
420990	SCAPT	1.01E+00	1.05E+00	1.04E+00	1.05E+00	1.08E+00	1.09E+00	1.01E+00	1.04E+00
430990	SCAPT	9.14E+00	9.83E+00	9.45E+00	9.30E+00	1.03E+01	1.01E+01	9.14E+00	9.70E+00
441010	SCAPT	2.95E+00	3.08E+00	3.04E+00	3.05E+00	3.18E+00	3.19E+00	2.95E+00	3.05E+00
481130	SCAPT	4.04E+03	3.67E+03	1.73E+03	3.99E+03	3.93E+03	2.73E+03	4.04E+03	3.54E+03
481150	SCAPT	8.22E+00	8.35E+00	7.46E+00	8.42E+00	8.70E+00	8.21E+00	8.22E+00	8.23E+00
491130	SCAPT	7.03E+00	7.22E+00	6.86E+00	7.22E+00	7.47E+00	7.31E+00	7.03E+00	7.09E+00
501150	SCAPT	4.55E+00	4.23E+00	2.29E+00	4.51E+00	4.53E+00	3.33E+00	4.55E+00	4.08E+00
501170	SCAPT	7.62E-01	7.80E-01	7.09E-01	7.80E-01	8.12E-01	7.73E-01	7.62E-01	7.76E-01
501180	SCAPT	2.27E-01	2.36E-01	2.49E-01	2.33E-01	2.40E-01	2.50E-01	2.27E-01	2.42E-01
501191	SCAPT	1.09E-01	1.03E-01	9.26E-02	1.15E-01	1.10E-01	9.78E-02	1.09E-01	1.01E-01
501190	SCAPT	3.02E-01	2.92E-01	2.10E-01	3.04E-01	3.09E-01	2.59E-01	3.02E-01	2.91E-01
501200	SCAPT	5.87E-02	5.95E-02	5.79E-02	6.04E-02	6.14E-02	6.06E-02	5.87E-02	6.50E-02
501230	SCAPT	1.08E-01	1.12E-01	1.20E-01	1.13E-01	1.15E-01	1.20E-01	1.08E-01	1.18E-01
501250	SCAPT	5.44E-01	5.65E-01	5.83E-01	5.71E-01	5.88E-01	6.03E-01	5.44E-01	5.70E-01
511210	SCAPT	5.44E+00	5.81E+00	5.75E+00	5.58E+00	6.02E+00	6.07E+00	5.44E+00	5.74E+00
511230	SCAPT	3.34E+00	3.64E+00	3.62E+00	3.43E+00	3.80E+00	3.83E+00	3.34E+00	3.58E+00
511240	SCAPT	1.37E+00	1.37E+00	1.17E+00	1.41E+00	1.44E+00	1.32E+00	1.37E+00	1.35E+00
511250	SCAPT	6.91E-01	7.15E-01	7.21E-01	7.15E-01	7.38E-01	7.46E-01	6.91E-01	7.18E-01
511260	SCAPT	1.87E+00	1.91E+00	1.76E+00	1.92E+00	1.98E+00	1.91E+00	1.87E+00	1.89E+00
521240	SCAPT	8.36E-01	7.98E-01	5.54E-01	8.39E-01	8.44E-01	6.94E-01	8.36E-01	7.85E-01
521250	SCAPT	9.00E-19	2.24E-19	2.20E-19	3.30E-19	3.56E-19	6.60E-19	9.00E-19	2.24E-01
521260	SCAPT	4.27E-01	4.35E-01	4.19E-01	4.38E-01	4.49E-01	4.42E-01	4.27E-01	4.37E-01
902300	SCAPT	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.33E+01
902300	SFISS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.91E-02
902320	SCAPT	3.05E+00	3.08E+00	2.93E+00	3.20E+00	3.17E+00	3.11E+00	3.05E+00	2.93E+00
902320	SFISS	2.22E-02	2.24E-02	2.51E-02	1.79E-02	1.81E-02	1.96E-02	2.22E-02	2.36E-02
912310	SCAPT	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.44E+01
912310	SFISS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.77E-01

912330	SCAPT	1.23E+011	.24E+011	.13E+011	.28E+011	.28E+011	.20E+011	.23E+011	.22E+01
912330	SFISS	1.46E-011	.47E-011	.65E-011	.19E-011	.20E-011	.30E-011	.46E-011	.47E-01
912330	SCAPT1	1.23E+011	.24E+011	.13E+011	.28E+011	.28E+011	.20E+011	.23E+011	.22E+01
922320	SCAPT	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.04E+01
922320	SFISS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.58E+01
922330	SCAPT	7.58E+007	.10E+005	.08E+007	.84E+007	.41E+006	.13E+007	.58E+006	.88E+00
922330	SFISS	6.23E+015	.71E+013	.68E+016	.44E+015	.98E+014	.70E+016	.23E+015	.52E+01
922340	SCAPT	1.92E+011	.99E+011	.72E+012	.09E+012	.04E+011	.88E+011	.92E+012	.06E+01
922340	SFISS	4.50E-014	.42E-014	.91E-013	.97E-014	.02E-014	.30E-014	.50E-014	.63E-01
922350	SCAPT	1.05E+019	.96E+006	.40E+001	.12E+011	.03E+018	.13E+001	.05E+019	.51E+00
922350	SFISS	4.67E+014	.32E+012	.27E+015	.00E+014	.49E+013	.20E+014	.67E+014	.10E+01
922360	SCAPT	7.54E+008	.28E+008	.49E+008	.25E+008	.38E+008	.62E+007	.54E+007	.61E+00
922360	SFISS	1.97E-011	.94E-012	.17E-011	.65E-011	.67E-011	.79E-011	.97E-012	.02E-01
922370	SCAPT	4.42E+013	.95E+012	.25E+014	.58E+014	.15E+013	.10E+014	.42E+013	.78E+01
922370	SFISS	2.32E-012	.14E-011	.44E-012	.41E-012	.24E-011	.81E-012	.32E-016	.14E-01
922380	SCAPT	9.02E-019	.37E-018	.72E-019	.19E-019	.03E-018	.72E-019	.02E-019	.22E-01
922380	SFISS	1.00E-019	.88E-021	.10E-018	.08E-028	.18E-028	.81E-021	.00E-019	.87E-02
922390	SCAPT	3.72E+003	.53E+002	.65E+003	.87E+003	.68E+003	.13E+003	.72E+000	.0
922390	SFISS	1.54E+001	.39E+008	.56E-011	.57E+001	.43E+001	.10E+001	.54E+000	.0
922400	SCAPT	5.45E-015	.41E-014	.97E-015	.68E-015	.60E-015	.34E-015	.45E-010	.0
922400	SFISS	7.29E-027	.36E-028	.25E-025	.91E-025	.98E-026	.45E-027	.29E-020	.0
932370	SCAPT	3.21E+013	.20E+012	.42E+013	.41E+013	.30E+012	.80E+013	.21E+013	.10E+01
932370	SFISS	5.24E-015	.15E-015	.71E-014	.61E-014	.68E-014	.99E-015	.24E-015	.21E-01
932380	SCAPT	1.79E+011	.58E+018	.42E+001	.85E+011	.67E+011	.20E+011	.79E+011	.46E+01
932380	SFISS	1.78E+021	.57E+028	.29E+011	.84E+021	.65E+021	.19E+021	.78E+021	.44E+02
942360	SCAPT	2.04E+011	.89E+011	.31E+012	.11E+011	.97E+011	.62E+012	.04E+011	.84E+01
942360	SFISS	2.06E+011	.91E+011	.34E+012	.14E+012	.00E+011	.65E+012	.06E+011	.94E+01
942380	SCAPT	3.47E+013	.15E+011	.53E+013	.73E+013	.29E+012	.26E+013	.47E+012	.98E+01
942380	SFISS	2.46E+002	.38E+002	.03E+002	.49E+002	.38E+002	.16E+002	.46E+002	.20E+00
942390	SCAPT	5.86E+015	.73E+012	.60E+016	.31E+015	.92E+013	.85E+015	.86E+015	.78E+01
942390	SFISS	1.06E+021	.02E+024	.64E+011	.14E+021	.06E+026	.92E+011	.06E+021	.02E+02
942400	SCAPT	1.04E+021	.37E+024	.39E+011	.11E+021	.36E+025	.26E+011	.04E+021	.43E+02
942400	SFISS	5.84E-015	.84E-016	.22E-015	.27E-015	.39E-015	.54E-015	.84E-015	.89E-01
942410	SCAPT	3.87E+013	.63E+011	.67E+014	.15E+013	.78E+012	.50E+013	.87E+013	.54E+01
942410	SFISS	1.18E+021	.11E+025	.45E+011	.26E+021	.15E+027	.88E+011	.18E+021	.07E+02
942420	SCAPT	2.94E+013	.02E+012	.39E+013	.05E+013	.10E+012	.45E+012	.94E+013	.15E+01
942420	SFISS	4.58E-014	.50E-015	.00E-013	.97E-014	.02E-014	.31E-014	.58E-014	.28E-01
942430	SCAPT	2.59E+012	.31E+011	.31E+012	.68E+012	.43E+011	.81E+012	.59E+011	.26E+01
942430	SFISS	2.09E+011	.91E+011	.24E+012	.17E+012	.00E+011	.59E+012	.09E+012	.60E+01
952410	SCAPT	9.92E+019	.96E+015	.76E+011	.07E+021	.03E+027	.36E+019	.92E+018	.33E+01
952410	SFISS	1.32E+001	.33E+001	.08E+001	.31E+001	.30E+001	.12E+001	.32E+001	.06E+00
952410	SCAPT1	1.23E+011	.23E+017	.12E+001	.32E+011	.27E+019	.10E+001	.23E+012	.08E+01
952421	SCAPT	1.55E+021	.34E+026	.41E+011	.59E+021	.42E+029	.80E+011	.55E+028	.32E+01
952421	SFISS	7.70E+026	.71E+023	.31E+027	.92E+027	.09E+024	.96E+027	.70E+023	.98E+02
952420	SCAPT	1.89E+011	.72E+018	.07E+002	.02E+011	.80E+011	.22E+011	.89E+011	.76E+01
952420	SFISS	1.71E+021	.55E+027	.32E+011	.83E+021	.62E+021	.10E+021	.71E+021	.58E+02
952430	SCAPT	3.80E+013	.94E+013	.05E+014	.04E+014	.04E+013	.28E+013	.80E+014	.97E+01
952430	SFISS	3.57E-013	.51E-013	.91E-013	.04E-013	.08E-013	.30E-013	.57E-014	.09E-01
962420	SCAPT	5.48E+005	.58E+005	.05E+005	.81E+005	.72E+005	.44E+005	.48E+005	.67E+00
962420	SFISS	2.20E-011	.99E-018	.82E-022	.36E-012	.09E-011	.38E-012	.20E-015	.32E-01
962430	SCAPT	9.92E+009	.71E+007	.16E+001	.05E+011	.00E+018	.34E+009	.92E+008	.07E+00
962430	SFISS	9.76E+019	.45E+016	.43E+011	.04E+029	.78E+017	.78E+019	.76E+016	.82E+01
962440	SCAPT	3.44E+003	.49E+003	.17E+003	.65E+003	.59E+003	.41E+003	.44E+001	.38E+01
962440	SFISS	8.53E-018	.43E-019	.02E-017	.79E-017	.84E-018	.18E-018	.53E-018	.88E-01
962450	SCAPT	1.84E+021	.59E+027	.48E+011	.90E+021	.68E+021	.16E+021	.84E+022	.49E+01
962450	SFISS	1.72E+021	.51E+027	.89E+011	.77E+021	.59E+021	.14E+021	.72E+021	.47E+02
962460	SCAPT	1.32E+001	.34E+001	.34E+001	.37E+001	.38E+001	.39E+001	.32E+002	.92E+00

962460	SFISS	4.90E-01	4.94E-01	5.46E-01	4.27E-01	4.31E-01	4.60E-01	4.90E-01	5.90E-01
962470	SCAPT	1.33E+01	1.19E+01	7.03E+00	1.37E+01	1.25E+01	9.49E+00	1.33E+01	1.58E+01
962470	SFISS	9.36E+00	8.66E+00	5.94E+00	9.69E+00	9.05E+00	7.38E+00	9.36E+00	2.44E+01
962480	SCAPT	4.02E+00	4.13E+00	4.21E+00	4.21E+00	4.24E+00	4.34E+00	4.02E+00	6.35E+00
962480	SFISS	5.70E-01	5.75E-01	6.28E-01	5.26E-01	5.32E-01	5.62E-01	5.70E-01	7.74E-01
972490	SCAPT	1.75E+02	1.59E+02	9.75E+01	1.81E+02	1.67E+02	1.28E+02	1.75E+02	2.11E+02
972490	SFISS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.95E-01
982490	SCAPT	2.17E+02	1.98E+02	1.15E+02	2.25E+02	2.07E+02	1.55E+02	2.17E+02	4.47E+01
982490	SFISS	1.74E+02	1.59E+02	9.56E+01	1.80E+02	1.67E+02	1.26E+02	1.74E+02	1.36E+02
982500	SCAPT	1.48E+02	1.30E+02	6.48E+01	1.53E+02	1.37E+02	9.64E+01	1.48E+02	4.25E+02
982500	SFISS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.19E-01
982510	SCAPT	6.32E+02	5.48E+02	2.58E+02	6.53E+02	5.79E+02	3.99E+02	6.32E+02	2.26E+02
982510	SFISS	4.06E+02	3.54E+02	1.71E+02	4.19E+02	3.74E+02	2.60E+02	4.06E+02	4.82E+02
982520	SCAPT	5.11E+00	4.56E+00	2.62E+00	5.29E+00	4.80E+00	3.59E+00	5.11E+00	2.70E+00
982520	SFISS	4.34E+00	4.06E+00	2.93E+00	4.51E+00	4.24E+00	3.55E+00	4.34E+00	5.96E+00
982530	SCAPT	9.19E+01	7.96E+01	3.77E+01	9.49E+01	8.42E+01	5.81E+01	9.19E+01	3.43E+01
982530	SFISS	1.13E+02	1.01E+02	5.92E+01	1.17E+02	1.06E+02	8.01E+01	1.13E+02	1.14E+02
982540	SCAPT	9.20E+00	8.13E+00	4.38E+00	9.51E+00	8.57E+00	6.23E+00	9.20E+00	0.0
992530	SCAPT	2.02E+01	1.96E+01	1.61E+01	2.10E+01	2.04E+01	1.83E+01	2.02E+01	1.14E+02
992530	SCAPT1	1.40E+01	1.37E+01	1.12E+01	1.46E+01	1.41E+01	1.27E+01	1.40E+01	7.91E+01
320720	SCAPT	1.23E-01	1.17E-01	8.23E-02	1.23E-01	1.24E-01	1.02E-01	1.23E-01	1.20E-01
320730	SCAPT	3.27E+00	3.30E+00	2.85E+00	3.33E+00	3.44E+00	3.18E+00	3.27E+00	3.25E+00
320740	SCAPT	5.51E-02	5.30E-02	4.00E-02	5.56E-02	5.59E-02	4.79E-02	5.51E-02	5.78E-02
320760	SCAPT	5.65E-02	5.76E-02	5.57E-02	5.86E-02	6.02E-02	5.92E-02	5.65E-02	6.30E-02
330750	SCAPT	2.15E+00	2.24E+00	2.16E+00	2.21E+00	2.31E+00	2.29E+00	2.15E+00	2.22E+00
340760	SCAPT	7.88E+00	7.33E+00	4.05E+00	7.81E+00	7.84E+00	5.81E+00	7.88E+00	7.08E+00
340770	SCAPT	4.38E+00	4.14E+00	2.55E+00	4.38E+00	4.41E+00	3.43E+00	4.38E+00	4.02E+00
340780	SCAPT	1.85E-01	1.90E-01	1.86E-01	1.90E-01	1.95E-01	1.93E-01	1.85E-01	1.94E-01
340800	SCAPT	8.99E-02	8.69E-02	6.63E-02	9.10E-02	9.18E-02	7.91E-02	8.99E-02	9.11E-02
340820	SCAPT	8.62E-03	8.35E-03	7.04E-03	8.71E-03	8.69E-03	7.88E-03	8.62E-03	1.40E-02
350790	SCAPT	4.80E+00	4.97E+00	4.78E+00	4.96E+00	5.16E+00	5.09E+00	4.80E+00	4.89E+00
350810	SCAPT	1.79E+00	1.85E+00	1.85E+00	1.85E+00	1.91E+00	1.93E+00	1.79E+00	1.84E+00
360800	SCAPT	2.82E+00	2.87E+00	2.42E+00	2.86E+00	3.00E+00	2.74E+00	2.82E+00	2.82E+00
360820	SCAPT	7.25E+00	7.44E+00	6.43E+00	7.36E+00	7.75E+00	7.18E+00	7.25E+00	7.30E+00
360830	SCAPT	2.13E+01	2.01E+01	1.25E+01	2.13E+01	2.14E+01	1.68E+01	2.13E+01	1.94E+01
360840	SCAPT	1.25E-01	1.30E-01	1.37E-01	1.31E-01	1.35E-01	1.40E-01	1.25E-01	1.31E-01
360850	SCAPT	1.84E-01	1.74E-01	1.13E-01	1.84E-01	1.86E-01	1.48E-01	1.84E-01	1.76E-01
360860	SCAPT	1.01E-02	9.85E-03	7.87E-03	1.03E-02	1.04E-02	9.18E-03	1.01E-02	1.03E-02
370850	SCAPT	2.52E-01	2.57E-01	2.57E-01	2.62E-01	2.68E-01	2.69E-01	2.52E-01	2.62E-01
370860	SCAPT	1.14E+00	1.14E+00	1.01E+00	1.16E+00	1.19E+00	1.11E+00	1.14E+00	1.13E+00
370870	SCAPT	7.69E-02	7.94E-02	7.97E-02	7.88E-02	8.12E-02	8.18E-02	7.69E-02	8.47E-02
380860	SCAPT	3.91E-01	3.79E-01	2.80E-01	3.96E-01	4.01E-01	3.40E-01	3.91E-01	3.76E-01
380870	SCAPT	4.30E+00	4.25E+00	3.65E+00	4.37E+00	4.44E+00	4.10E+00	4.30E+00	4.14E+00
380880	SCAPT	1.24E-03	1.20E-03	1.05E-03	1.23E-03	1.21E-03	1.11E-03	1.24E-03	6.92E-03
380890	SCAPT	5.26E-02	5.03E-02	3.53E-02	5.28E-02	5.31E-02	4.39E-02	5.26E-02	5.53E-02
380900	SCAPT	8.74E-02	8.15E-02	4.72E-02	8.68E-02	8.71E-02	6.58E-02	8.74E-02	8.51E-02
390890	SCAPT	1.33E-01	1.25E-01	7.72E-02	1.33E-01	1.33E-01	1.04E-01	1.33E-01	1.27E-01
390900	SCAPT	4.33E-01	4.15E-01	2.89E-01	4.36E-01	4.40E-01	3.63E-01	4.33E-01	4.11E-01
390910	SCAPT	1.65E-01	1.57E-01	1.06E-01	1.66E-01	1.67E-01	1.36E-01	1.65E-01	1.60E-01
400900	SCAPT	2.51E-02	2.47E-02	2.23E-02	2.54E-02	2.55E-02	2.40E-02	2.51E-02	3.02E-02
400910	SCAPT	2.75E-01	2.77E-01	2.52E-01	2.81E-01	2.88E-01	2.73E-01	2.75E-01	2.80E-01
400920	SCAPT	5.43E-02	5.36E-02	4.62E-02	5.56E-02	5.62E-02	5.18E-02	5.43E-02	5.87E-02
400930	SCAPT	1.03E+00	1.07E+00	1.03E+00	1.06E+00	1.11E+00	1.09E+00	1.03E+00	1.06E+00
400940	SCAPT	1.94E-02	1.94E-02	1.85E-02	2.01E-02	2.03E-02	1.98E-02	1.94E-02	2.51E-02
400950	SCAPT	2.32E-01	2.37E-01	2.33E-01	2.40E-01	2.46E-01	2.45E-01	2.32E-01	2.42E-01
400960	SCAPT	1.77E-01	1.85E-01	1.98E-01	1.81E-01	1.87E-01	1.96E-01	1.77E-01	1.91E-01
410930	SCAPT	4.10E-01	4.17E-01	3.96E-01	4.23E-01	4.35E-01	4.24E-01	4.10E-01	4.14E-01

410931	SCAPT	4.32E-014.23E-014.03E-014.32E-014.28E-014.17E-014.32E-014.23E-01
410950	SCAPT	8.51E-018.75E-018.72E-018.79E-019.06E-019.08E-018.51E-018.77E-01
420950	SCAPT	4.22E+004.36E+003.91E+004.30E+004.53E+004.29E+004.22E+004.30E+00
420960	SCAPT	6.87E-017.12E-017.13E-017.10E-017.35E-017.41E-016.87E-017.12E-01
420970	SCAPT	6.93E-017.05E-016.55E-017.14E-017.35E-017.08E-016.93E-017.05E-01
420980	SCAPT	2.37E-012.46E-012.56E-012.47E-012.54E-012.62E-012.37E-012.51E-01
420990	SCAPT	1.01E+001.05E+001.04E+001.05E+001.08E+001.09E+001.01E+001.04E+00
421000	SCAPT	1.48E-011.53E-011.55E-011.53E-011.57E-011.60E-011.48E-011.58E-01
430990	SCAPT	9.14E+009.83E+009.45E+009.30E+001.03E+011.01E+019.14E+009.70E+00
440990	SCAPT	4.31E+004.45E+004.44E+004.46E+004.60E+004.65E+004.31E+004.37E+00
441000	SCAPT	7.80E-017.53E-015.48E-017.87E-017.96E-016.71E-017.80E-017.42E-01
441010	SCAPT	2.95E+003.08E+003.04E+003.05E+003.18E+003.19E+002.95E+003.05E+00
441020	SCAPT	2.62E-012.58E-012.20E-012.66E-012.69E-012.46E-012.62E-012.61E-01
441030	SCAPT	2.67E+002.74E+002.55E+002.74E+002.85E+002.75E+002.67E+002.70E+00
441040	SCAPT	2.62E-012.69E-012.69E-012.71E-012.78E-012.80E-012.62E-012.73E-01
441050	SCAPT	2.91E-013.00E-013.15E-013.05E-013.12E-013.22E-012.91E-013.07E-01
441060	SCAPT	8.90E-029.10E-029.19E-029.29E-029.51E-029.59E-028.90E-029.72E-02
451030	SCAPT	3.68E+013.75E+012.69E+013.71E+013.93E+013.17E+013.68E+013.67E+01
451050	SCAPT	1.02E+039.80E+025.97E+021.01E+031.04E+037.98E+021.02E+039.47E+02
461040	SCAPT	6.46E-016.71E-017.02E-016.70E-016.89E-017.12E-016.46E-016.74E-01
461050	SCAPT	3.83E+003.85E+003.42E+003.92E+004.02E+003.79E+003.83E+003.79E+00
461060	SCAPT	2.75E-012.84E-012.95E-012.83E-012.91E-012.99E-012.75E-012.89E-01
461070	SCAPT	2.82E+002.87E+002.59E+002.88E+002.99E+002.84E+002.82E+002.83E+00
461080	SCAPT	7.09E+007.42E+007.29E+007.35E+007.70E+007.70E+007.09E+007.28E+00
461100	SCAPT	2.59E-012.69E-012.80E-012.70E-012.78E-012.86E-012.59E-012.75E-01
471070	SCAPT	6.25E+006.16E+004.83E+006.33E+006.47E+005.68E+006.25E+006.02E+00
471090	SCAPT	3.89E+014.12E+013.88E+013.95E+014.29E+014.19E+013.89E+014.06E+01
471110	SCAPT	3.46E+003.61E+003.70E+003.59E+003.73E+003.81E+003.46E+003.60E+00
481080	SCAPT	2.64E-012.62E-012.34E-012.69E-012.71E-012.54E-012.64E-012.66E-01
481100	SCAPT	2.24E+002.21E+001.85E+002.28E+002.31E+002.10E+002.24E+002.18E+00
481110	SCAPT	3.55E+003.46E+002.60E+003.59E+003.64E+003.13E+003.55E+003.38E+00
481120	SCAPT	6.33E-016.41E-015.84E-016.49E-016.66E-016.34E-016.33E-016.40E-01
481130	SCAPT	4.04E+033.67E+031.73E+033.99E+033.93E+032.73E+034.04E+033.54E+03
481140	SCAPT	6.45E-016.83E-017.16E-016.68E-017.03E-017.28E-016.45E-016.87E-01
481150	SCAPT	8.22E+008.35E+007.46E+008.42E+008.70E+008.21E+008.22E+008.23E+00
481160	SCAPT	1.05E-011.07E-011.13E-011.09E-011.11E-011.14E-011.05E-011.13E-01
491130	SCAPT	7.03E+007.22E+006.86E+007.22E+007.47E+007.31E+007.03E+007.09E+00
491150	SCAPT	9.61E+019.78E+018.21E+019.79E+011.02E+029.02E+019.61E+019.56E+01
501150	SCAPT	4.55E+004.23E+002.29E+004.51E+004.53E+003.33E+004.55E+004.08E+00
501160	SCAPT	3.45E-013.74E-013.92E-013.57E-013.85E-014.00E-013.45E-013.78E-01
501170	SCAPT	7.62E-017.80E-017.09E-017.80E-018.12E-017.73E-017.62E-017.76E-01
501180	SCAPT	2.27E-012.36E-012.49E-012.33E-012.40E-012.50E-012.27E-012.42E-01
501190	SCAPT	3.02E-012.92E-012.10E-013.04E-013.09E-012.59E-013.02E-012.91E-01
501191	SCAPT	1.09E-011.03E-019.26E-021.15E-011.10E-019.78E-021.09E-011.01E-01
501200	SCAPT	5.87E-025.95E-025.79E-026.04E-026.14E-026.06E-025.87E-026.50E-02
501220	SCAPT	4.22E-024.22E-023.73E-024.29E-024.39E-024.10E-024.22E-024.75E-02
501230	SCAPT	1.08E-011.12E-011.20E-011.13E-011.15E-011.20E-011.08E-011.18E-01
501240	SCAPT	2.22E-012.35E-012.39E-012.30E-012.42E-012.47E-012.22E-012.39E-01
501250	SCAPT	5.44E-015.65E-015.83E-015.71E-015.88E-016.03E-015.44E-015.70E-01
501260	SCAPT	3.08E-022.88E-021.76E-023.06E-023.06E-022.36E-023.08E-023.42E-02
511210	SCAPT	5.44E+005.81E+005.75E+005.58E+006.02E+006.07E+005.44E+005.74E+00
511230	SCAPT	3.34E+003.64E+003.62E+003.43E+003.80E+003.83E+003.34E+003.58E+00
511240	SCAPT	1.37E+001.37E+001.17E+001.41E+001.44E+001.32E+001.37E+001.35E+00
511250	SCAPT	6.91E-017.15E-017.21E-017.15E-017.38E-017.46E-016.91E-017.18E-01
511260	SCAPT	1.87E+001.91E+001.76E+001.92E+001.98E+001.91E+001.87E+001.89E+00
521220	SCAPT	2.42E+002.55E+002.55E+002.51E+002.63E+002.65E+002.42E+002.54E+00
521230	SCAPT	1.73E+021.73E+021.49E+021.76E+021.80E+021.63E+021.73E+021.68E+02

521240	SCAPT	8.36E-017.98E-015.54E-018.39E-018.44E-016.94E-018.36E-017.85E-01
521250	SCAPT	9.00E-019.24E-019.20E-019.30E-019.56E-019.60E-019.00E-019.24E-01
521260	SCAPT	4.27E-014.35E-014.19E-014.38E-014.49E-014.42E-014.27E-014.37E-01
521270	SCAPT	2.05E+002.06E+001.77E+002.09E+002.15E+001.98E+002.05E+002.03E+00
521280	SCAPT	1.15E-011.17E-011.16E-011.18E-011.19E-011.20E-011.15E-011.22E-01
521290	SCAPT	2.90E-012.91E-012.64E-012.98E-013.04E-012.88E-012.90E-012.95E-01
521300	SCAPT	3.58E-023.41E-022.38E-023.60E-023.62E-022.98E-023.58E-023.91E-02
521320	SCAPT	4.89E-044.71E-044.26E-044.75E-044.65E-044.36E-044.89E-046.63E-03
531270	SCAPT	4.85E+005.11E+005.10E+005.01E+005.29E+005.33E+004.85E+005.06E+00
531290	SCAPT	3.22E+003.09E+002.10E+003.24E+003.27E+002.67E+003.22E+003.01E+00
531300	SCAPT	6.66E+006.85E+006.44E+006.85E+007.11E+006.92E+006.66E+006.76E+00
531310	SCAPT	3.23E-013.31E-013.24E-013.34E-013.43E-013.41E-013.23E-013.35E-01
531350	SCAPT	2.12E-031.99E-031.24E-032.09E-032.10E-031.63E-032.12E-038.10E-03
541280	SCAPT	6.54E-016.46E-015.36E-016.64E-016.76E-016.10E-016.54E-016.36E-01
541290	SCAPT	8.45E+008.58E+008.14E+008.69E+008.90E+008.76E+008.45E+008.33E+00
541300	SCAPT	6.26E-015.87E-013.52E-016.24E-016.26E-014.81E-016.26E-015.70E-01
541310	SCAPT	3.05E+013.09E+012.75E+013.12E+013.21E+013.04E+013.05E+013.03E+01
541320	SCAPT	1.02E-011.02E-018.94E-021.05E-011.06E-019.92E-021.02E-011.01E-01
541330	SCAPT	2.44E+012.36E+011.66E+012.45E+012.50E+012.08E+012.44E+012.30E+01
541340	SCAPT	4.42E-024.30E-023.52E-024.49E-024.52E-024.05E-024.42E-024.30E-02
541350	SCAPT	2.45E+052.17E+051.00E+052.41E+052.35E+051.63E+052.45E+052.07E+05
541360	SCAPT	1.66E-021.56E-029.64E-031.66E-021.67E-021.30E-021.66E-021.58E-02
551330	SCAPT	1.07E+011.14E+011.06E+011.09E+011.19E+011.15E+011.07E+011.12E+01
551340	SCAPT	1.68E+011.61E+011.08E+011.68E+011.70E+011.38E+011.68E+011.56E+01
551350	SCAPT	2.39E+002.43E+002.16E+002.45E+002.53E+002.39E+002.39E+002.39E+00
551360	SCAPT	1.34E+001.40E+001.42E+001.39E+001.44E+001.47E+001.34E+001.39E+00
551370	SCAPT	2.56E-022.55E-022.26E-022.63E-022.68E-022.51E-022.56E-023.15E-02
561340	SCAPT	8.53E-018.94E-018.54E-018.76E-019.28E-019.10E-018.53E-018.90E-01
561350	SCAPT	3.48E+003.59E+003.54E+003.60E+003.71E+003.71E+003.48E+003.55E+00
561360	SCAPT	1.02E-011.02E-019.15E-021.05E-011.07E-011.01E-011.02E-011.07E-01
561370	SCAPT	5.51E-015.22E-013.31E-015.50E-015.55E-014.37E-015.51E-015.12E-01
561380	SCAPT	3.56E-023.32E-022.01E-023.52E-023.52E-022.70E-023.56E-023.80E-02
561400	SCAPT	5.28E-015.38E-015.04E-015.39E-015.54E-015.36E-015.28E-015.38E-01
571390	SCAPT	1.04E+001.00E+006.63E-011.05E+001.06E+008.55E-011.04E+009.79E-01
571400	SCAPT	2.21E+002.31E+002.33E+002.29E+002.38E+002.41E+002.21E+002.30E+00
581400	SCAPT	6.31E-025.94E-023.84E-026.27E-026.28E-024.98E-026.31E-026.40E-02
581410	SCAPT	2.97E+002.80E+001.70E+002.96E+002.98E+002.30E+002.97E+002.72E+00
581420	SCAPT	1.06E-011.00E-016.53E-021.06E-011.06E-018.47E-021.06E-011.04E-01
581430	SCAPT	1.75E+001.79E+001.64E+001.80E+001.86E+001.78E+001.75E+001.77E+00
581440	SCAPT	1.49E-011.45E-011.11E-011.51E-011.53E-011.32E-011.49E-011.48E-01
591410	SCAPT	1.51E+001.45E+001.04E+001.51E+001.53E+001.28E+001.51E+001.42E+00
591420	SCAPT	5.59E+005.71E+005.13E+005.72E+005.96E+005.65E+005.59E+005.61E+00
591430	SCAPT	1.21E+011.18E+018.61E+001.22E+011.25E+011.06E+011.21E+011.15E+01
601420	SCAPT	1.71E+001.59E+008.65E-011.69E+001.70E+001.25E+001.71E+001.54E+00
601430	SCAPT	2.87E+012.65E+011.43E+012.84E+012.84E+012.09E+012.87E+012.56E+01
601440	SCAPT	4.71E-014.54E-013.26E-014.73E-014.78E-013.99E-014.71E-014.49E-01
601450	SCAPT	9.35E+009.35E+007.86E+009.48E+009.77E+008.91E+009.35E+009.15E+00
601460	SCAPT	2.39E-012.33E-011.89E-012.41E-012.44E-012.16E-012.39E-012.30E-01
601470	SCAPT	2.01E+012.09E+011.96E+012.05E+012.18E+012.12E+012.01E+012.05E+01
601480	SCAPT	8.67E-018.80E-018.30E-018.86E-019.08E-018.81E-018.67E-018.72E-01
601500	SCAPT	6.43E-016.62E-016.50E-016.65E-016.85E-016.82E-016.43E-016.58E-01
611470	SCAPT	6.19E+016.61E+016.13E+016.28E+016.90E+016.69E+016.19E+016.50E+01
611480	SCAPT	1.17E+031.22E+039.65E+021.19E+031.27E+031.09E+031.17E+031.20E+03
611481	SCAPT	2.92E+032.65E+031.29E+032.89E+032.84E+031.99E+032.92E+032.56E+03
611490	SCAPT	1.32E+021.24E+027.00E+011.32E+021.32E+029.90E+011.32E+021.19E+02
611510	SCAPT	1.03E+021.01E+027.41E+011.04E+021.07E+029.07E+011.03E+029.79E+01
621470	SCAPT	2.39E+012.48E+012.32E+012.45E+012.58E+012.51E+012.39E+012.43E+01

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621480   SCAPT    1.14E+001.16E+001.12E+001.17E+001.20E+001.18E+001.14E+001.16E+00
621490   SCAPT    7.28E+036.44E+033.02E+037.19E+036.94E+034.84E+037.28E+036.16E+03
621500   SCAPT    1.53E+011.54E+011.19E+011.54E+011.63E+011.42E+011.53E+011.50E+01
621510   SCAPT    7.26E+026.75E+023.49E+027.16E+027.30E+025.27E+027.26E+026.45E+02
621520   SCAPT    7.56E+018.22E+017.49E+017.67E+018.67E+018.34E+017.56E+018.02E+01
621530   SCAPT    8.98E+019.34E+018.24E+019.11E+019.78E+019.23E+018.98E+019.12E+01
621540   SCAPT    1.52E+001.52E+001.37E+001.56E+001.59E+001.50E+001.52E+001.51E+00
631510   SCAPT    7.41E+027.11E+024.09E+027.36E+027.58E+025.71E+027.41E+026.92E+02
631520   SCAPT    1.91E+021.89E+021.28E+021.92E+022.00E+021.61E+021.91E+021.83E+02
631540   SCAPT    1.29E+021.27E+028.85E+011.29E+021.34E+021.10E+021.29E+021.23E+02
631550   SCAPT    3.66E+023.39E+021.82E+023.62E+023.63E+022.66E+023.66E+023.27E+02
631560   SCAPT    7.38E+017.23E+015.28E+017.42E+017.62E+016.43E+017.38E+017.01E+01
631570   SCAPT    4.61E+014.64E+013.84E+014.67E+014.85E+014.37E+014.61E+014.52E+01
641540   SCAPT    1.35E+011.33E+011.04E+011.36E+011.41E+011.23E+011.35E+011.30E+01
641550   SCAPT    2.76E+032.48E+031.15E+032.71E+032.69E+031.88E+032.76E+032.36E+03
641560   SCAPT    3.77E+004.00E+004.18E+003.93E+004.13E+004.29E+003.77E+003.95E+00
641570   SCAPT    1.17E+041.05E+044.82E+031.15E+041.14E+047.91E+031.17E+041.00E+04
641580   SCAPT    1.79E+001.93E+001.92E+001.85E+002.00E+002.02E+001.79E+001.90E+00
641600   SCAPT    3.70E-013.77E-013.72E-013.82E-013.90E-013.88E-013.70E-013.81E-01
651590   SCAPT    1.48E+011.53E+011.47E+011.53E+011.58E+011.57E+011.48E+011.50E+01
651600   SCAPT    6.86E+016.63E+014.56E+016.88E+017.02E+015.74E+016.86E+016.43E+01
661600   SCAPT    4.94E+015.10E+014.95E+015.08E+015.28E+015.24E+014.94E+014.98E+01
661610   SCAPT    7.62E+017.35E+015.23E+017.66E+017.77E+016.47E+017.62E+017.14E+01
661620   SCAPT    7.45E+017.95E+017.41E+017.56E+018.29E+018.05E+017.45E+017.83E+01
661630   SCAPT    4.82E+014.83E+014.22E+014.91E+015.03E+014.64E+014.82E+014.73E+01
661640   SCAPT    1.94E+021.77E+028.64E+011.91E+021.91E+021.35E+021.94E+021.70E+02
671650   SCAPT    2.55E+012.59E+012.39E+012.61E+012.70E+012.60E+012.55E+012.54E+01
681660   SCAPT    6.74E+006.70E+005.45E+006.85E+007.01E+006.28E+006.74E+006.57E+00
681670   SCAPT    2.13E+022.15E+021.54E+022.15E+022.25E+021.87E+022.13E+022.11E+02
     0
                                                  33P3U   ORNL 78
    18
0.0      5.71E-011.15E+001.74E+002.33E+002.92E+002.92E+003.46E+003.99E+004.53E+00
5.06E+005.59E+005.59E+006.08E+006.56E+007.04E+007.52E+008.00E+00
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
922340     SCAPT
1.94E+011.93E+011.94E+011.94E+011.94E+011.94E+011.93E+011.93E+011.93E+011.93E+01
1.93E+011.93E+011.93E+011.93E+011.93E+011.93E+011.94E+011.94E+01
922350     SCAPT
1.03E+011.02E+011.02E+011.02E+011.02E+011.02E+011.04E+011.04E+011.03E+011.03E+01
1.03E+011.04E+011.06E+011.06E+011.06E+011.06E+011.06E+011.06E+01
922350     SFISS
4.58E+014.55E+014.53E+014.53E+014.53E+014.53E+014.56E+014.63E+014.61E+014.61E+01
4.61E+014.61E+014.78E+014.75E+014.73E+014.73E+014.73E+014.73E+01
922360     SCAPT
7.69E+007.71E+007.73E+007.75E+007.76E+007.77E+007.57E+007.59E+007.61E+007.63E+00
7.64E+007.65E+007.51E+007.53E+007.55E+007.56E+007.58E+007.59E+00
922380     SCAPT
8.88E-018.88E-018.89E-018.90E-018.90E-018.91E-018.94E-018.94E-018.95E-018.96E-01
8.96E-018.97E-019.03E-019.03E-019.04E-019.04E-019.05E-019.06E-01
932370     SCAPT
3.29E+013.27E+013.26E+013.26E+013.25E+013.26E+013.32E+013.31E+013.30E+013.30E+01
3.29E+013.29E+013.33E+013.32E+013.31E+013.30E+013.30E+013.30E+01
942380     SCAPT
3.31E+013.29E+013.28E+013.27E+013.28E+013.28E+013.42E+013.40E+013.38E+013.38E+01
3.38E+013.38E+013.54E+013.52E+013.50E+013.50E+013.50E+013.50E+01
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942380     SFISS
2.40E+002.40E+002.39E+002.39E+002.39E+002.40E+002.43E+002.43E+002.43E+002.43E+00
2.43E+002.43E+002.47E+002.47E+002.47E+002.46E+002.46E+002.47E+00
942390     SCAPT
6.56E+016.52E+016.49E+016.48E+016.48E+016.49E+016.21E+016.17E+016.14E+016.13E+01
6.13E+016.14E+016.11E+016.06E+016.04E+016.03E+016.03E+016.04E+01
942390     SFISS
1.15E+021.15E+021.14E+021.14E+021.14E+021.14E+021.11E+021.10E+021.10E+021.10E+02
1.10E+021.10E+021.10E+021.10E+021.09E+021.09E+021.09E+021.09E+02
942400     SCAPT
1.85E+021.84E+021.83E+021.81E+021.80E+021.79E+021.30E+021.29E+021.29E+021.28E+02
1.27E+021.26E+021.07E+021.07E+021.06E+021.05E+021.05E+021.04E+02
942410     SCAPT
4.03E+014.00E+013.98E+013.98E+013.98E+013.98E+013.96E+013.93E+013.92E+013.91E+01
3.91E+013.92E+013.99E+013.96E+013.94E+013.94E+013.94E+013.94E+01
942410     SFISS
1.21E+021.21E+021.20E+021.20E+021.20E+021.20E+021.20E+021.19E+021.19E+021.19E+02
1.19E+021.19E+021.21E+021.21E+021.20E+021.20E+021.20E+021.20E+02
942420     SCAPT
2.94E+012.95E+012.96E+012.96E+012.97E+012.97E+012.93E+012.94E+012.95E+012.95E+01
2.96E+012.96E+012.89E+012.90E+012.91E+012.91E+012.92E+012.92E+01
952410     SCAPT
1.09E+021.08E+021.08E+021.08E+021.08E+021.08E+021.05E+021.05E+021.04E+021.04E+02
1.04E+021.04E+021.04E+021.03E+021.03E+021.02E+021.02E+021.02E+02
952410     SCAPT1
1.35E+011.34E+011.33E+011.33E+011.33E+011.33E+011.30E+011.29E+011.29E+011.28E+01
1.28E+011.28E+011.28E+011.27E+011.27E+011.27E+011.26E+011.26E+01
952430     SCAPT
3.83E+013.81E+013.80E+013.78E+013.76E+013.74E+013.89E+013.87E+013.85E+013.83E+01
3.82E+013.80E+013.92E+013.91E+013.88E+013.86E+013.84E+013.82E+01
962420     SCAPT
5.42E+005.43E+005.43E+005.44E+005.44E+005.45E+005.45E+005.45E+005.46E+005.47E+00
5.47E+005.48E+005.49E+005.50E+005.50E+005.51E+005.52E+005.52E+00
962440     SCAPT
3.40E+003.40E+003.41E+003.41E+003.41E+003.42E+003.42E+003.42E+003.43E+003.43E+00
3.43E+003.44E+003.45E+003.45E+003.46E+003.46E+003.46E+003.46E+00
     0
                                                  33P3RU  ORNL 78
    18
0.0      8.54E-011.66E+002.42E+003.14E+003.82E+003.82E+004.30E+004.79E+005.28E+00
5.78E+006.29E+006.29E+006.56E+006.85E+007.19E+007.55E+007.94E+00
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
922340     SCAPT
2.03E+012.01E+012.01E+012.01E+012.01E+012.01E+011.99E+012.00E+012.00E+012.00E+01
2.00E+012.01E+012.02E+012.02E+012.02E+012.02E+012.02E+012.02E+01
922350     SCAPT
1.05E+011.02E+011.01E+019.99E+009.94E+009.93E+009.99E+009.98E+009.98E+009.99E+00
1.00E+011.00E+011.02E+011.02E+011.02E+011.02E+011.03E+011.03E+01
922350     SFISS
4.66E+014.50E+014.40E+014.35E+014.32E+014.30E+014.33E+014.32E+014.32E+014.32E+01
4.33E+014.34E+014.44E+014.44E+014.45E+014.45E+014.46E+014.48E+01
922360     SCAPT
8.16E+008.23E+008.28E+008.33E+008.36E+008.38E+008.28E+008.30E+008.31E+008.32E+00
8.33E+008.33E+008.31E+008.31E+008.31E+008.31E+008.31E+008.31E+00
922380     SCAPT
9.32E-019.30E-019.31E-019.32E-019.33E-019.34E-019.38E-019.39E-019.39E-019.40E-01
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9.41E-019.41E-019.51E-019.51E-019.51E-019.51E-019.52E-019.52E-01
932370     SCAPT
3.35E+013.28E+013.23E+013.21E+013.19E+013.18E+013.21E+013.20E+013.20E+013.19E+01
3.20E+013.20E+013.23E+013.23E+013.23E+013.23E+013.23E+013.24E+01
942380     SCAPT
3.39E+013.26E+013.19E+013.15E+013.12E+013.12E+013.16E+013.16E+013.15E+013.16E+01
3.16E+013.17E+013.26E+013.26E+013.27E+013.27E+013.28E+013.29E+01
942380     SFISS
2.42E+002.40E+002.38E+002.37E+002.37E+002.37E+002.39E+002.38E+002.38E+002.39E+00
2.39E+002.39E+002.41E+002.42E+002.42E+002.42E+002.42E+002.42E+00
942390     SCAPT
6.61E+016.36E+016.20E+016.12E+016.07E+016.05E+015.75E+015.74E+015.74E+015.74E+01
5.75E+015.77E+015.71E+015.72E+015.72E+015.73E+015.75E+015.77E+01
942390     SFISS
1.16E+021.12E+021.09E+021.08E+021.07E+021.06E+021.02E+021.02E+021.02E+021.02E+02
1.02E+021.03E+021.02E+021.03E+021.03E+021.03E+021.03E+021.04E+02
942400     SCAPT
1.95E+021.94E+021.91E+021.87E+021.85E+021.83E+021.37E+021.36E+021.35E+021.34E+02
1.34E+021.33E+021.14E+021.14E+021.13E+021.13E+021.13E+021.12E+02
942410     SCAPT
4.07E+013.91E+013.82E+013.76E+013.74E+013.73E+013.64E+013.64E+013.64E+013.64E+01
3.65E+013.66E+013.68E+013.69E+013.69E+013.70E+013.71E+013.72E+01
942410     SFISS
1.22E+021.18E+021.15E+021.14E+021.13E+021.13E+021.11E+021.11E+021.11E+021.11E+02
1.11E+021.11E+021.12E+021.13E+021.13E+021.13E+021.13E+021.13E+02
942420     SCAPT
2.94E+012.97E+012.99E+013.00E+013.02E+013.03E+013.02E+013.02E+013.03E+013.03E+01
3.03E+013.04E+013.02E+013.02E+013.02E+013.01E+013.01E+013.01E+01
952410     SCAPT
1.11E+021.08E+021.05E+021.04E+021.03E+021.03E+029.99E+019.97E+019.95E+019.95E+01
9.96E+019.97E+019.92E+019.93E+019.93E+019.94E+019.95E+019.97E+01
952410     SCAPT1
1.37E+011.33E+011.30E+011.29E+011.28E+011.27E+011.23E+011.23E+011.23E+011.23E+01
1.23E+011.23E+011.23E+011.23E+011.23E+011.23E+011.23E+011.23E+01
952430     SCAPT
4.00E+013.98E+013.94E+013.88E+013.84E+013.81E+013.95E+013.93E+013.91E+013.88E+01
3.86E+013.85E+013.95E+013.94E+013.93E+013.92E+013.91E+013.89E+01
962420     SCAPT
5.56E+005.55E+005.56E+005.56E+005.57E+005.58E+005.59E+005.60E+005.60E+005.61E+00
5.61E+005.62E+005.67E+005.67E+005.67E+005.67E+005.67E+005.67E+00
962440     SCAPT
3.48E+003.48E+003.47E+003.48E+003.48E+003.48E+003.49E+003.50E+003.50E+003.50E+00
3.51E+003.51E+003.54E+003.54E+003.54E+003.54E+003.54E+003.54E+00
     0
                                                  33P3RM  ORNL 78
    18
0.0      3.06E-016.53E-011.04E+001.47E+001.93E+001.93E+002.45E+003.00E+003.55E+00
4.12E+004.70E+004.70E+005.50E+006.27E+007.00E+007.70E+008.38E+00
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
922340     SCAPT
1.66E+011.66E+011.66E+011.66E+011.66E+011.67E+011.72E+011.73E+011.73E+011.74E+01
1.74E+011.75E+011.76E+011.77E+011.78E+011.79E+011.80E+011.81E+01
922350     SCAPT
5.95E+005.97E+005.99E+006.02E+006.06E+006.10E+006.44E+006.50E+006.55E+006.62E+00
6.69E+006.76E+006.94E+007.03E+007.13E+007.23E+007.33E+007.44E+00
922350     SFISS
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2.03E+012.04E+012.06E+012.07E+012.09E+012.12E+012.27E+012.31E+012.34E+012.38E+01
2.42E+012.46E+012.56E+012.61E+012.66E+012.72E+012.77E+012.84E+01
922360     SCAPT
8.35E+008.35E+008.34E+008.34E+008.33E+008.33E+008.50E+008.50E+008.50E+008.50E+00
8.50E+008.49E+008.51E+008.53E+008.54E+008.54E+008.55E+008.55E+00
922380     SCAPT
8.57E-018.57E-018.57E-018.57E-018.57E-018.58E-018.78E-018.79E-018.80E-018.81E-01
8.82E-018.83E-018.91E-018.94E-018.96E-018.98E-019.00E-019.03E-01
932370     SCAPT
2.24E+012.25E+012.26E+012.26E+012.28E+012.29E+012.35E+012.37E+012.39E+012.41E+01
2.43E+012.46E+012.46E+012.49E+012.52E+012.55E+012.59E+012.62E+01
942380     SCAPT
1.34E+011.35E+011.37E+011.38E+011.39E+011.41E+011.53E+011.56E+011.59E+011.62E+01
1.65E+011.68E+011.76E+011.79E+011.84E+011.88E+011.93E+011.97E+01
942380     SFISS
2.00E+002.00E+002.00E+002.01E+002.01E+002.01E+002.04E+002.05E+002.05E+002.06E+00
2.06E+002.07E+002.09E+002.10E+002.11E+002.12E+002.12E+002.13E+00
942390     SCAPT
2.27E+012.29E+012.31E+012.33E+012.36E+012.40E+012.62E+012.67E+012.72E+012.77E+01
2.83E+012.89E+013.02E+013.09E+013.16E+013.24E+013.33E+013.41E+01
942390     SFISS
4.03E+014.07E+014.11E+014.16E+014.21E+014.27E+014.67E+014.76E+014.85E+014.94E+01
5.05E+015.16E+015.38E+015.51E+015.64E+015.79E+015.94E+016.10E+01
942400     SCAPT
4.01E+014.02E+014.03E+014.03E+014.05E+014.06E+014.34E+014.36E+014.39E+014.41E+01
4.44E+014.48E+014.70E+014.76E+014.80E+014.85E+014.90E+014.96E+01
942410     SCAPT
1.46E+011.47E+011.49E+011.50E+011.52E+011.54E+011.68E+011.71E+011.74E+011.78E+01
1.81E+011.85E+011.94E+011.98E+012.03E+012.08E+012.13E+012.19E+01
942410     SFISS
4.83E+014.87E+014.90E+014.95E+015.00E+015.07E+015.48E+015.57E+015.66E+015.76E+01
5.86E+015.97E+016.22E+016.35E+016.49E+016.63E+016.79E+016.95E+01
942420     SCAPT
2.34E+012.34E+012.34E+012.33E+012.33E+012.33E+012.35E+012.35E+012.35E+012.35E+01
2.35E+012.35E+012.32E+012.33E+012.33E+012.34E+012.34E+012.34E+01
952410     SCAPT
5.16E+015.19E+015.22E+015.26E+015.30E+015.36E+015.62E+015.70E+015.77E+015.85E+01
5.94E+016.03E+016.13E+016.23E+016.35E+016.47E+016.60E+016.73E+01
952410     SCAPT1
6.38E+006.41E+006.45E+006.50E+006.55E+006.62E+006.95E+007.04E+007.13E+007.23E+00
7.34E+007.45E+007.57E+007.71E+007.85E+008.00E+008.15E+008.32E+00
952430     SCAPT
2.86E+012.86E+012.85E+012.85E+012.86E+012.86E+012.96E+012.97E+012.97E+012.98E+01
2.98E+012.99E+013.00E+013.02E+013.04E+013.05E+013.07E+013.08E+01
962420     SCAPT
4.93E+004.93E+004.93E+004.94E+004.94E+004.94E+005.08E+005.09E+005.10E+005.11E+00
5.11E+005.12E+005.17E+005.19E+005.20E+005.22E+005.24E+005.25E+00
962440     SCAPT
3.11E+003.11E+003.11E+003.11E+003.11E+003.11E+003.19E+003.19E+003.20E+003.20E+00
3.21E+003.21E+003.24E+003.25E+003.26E+003.27E+003.28E+003.29E+00
     0
                                                  27B4U   ORNL 78
    24
0.0      3.48E-017.02E-011.06E+001.42E+001.79E+001.79E+002.13E+002.48E+002.82E+00
3.17E+003.51E+003.51E+003.84E+004.16E+004.48E+004.81E+005.13E+005.13E+005.43E+00
5.73E+006.04E+006.34E+006.64E+00
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
```

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0.0     0.0     0.0     0.0     0.0     0.0     0.0     0.0     0.0     0.0
0.0     0.0     0.0     0.0
922340     SCAPT
2.09E+012.09E+012.09E+012.09E+012.09E+012.09E+012.08E+012.08E+012.08E+012.08E+01
2.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.09E+012.09E+01
2.08E+012.08E+012.08E+012.08E+01
922350     SCAPT
1.10E+011.09E+011.09E+011.09E+011.08E+011.08E+011.11E+011.10E+011.10E+011.09E+01
1.09E+011.09E+011.12E+011.11E+011.11E+011.11E+011.10E+011.10E+011.13E+011.13E+01
1.12E+011.12E+011.12E+011.12E+01
922350     SFISS
4.91E+014.88E+014.85E+014.83E+014.82E+014.81E+014.94E+014.91E+014.89E+014.87E+01
4.85E+014.84E+015.01E+014.97E+014.95E+014.93E+014.91E+014.90E+015.08E+015.05E+01
5.03E+015.01E+014.99E+014.98E+01
922360     SCAPT
8.33E+008.35E+008.37E+008.38E+008.40E+008.41E+008.25E+008.27E+008.29E+008.30E+00
8.31E+008.32E+008.20E+008.22E+008.23E+008.25E+008.26E+008.27E+008.16E+008.18E+00
8.20E+008.21E+008.22E+008.23E+00
922380     SCAPT
9.01E-019.01E-019.01E-019.01E-019.01E-019.01E-019.03E-019.03E-019.03E-019.04E-01
9.04E-019.04E-019.08E-019.08E-019.08E-019.08E-019.08E-019.13E-019.13E-01
9.13E-019.13E-019.14E-019.14E-01
932370     SCAPT
3.40E+013.38E+013.37E+013.36E+013.35E+013.35E+013.44E+013.42E+013.41E+013.40E+01
3.39E+013.39E+013.45E+013.44E+013.43E+013.41E+013.41E+013.40E+013.45E+013.44E+01
3.43E+013.42E+013.41E+013.40E+01
942380     SCAPT
3.59E+013.57E+013.55E+013.54E+013.52E+013.52E+013.65E+013.62E+013.60E+013.59E+01
3.58E+013.57E+013.71E+013.69E+013.67E+013.65E+013.64E+013.63E+013.79E+013.76E+01
3.74E+013.73E+013.72E+013.71E+01
942380     SFISS
2.44E+002.43E+002.43E+002.43E+002.42E+002.42E+002.45E+002.45E+002.45E+002.44E+00
2.44E+002.44E+002.48E+002.47E+002.47E+002.46E+002.46E+002.46E+002.50E+002.49E+00
2.49E+002.49E+002.49E+002.48E+00
942390     SCAPT
6.91E+016.86E+016.82E+016.79E+016.77E+016.76E+016.62E+016.58E+016.54E+016.51E+01
6.49E+016.47E+016.49E+016.45E+016.41E+016.38E+016.36E+016.35E+016.42E+016.38E+01
6.34E+016.31E+016.29E+016.28E+01
942390     SFISS
1.22E+021.21E+021.20E+021.20E+021.19E+021.19E+021.18E+021.17E+021.17E+021.16E+02
1.16E+021.16E+021.17E+021.16E+021.15E+021.15E+021.14E+021.14E+021.16E+021.15E+02
1.15E+021.14E+021.14E+021.14E+02
942400     SCAPT
2.06E+022.05E+022.04E+022.03E+022.01E+022.00E+021.58E+021.57E+021.56E+021.55E+02
1.54E+021.53E+021.31E+021.30E+021.30E+021.29E+021.28E+021.27E+021.15E+021.14E+02
1.14E+021.13E+021.12E+021.12E+02
942410     SCAPT
4.29E+014.26E+014.24E+014.22E+014.21E+014.20E+014.22E+014.19E+014.17E+014.15E+01
4.14E+014.13E+014.21E+014.18E+014.16E+014.14E+014.13E+014.12E+014.22E+014.19E+01
4.17E+014.15E+014.14E+014.13E+01
942410     SFISS
1.29E+021.28E+021.28E+021.27E+021.27E+021.26E+021.28E+021.27E+021.26E+021.26E+02
1.25E+021.25E+021.28E+021.27E+021.26E+021.26E+021.25E+021.25E+021.28E+021.28E+02
1.27E+021.26E+021.26E+021.26E+02
942420     SCAPT
3.06E+013.07E+013.08E+013.08E+013.09E+013.09E+013.06E+013.07E+013.08E+013.08E+01
3.09E+013.09E+013.05E+013.06E+013.06E+013.07E+013.07E+013.08E+013.02E+013.02E+01
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3.03E+013.03E+013.04E+013.04E+01
952410     SCAPT
1.14E+021.13E+021.13E+021.12E+021.12E+021.12E+021.11E+021.10E+021.10E+021.09E+02
1.09E+021.09E+021.10E+021.09E+021.08E+021.08E+021.08E+021.07E+021.08E+021.08E+02
1.07E+021.07E+021.07E+021.06E+02
952410     SCAPT1
1.41E+011.40E+011.39E+011.39E+011.38E+011.38E+011.37E+011.36E+011.36E+011.35E+01
1.35E+011.34E+011.35E+011.35E+011.34E+011.33E+011.33E+011.33E+011.34E+011.33E+01
1.33E+011.32E+011.32E+011.31E+01
952430     SCAPT
4.04E+014.03E+014.01E+014.00E+013.98E+013.96E+014.11E+014.09E+014.07E+014.05E+01
4.04E+014.02E+014.14E+014.13E+014.11E+014.08E+014.07E+014.05E+014.16E+014.14E+01
4.13E+014.11E+014.08E+014.06E+01
962420     SCAPT
5.73E+005.73E+005.74E+005.74E+005.74E+005.75E+005.74E+005.75E+005.75E+005.75E+00
5.75E+005.76E+005.76E+005.77E+005.77E+005.77E+005.77E+005.78E+005.79E+005.79E+00
5.80E+005.80E+005.80E+005.80E+00
962440     SCAPT
3.60E+003.60E+003.60E+003.60E+003.60E+003.60E+003.61E+003.61E+003.61E+003.61E+00
3.61E+003.61E+003.62E+003.62E+003.62E+003.62E+003.62E+003.63E+003.64E+003.64E+00
3.64E+003.64E+003.64E+003.64E+00
     0
                                                  27B4RU  ORNL 78
    24
0.0      4.07E-018.12E-011.21E+001.61E+002.00E+002.00E+002.34E+002.68E+003.02E+00
3.36E+003.70E+003.70E+003.97E+004.25E+004.53E+004.81E+005.10E+005.10E+005.33E+00
5.57E+005.81E+006.06E+006.32E+00
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0
922340     SCAPT
2.07E+012.06E+012.05E+012.05E+012.05E+012.05E+012.02E+012.02E+012.02E+012.02E+01
2.03E+012.03E+012.00E+012.00E+012.00E+012.00E+012.00E+012.00E+012.00E+012.00E+01
2.00E+012.00E+012.00E+012.00E+01
922350     SCAPT
1.07E+011.05E+011.04E+011.03E+011.03E+011.03E+011.01E+011.01E+011.01E+011.01E+01
1.01E+011.01E+019.86E+009.86E+009.86E+009.86E+009.87E+009.88E+009.87E+009.88E+00
9.88E+009.89E+009.90E+009.91E+00
922350     SFISS
4.74E+014.65E+014.58E+014.53E+014.49E+014.47E+014.39E+014.37E+014.36E+014.35E+01
4.35E+014.35E+014.21E+014.21E+014.21E+014.21E+014.22E+014.22E+014.20E+014.20E+01
4.20E+014.21E+014.21E+014.22E+01
922360     SCAPT
8.34E+008.37E+008.40E+008.42E+008.45E+008.46E+008.40E+008.42E+008.43E+008.44E+00
8.45E+008.46E+008.42E+008.43E+008.44E+008.44E+008.44E+008.44E+008.44E+008.44E+00
8.44E+008.44E+008.44E+008.44E+00
922380     SCAPT
9.00E-018.98E-018.97E-018.97E-018.97E-018.98E-018.98E-018.98E-018.99E-018.99E-01
8.99E-019.00E-018.99E-019.00E-019.00E-019.01E-019.01E-019.05E-019.06E-01
9.06E-019.06E-019.06E-019.06E-01
932370     SCAPT
3.40E+013.36E+013.33E+013.30E+013.28E+013.27E+013.25E+013.24E+013.24E+013.23E+01
3.23E+013.22E+013.16E+013.16E+013.16E+013.16E+013.16E+013.16E+013.14E+013.14E+01
3.14E+013.14E+013.14E+013.14E+01
942380     SCAPT
3.45E+013.38E+013.33E+013.29E+013.26E+013.24E+013.20E+013.19E+013.18E+013.17E+01
3.17E+013.17E+013.07E+013.07E+013.07E+013.08E+013.08E+013.08E+013.07E+013.08E+01
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3.08E+013.08E+013.09E+013.09E+01
942380     SFISS
2.41E+002.39E+002.38E+002.37E+002.37E+002.36E+002.36E+002.36E+002.36E+002.35E+00
2.35E+002.35E+002.34E+002.34E+002.34E+002.34E+002.34E+002.34E+002.34E+002.34E+00
2.35E+002.35E+002.35E+002.35E+00
942390     SCAPT
6.73E+016.58E+016.47E+016.40E+016.34E+016.30E+015.87E+015.85E+015.83E+015.82E+01
5.81E+015.81E+015.43E+015.43E+015.44E+015.44E+015.44E+015.45E+015.29E+015.29E+01
5.30E+015.31E+015.32E+015.33E+01
942390     SFISS
1.18E+021.16E+021.14E+021.13E+021.11E+021.11E+021.04E+021.04E+021.04E+021.03E+02
1.03E+021.03E+029.72E+019.72E+019.72E+019.73E+019.74E+019.75E+019.50E+019.52E+01
9.53E+019.54E+019.55E+019.57E+01
942400     SCAPT
2.05E+022.05E+022.04E+022.02E+022.00E+021.98E+021.53E+021.52E+021.52E+021.51E+02
1.50E+021.49E+021.24E+021.23E+021.23E+021.22E+021.22E+021.21E+021.07E+021.07E+02
1.07E+021.06E+021.06E+021.06E+02
942410     SCAPT
4.14E+014.05E+013.99E+013.94E+013.90E+013.88E+013.71E+013.70E+013.68E+013.68E+01
3.67E+013.67E+013.49E+013.49E+013.49E+013.49E+013.50E+013.50E+013.44E+013.44E+01
3.45E+013.45E+013.46E+013.46E+01
942410     SFISS
1.25E+021.22E+021.20E+021.19E+021.18E+021.17E+021.13E+021.13E+021.12E+021.12E+02
1.12E+021.12E+021.07E+021.07E+021.07E+021.07E+021.07E+021.07E+021.06E+021.06E+02
1.06E+021.06E+021.06E+021.06E+02
942420     SCAPT
3.04E+013.06E+013.07E+013.07E+013.08E+013.09E+013.09E+013.10E+013.10E+013.11E+01
3.11E+013.12E+013.11E+013.11E+013.11E+013.11E+013.11E+013.11E+013.09E+013.09E+01
3.09E+013.09E+013.09E+013.09E+01
952410     SCAPT
1.13E+021.11E+021.09E+021.08E+021.07E+021.07E+021.02E+021.01E+021.01E+021.01E+02
1.01E+021.01E+029.59E+019.59E+019.58E+019.58E+019.58E+019.59E+019.38E+019.39E+01
9.39E+019.39E+019.40E+019.41E+01
952410     SCAPT1
1.39E+011.37E+011.35E+011.34E+011.33E+011.32E+011.26E+011.25E+011.25E+011.25E+01
1.24E+011.24E+011.19E+011.18E+011.18E+011.18E+011.18E+011.19E+011.16E+011.16E+01
1.16E+011.16E+011.16E+011.16E+01
952430     SCAPT
4.08E+014.07E+014.05E+014.02E+014.00E+013.97E+014.04E+014.02E+014.00E+013.98E+01
3.96E+013.95E+013.98E+013.96E+013.95E+013.94E+013.93E+013.92E+013.95E+013.94E+01
3.94E+013.93E+013.92E+013.91E+01
962420     SCAPT
5.70E+005.69E+005.69E+005.69E+005.69E+005.70E+005.69E+005.69E+005.70E+005.70E+00
5.70E+005.71E+005.69E+005.70E+005.70E+005.70E+005.70E+005.71E+005.72E+005.73E+00
5.73E+005.73E+005.73E+005.73E+00
962440     SCAPT
3.58E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+00
3.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.57E+003.59E+003.59E+00
3.59E+003.59E+003.59E+003.59E+00
     0
                                                  27B4RM  ORNL 78
    24
0.0       3.17E-016.38E-019.66E-011.30E+001.64E+001.64E+001.99E+002.34E+002.69E+00
3.04E+003.40E+003.40E+003.79E+004.19E+004.58E+004.96E+005.35E+005.35E+005.77E+00
6.19E+006.60E+007.01E+007.40E+00
0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0
0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0       0.0
```

```
0.0     0.0     0.0     0.0
922340     SCAPT
1.75E+011.76E+011.76E+011.77E+011.77E+011.77E+011.80E+011.80E+011.81E+011.81E+01
1.82E+011.82E+011.86E+011.87E+011.87E+011.88E+011.88E+011.89E+011.91E+011.91E+01
1.92E+011.92E+011.93E+011.94E+01
922350     SCAPT
6.88E+006.92E+006.97E+007.01E+007.05E+007.10E+007.33E+007.38E+007.43E+007.48E+00
7.54E+007.59E+008.00E+008.06E+008.12E+008.18E+008.24E+008.31E+008.63E+008.67E+00
8.71E+008.76E+008.81E+008.87E+00
922350     SFISS
2.54E+012.57E+012.59E+012.62E+012.64E+012.67E+012.78E+012.81E+012.84E+012.87E+01
2.90E+012.93E+013.15E+013.18E+013.22E+013.25E+013.29E+013.33E+013.52E+013.53E+01
3.56E+013.58E+013.61E+013.64E+01
922360     SCAPT
8.45E+008.45E+008.45E+008.45E+008.45E+008.49E+008.50E+008.50E+008.50E+00
8.50E+008.50E+008.53E+008.53E+008.54E+008.54E+008.54E+008.54E+008.48E+008.49E+00
8.51E+008.52E+008.53E+008.53E+00
922380     SCAPT
8.30E-018.31E-018.31E-018.32E-018.33E-018.34E-018.44E-018.45E-018.46E-018.47E-01
8.48E-018.49E-018.60E-018.61E-018.63E-018.64E-018.66E-018.67E-018.73E-018.74E-01
8.76E-018.77E-018.79E-018.80E-01
932370     SCAPT
2.52E+012.53E+012.55E+012.56E+012.58E+012.59E+012.63E+012.64E+012.66E+012.67E+01
2.69E+012.71E+012.80E+012.82E+012.84E+012.85E+012.87E+012.89E+012.94E+012.96E+01
2.97E+012.98E+013.00E+013.02E+01
942360     SCAPT
1.75E+011.77E+011.79E+011.81E+011.83E+011.85E+011.94E+011.96E+011.98E+012.01E+01
2.03E+012.05E+012.23E+012.25E+012.28E+012.31E+012.34E+012.36E+012.51E+012.53E+01
2.55E+012.57E+012.59E+012.61E+01
942380     SFISS
2.04E+002.05E+002.05E+002.06E+002.06E+002.06E+002.08E+002.09E+002.09E+002.10E+00
2.10E+002.11E+002.15E+002.15E+002.16E+002.16E+002.17E+002.17E+002.21E+002.21E+00
2.21E+002.22E+002.22E+002.23E+00
942390     SCAPT
2.89E+012.93E+012.96E+013.00E+013.03E+013.07E+013.25E+013.29E+013.33E+013.37E+01
3.42E+013.46E+013.78E+013.83E+013.88E+013.93E+013.98E+014.03E+014.30E+014.32E+01
4.36E+014.39E+014.43E+014.47E+01
942390     SFISS
5.21E+015.27E+015.33E+015.39E+015.46E+015.53E+015.84E+015.92E+015.99E+016.07E+01
6.15E+016.23E+016.80E+016.89E+016.97E+017.06E+017.15E+017.25E+017.74E+017.78E+01
7.84E+017.91E+017.98E+018.05E+01
942400     SCAPT
4.42E+014.43E+014.45E+014.46E+014.47E+014.49E+014.65E+014.67E+014.69E+014.71E+01
4.72E+014.74E+015.06E+015.08E+015.10E+015.13E+015.15E+015.17E+015.47E+015.49E+01
5.51E+015.53E+015.55E+015.57E+01
942410     SCAPT
1.90E+011.92E+011.94E+011.96E+011.99E+012.01E+012.12E+012.15E+012.17E+012.20E+01
2.23E+012.25E+012.45E+012.49E+012.52E+012.55E+012.58E+012.61E+012.79E+012.80E+01
2.82E+012.85E+012.87E+012.90E+01
942410     SFISS
6.14E+016.20E+016.26E+016.33E+016.39E+016.46E+016.77E+016.85E+016.92E+017.00E+01
7.08E+017.16E+017.74E+017.83E+017.92E+018.01E+018.11E+018.20E+018.70E+018.75E+01
8.81E+018.87E+018.95E+019.03E+01
942420     SCAPT
2.46E+012.46E+012.46E+012.46E+012.46E+012.46E+012.45E+012.46E+012.46E+012.46E+01
2.46E+012.46E+012.45E+012.45E+012.45E+012.46E+012.46E+012.46E+012.41E+012.42E+01
2.42E+012.43E+012.43E+012.43E+01
```

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952410     SCAPT
6.13E+016.18E+016.23E+016.28E+016.33E+016.39E+016.59E+016.65E+016.71E+016.77E+01
6.83E+016.90E+017.31E+017.38E+017.45E+017.53E+017.60E+017.67E+017.99E+018.03E+01
8.08E+018.13E+018.19E+018.25E+01
952410     SCAPT1
7.58E+007.64E+007.70E+007.76E+007.83E+007.89E+008.15E+008.22E+008.29E+008.37E+00
8.44E+008.52E+009.04E+009.13E+009.21E+009.30E+009.39E+009.48E+009.87E+009.92E+00
9.98E+001.00E+011.01E+011.02E+01
952430     SCAPT
3.11E+013.12E+013.12E+013.12E+013.12E+013.12E+013.17E+013.17E+013.17E+013.18E+01
3.18E+013.18E+013.26E+013.27E+013.27E+013.28E+013.28E+013.29E+013.32E+013.33E+01
3.34E+013.34E+013.35E+013.36E+01
962420     SCAPT
5.13E+005.14E+005.15E+005.15E+005.16E+005.16E+005.24E+005.25E+005.25E+005.26E+00
5.27E+005.28E+005.36E+005.37E+005.38E+005.39E+005.40E+005.41E+005.43E+005.45E+00
5.46E+005.47E+005.48E+005.49E+00
962440     SCAPT
3.24E+003.24E+003.25E+003.25E+003.25E+003.26E+003.30E+003.30E+003.31E+003.31E+00
3.32E+003.32E+003.37E+003.37E+003.38E+003.39E+003.39E+003.40E+003.42E+003.42E+00
3.43E+003.44E+003.44E+003.45E+00
     0
                                                  30P3U 31 KFK KWO
    14
        0.042    0.62     0.87     0.91     1.35     2.12     2.17     2.99     3.99
4.99    5.12     5.19     5.99


922350     SCAPT
10.3    10.3     10.1     10.2     10.2     10.2     9.92     9.87     9.92     10.0
9.99    9.92     9.90     10.1
922350     SFISS
46.3    45.7     44.9     45.6     44.9     45.4     43.1     42.6     42.9     43.6
43.1    42.5     42.3     43.8
922360     SCAPT
8.62    8.63     8.22     8.0      8.0      7.76     7.52     7.53     7.18     6.86
6.64    6.63     6.64     6.41
922360     SFISS
.222    .222     .224     .224     .224     .224     .224     .223     .223     .222
.221    .220     .220     .220
922380     SCAPT
.830    .830     .827     .826     .826     .826     .826     .826     .827     .829
.831    .831     .831     .833
922380     SFISS
.114    .114     .114     .115     .114     .115     .114     .114     .114     .113
.113    .112     .112     .112
932370     SCAPT
32.1    32.2     31.8     31.7     31.7     31.7     31.1     31.1     31.0     30.9
30.8    30.6     30.6     30.8
932370     SFISS
.57     .57      .57      .57      .57      .57      .57      .57      .57      .57
.57     .57      .57      .56
942380     SCAPT
35.3    34.8     34.3     35.0     34.5     35.0     33.1     32.8     33.1     33.4
33.4    33.0     32.8     34.1
942380     SFISS
2.46    2.45     2.44     2.46     2.45     2.47     2.42     2.41     2.43     2.45
2.44    2.43     2.42     2.46
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942390    SCAPT
67.8     67.3     63.2     62.5     62.1     61.1     56.7     56.5     55.4     55.1
54.0     53.1     53.1     54.1
942390    SFISS
123.     122.     115.0    114.0    113.0    112.0    104.     104.     102.     102.
99.9     98.2     98.1     100.
942400    SCAPT
210.     209.0    200.0    187.0    183.0    177.0    162.0    153.0    145.0    129.0
115.0    109.0    108.0    104.0
942400    SFISS
.617     .617     .618     .619     .618     .617     .615     .614     .611     .608
.604     .604     .604     .60
942410    SCAPT
47.4     47.0     45.1     45.2     44.7     44.6     41.7     41.5     41.1     41.3
40.6     39.9     39.9     40.9
942410    SFISS
124.0    123.0    118.0    119.0    118.0    118.0    111.0    110.0    109.0    110.0
108.0    107.0    106.0    109.0
942420    SCAPT
27.0     27.2     27.2     26.9     27.1     27.0     27.2     27.3     27.0     26.6
26.2     26.2     26.3     25.6
942420    SFISS
.45      .45      .45      .46      .46      .46      .46      .45      .45      .45
.45      .45      .45      .45
952410    SCAPT
108.0    107.0    102.     101.     100.     99.1     93.9     93.7     92.1     91.5
90.1     88.9     89.0     89.9
952410    SCAPT1
13.3     13.3     12.6     12.5     12.4     12.2     11.6     11.6     11.4     11.3
11.1     11.0     11.0     11.1
952410    SFISS
1.27     1.26     1.23     1.22     1.22     1.21     1.17     1.17     1.15     1.13
1.13     1.13     1.13     1.13
952430    SCAPT
50.4     50.7     50.2     49.7     49.9     49.6     49.3     49.5     49.0     48.3
47.9     47.8     48.0     47.3
952430    SFISS
.68      .68      .68      .69      .68      .69      .68      .68      .68      .68
.67      .67      .67      .67
962440    SCAPT
17.2     17.2     17.2     17.1     17.2     17.1     17.3     17.4     17.3     17.2
17.2     17.3     17.3     17.1
962440    SFISS
1.65     1.65     1.65     1.65     1.65     1.65     1.66     1.66     1.65     1.65
1.65     1.65     1.65     1.64
    0
                                                   50P5U    ORNL 80
    30
0.0      5.79E-011.16E+001.76E+002.35E+002.95E+002.95E+003.50E+004.05E+004.60E+00
5.14E+005.69E+005.69E+006.18E+006.68E+007.18E+007.67E+008.16E+008.16E+008.61E+00
9.06E+009.52E+009.97E+001.04E+011.04E+011.08E+011.12E+011.17E+011.21E+011.25E+01
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0      0.0
922340    SCAPT
2.07E+012.07E+012.07E+012.07E+012.08E+012.08E+012.06E+012.07E+012.07E+012.07E+01
2.07E+012.08E+012.07E+012.07E+012.07E+012.07E+012.08E+012.08E+012.07E+012.07E+01
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2.07E+012.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.08E+012.09E+01
922350     SCAPT
9.37E+009.37E+009.38E+009.39E+009.41E+009.43E+009.49E+009.49E+009.49E+009.50E+00
9.52E+009.55E+009.62E+009.62E+009.62E+009.63E+009.65E+009.68E+009.76E+009.76E+00
9.77E+009.78E+009.80E+009.82E+009.93E+009.92E+009.93E+009.94E+009.96E+009.99E+00
922350     SFISS
4.02E+014.02E+014.02E+014.03E+014.03E+014.05E+014.08E+014.08E+014.08E+014.08E+01
4.09E+014.11E+014.15E+014.14E+014.14E+014.15E+014.16E+014.17E+014.22E+014.22E+01
4.22E+014.22E+014.23E+014.25E+014.30E+014.30E+014.30E+014.30E+014.31E+014.33E+01
922360     SCAPT
8.11E+008.12E+008.13E+008.14E+008.14E+008.15E+007.67E+007.68E+007.69E+007.70E+00
7.70E+007.71E+007.30E+007.31E+007.32E+007.33E+007.34E+007.34E+006.99E+007.00E+00
7.01E+007.02E+007.02E+007.03E+006.72E+006.73E+006.74E+006.75E+006.75E+006.76E+00
922380     SCAPT
9.20E-019.20E-019.21E-019.22E-019.23E-019.24E-019.25E-019.26E-019.26E-019.27E-01
9.28E-019.29E-019.31E-019.31E-019.32E-019.33E-019.34E-019.35E-019.37E-019.38E-01
9.39E-019.39E-019.40E-019.41E-019.44E-019.44E-019.45E-019.46E-019.47E-019.48E-01
932370     SCAPT
3.08E+013.07E+013.07E+013.07E+013.08E+013.08E+013.09E+013.09E+013.09E+013.09E+01
3.09E+013.10E+013.10E+013.10E+013.10E+013.10E+013.10E+013.11E+013.11E+013.11E+01
3.11E+013.11E+013.11E+013.11E+013.12E+013.11E+013.11E+013.11E+013.12E+013.12E+01
942380     SCAPT
2.90E+012.90E+012.90E+012.90E+012.91E+012.92E+012.96E+012.96E+012.96E+012.97E+01
2.97E+012.98E+013.03E+013.03E+013.03E+013.04E+013.04E+013.05E+013.11E+013.10E+01
3.11E+013.11E+013.12E+013.13E+013.19E+013.19E+013.19E+013.19E+013.20E+013.21E+01
942380     SFISS
2.18E+002.18E+002.18E+002.18E+002.18E+002.18E+002.20E+002.20E+002.20E+002.20E+00
2.20E+002.20E+002.22E+002.22E+002.22E+002.22E+002.22E+002.23E+002.24E+002.24E+00
2.24E+002.25E+002.25E+002.25E+002.27E+002.27E+002.27E+002.27E+002.27E+002.28E+00
942390     SCAPT
5.89E+015.89E+015.89E+015.90E+015.92E+015.94E+015.74E+015.73E+015.74E+015.75E+01
5.76E+015.78E+015.60E+015.59E+015.60E+015.61E+015.62E+015.64E+015.47E+015.46E+01
5.47E+015.48E+015.49E+015.51E+015.35E+015.34E+015.34E+015.35E+015.37E+015.39E+01
942390     SFISS
1.03E+021.03E+021.03E+021.03E+021.03E+021.01E+021.01E+021.01E+021.01E+02
1.01E+021.02E+029.93E+019.92E+019.92E+019.94E+019.96E+011.00E+029.79E+019.78E+01
9.78E+019.80E+019.83E+019.86E+019.67E+019.66E+019.66E+019.68E+019.70E+019.74E+01
942400     SCAPT
1.77E+021.76E+021.75E+021.74E+021.73E+021.72E+021.43E+021.43E+021.42E+021.41E+02
1.41E+021.40E+021.24E+021.24E+021.23E+021.23E+021.22E+021.21E+021.12E+021.11E+02
1.11E+021.10E+021.10E+021.09E+021.03E+021.03E+021.02E+021.02E+021.01E+021.01E+02
942410     SCAPT
3.53E+013.53E+013.53E+013.54E+013.55E+013.56E+013.52E+013.51E+013.51E+013.52E+01
3.53E+013.54E+013.51E+013.50E+013.50E+013.51E+013.52E+013.53E+013.50E+013.50E+01
3.50E+013.51E+013.52E+013.53E+013.50E+013.50E+013.50E+013.51E+013.52E+013.53E+01
942410     SFISS
1.07E+021.06E+021.07E+021.07E+021.07E+021.07E+021.06E+021.06E+021.06E+021.07E+02
1.07E+021.07E+021.07E+021.07E+021.07E+021.07E+021.07E+021.07E+021.07E+021.07E+02
1.07E+021.07E+021.07E+021.08E+021.07E+021.07E+021.07E+021.07E+021.08E+021.08E+02
942420     SCAPT
3.03E+013.03E+013.31E+013.31E+013.31E+013.32E+013.18E+013.19E+013.19E+013.20E+01
3.20E+013.20E+013.08E+013.08E+013.09E+013.09E+013.09E+013.09E+012.98E+012.99E+01
2.99E+012.99E+013.00E+013.00E+012.89E+012.90E+012.90E+012.91E+012.91E+012.91E+01
952410     SCAPT
9.37E+019.36E+019.36E+019.36E+019.38E+019.40E+019.22E+019.21E+019.21E+019.21E+01
9.23E+019.25E+019.08E+019.07E+019.06E+019.07E+019.08E+019.11E+018.94E+018.93E+01
8.93E+018.93E+018.95E+018.97E+018.81E+018.80E+018.79E+018.80E+018.81E+018.83E+01
```

```
952410     SCAPT1
1.16E+011.16E+011.16E+011.16E+011.16E+011.16E+011.14E+011.14E+011.14E+011.14E+01
1.14E+011.14E+011.12E+011.12E+011.12E+011.12E+011.12E+011.13E+011.10E+011.10E+01
1.10E+011.10E+011.11E+011.11E+011.09E+011.09E+011.09E+011.09E+011.09E+011.09E+01
952430     SCAPT
4.91E+014.89E+014.87E+014.85E+014.83E+014.82E+014.98E+014.96E+014.95E+014.93E+01
4.91E+014.88E+015.01E+015.00E+014.98E+014.96E+014.94E+014.92E+015.03E+015.02E+01
5.00E+014.98E+014.96E+014.94E+015.05E+015.04E+015.02E+015.00E+014.98E+014.96E+01
962420     SCAPT
5.67E+005.67E+005.68E+005.69E+005.69E+005.70E+005.69E+005.69E+005.70E+005.70E+00
5.71E+005.72E+005.70E+005.71E+005.72E+005.72E+005.73E+005.73E+005.72E+005.73E+00
5.73E+005.74E+005.75E+005.75E+005.74E+005.75E+005.75E+005.76E+005.76E+005.77E+00
962440     SCAPT
1.38E+011.39E+011.39E+011.39E+011.39E+011.39E+011.39E+011.39E+011.39E+011.39E+01
1.39E+011.39E+011.39E+011.39E+011.39E+011.39E+011.40E+011.40E+011.39E+011.39E+01
1.39E+011.40E+011.40E+011.40E+011.39E+011.39E+011.40E+011.40E+011.40E+011.40E+01
     0
                                                  33P3U 32 KFK BIB
    15
0.0       0.035     0.39      1.0       1.8       2.6       2.65      2.9       3.65      4.43
5.22      5.26      5.6       6.25      7.0


922350     SCAPT
9.98E+009.89E+009.83E+009.77E+009.79E+009.51E+009.44E+009.49E+009.57E+009.68E+00
9.45E+009.39E+009.46E+009.56E+009.69E+00
922350     SFISS
4.40E+014.31E+014.28E+014.24E+014.26E+014.03E+013.96E+014.00E+014.06E+014.14E+01
3.95E+013.89E+013.94E+014.02E+014.11E+01
922360     SCAPT
8.58E+008.61E+008.34E+007.93E+007.52E+007.30E+007.32E+007.17E+006.93E+006.68E+00
6.57E+006.59E+006.49E+006.33E+006.16E+00
922360     SFISS
2.24E-012.24E-012.25E-012.26E-012.26E-012.25E-012.24E-012.24E-012.24E-012.24E-01
2.22E-012.22E-012.22E-012.21E-012.21E-01
922380     SCAPT
8.25E-018.25E-018.23E-018.21E-018.21E-018.21E-018.22E-018.22E-018.23E-018.24E-01
8.25E-018.25E-018.26E-018.27E-018.28E-01
922380     SFISS
1.14E-011.14E-011.15E-011.15E-011.15E-011.15E-011.14E-011.14E-011.14E-011.14E-01
1.13E-011.13E-011.13E-011.13E-011.12E-01
932370     SCAPT
3.14E+013.14E+013.12E+013.08E+013.07E+013.01E+013.01E+013.00E+013.00E+013.01E+01
2.96E+012.96E+012.96E+012.97E+012.97E+01
932370     SFISS
5.73E-015.72E-015.75E-015.77E-015.77E-015.77E-015.76E-015.76E-015.75E-015.74E-01
5.72E-015.72E-015.71E-015.70E-015.69E-01
942380     SCAPT
3.34E+013.27E+013.25E+013.24E+013.27E+013.09E+013.04E+013.07E+013.13E+013.20E+01
3.04E+013.00E+013.04E+013.11E+013.19E+01
942380     SFISS
2.42E+002.40E+002.40E+002.40E+002.41E+002.37E+002.36E+002.37E+002.38E+002.40E+00
2.37E+002.35E+002.37E+002.38E+002.41E+00
942390     SCAPT
6.47E+016.40E+016.15E+015.84E+015.64E+015.25E+015.21E+015.19E+015.17E+015.18E+01
4.92E+014.89E+014.92E+014.95E+015.02E+01
942390     SFISS
```

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1.17E+021.16E+021.12E+021.07E+021.03E+029.63E+019.53E+019.53E+019.52E+019.58E+01
9.08E+019.01E+019.07E+019.16E+019.30E+01
942400     SCAPT
206.     199.5    185.5    169.5    151.5    140.5    137.     129.     119.     110.
105.5    103.5    99.5     94.      88.5
942400     SFISS
6.21E-016.21E-016.23E-016.22E-016.20E-016.18E-016.17E-016.16E-016.14E-016.11E-01
6.09E-016.09E-016.08E-016.06E-016.04E-01
942410     SCAPT
4.52E+014.45E+014.34E+014.21E+014.14E+013.87E+013.83E+013.84E+013.86E+013.89E+01
3.70E+013.66E+013.70E+013.74E+013.81E+01
942410     SFISS
1.18E+021.16E+021.14E+021.11E+021.10E+021.03E+021.02E+021.02E+021.03E+021.04E+02
9.88E+019.78E+019.87E+011.00E+021.02E+02
942420     SCAPT
2.69E+012.71E+012.70E+012.69E+012.68E+012.69E+012.71E+012.69E+012.66E+012.62E+01
2.60E+012.61E+012.59E+012.54E+012.47E+01
942420     SFISS
4.56E-014.55E-014.57E-014.59E-014.60E-0'4.59E-014.58E-014.58E-014.58E-014.57E-01
4.55E-014.54E-014.54E-014.53E-014.52E-01
952410     SCAPT
104.13   103.24   99.68    96.12    93.45    88.29    87.93    87.58    87.13    87.04
83.93    83.66    83.84    84.02    84.55
952410     SCAPT1
12.87    12.76    12.32    11.88    11.55    10.91    19.87    10.82    10.77    10.76
10.37    10.34    10.36    10.38    10.45
952410     SFISS
1.24E+001.24E+001.22E+001.19E+001.17E+001.14E+001.13E+001.13E+001.12E+001.12E+00
1.10E+001.10E+001.10E+001.10E+001.10E+00
952430     SCAPT
5.00E+015.02E+014.99E+014.94E+014.88E+014.86E+014.88E+014.85E+014.80E+014.75E+01
4.72E+014.74E+014.71E+014.66E+014.61E+01
952430     SFISS
6.85E-016.84E-016.87E-016.89E-016.90E-016.87E-016.86E-016.86E-016.85E-016.84E-01
6.80E-016.79E-016.79E-016.78E-016.76E-01
962440     SCAPT
1.71E+011.72E+011.72E+011.71E+011.70E+011.72E+011.73E+011.73E+011.72E+011.70E+01
1.72E+011.73E+011.72E+011.71E+011.70E+01
962440     SFISS
1.65E+001.65E+001.65E+001.65E+001.65E+001.66E+001.65E+001.65E+001.64E+00
1.65E+001.65E+001.65E+001.64E+001.63E+00
     0
                                                  30P4U 31 KFK TRI
    17
0.0      0.032    0.42     0.81     1.25     1.29     1.6      2.15     2.71     2.75
3.22     3.78     4.33     4.37     4.92     5.48     6.13     6.60


922350     SCAPT
9.24E+009.16E+008.99E+008.87E+008.84E+008.77E+008.72E+008.66E+008.68E+008.62E+00
8.60E+008.59E+008.53E+008.48E+008.47E+008.45E+008.46E+00
922350     SFISS
3.90E+013.83E+013.71E+013.63E+013.62E+013.55E+013.52E+013.48E+013.50E+013.44E+01
3.43E+013.41E+013.37E+013.32E+013.31E+013.30E+013.30E+01
922360     SCAPT
8.53E+008.56E+008.28E+008.03E+007.74E+007.76E+007.60E+007.34E+007.08E+007.10E+00
6.93E+006.75E+006.61E+006.62E+006.47E+006.34E+006.21E+00
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922360     SFISS
2.25E-012.24E-012.26E-012.26E-012.27E-012.27E-012.27E-012.27E-012.27E-012.27E-01
2.27E-012.27E-012.26E-012.26E-012.25E-012.25E-012.25E-01
922380     SCAPT
8.05E-018.05E-018.02E-018.00E-017.98E-017.98E-017.98E-017.97E-017.95E-017.96E-01
7.95E-017.95E-017.96E-017.96E-017.96E-017.95E-017.95E-01
922380     SFISS
1.15E-011.14E-011.15E-011.15E-011.16E-011.16E-011.16E-011.16E-011.16E-011.16E-01
1.16E-011.15E-011.15E-011.15E-011.15E-011.15E-011.15E-01
932370     SCAPT
2.99E+012.98E+012.94E+012.91E+012.88E+012.87E+012.85E+012.83E+012.81E+012.80E+01
2.79E+012.77E+012.75E+012.75E+012.73E+012.72E+012.71E+01
932370     SFISS
5.77E-015.77E-015.80E-015.82E-015.84E-015.83E-015.84E-015.84E-015.85E-015.85E-01
5.85E-015.84E-015.83E-015.83E-015.82E-015.82E-015.82E-01
942380     SCAPT
2.93E+012.87E+012.79E+012.74E+012.74E+012.69E+012.67E+012.64E+012.67E+012.62E+01
2.62E+012.61E+012.57E+012.53E+012.53E+012.52E+012.53E+01
942380     SFISS
2.31E+002.29E+002.28E+002.27E+002.27E+002.26E+002.26E+002.25E+002.26E+002.25E+00
2.25E+002.25E+002.24E+002.23E+002.23E+002.23E+002.24E+00
942390     SCAPT
5.84E+015.77E+015.39E+015.13E+014.92E+014.87E+014.75E+014.58E+014.49E+014.45E+01
4.38E+014.31E+014.20E+014.17E+014.11E+014.06E+014.03E+01
942390     SFISS
1.05E+021.04E+029.73E+019.28E+018.95E+018.84E+018.64E+018.36E+018.22E+018.14E+01
8.02E+017.90E+017.71E+017.64E+017.55E+017.47E+017.42E+01
942400     SCAPT
200.     192.5    180.5    170.     163.5    159.5    151.     139.5    138.5    134.5
122.     115.     111.5    108.     102.5    98.0     93.5
942400     SFISS
6.25E-016.25E-016.27E-016.28E-016.28E-016.27E-016.27E-016.26E-016.25E-016.24E-01
6.23E-016.22E-016.20E-016.20E-016.19E-016.18E-016.17E-01
942410     SCAPT
4.03E+013.98E+013.78E+013.64E+013.56E+013.51E+013.45E+013.36E+013.33E+013.29E+01
3.25E+013.22E+013.15E+013.12E+013.09E+013.06E+013.05E+01
942410     SFISS
1.05E+021.04E+029.94E+019.60E+019.42E+019.28E+019.15E+018.94E+018.89E+018.77E+01
8.69E+018.61E+018.44E+018.35E+018.29E+018.22E+018.20E+01
942420     SCAPT
2.65E+012.66E+012.66E+012.66E+012.64E+012.66E+012.66E+012.65E+012.62E+012.63E+01
2.61E+012.59E+012.57E+012.58E+012.55E+012.51E+012.48E+01
942420     SFISS
4.59E-014.58E-014.61E-014.62E-014.64E-014.64E-014.64E-014.64E-014.65E-014.65E-01
4.65E-014.64E-014.63E-014.63E-014.63E-014.62E-014.62E-01
952410     SCAPT
9.58E+019.51E+019.03E+018.69E+018.40E+018.35E+018.20E+017.97E+017.83E+017.79E+01
7.69E+017.58E+017.43E+017.40E+017.32E+017.24E+017.18E+01
952410     SCAPT1
1.18E+011.18E+011.12E+011.07E+011.04E+011.03E+011.01E+019.86E+009.67E+009.63E+00
9.50E+009.37E+009.18E+009.15E+009.05E+008.95E+008.88E+00
952410     SFISS
1.19E+001.19E+001.16E+001.14E+001.12E+001.11E+001.10E+001.09E+001.08E+001.07E+00
1.07E+001.06E+001.05E+001.04E+001.04E+001.03E+001.03E+00
952430     SCAPT
4.90E+014.92E+014.89E+014.85E+014.79E+014.81E+014.79E+014.75E+014.68E+014.70E+01
4.66E+014.62E+014.58E+014.60E+014.56E+014.51E+014.47E+01
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952430     SFISS
6.88E-016.87E-016.90E-016.92E-016.95E-016.94E-016.94E-016.95E-016.96E-016.95E-01
6.94E-016.94E-016.92E-016.91E-016.90E-016.90E-016.89E-01
962440     SCAPT
1.70E+011.71E+011.71E+011.71E+011.70E+011.71E+011.71E+011.71E+011.69E+011.70E+01
1.70E+011.69E+011.69E+011.70E+011.70E+011.69E+011.68E+01
962440     SFISS
1.64E+001.65E+001.65E+001.65E+001.65E+001.65E+001.65E+001.64E+001.65E+00
1.65E+001.64E+001.64E+001.64E+001.64E+001.64E+001.63E+00
     0
                                                  38P3U 30 KFK KWO
    13
0.0      0.092    0.91     1.97     3.02     3.12     3.87     4.7      5.5      6.32
6.45     7.27     8.05     9.0


922350     SCAPT
1.02E+011.01E+019.90E+009.89E+009.62E+009.57E+009.61E+009.68E+009.78E+009.56E+00
9.55E+009.63E+009.74E+00
922350     SFISS
4.50E+014.42E+014.31E+014.30E+014.08E+014.03E+014.06E+014.11E+014.18E+014.00E+01
3.98E+014.04E+014.12E+01
922360     SCAPT
8.62E+008.60E+008.03E+007.48E+007.15E+007.16E+006.89E+006.65E+006.43E+006.33E+00
6.33E+006.16E+006.01E+00
922360     SFISS
2.22E-012.21E-012.23E-012.23E-012.22E-012.22E-012.21E-012.21E-012.20E-012.19E-01
2.18E-012.18E-012.17E-01
922380     SCAPT
8.16E-018.16E-018.12E-018.12E-018.11E-018.12E-018.12E-018.13E-018.15E-018.14E-01
8.15E-018.16E-018.17E-01
922380     SFISS
1.13E-011.13E-011.14E-011.14E-011.13E-011.13E-011.13E-011.12E-011.12E-011.11E-01
1.11E-011.11E-011.11E-01
932370     SCAPT
3.18E+013.18E+013.12E+013.09E+013.03E+013.03E+013.02E+013.01E+013.01E+012.97E+01
2.97E+012.97E+012.98E+01
932370     SFISS
5.67E-015.67E-015.71E-015.73E-015.72E-015.71E-015.70E-015.69E-015.68E-015.66E-01
5.65E-015.64E-015.63E-01
942380     SCAPT
3.42E+013.36E+013.30E+013.31E+013.14E+013.10E+013.13E+013.18E+013.24E+013.10E+01
3.08E+013.13E+013.20E+01
942380     SFISS
2.43E+002.42E+002.41E+002.42E+002.38E+002.37E+002.38E+002.40E+002.42E+002.38E+00
2.37E+002.39E+002.41E+00
942390     SCAPT
6.62E+016.53E+015.97E+015.66E+015.24E+015.22E+015.16E+015.13E+015.15E+014.91E+01
4.92E+014.94E+014.99E+01
942390     SFISS
1.20E+021.18E+021.09E+021.04E+029.64E+019.58E+019.50E+019.49E+019.54E+019.08E+01
9.08E+019.14E+019.26E+01
942400     SCAPT
206.     192.5    168.     144.5    133.5    127.     116.5    108.     100.5    96.5
93.      88.5     85.5
942400     SFISS
6.16E-016.16E-016.18E-016.15E-016.12E-016.11E-016.09E-016.06E-016.04E-016.02E-01
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6.01E-016.00E-015.98E-01
942410     SCAPT
4.62E+014.55E+014.29E+014.17E+013.89E+013.86E+013.85E+013.86E+013.90E+013.72E+01
3.71E+013.74E+013.80E+01
942410     SFISS
1.21E+021.19E+021.13E+021.10E+021.04E+021.03E+021.03E+021.03E+021.04E+029.94E+01
9.92E+011.00E+021.02E+02
942420     SCAPT
2.70E+012.72E+012.71E+012.69E+012.69E+012.71E+012.67E+012.62E+012.56E+012.52E+01
2.52E+012.45E+012.38E+01
942420     SFISS
4.51E-014.51E-014.55E-014.55E-014.54E-014.54E-014.53E-014.52E-014.51E-014.50E-01
4.49E-014.48E-014.47E-01
952410     SCAPT
1.06E+021.05E+029.76E+019.34E+018.83E+018.81E+018.71E+018.66E+018.65E+018.35E+01
8.37E+018.37E+018.41E+01
952410     SCAPT1
1.31E+011.29E+011.21E+011.15E+011.09E+011.09E+011.08E+011.07E+011.07E+011.03E+01
1.03E+011.03E+011.04E+01
952410     SFISS
1.25E+001.24E+001.20E+001.17E+001.13E+001.13E+001.12E+001.11E+001.11E+001.09E+00
1.09E+001.08E+001.08E+00
952430     SCAPT
5.03E+015.05E+014.98E+014.90E+014.85E+014.87E+014.82E+014.76E+014.70E+014.66E+01
4.68E+014.62E+014.56E+01
952430     SFISS
6.76E-016.76E-016.81E-016.82E-016.79E-016.77E-016.77E-016.75E-016.73E-016.70E-01
6.68E-016.67E-016.66E-01
962440     SCAPT
1.72E+011.73E+011.72E+011.71E+011.73E+011.74E+011.73E+011.72E+011.71E+011.72E+01
1.73E+011.71E+011.70E+01
962440     SFISS
1.64E+001.65E+001.65E+001.65E+001.65E+001.66E+001.65E+001.64E+001.64E+001.64E+00
1.64E+001.63E+001.62E+00
     0
                                                  38P3U 28 KFK KWO
    14
0.0      0.55     0.83     1.61     2.40     3.3      3.39     4.12     4.9      5.68
6.47     6.57     7.35     8.13     9.0

922350     SCAPT
1.06E+011.05E+011.03E+011.02E+011.02E+019.94E+009.89E+009.94E+001.00E+011.01E+01
9.89E+009.87E+009.95E+001.01E+01
922350     SFISS
4.76E+014.68E+014.54E+014.48E+014.52E+014.28E+014.23E+014.27E+014.33E+014.40E+01
4.21E+014.18E+014.25E+014.34E+01
922360     SCAPT
8.63E+008.65E+008.13E+007.71E+007.35E+007.13E+007.14E+006.90E+006.68E+006.49E+00
6.40E+006.41E+006.25E+006.10E+00
922360     SFISS
2.19E-012.19E-012.21E-012.21E-012.21E-012.20E-012.20E-012.19E-012.19E-012.18E-01
2.17E-012.16E-012.16E-012.16E-01
922380     SCAPT
8.30E-018.30E-018.25E-018.24E-018.24E-018.24E-018.24E-018.25E-018.26E-018.28E-01
8.27E-018.28E-018.28E-018.30E-01
922380     SFISS
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1.12E-011.12E-011.13E-011.13E-011.13E-011.12E-011.12E-011.12E-011.11E-011.11E-01
1.10E-011.10E-011.10E-011.10E-01
932370     SCAPT
3.26E+013.26E+013.20E+013.16E+013.15E+013.09E+013.09E+013.08E+013.08E+013.09E+01
3.04E+013.04E+013.04E+013.05E+01
932370     SFISS
5.61E-015.61E-015.66E-015.67E-015.67E-015.66E-015.65E-015.65E-015.64E-015.62E-01
5.61E-015.60E-015.59E-015.58E-01
942380     SCAPT
3.63E+013.56E+013.48E+013.45E+013.50E+013.31E+013.26E+013.30E+013.36E+013.43E+01
3.26E+013.24E+013.30E+013.38E+01
942380     SFISS
2.49E+002.47E+002.46E+002.45E+002.47E+002.43E+002.41E+002.43E+002.44E+002.46E+00
2.42E+002.42E+002.43E+002.46E+00
942390     SCAPT
6.95E+016.87E+016.27E+015.95E+015.83E+015.44E+015.41E+015.38E+015.37E+015.40E+01
5.15E+015.16E+015.19E+015.25E+01
942390     SFISS
1.26E+021.24E+021.14E+021.09E+021.07E+021.00E+029.97E+019.93E+019.95E+011.00E+02
9.54E+019.54E+019.62E+019.76E+01
942400     SCAPT
209.     196.5    175.     155.     136.5    131.     125.5    116.5    113.     105.5
96.5     93.5     89.5     87.0
942400     SFISS
6.10E-016.10E-016.11E-016.08E-016.06E-016.05E-016.03E-016.01E-015.98E-01
5.97E-015.96E-015.95E-015.93E-01
942410     SCAPT
4.87E+014.81E+014.51E+014.36E+014.33E+014.06E+014.03E+014.03E+014.05E+014.10E+01
3.90E+013.90E+013.93E+014.00E+01
942410     SFISS
1.27E+021.26E+021.19E+021.15E+021.15E+021.08E+021.07E+021.07E+021.08E+021.09E+02
1.04E+021.04E+021.05E+021.07E+02
942420     SCAPT
2.71E+012.73E+012.73E+012.72E+012.70E+012.70E+012.71E+012.67E+012.61E+012.55E+01
2.51E+012.52E+012.45E+012.38E+01
942420     SFISS
4.46E-014.46E-014.50E-014.51E-014.51E-014.50E-014.49E-014.49E-014.48E-014.47E-01
4.46E-014.45E-014.44E-014.43E-01
952410     SCAPT
1.10E+021.09E+021.01E+029.73E+019.55E+019.09E+019.07E+019.00E+018.97E+018.98E+01
8.67E+018.69E+018.70E+018.75E+01
952410     SCAPT1
1.36E+011.35E+011.25E+011.20E+011.18E+011.12E+011.12E+011.11E+011.11E+011.11E+01
1.07E+011.07E+011.08E+011.08E+01
952410     SFISS
1.27E+001.27E+001.22E+001.19E+001.17E+001.14E+001.14E+001.13E+001.13E+001.12E+00
1.10E+001.10E+001.10E+001.10E+00
952430     SCAPT
5.06E+015.09E+015.03E+014.97E+014.91E+014.88E+014.90E+014.85E+014.79E+014.74E+01
4.70E+014.71E+014.66E+014.60E+01
952430     SFISS
6.70E-016.68E-016.74E-016.76E-016.76E-016.73E-016.72E-016.71E-016.69E-016.67E-01
6.65E-016.63E-016.62E-016.60E-01
962440     SCAPT
1.72E+011.73E+011.73E+011.73E+011.72E+011.74E+011.74E+011.74E+011.73E+011.71E+01
1.73E+011.73E+011.72E+011.71E+01
962440     SFISS
```

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1.64E+001.65E+001.65E+001.65E+001.64E+001.65E+001.65E+001.65E+001.64E+001.64E+00
1.64E+001.64E+001.63E+001.62E+00
     0
                                                  36P3U 34 KFK BIB
    15
0.0      0.042    0.42     1.13     1.98     2.84     2.89     3.27     3.98     4.84
5.70     5.74     6.12     6.83     7.69     8.55


922350     SCAPT
9.70E+009.61E+009.55E+009.50E+009.52E+009.27E+009.21E+009.26E+009.34E+009.46E+00
9.25E+009.19E+009.26E+009.36E+009.49E+00
922350     SFISS
4.22E+014.14E+014.10E+014.07E+014.09E+013.88E+013.82E+013.86E+013.91E+014.00E+01
3.82E+013.77E+013.82E+013.89E+013.99E+01
922360     SCAPT
8.57E+008.60E+008.30E+007.86E+007.42E+007.18E+007.20E+007.04E+006.79E+006.54E+00
6.41E+006.43E+006.33E+006.17E+006.01E+00
922360     SFISS
2.26E-012.26E-012.27E-012.28E-012.28E-012.27E-012.26E-012.26E-012.26E-012.25E-01
2.24E-012.23E-012.23E-012.22E-012.22E-01
932370     SCAPT
3.09E+013.08E+013.06E+013.03E+013.01E+012.96E+012.95E+012.95E+012.95E+012.95E+01
2.91E+012.91E+012.91E+012.92E+012.93E+01
932370     SFISS
5.78E-015.77E-015.80E-015.82E-015.82E-015.81E-015.80E-015.80E-015.79E-015.78E-01
5.76E-015.75E-015.74E-015.73E-015.72E-01
922380     SCAPT
8.18E-018.18E-018.16E-018.14E-018.15E-018.15E-018.16E-018.16E-018.17E-018.18E-01
8.19E-018.20E-018.20E-018.22E-018.23E-01
922380     SFISS
1.16E-011.15E-011.16E-011.16E-011.16E-011.16E-011.15E-011.15E-011.15E-011.15E-01
1.14E-011.14E-011.14E-011.13E-011.13E-01
942380     SCAPT
3.19E+013.13E+013.11E+013.10E+013.13E+012.97E+012.92E+012.96E+013.01E+013.09E+01
2.94E+012.90E+012.94E+013.01E+013.09E+01
942380     SFISS
2.38E+002.36E+002.36E+002.37E+002.38E+002.34E+002.33E+002.34E+002.35E+002.38E+00
2.34E+002.33E+002.34E+002.36E+002.38E+00
942390     SCAPT
6.25E+016.18E+015.93E+015.62E+015.43E+015.05E+015.02E+015.00E+014.99E+015.00E+01
4.76E+014.73E+014.76E+014.80E+014.86E+01
942390     SFISS
1.13E+021.11E+021.07E+021.02E+029.94E+019.26E+019.18E+019.17E+019.17E+019.23E+01
8.77E+018.71E+018.77E+018.86E+019.01E+01
942400     SCAPT
204.5    197.5    181.     161.5    143.5    134.5    131.     123.     113.     104.
99.5     97.      92.5     88.      85.
942400     SFISS
6.26E-016.26E-016.27E-016.27E-016.24E-016.21E-016.21E-016.19E-016.17E-016.14E-01
6.12E-016.12E-016.11E-016.09E-016.06E-01
942410     SCAPT
4.34E+014.29E+014.17E+014.04E+013.97E+013.73E+013.69E+013.70E+013.72E+013.76E+01
3.58E+013.54E+013.58E+013.62E+013.69E+01
942410     SFISS
1.14E+021.12E+021.10E+021.07E+021.05E+029.91E+019.80E+019.84E+019.90E+011.00E+02
9.56E+019.46E+019.55E+019.69E+019.87E+01
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942420     SCAPT
2.68E+012.70E+012.69E+012.68E+012.67E+012.68E+012.70E+012.68E+012.65E+012.60E+01
2.58E+012.59E+012.56E+012.51E+012.44E+01
942420     SFISS
4.60E-014.60E-014.62E-014.63E-014.63E-014.62E-014.61E-014.61E-014.61E-014.60E-01
4.58E-014.57E-014.56E-014.56E-014.54E-01
952410     SCAPT
1.01E+021.00E+029.70E+019.29E+019.02E+018.57E+018.54E+018.51E+018.47E+018.46E+01
8.17E+018.14E+018.16E+018.19E+018.24E+01
952410     SCAPT1
1.25E+011.24E+011.20E+011.15E+011.12E+011.06E+011.06E+011.05E+011.05E+011.05E+01
1.01E+011.01E+011.01E+011.01E+011.02E+01
952410     SFISS
1.23E+001.23E+001.20E+001.18E+001.16E+001.12E+001.12E+001.12E+001.11E+001.11E+00
1.09E+001.08E+001.08E+001.08E+001.08E+00
952430     SCAPT
4.97E+015.00E+014.96E+014.91E+014.85E+014.82E+014.84E+014.81E+014.76E+014.70E+01
4.67E+014.69E+014.66E+014.61E+014.55E+01
952430     SFISS
6.92E-016.90E-016.93E-016.95E-016.96E-016.93E-016.91E-016.91E-016.90E-016.88E-01
6.84E-016.83E-016.82E-016.81E-016.79E-01
962440     SCAPT
1.71E+011.72E+011.71E+011.71E+011.70E+011.72E+011.73E+011.72E+011.71E+011.70E+01
1.71E+011.72E+011.71E+011.70E+011.69E+01
962440     SFISS
1.65E+001.65E+001.65E+001.65E+001.65E+001.66E+001.66E+001.66E+001.65E+001.64E+00
1.65E+001.65E+001.64E+001.64E+001.63E+00
     0
                                                  40P3U 36 KFK BIB
    15
0.0      0.047    0.47     1.25     2.21     3.16     3.21     3.64     4.43     5.38
6.33     6.38     6.81     7.60     8.55     9.50


922350     SCAPT
9.44E+009.35E+009.29E+009.25E+009.28E+009.06E+008.99E+009.05E+009.13E+009.26E+00
9.07E+009.02E+009.09E+009.19E+009.32E+00
922350     SFISS
4.05E+013.98E+013.94E+013.91E+013.93E+013.74E+013.68E+013.72E+013.78E+013.87E+01
3.70E+013.66E+013.71E+013.78E+013.87E+01
922360     SCAPT
8.56E+008.58E+008.25E+007.77E+007.30E+007.02E+007.04E+006.88E+006.62E+006.36E+00
6.23E+006.25E+006.15E+005.99E+005.82E+00
922360     SFISS
2.29E-012.28E-012.29E-012.30E-012.29E-012.28E-012.28E-012.27E-012.27E-012.26E-01
2.25E-012.24E-012.24E-012.23E-012.22E-01
932370     SCAPT
3.03E+013.03E+013.00E+012.97E+012.96E+012.91E+012.90E+012.90E+012.90E+012.91E+01
2.87E+012.86E+012.87E+012.87E+012.88E+01
932370     SFISS
5.83E-015.82E-015.85E-015.86E-015.86E-015.85E-015.84E-015.84E-015.83E-015.81E-01
5.78E-015.77E-015.77E-015.76E-015.74E-01
922380     SCAPT
8.11E-018.11E-018.10E-018.08E-018.09E-018.10E-018.10E-018.11E-018.12E-018.14E-01
8.15E-018.16E-018.16E-018.17E-018.19E-01
922380     SFISS
1.17E-011.16E-011.17E-011.17E-011.17E-011.16E-011.16E-011.16E-011.16E-011.15E-01
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1.14E-011.14E-011.14E-011.14E-011.13E-01
942380     SCAPT
3.06E+013.00E+012.98E+012.97E+013.01E+012.86E+012.81E+012.85E+012.90E+012.98E+01
2.85E+012.81E+012.85E+012.92E+013.00E+01
942380     SFISS
2.35E+002.33E+002.33E+002.33E+002.35E+002.31E+002.30E+002.31E+002.33E+002.35E+00
2.32E+002.31E+002.32E+002.34E+002.36E+00
942390     SCAPT
6.04E+015.97E+015.71E+015.40E+015.22E+014.86E+014.83E+014.82E+014.81E+014.83E+01
4.60E+014.58E+014.61E+014.65E+014.72E+01
942390     SFISS
1.09E+021.08E+021.03E+029.83E+019.54E+018.91E+018.83E+018.83E+018.84E+018.91E+01
8.48E+018.42E+018.48E+018.58E+018.73E+01
942400     SCAPT
203.     194.     176.     155.5    136.5    127.     123.     115.     110.5    102.
94.      91.5     88.      84.5     82.
942400     SFISS
6.31E-016.30E-016.32E-016.31E-016.27E-016.24E-016.24E-016.22E-016.20E-016.17E-01
6.14E-016.14E-016.13E-016.11E-016.08E-01
942410     SCAPT
4.19E+014.13E+014.02E+013.88E+013.82E+013.59E+013.55E+013.56E+013.58E+013.63E+01
3.46E+013.43E+013.46E+013.51E+013.58E+01
942410     SFISS
1.10E+021.08E+021.05E+021.03E+021.01E+029.55E+019.45E+019.49E+019.56E+019.69E+01
9.26E+019.17E+019.26E+019.39E+019.58E+01
942420     SCAPT
2.67E+012.69E+012.68E+012.68E+012.66E+012.67E+012.68E+012.66E+012.63E+012.57E+01
2.54E+012.55E+012.52E+012.46E+012.39E+01
942420     SFISS
4.64E-014.64E-014.66E-014.67E-014.67E-014.65E-014.64E-014.64E-014.63E-014.62E-01
4.60E-014.59E-014.58E-014.57E-014.56E-01
952410     SCAPT
9.83E+019.77E+019.43E+019.01E+018.75E+018.32E+018.29E+018.27E+018.23E+018.23E+01
7.95E+017.93E+017.95E+017.98E+018.04E+01
952410     SCAPT1
1.21E+011.21E+011.17E+011.11E+011.08E+011.03E+011.03E+011.02E+011.02E+011.02E+01
9.83E+009.80E+009.83E+009.86E+009.93E+00
952410     SFISS
1.22E+001.21E+001.19E+001.16E+001.14E+001.11E+001.11E+001.10E+001.10E+001.10E+00
1.07E+001.07E+001.07E+001.07E+001.07E+00
952430     SCAPT
4.95E+014.97E+014.93E+014.88E+014.81E+014.77E+014.79E+014.76E+014.71E+014.65E+01
4.61E+014.63E+014.60E+014.55E+014.49E+01
952430     SFISS
6.98E-016.97E-016.99E-017.01E-017.01E-016.97E-016.96E-016.96E-016.94E-016.92E-01
6.87E-016.86E-016.85E-016.83E-016.81E-01
962440     SCAPT
1.70E+011.71E+011.70E+011.69E+011.71E+011.72E+011.71E+011.70E+011.69E+01
1.71E+011.71E+011.71E+011.69E+011.68E+01
962440     SFISS
1.65E+001.66E+001.66E+001.65E+001.65E+001.66E+001.66E+001.66E+001.65E+001.64E+00
1.64E+001.65E+001.64E+001.63E+001.63E+00
     0
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C.2 Graphic Comparison of the Burnup-Dependent Actinide Cross Sections for the PWR BIBLIS (3.2 w/o U-235, Burnup 33 GWd/tU), Generated with KfK Data and Methods, with Corresponding Cross Sections for the ORNL Reference Reactor (3.2 w/o U-235, Burnup 33 GWd/tU), Generated with the ORNL Method and ENDFB/B-IV Data (cf. Chapter 3)

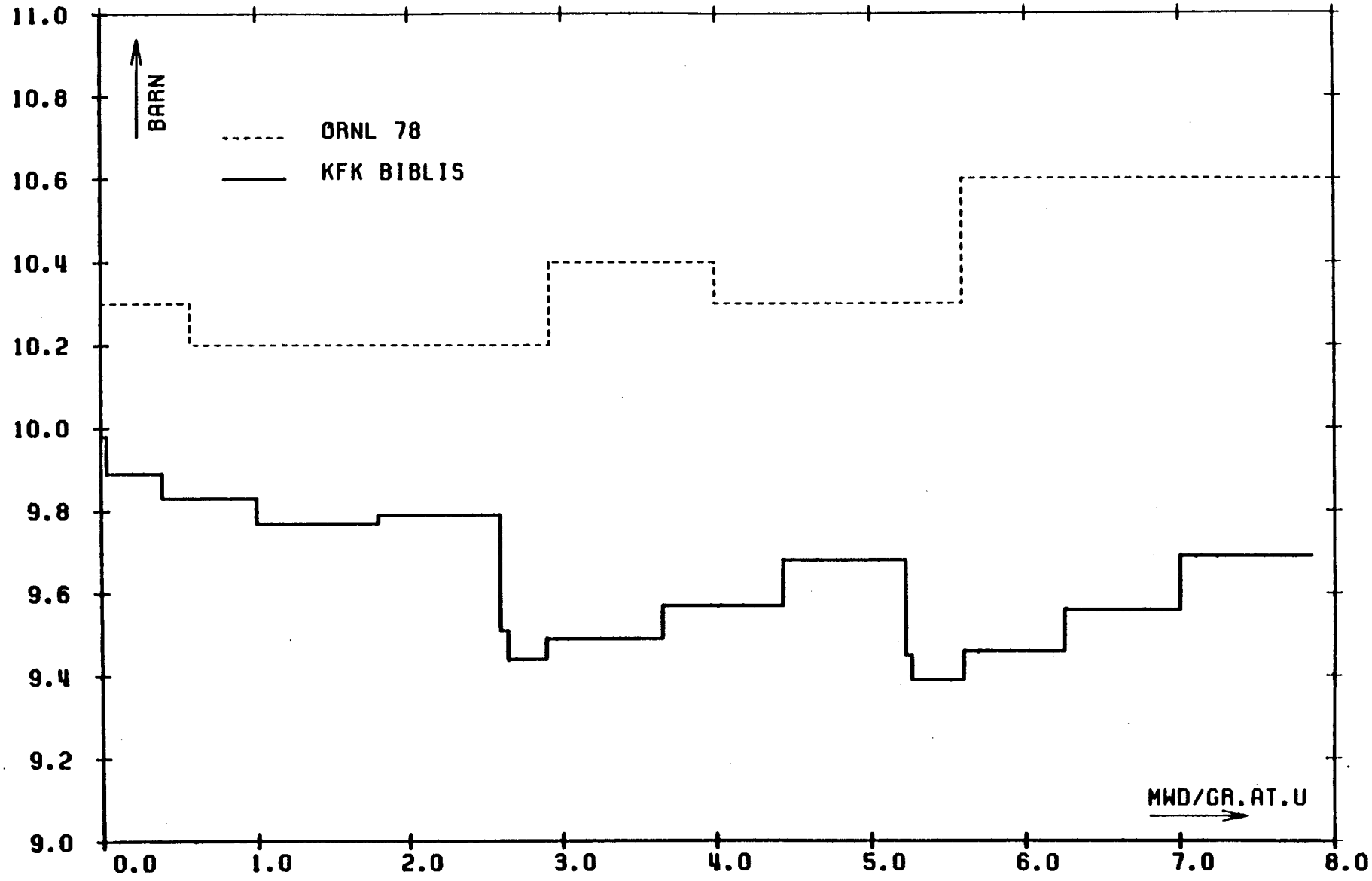

Abb.C.2-1 U 235 SCAPT

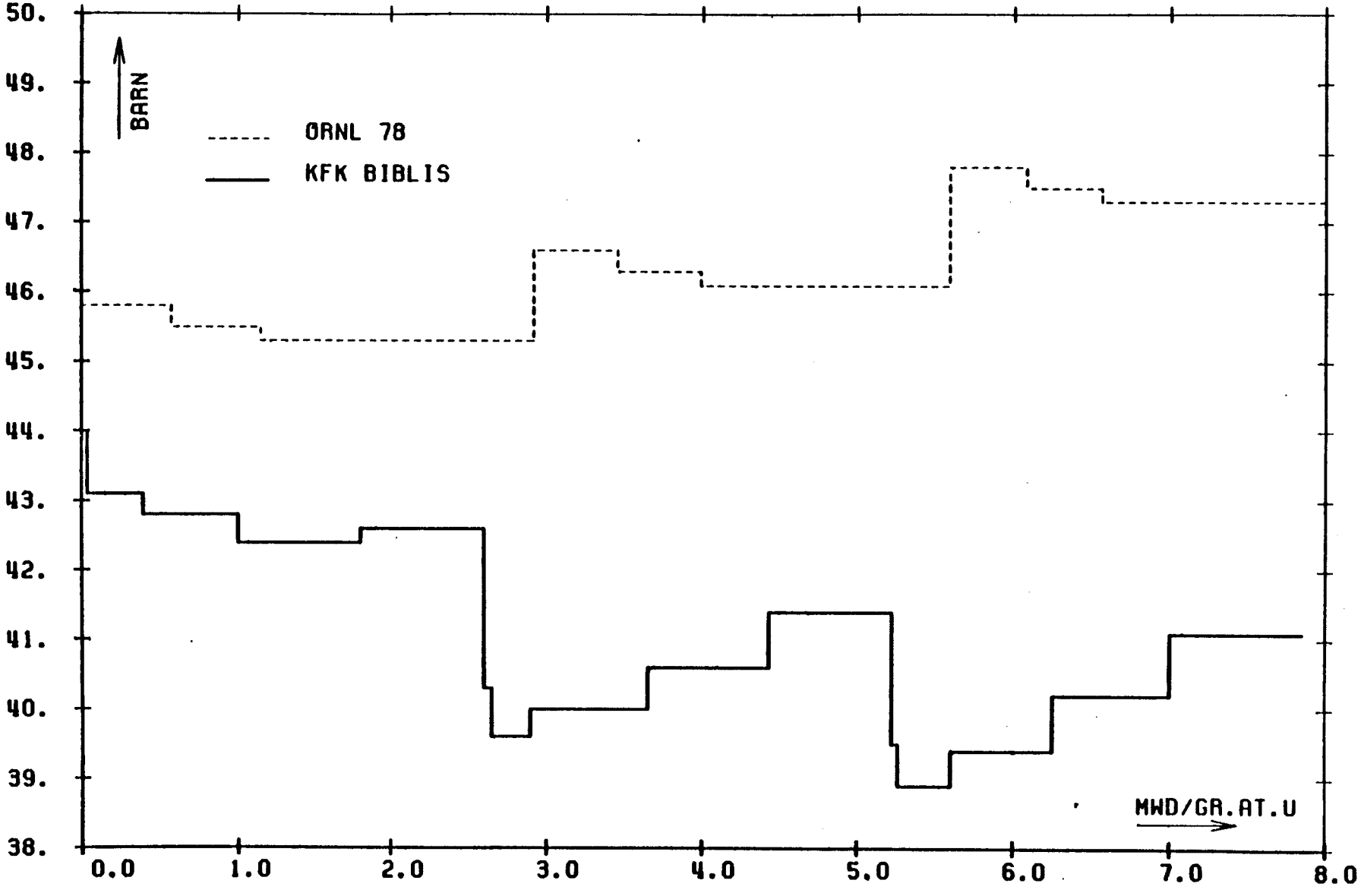

Abb.C.2-2 U 235 SFISS

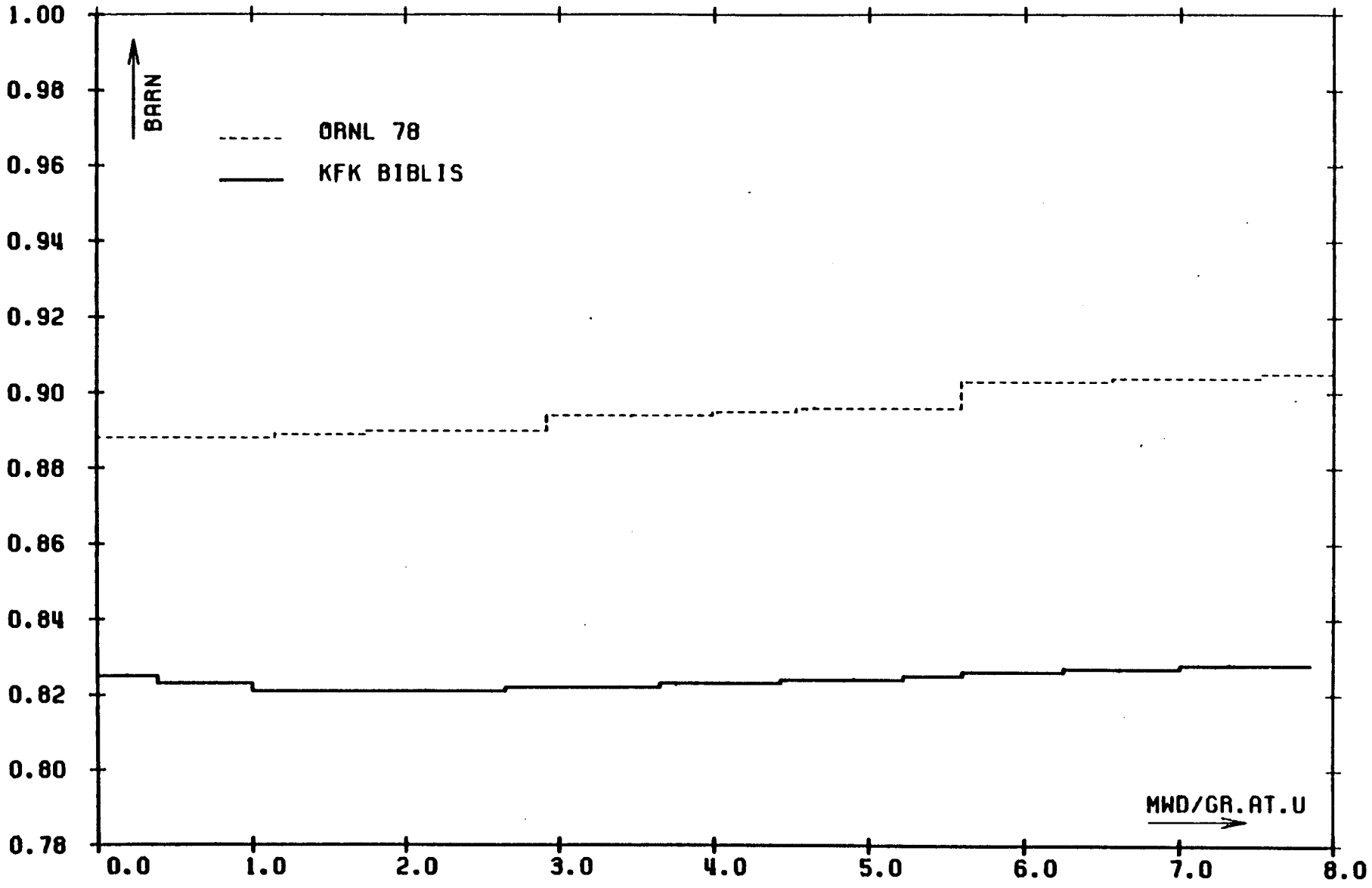

Abb.C.2-3 U 238 SCAPT

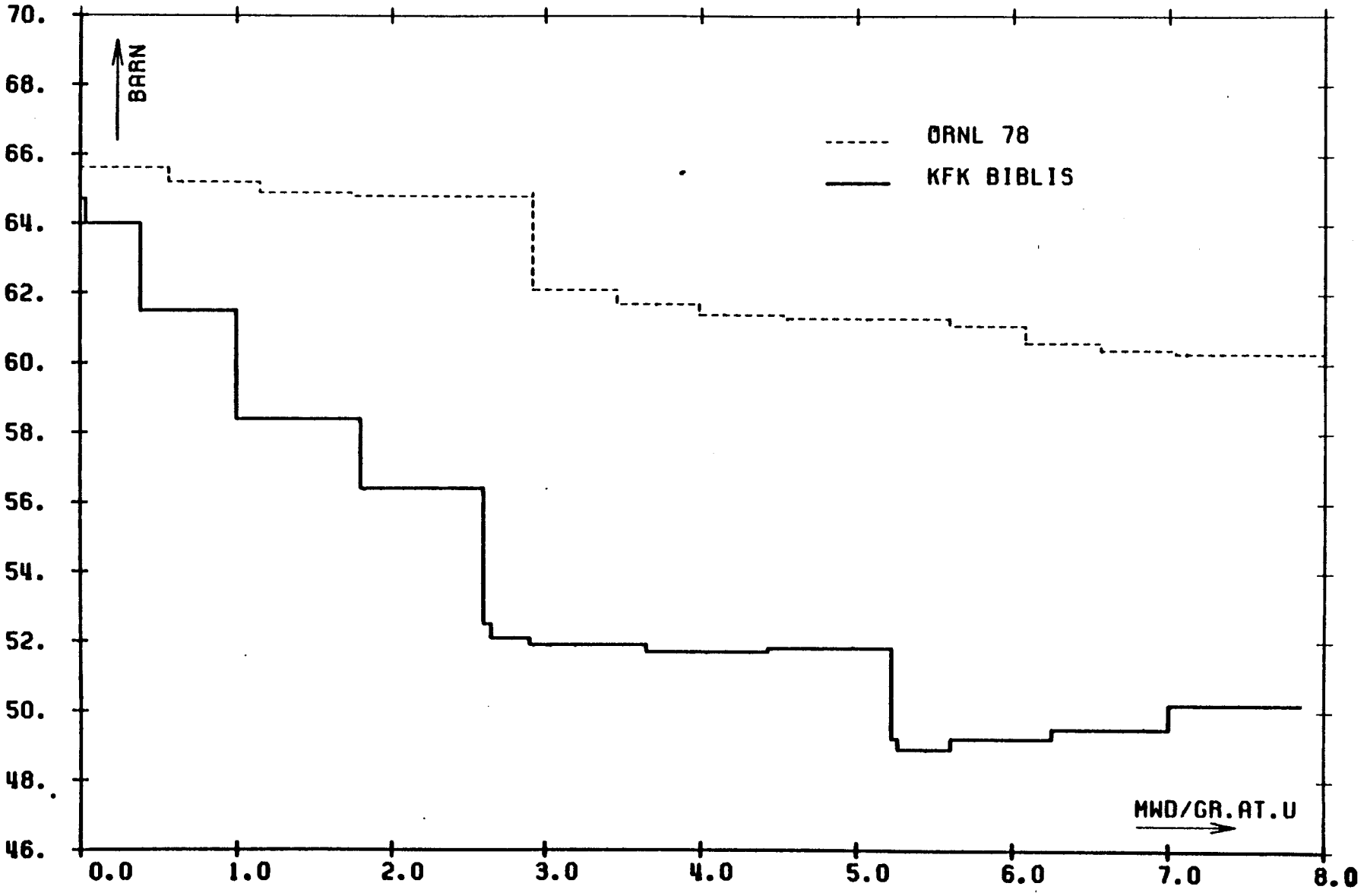

Abb.C.2-4 PU239 SCAPT

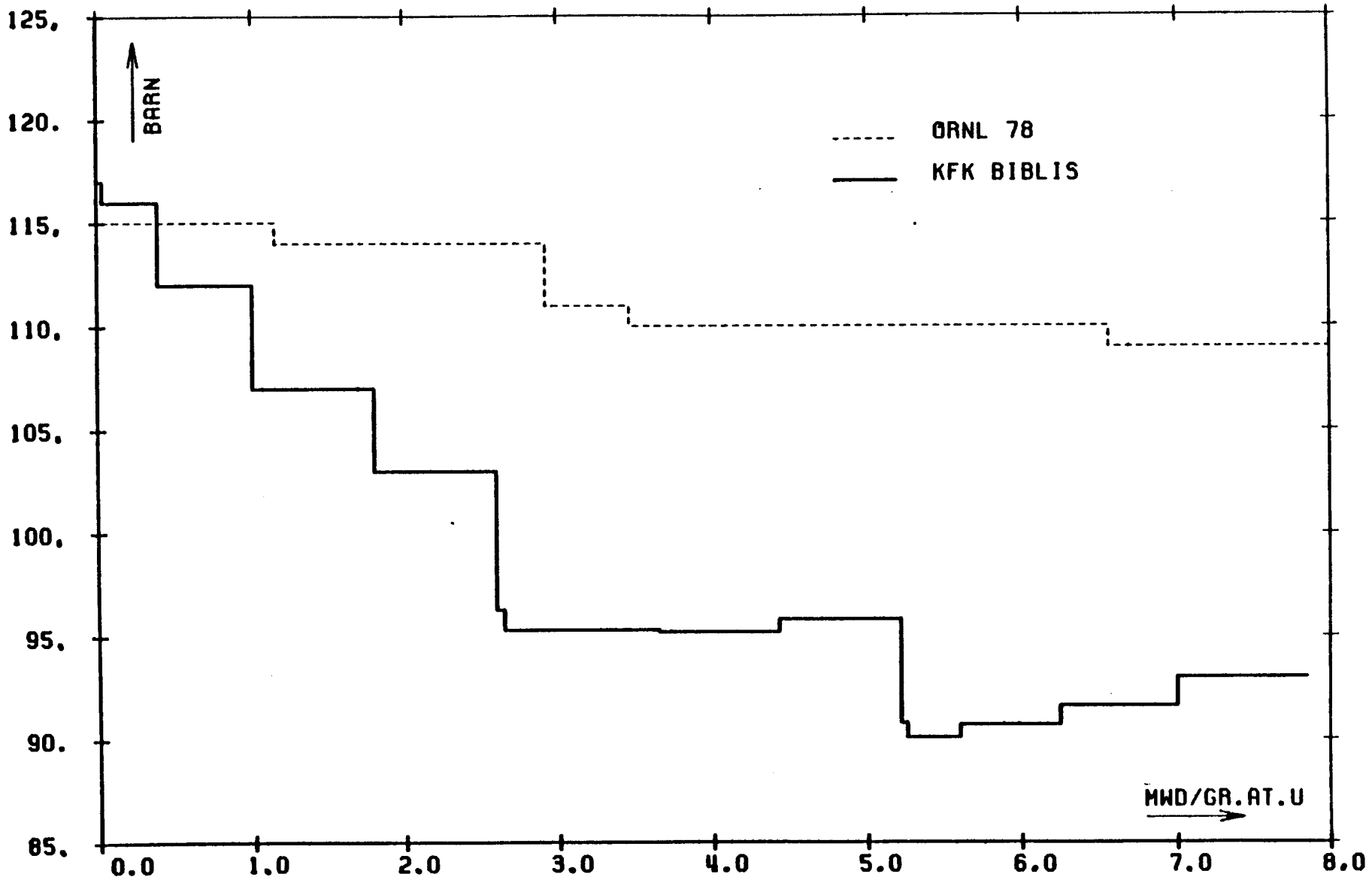

Abb.C.2-5 PU239 SFISS

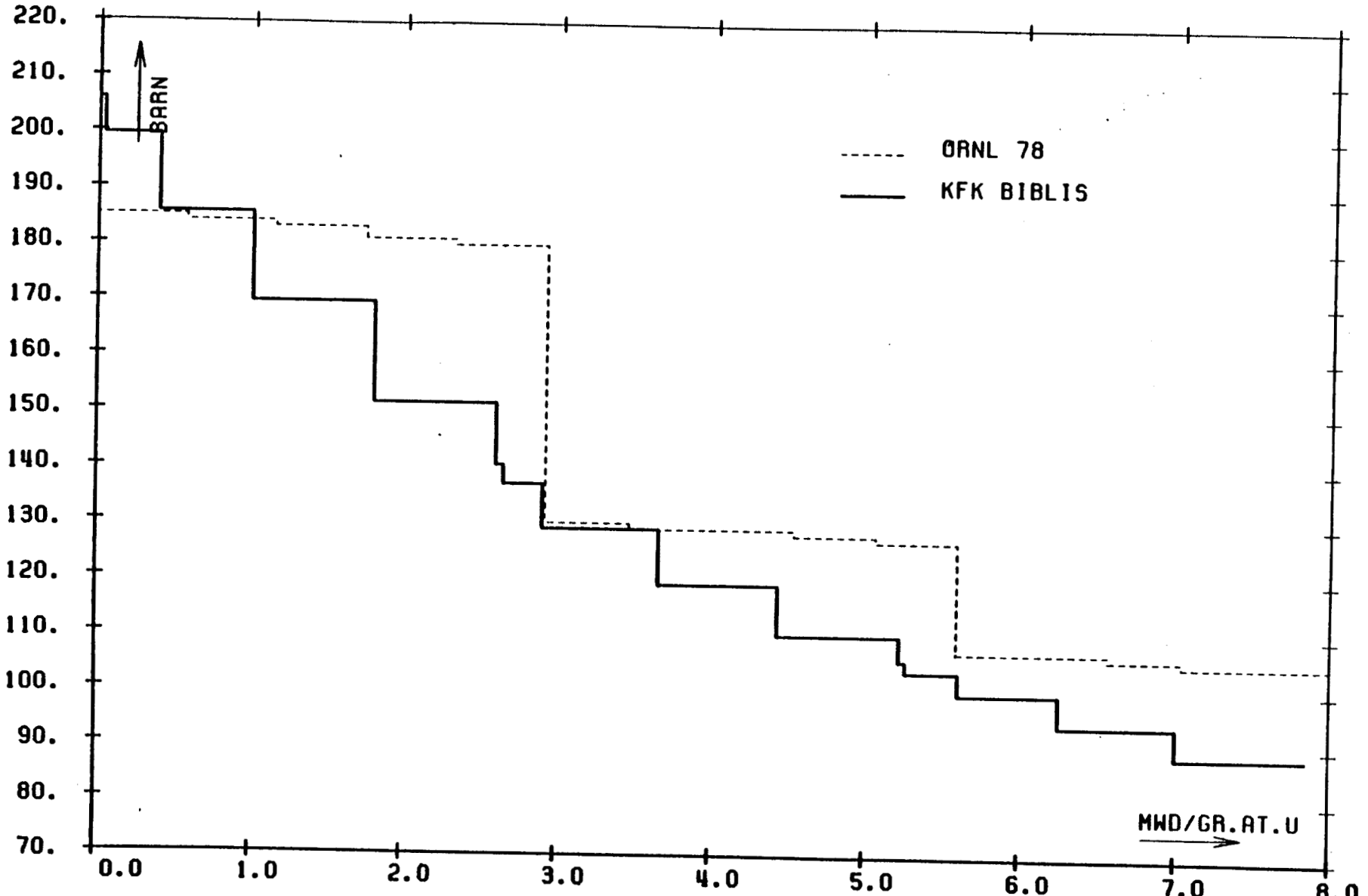

Abb.C.2-6 PU240 SCAPT

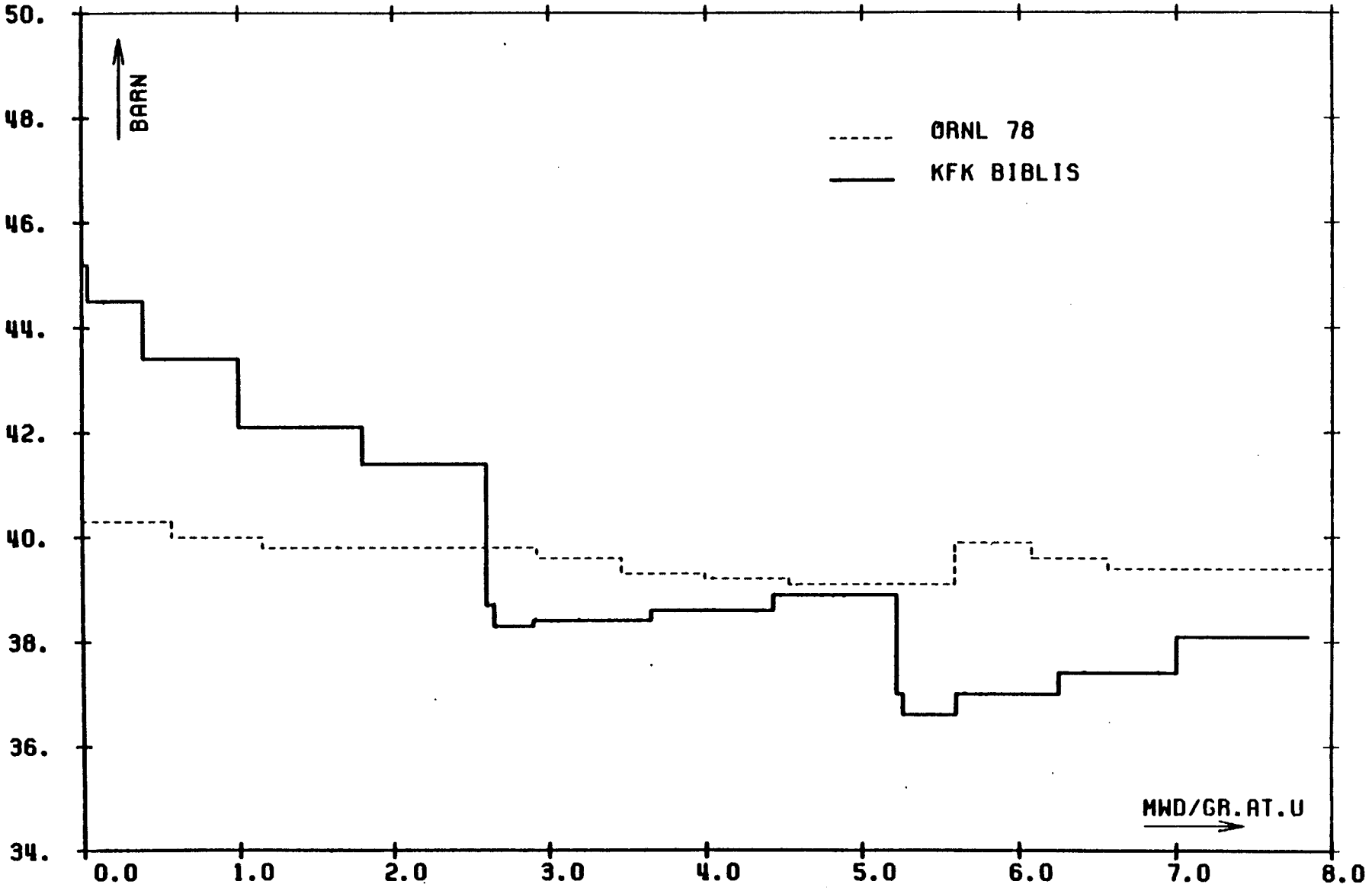

Abb.C.2-7 PU241 SCAPT

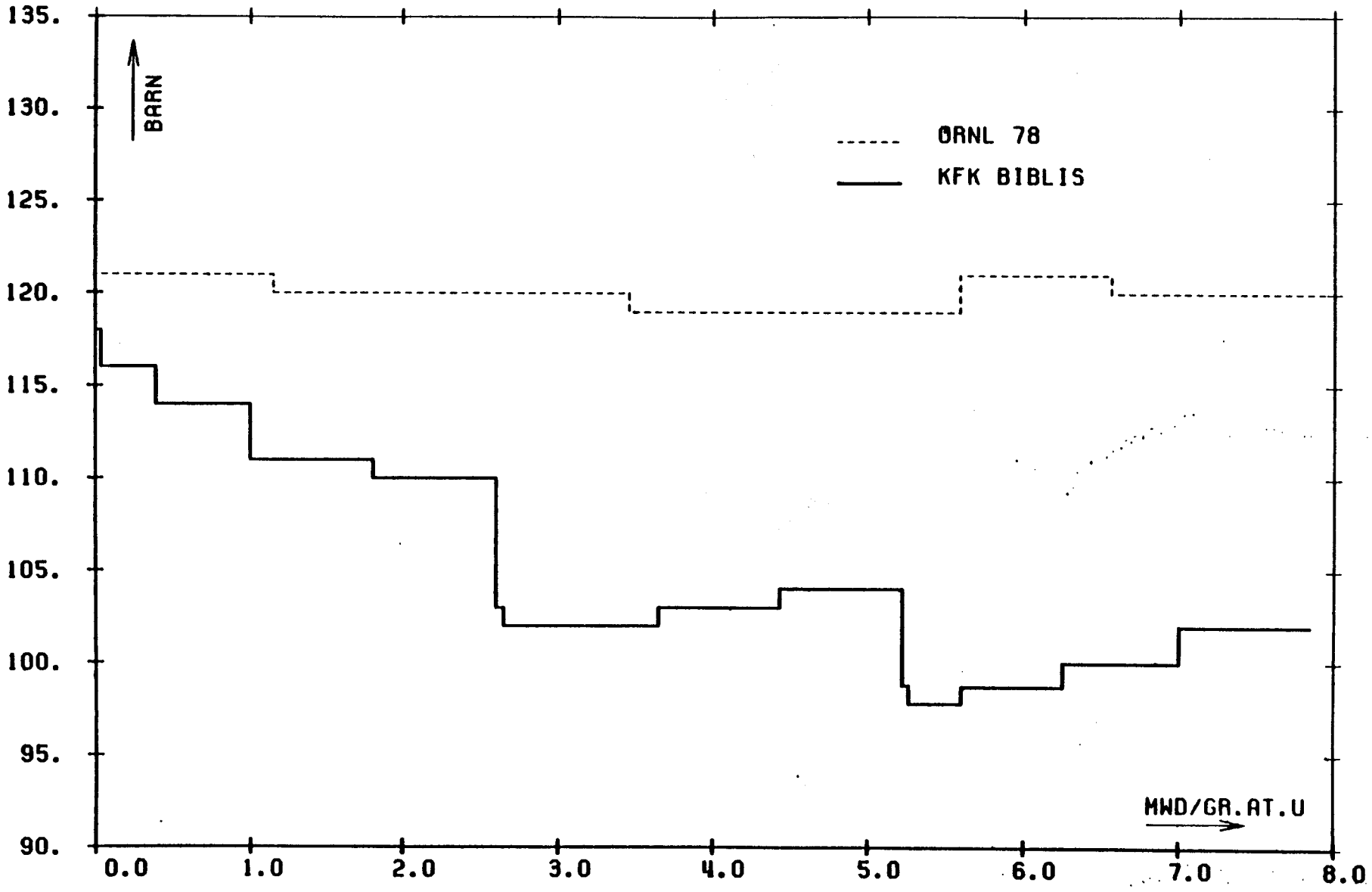

Abb.C.2-8 PU241 SFISS

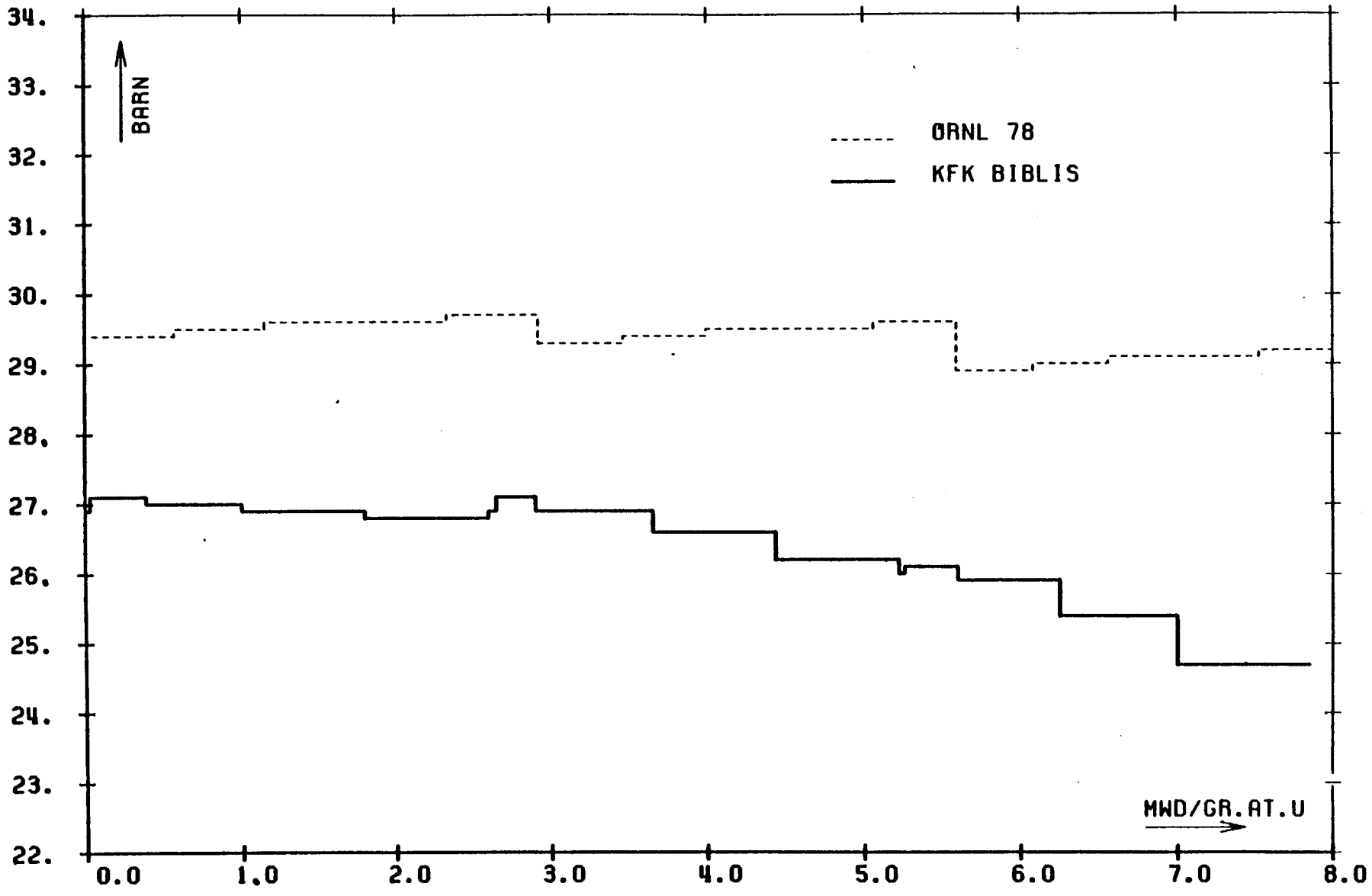

Abb.C.2-9 PU242 SCAPT

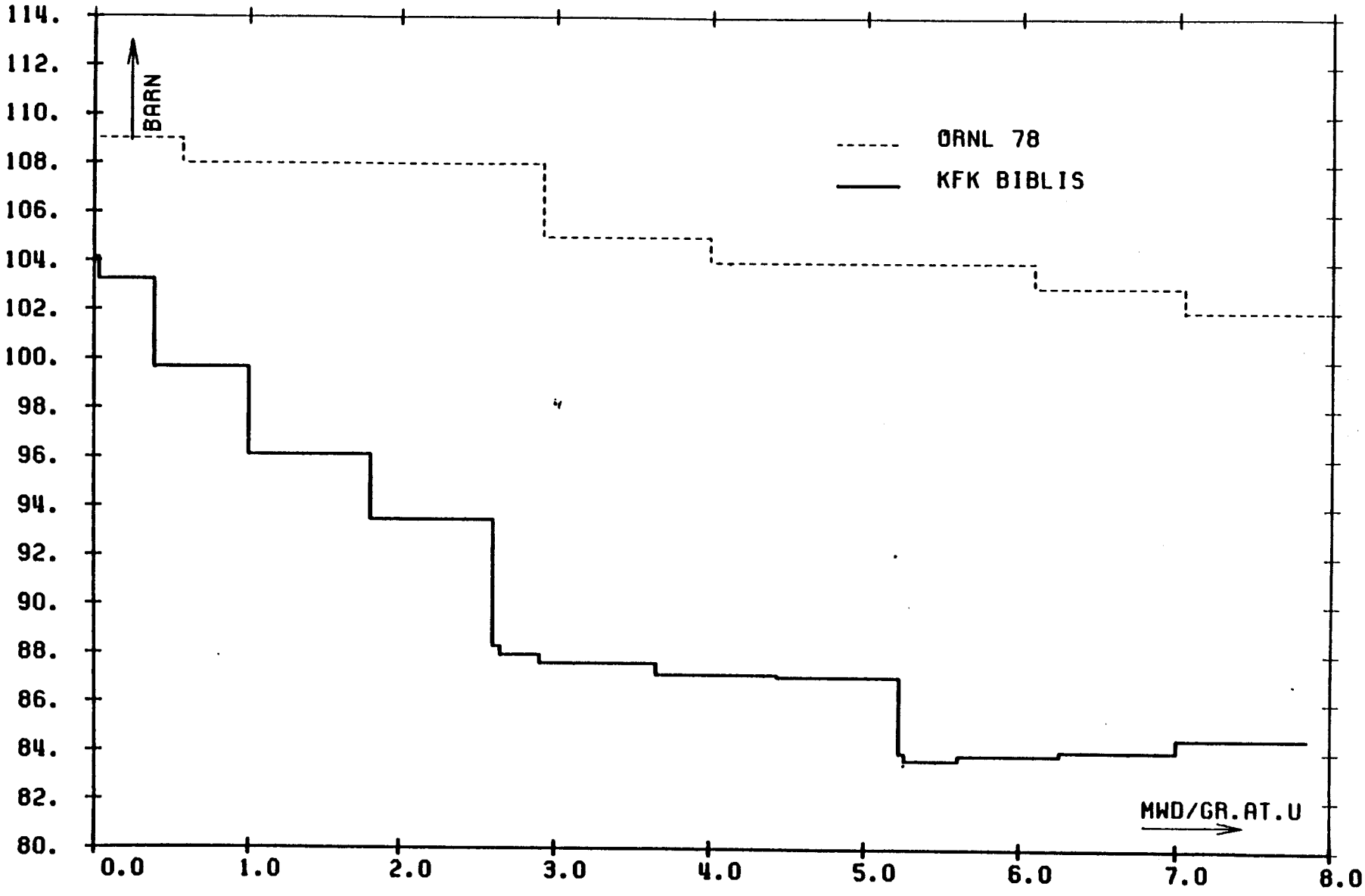

Abb.C.2-10 AM241 SCAPT

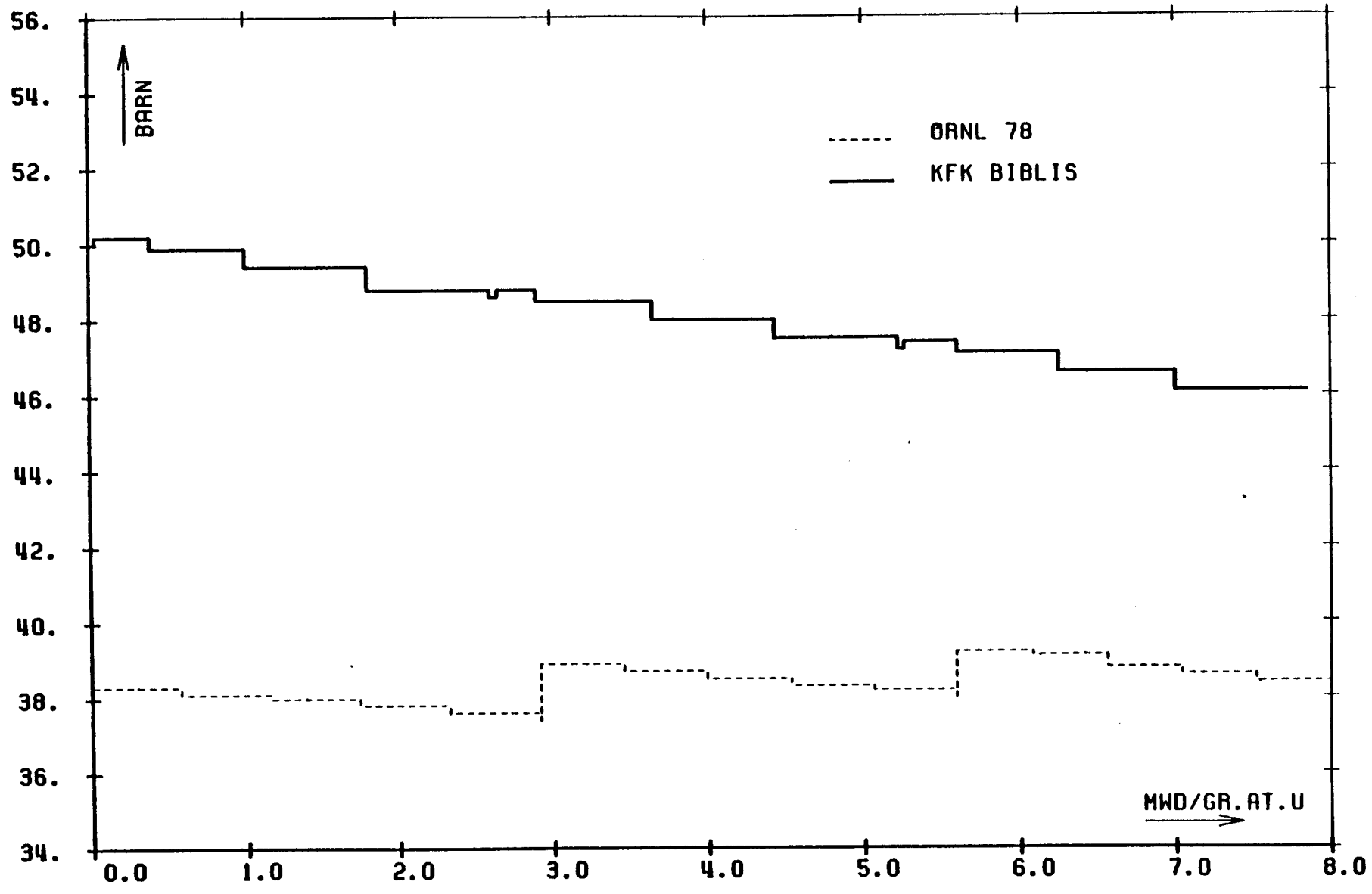

Abb.C.2-11 AM243 SCAPT

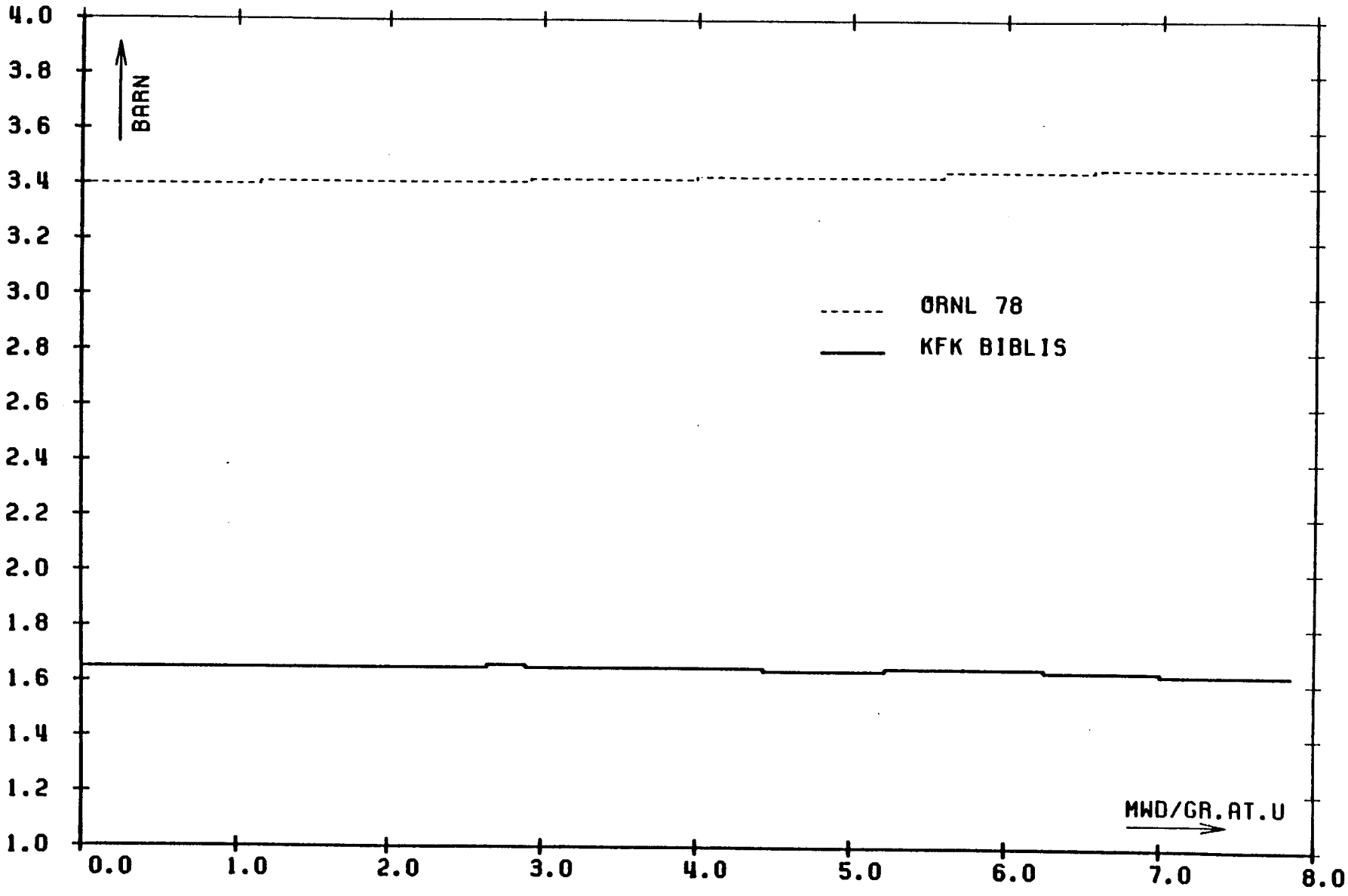

Abb.C.2-12 CM244 SCAPT

D. Consistently Calculated Fuel and Waste Inventories for PWR of the BIBLIS Type

In the following the results of consistent KORIGEN calculations for PWR of the Biblis type with an initial enrichment of 3.2, 3.4 and 3.6 w/o U-235 (discharge burnups used as a basis: 33, 36 and 40 GWd/tU) are listed. The fuel and waste inventories are to be considered representative for German PWR with regard to future reprocessing. The fuel exists in each case in the form of UO_2. The initial inventory is in each case 100 t U. The initial inventory of cladding and structural materials was compiled according to DWK data [55] and can be found in Table D.1. The thermal power is 3300, 3600 and 4000 MW_{th} with operation of 3 x 333 d full load and 40 d each zero load (cf. Chap. 3).

For each fuel type there follow 16 pages of tables. Listed is in each case the decay of the fuel from discharge from the reactor to reprocessing (7 years later) and the decay of the waste up to 10^7 years after discharge, whereby it is assumed that 1.0% of the U or Pu goes into the waste in each case.

For the purpose of limiting the extent of the results presented, the most important tables were selected, on the one hand (see summary table D.3), and on the other hand the nuclides or elements within the tables were subject to a cutoff (Table D.2).

Table D.1. Cladding and structural materials from the PWR BIBLIS that were used [55]

Element	Concentration [g/tU]	Element	Concentration [g/tU]
B	0.935	Fe	75494.
C	110.7	Co	119.8
O*)	1542.8	Ni	25626.8
Al	1185.8	Cu	9.7
Si	1079.5	Zr	288600.
P	48.8	Nb	508.7
S	32.9	Mo	504.8
Ti	846.9	Sn	4269.
Cr	23390.	Ta	408.8
Mn	2096.		

* In the calculations the oxygen of the fuel is added.

Table D.2. Cut-off used (is used in each case for the first time interval).

Property	Cut-Off
Concentration	10^{-3} g/tU
Activity	10^{-3} Ci/tU
Alpha activity	10^{-3} Ci/tU
Total thermal power	10^{-3} W/tU
Gamma power	10^{-3} W/tU

Table D.3. Overview of the tables of results listed for each fuel type

	Actinides	Fission Products	Light Elements	Total
Nuclide concentration	X			X
Element concentration	X			
Total thermal power	X	X	X	X
Gamma power	X	X	X	X
Total activity	X	X	X	X
Alpha activity	X			X
Gamma spectrum	X	X	X	
n-emission rate	X			

D.1 BIBLIS Type 3.2 w/o U-235, Burnup 33 GWd/tU

[see following tables]

FUEL DECAY BIBLIS-TYPE 3.2% U235

ACTINIDES

BURNUP=3.30E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE 4	0.0	2.79E-01	4.83E-01	5.61E-01	7.05E-01	7.98E-01
TH230	0.0	8.13E-04	1.22E-03	1.63E-03	2.87E-03	3.71E-03
U 233	0.0	7.03E-03	7.18E-03	7.34E-03	7.77E-03	8.06E-03
U 234	2.43E+02	1.45E+02	1.46E+02	1.47E+02	1.50E+02	1.53E+02
U 235	3.20E+04	8.25E+03	8.25E+03	8.25E+03	8.25E+03	8.26E+03
U 236	0.0	4.05E+03	4.05E+03	4.05E+03	4.05E+03	4.05E+03
U 238	9.68E+05	9.43E+05	9.43E+05	9.43E+05	9.43E+05	9.43E+05
NP236M	0.0	1.95E-03	1.95E-03	1.95E-03	1.95E-03	1.95E-03
NP237	0.0	4.26E+02	4.35E+02	4.35E+02	4.37E+02	4.38E+02
PU236	0.0	1.57E-03	1.23E-03	9.68E-04	4.67E-04	2.87E-04
PU238	0.0	1.32E+02	1.42E+02	1.43E+02	1.40E+02	1.38E+02
PU239	0.0	5.39E+03	5.47E+03	5.47E+03	5.47E+03	5.47E+03
PU240	0.0	2.23E+03	2.23E+03	2.23E+03	2.23E+03	2.23E+03
PU241	0.0	1.22E+03	1.16E+03	1.10E+03	9.56E+02	8.68E+02
PU242	0.0	4.86E+02	4.86E+02	4.86E+02	4.86E+02	4.86E+02
PU244	0.0	2.58E-02	2.58E-02	2.58E-02	2.58E-02	2.58E-02
AM241	0.0	3.70E+01	9.41E+01	1.48E+02	2.96E+02	3.82E+02
AM242M	0.0	3.99E-01	3.97E-01	3.95E-01	3.90E-01	3.86E-01
AM243	0.0	8.37E+01	8.38E+01	8.38E+01	8.38E+01	8.38E+01
CM242	0.0	1.24E+01	2.65E+00	5.63E-01	6.30E-03	1.18E-03
CM243	0.0	3.24E-01	3.16E-01	3.09E-01	2.87E-01	2.73E-01
CM244	0.0	2.39E+01	2.31E+01	2.22E+01	1.98E+01	1.84E+01
CM245	0.0	8.04E-01	8.04E-01	8.04E-01	8.04E-01	8.04E-01
CM246	0.0	7.32E-01	7.31E-01	7.31E-01	7.31E-01	7.31E-01
CM247	0.0	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.65E-03
TOTAL	1.00E+06	9.66E+05	9.66E+05	9.66E+05	9.66E+05	9.66E+05

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE	0.0	2.79E-01	4.83E-01	5.61E-01	7.05E-01	7.98E-01
TL	0.0	2.99E-12	7.42E-12	1.28E-11	2.80E-11	3.54E-11
PB	0.0	7.26E-07	2.40E-06	5.72E-06	2.63E-05	4.74E-05
BI	0.0	3.22E-10	6.19E-10	9.53E-10	1.92E-09	2.43E-09
PO	0.0	7.99E-14	1.89E-13	3.58E-13	1.43E-12	2.80E-12
AT	0.0	3.28E-19	5.00E-20	5.41E-20	6.70E-20	7.60E-20
RN	0.0	2.75E-12	6.75E-12	1.16E-11	2.55E-11	3.24E-11
FR	0.0	3.34E-15	1.10E-15	1.41E-15	2.33E-15	2.92E-15
RA	0.0	3.07E-08	6.24E-08	1.03E-07	2.41E-07	3.37E-07
AC	0.0	1.38E-08	2.45E-08	3.51E-08	6.57E-08	8.54E-08
TH	0.0	1.04E-03	1.56E-03	2.09E-03	3.71E-03	4.79E-03
PA	0.0	5.55E-04	5.64E-04	5.72E-04	5.96E-04	6.12E-04
U	1.00E+06	9.56E+05	9.56E+05	9.56E+05	9.56E+05	9.56E+05
NP	0.0	5.11E+02	4.35E+02	4.35E+02	4.37E+02	4.38E+02
PU	0.0	9.45E+03	9.49E+03	9.43E+03	9.28E+03	9.19E+03
AM	0.0	1.21E+02	1.78E+02	2.33E+02	3.80E+02	4.66E+02
CM	0.0	3.82E+01	2.76E+01	2.46E+01	2.17E+01	2.02E+01
BK	0.0	1.22E-06	5.53E-07	2.51E-07	2.34E-08	4.80E-09
CF	0.0	7.21E-07	1.35E-06	1.61E-06	1.74E-06	1.71E-06
ES	0.0	1.70E-10	6.50E-15	3.16E-20	2.51E-36	4.56E-47
TOTALS	1.00E+06	9.66E+05	9.66E+05	9.66E+05	9.66E+05	9.66E+05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 237	0.0	6.42E+02	2.59E-03	2.47E-03	2.13E-03	1.94E-03
NP239	0.0	1.97E+04	1.69E-02	1.69E-02	1.69E-02	1.69E-02
PU238	0.0	2.14E-02	2.29E-02	2.31E-02	2.26E-02	2.23E-02
PU239	0.0	1.24E-03	1.26E-03	1.26E-03	1.26E-03	1.26E-03
PU240	0.0	4.60E-03	4.61E-03	4.61E-03	4.61E-03	4.61E-03
PU241	0.0	5.65E-03	5.38E-03	5.13E-03	4.44E-03	4.03E-03
AM241	0.0	1.84E-02	4.67E-02	7.36E-02	1.47E-01	1.90E-01
AM243	0.0	5.46E-03	5.47E-03	5.47E-03	5.47E-03	5.46E-03
CM242	0.0	4.03E-01	8.59E-02	1.82E-02	2.04E-04	3.81E-05
CM243	0.0	1.32E-02	1.29E-02	1.26E-02	1.17E-02	1.11E-02
CM244	0.0	1.75E-02	1.69E-02	1.63E-02	1.45E-02	1.35E-02
TOTAL	7.91E-05	3.01E+04	2.23E-01	1.81E-01	2.32E-01	2.73E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PA234M	0.0	1.59E-03	1.57E-03	1.57E-03	1.57E-03	1.57E-03
U 234	4.37E-02	2.61E-02	2.63E-02	2.65E-02	2.71E-02	2.74E-02
U 236	0.0	7.10E-03	7.10E-03	7.10E-03	7.10E-03	7.10E-03
U 237	0.0	1.40E+03	5.66E-03	5.39E-03	4.67E-03	4.24E-03
U 238	8.24E-03	8.03E-03	8.03E-03	8.03E-03	8.03E-03	8.03E-03
NP237	0.0	8.83E-03	9.02E-03	9.02E-03	9.05E-03	9.07E-03
NP239	0.0	4.71E+04	4.04E-02	4.04E-02	4.04E-02	4.04E-02
PU236	0.0	2.89E-02	2.28E-02	1.79E-02	8.63E-03	5.31E-03
PU238	0.0	7.48E+01	8.04E+01	8.09E+01	7.93E+01	7.81E+01
PU239	0.0	1.04E+01	1.06E+01	1.06E+01	1.06E+01	1.06E+01
PU240	0.0	1.58E+01	1.58E+01	1.58E+01	1.58E+01	1.59E+01
PU241	0.0	3.86E+00	3.68E+00	3.51E+00	3.04E+00	2.76E+00
PU242	0.0	5.48E-02	5.48E-02	5.48E-02	5.48E-02	5.48E-02
AM241	0.0	4.25E+00	1.08E+01	1.70E+01	3.39E+01	4.39E+01
AM242M	0.0	1.53E-03	1.52E-03	1.52E-03	1.50E-03	1.48E-03
AM242	0.0	8.90E+01	4.46E-03	4.44E-03	4.38E-03	4.34E-03
AM243	0.0	5.38E-01	5.39E-01	5.39E-01	5.39E-01	5.39E-01
CM242	0.0	1.52E+03	3.23E+02	6.87E+01	7.68E-01	1.43E-01
CM243	0.0	6.14E-01	5.99E-01	5.85E-01	5.44E-01	5.18E-01
CM244	0.0	6.78E+01	6.54E+01	6.30E+01	5.61E+01	5.20E+01
CM245	0.0	4.60E-03	4.60E-03	4.60E-03	4.60E-03	4.60E-03
CM246	0.0	7.36E-03	7.36E-03	7.36E-03	7.35E-03	7.35E-03
TOTAL	5.39E-02	1.06E+05	5.11E+02	2.61E+02	2.01E+02	2.05E+02

FUEL DECAY BIBLIS-TYPE 3.2% U235

BURNUP=3.30E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TL208	0.0	8.80E-04	2.18E-03	3.77E-03	8.25E-03	1.04E-02
PB212	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
BI212	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
PO212	0.0	1.57E-03	3.90E-03	6.72E-03	1.47E-02	1.86E-02
PO216	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
RN220	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
RA224	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
TH228	0.0	2.44E-03	6.07E-03	1.05E-02	2.30E-02	2.89E-02
TH231	0.0	1.01E+00	1.78E-02	1.78E-02	1.79E-02	1.79E-02
TH234	0.0	3.18E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01
PA233	0.0	2.99E-01	3.07E-01	3.07E-01	3.08E-01	3.09E-01
PA234M	0.0	3.22E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01
U 232	0.0	1.05E-02	1.75E-02	2.30E-02	3.27E-02	3.58E-02
U 234	1.52E+00	9.06E-01	9.13E-01	9.19E-01	9.40E-01	9.54E-01
U 235	6.92E-02	1.78E-02	1.78E-02	1.78E-02	1.79E-02	1.79E-02
U 236	0.0	2.62E-01	2.62E-01	2.62E-01	2.62E-01	2.62E-01
U 237	0.0	7.42E+05	2.99E+00	2.85E+00	2.47E+00	2.24E+00
U 238	3.26E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01
NP237	0.0	3.00E-01	3.07E-01	3.07E-01	3.08E-01	3.09E-01
NP238	0.0	3.13E+05	1.93E-02	1.92E-02	1.89E-02	1.88E-02
NP239	0.0	1.95E+07	1.67E+01	1.67E+01	1.67E+01	1.67E+01
PU236	0.0	8.32E-01	6.56E-01	5.14E-01	2.48E-01	1.53E-01
PU238	0.0	2.26E+03	2.43E+03	2.44E+03	2.39E+03	2.36E+03
PU239	0.0	3.35E+02	3.40E+02	3.40E+02	3.40E+02	3.40E+02
PU240	0.0	5.08E+02	5.08E+02	5.08E+02	5.09E+02	5.09E+02
PU241	0.0	1.25E+05	1.19E+05	1.14E+05	9.85E+04	8.95E+04
PU242	0.0	1.86E+00	1.86E+00	1.86E+00	1.86E+00	1.86E+00
AM241	0.0	1.27E+02	3.23E+02	5.09E+02	1.02E+03	1.31E+03
AM242M	0.0	3.88E+00	3.86E+00	3.84E+00	3.79E+00	3.75E+00
AM242	0.0	7.66E+04	3.84E+00	3.82E+00	3.77E+00	3.74E+00
AM243	0.0	1.67E+01	1.67E+01	1.67E+01	1.67E+01	1.67E+01
CM242	0.0	4.11E+04	8.78E+03	1.86E+03	2.09E+01	3.89E+00
CM243	0.0	1.67E+01	1.63E+01	1.59E+01	1.48E+01	1.41E+01
CM244	0.0	1.94E+03	1.87E+03	1.80E+03	1.60E+03	1.49E+03
CM245	0.0	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01
CM246	0.0	2.25E-01	2.25E-01	2.25E-01	2.25E-01	2.25E-01
TOTAL	1.91E+00	4.07E+07	1.34E+05	1.21E+05	1.04E+05	9.55E+04

ACTINIDES

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
BI212	0.0	8.80E-04	2.18E-03	3.77E-03	8.25E-03	1.04E-02
PO212	0.0	1.57E-03	3.90E-03	6.72E-03	1.47E-02	1.86E-02
PO216	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
RN220	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
RA224	0.0	2.45E-03	6.08E-03	1.05E-02	2.30E-02	2.90E-02
TH228	0.0	2.44E-03	6.07E-03	1.05E-02	2.30E-02	2.89E-02
U 232	0.0	1.05E-02	1.75E-02	2.30E-02	3.27E-02	3.58E-02
U 234	1.52E+00	9.06E-01	9.13E-01	9.19E-01	9.40E-01	9.54E-01
U 235	6.92E-02	1.78E-02	1.78E-02	1.78E-02	1.79E-02	1.79E-02
U 236	0.0	2.62E-01	2.62E-01	2.62E-01	2.62E-01	2.62E-01
U 238	3.26E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01	3.17E-01
NP237	0.0	3.00E-01	3.07E-01	3.07E-01	3.08E-01	3.09E-01
PU236	0.0	8.32E-01	6.56E-01	5.14E-01	2.48E-01	1.53E-01
PU238	0.0	2.26E+03	2.43E+03	2.44E+03	2.39E+03	2.36E+03
PU239	0.0	3.35E+02	3.40E+02	3.40E+02	3.40E+02	3.40E+02
PU240	0.0	5.08E+02	5.08E+02	5.08E+02	5.09E+02	5.09E+02
PU241	0.0	3.13E+00	2.99E+00	2.85E+00	2.46E+00	2.24E+00
PU242	0.0	1.86E+00	1.86E+00	1.86E+00	1.86E+00	1.86E+00
AM241	0.0	1.27E+02	3.23E+02	5.09E+02	1.02E+03	1.31E+03
AM242M	0.0	1.94E-02	1.93E-02	1.92E-02	1.89E-02	1.88E-02
AM243	0.0	1.67E+01	1.67E+01	1.67E+01	1.67E+01	1.67E+01
CM242	0.0	4.11E+04	8.78E+03	1.86E+03	2.09E+01	3.89E+00
CM243	0.0	1.67E+01	1.63E+01	1.59E+01	1.48E+01	1.41E+01
CM244	0.0	1.94E+03	1.87E+03	1.80E+03	1.60E+03	1.49E+03
CM245	0.0	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.38E-01
CM246	0.0	2.25E-01	2.25E-01	2.25E-01	2.25E-01	2.25E-01
TOTAL	1.91E+00	4.63E+04	1.43E+04	7.50E+03	5.92E+03	6.05E+03

FUEL DECAY BIBLIS-TYPE 3.2% U235

FISSION PRODUCTS

BURNUP=3.30E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	1.68E-02	1.59E-02	1.50E-02	1.27E-02	1.14E-02
KR 85	0.0	1.41E+01	1.32E+01	1.24E+01	1.02E+01	8.96E+00
SR 89	0.0	2.70E+03	1.80E+01	1.20E-01	3.52E-08	1.56E-12
SR 90	0.0	8.67E+01	8.47E+01	8.27E+01	7.70E+01	7.34E+01
Y 90	0.0	4.36E+02	4.04E+02	3.95E+02	3.68E+02	3.51E+02
Y 91	0.0	3.68E+03	4.89E+01	6.47E-01	1.49E-06	2.61E-10
ZR 95	0.0	7.43E+03	1.42E+02	2.72E+00	1.91E-05	6.99E-09
NB 95	0.0	7.02E+03	3.03E+02	5.93E+00	4.01E-05	1.47E-08
NB 95M	0.0	1.47E+01	2.89E-01	5.54E-03	3.88E-08	1.42E-11
TC 99	0.0	6.58E-03	6.61E-03	6.61E-03	6.61E-03	6.61E-03
RU103	0.0	5.05E+03	8.08E+00	1.28E-02	5.15E-11	1.30E-16
RU106	0.0	2.96E+01	1.49E+01	7.49E+00	9.53E-01	2.41E-01
RH103M	0.0	3.12E+02	4.99E-01	7.94E-04	3.18E-12	8.05E-18
RH106	0.0	5.24E+03	2.41E+03	1.21E+03	1.54E+02	3.90E+01
AG110	0.0	9.35E+02	1.01E-01	3.67E-02	1.76E-03	2.32E-04
AG110M	0.0	4.86E+01	1.76E+01	6.41E+00	3.07E-01	4.05E-02
CD113M	0.0	5.97E-02	5.70E-02	5.43E-02	4.71E-02	4.29E-02
CD115M	0.0	2.75E+00	9.40E-03	3.23E-05	1.30E-12	1.53E-17
SN119M	0.0	5.69E-02	2.03E-02	7.21E-03	3.25E-04	4.12E-05
SN123	0.0	6.11E+00	8.61E-01	1.21E-01	3.40E-04	6.75E-06
SB124	0.0	8.70E+00	1.30E-01	1.94E-03	6.44E-09	1.44E-12
SB125	0.0	2.79E+01	2.19E+01	1.71E+01	8.04E+00	4.86E+00
SB126	0.0	1.38E+01	1.40E-03	1.40E-03	1.40E-03	1.40E-03
SB126M	0.0	7.90E+00	6.90E-03	6.90E-03	6.90E-03	6.90E-03
TE123M	0.0	1.0[illegible]E-02	1.21E-03	1.46E-04	2.57E-07	3.75E-09
TE125M	0.0	1.57E+00	1.44E+00	1.12E+00	5.27E-01	3.19E-01
TE127	0.0	1.20E+02	1.56E+00	1.53E-01	1.44E-04	1.39E-06
TE127M	0.0	6.21E+00	6.34E-01	6.21E-02	5.86E-05	5.64E-07
TE129	0.0	1.07E+03	5.68E-02	3.04E-05	4.66E-15	1.33E-21
TE129M	0.0	7.97E+01	4.29E-02	2.29E-05	3.51E-15	1.01E-21
CS134	0.0	1.59E+03	1.14E+03	8.14E+02	2.97E+02	1.52E+02
CS137	0.0	1.18E+02	1.15E+02	1.13E+02	1.05E+02	1.00E+02
BA137M	0.0	3.97E+02	3.87E+02	3.78E+02	3.53E+02	3.37E+02
CE141	0.0	2.15E+03	9.01E-01	3.75E-04	2.69E-14	4.66E-21
CE144	0.0	7.28E+02	2.99E+02	1.23E+02	8.48E+00	1.43E+00
PR144	0.0	8.14E+03	3.31E+03	1.36E+03	9.40E+01	1.58E+01
PR144M	0.0	4.51E+00	1.85E+00	7.59E-01	5.25E-02	8.84E-03
PM147	0.0	5.50E+01	4.41E+01	3.39E+01	1.53E+01	9.05E+00
PM148	0.0	1.34E+03	2.90E-02	6.39E-05	6.62E-13	3.14E-18
PM148M	0.0	3.90E+02	8.46E-01	1.87E-03	1.93E-11	9.18E-17
SM151	0.0	3.87E-02	3.92E-02	3.89E-02	3.80E-02	3.74E-02
EU152	0.0	3.62E-02	3.44E-02	3.27E-02	2.81E-02	2.53E-02
EU154	0.0	1.10E+02	1.01E+02	9.35E+01	7.34E+01	6.25E+01
EU155	0.0	5.36E+00	4.66E+00	4.05E+00	2.66E+00	2.01E+00
GD153	0.0	1.58E-02	5.56E-03	1.95E-03	8.48E-05	1.05E-05
TB160	0.0	6.21E+00	1.88E-01	5.66E-03	1.55E-07	1.42E-10
TOTAL	0.0	1.80E+06	8.90E+03	4.66E+03	1.57E+03	1.16E+03

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
KR 85	0.0	5.88E-01	5.51E-01	5.17E-01	4.25E-01	3.74E-01
SR 89	0.0	1.58E+02	1.05E+00	7.01E-03	2.07E-09	9.15E-14
SR 90	0.0	4.87E+00	4.75E+00	4.64E+00	4.32E+00	4.12E+00
Y 90	0.0	6.98E+01	6.48E+01	6.32E+01	5.89E+01	5.61E+01
Y 91	0.0	2.40E+02	3.20E+00	4.22E-02	9.76E-08	1.70E-11
ZR 95	0.0	6.44E+03	1.23E+02	2.36E+00	1.65E-05	6.06E-09
NB 95	0.0	6.65E+03	2.87E+02	5.61E+00	3.80E-05	1.39E-08
NB 95M	0.0	4.31E+00	8.48E-02	1.62E-03	1.14E-08	4.17E-12
RU103	0.0	4.38E+03	7.00E+00	1.11E-02	4.47E-11	1.13E-16
RU106	0.0	1.23E-01	6.17E-02	3.10E-02	3.94E-03	9.97E-04
RH103M	0.0	1.37E+01	2.20E-02	3.50E-05	1.40E-13	3.55E-19
RH106	0.0	1.15E+03	5.28E+02	2.66E+02	3.38E+01	8.54E+00
AG110	0.0	1.08E+02	1.16E-02	4.22E-03	2.02E-04	2.67E-05
AG110M	0.0	4.74E+01	1.72E+01	6.25E+00	2.99E-01	3.95E-02
CD113M	0.0	1.07E-03	1.02E-03	9.76E-04	8.46E-04	7.70E-04
SN119M	0.0	7.44E-03	2.65E-03	9.42E-04	4.25E-05	5.38E-06
SN123	0.0	4.12E-01	5.80E-02	8.17E-03	2.29E-05	4.54E-07
SB124	0.0	7.30E+00	1.09E-01	1.63E-03	5.41E-09	1.21E-12
SB125	0.0	2.29E+01	1.80E+01	1.40E+01	6.59E+00	3.99E+00
SB126	0.0	1.26E+01	1.28E-03	1.28E-03	1.28E-03	1.28E-03
SB126M	0.0	5.82E+00	5.08E-03	5.08E-03	5.08E-03	5.08E-03
TE125M	0.0	3.95E-01	3.60E-01	2.81E-01	1.32E-01	8.00E-02
TE127	0.0	6.20E+00	8.08E-02	7.93E-03	7.48E-06	7.20E-08
TE127M	0.0	7.23E-01	7.38E-02	7.24E-03	6.83E-06	6.57E-08
TE129	0.0	1.56E+02	8.25E-03	4.42E-06	6.76E-16	1.94E-22
TE129M	0.0	1.12E+01	6.04E-03	3.23E-06	4.95E-16	1.42E-22
CS134	0.0	1.45E+03	1.03E+03	7.38E+02	2.69E+02	1.38E+02
CS137	0.0	2.97E+00	2.91E+00	2.84E+00	2.65E+00	2.53E+00
BA137M	0.0	3.56E+02	3.47E+02	3.39E+02	3.17E+02	3.02E+02
CE141	0.0	6.95E+02	2.91E-01	1.21E-04	8.69E-15	1.50E-21
CE144	0.0	1.41E+02	5.81E+01	2.38E+01	1.65E+00	2.78E-01
PR144	0.0	9.58E+02	3.90E+02	1.60E+02	1.11E+01	1.87E+00
PR144M	0.0	9.92E-01	4.07E-01	1.67E-01	1.15E-02	1.95E-03
PM147	0.0	7.18E-01	5.76E-01	4.42E-01	2.00E-01	1.18E-01
PM148	0.0	6.99E+02	1.52E-02	3.34E-05	3.46E-13	1.64E-18
PM148M	0.0	3.62E+02	7.86E-01	1.74E-03	1.80E-11	8.53E-17
EU152	0.0	3.25E-02	3.09E-02	2.93E-02	2.52E-02	2.27E-02
EU154	0.0	9.00E+01	8.30E+01	7.66E+01	6.02E+01	5.12E+01
EU155	0.0	2.67E+00	2.32E+00	2.02E+00	1.33E+00	1.00E+00
GD153	0.0	1.15E-02	4.04E-03	1.42E-03	6.16E-05	7.60E-06
TB160	0.0	5.12E+00	1.54E-01	4.66E-03	1.28E-07	1.17E-10
TOTAL	0.0	5.65E+05	2.97E+03	1.71E+03	7.67E+02	5.70E+02

FUEL DECAY BIBLIS-TYPE 3.2% U235 FISSION PRODUCTS

BURNUP=3.30E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	4.98E+02	4.70E+02	4.45E+02	3.76E+02	3.36E+02
SE 79	0.0	3.69E-01	3.69E-01	3.69E-01	3.69E-01	3.69E-01
KR 85	0.0	9.40E+03	8.82E+03	8.27E+03	6.81E+03	5.98E+03
RB 86	0.0	1.81E+03	2.32E-03	2.98E-09	6.30E-27	1.04E-38
SR 89	0.0	7.81E+05	5.19E+03	3.46E+01	1.02E-05	4.51E-10
SR 90	0.0	7.47E+04	7.30E+04	7.12E+04	6.63E+04	6.32E+04
Y 90	0.0	7.86E+04	7.30E+04	7.13E+04	6.63E+04	6.33E+04
Y 91	0.0	1.02E+06	1.36E+04	1.80E+02	4.16E-04	7.26E-08
ZR 93	0.0	1.79E+00	1.79E+00	1.79E+00	1.79E+00	1.79E+00
ZR 95	0.0	1.47E+06	2.81E+04	5.37E+02	3.77E-03	1.38E-06
NB 93M	0.0	1.31E-01	2.09E-01	2.83E-01	4.84E-01	6.02E-01
NB 95	0.0	1.46E+06	6.32E+04	1.24E+03	8.36E-03	3.06E-06
NB 95M	0.0	1.06E+04	2.08E+02	3.99E+00	2.79E-05	1.02E-08
TC 99	0.0	1.31E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01
RU103	0.0	1.50E+06	2.41E+03	3.81E+00	1.53E-08	3.88E-14
RU106	0.0	5.00E+05	2.51E+05	1.26E+05	1.61E+04	4.06E+03
RH103M	0.0	1.36E+06	2.17E+03	3.45E+00	1.38E-08	3.50E-14
RH106	0.0	5.46E+05	2.51E+05	1.26E+05	1.61E+04	4.06E+03
PD107	0.0	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01
AG110	0.0	1.30E+05	1.41E+01	5.10E+00	2.44E-01	3.22E-02
AG110M	0.0	2.91E+03	1.06E+03	3.84E+02	1.84E+01	2.42E+00
CD113M	0.0	3.55E+01	3.39E+01	3.23E+01	2.80E+01	2.55E+01
CD115M	0.0	7.37E+02	2.52E+00	8.65E-03	3.49E-10	4.10E-15
IN114	0.0	2.56E+00	1.06E-02	6.38E-05	1.39E-11	5.06E-16
IN114M	0.0	1.84E+00	1.11E-02	6.67E-05	1.46E-11	5.29E-16
SN119M	0.0	1.10E+02	3.92E+01	1.39E+01	6.29E-01	7.97E-02
SN121M	0.0	1.57E-01	1.54E-01	1.52E-01	1.46E-01	1.42E-01
SN123	0.0	1.96E+03	2.76E+02	3.89E+01	1.09E-01	2.16E-03
SN126	0.0	5.42E-01	5.42E-01	5.42E-01	5.42E-01	5.42E-01
SB124	0.0	6.55E+02	9.78E+00	1.46E-01	4.85E-07	1.08E-10
SB125	0.0	8.92E+03	7.02E+03	5.46E+03	2.57E+03	1.56E+03
SB126	0.0	7.46E+02	7.59E-02	7.59E-02	7.59E-02	7.59E-02
SB126M	0.0	6.20E+02	5.42E-01	5.42E-01	5.42E-01	5.42E-01
TE123M	0.0	6.89E+00	8.32E-01	1.00E-01	1.77E-04	2.57E-06
TE125M	0.0	1.87E+03	1.71E+03	1.33E+03	6.27E+02	3.80E+02
TE127	0.0	8.86E+04	1.15E+03	1.13E+02	1.07E-01	1.03E-03
TE127M	0.0	1.16E+04	1.18E+03	1.16E+02	1.09E-01	1.05E-03
TE129	0.0	3.00E+05	1.59E+01	8.51E-03	1.30E-12	3.73E-19
TE129M	0.0	4.55E+04	2.44E+01	1.31E-02	2.00E-12	5.73E-19
I 129	0.0	3.09E-02	3.12E-02	3.12E-02	3.12E-02	3.12E-02
CS134	0.0	1.57E+05	1.12E+05	7.99E+04	2.92E+04	1.49E+04
CS135	0.0	3.51E-01	3.52E-01	3.52E-01	3.52E-01	3.52E-01
CS137	0.0	1.07E+05	1.04E+05	1.02E+05	9.49E+04	9.06E+04
BA137M	0.0	1.01E+05	9.85E+04	9.62E+04	8.98E+04	8.57E+04
BA140	0.0	1.55E+06	3.94E-03	1.00E-11	0.0	0.0
LA140	0.0	1.61E+06	4.53E-03	1.15E-11	0.0	0.0
CE141	0.0	1.47E+06	6.15E+02	2.56E-01	1.84E-11	3.18E-18
CE144	0.0	1.10E+06	4.50E+05	1.85E+05	1.28E+04	2.15E+03
PR143	0.0	1.32E+06	1.16E-02	9.10E-11	0.0	0.0
PR144	0.0	1.11E+06	4.50E+05	1.85E+05	1.28E+04	2.15E+03
PR144M	0.0	1.32E+04	5.41E+03	2.22E+03	1.53E+02	2.59E+01
PM147	0.0	1.53E+05	1.23E+05	9.45E+04	4.28E+04	2.52E+04

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PM148	0.0	1.74E+05	3.77E+00	8.30E-03	8.59E-11	4.08E-16
PM148M	0.0	3.07E+04	6.67E+01	1.48E-01	1.53E-09	7.24E-15
SM151	0.0	3.30E+02	3.34E+02	3.31E+02	3.24E+02	3.19E+02
EU152	0.0	4.78E+00	4.54E+00	4.31E+00	3.70E+00	3.34E+00
EU154	0.0	1.23E+04	1.13E+04	1.04E+04	8.20E+03	6.98E+03
EU155	0.0	7.37E+03	6.41E+03	5.57E+03	3.66E+03	2.77E+03
EU156	0.0	2.25E+05	1.30E-02	7.49E-10	0.0	0.0
GD153	0.0	1.84E+01	6.45E+00	2.27E+00	9.84E-02	1.21E-02
TB160	0.0	7.63E+02	2.30E+01	6.95E-01	1.91E-05	1.74E-08
TOTAL	0.0	1.52E+08	2.15E+06	1.17E+06	4.70E+05	3.74E+05

******** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS ********

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
			CONCENTRATIONS ,GRAMS			
TOTAL	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06
			RADIOACTIVITY ,CURIES			
TOTAL	1.91E+00	1.94E+08	2.38E+06	1.36E+06	6.04E+05	4.90E+05
			THERMAL POWER ,WATTS			
TOTAL	5.39E-02	1.92E+06	9.98E+03	5.25E+03	1.96E+03	1.50E+03
			GAMMA POWER ,WATTS			
TOTAL	7.91E-05	6.02E+05	3.44E+03	1.97E+03	9.26E+02	6.92E+02
			ALPHA-ACTIVITY ,CURIES			
TOTAL	1.91E+00	4.63E+04	1.43E+04	7.50E+03	5.92E+03	6.05E+03

FUEL DECAY BIBLIS-TYPE 3.2% U235

BURNUP=3.30E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	3.85E-01	1.88E-02	9.15E-04	1.06E-07	2.52E-10
CR 51	0.0	4.14E+01	4.46E-03	4.81E-07	6.04E-19	7.02E-27
MN 54	0.0	3.55E+01	1.58E+01	7.03E+00	6.19E-01	1.22E-01
FE 55	0.0	8.54E+01	6.54E+01	5.01E+01	2.25E+01	1.32E+01
FE 59	0.0	4.63E+01	1.67E-01	6.04E-04	2.84E-11	3.69E-16
CO 58	0.0	1.89E+02	5.28E+00	1.48E-01	3.24E-06	2.53E-09
CO 60	0.0	3.16E+02	2.77E+02	2.43E+02	1.64E+02	1.26E+02
NI 59	0.0	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01
NI 63	0.0	9.85E-01	9.77E-01	9.70E-01	9.48E-01	9.34E-01
SR 89	0.0	2.42E-01	1.61E-03	1.07E-05	3.16E-12	1.40E-16
Y 91	0.0	6.63E-01	8.76E-03	1.16E-04	2.68E-10	4.67E-14
ZR 95	0.0	3.15E+02	6.04E+00	1.16E-01	8.10E-07	2.97E-10
NB 94	0.0	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.06E-02
NB 95	0.0	2.93E+02	1.28E+01	2.51E-01	1.69E-06	6.19E-10
SN119M	0.0	5.05E+00	1.80E+00	6.40E-01	2.89E-02	3.66E-03
SN121M	0.0	1.85E-03	1.82E-03	1.80E-03	1.73E-03	1.68E-03
SB124	0.0	1.91E-01	2.85E-03	4.25E-05	1.41E-10	3.15E-14
SB125	0.0	2.43E-01	1.89E-01	1.47E-01	6.93E-02	4.20E-02
TE125M	0.0	1.39E-02	1.24E-02	9.66E-03	4.55E-03	2.75E-03
TA182	0.0	1.69E+03	1.87E+02	2.07E+01	2.75E-02	3.37E-04
TOTAL	1.08E-14	8.41E+03	5.73E+02	3.23E+02	1.88E+02	1.40E+02

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	3.65E-01	1.78E-02	8.69E-04	1.01E-07	2.40E-10
CR 51	0.0	3.72E+01	4.01E-03	4.33E-07	5.43E-19	6.31E-27
MN 54	0.0	3.53E+01	1.57E+01	6.99E+00	6.16E-01	1.22E-01
CO 58	0.0	1.83E+02	5.13E+00	1.43E-01	3.14E-06	2.46E-09
CO 60	0.0	3.04E+02	2.67E+02	2.34E+02	1.58E+02	1.21E+02
ZR 95	0.0	2.74E+02	5.24E+00	1.00E-01	7.02E-07	2.57E-10
NB 94	0.0	9.75E-03	9.75E-03	9.74E-03	9.74E-03	9.74E-03
NB 95	0.0	2.77E+02	1.21E+01	2.37E-01	1.60E-06	5.86E-10
SN119M	0.0	6.61E-01	2.35E-01	8.37E-02	3.77E-03	4.78E-04
SB124	0.0	1.60E-01	2.39E-03	3.57E-05	1.19E-10	2.64E-14
SB125	0.0	1.99E-01	1.55E-01	1.21E-01	5.68E-02	3.44E-02
TE125M	0.0	3.49E-03	3.11E-03	2.42E-03	1.14E-03	6.90E-04
TA182	0.0	1.48E+03	1.64E+02	1.81E+01	2.40E-02	2.95E-04
TOTAL	0.0	6.29E+03	4.69E+02	2.60E+02	1.58E+02	1.21E+02

LIGHT ELEMENTS

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	1.50E-03	1.42E-03	1.34E-03	1.13E-03	1.01E-03
C 14	0.0	1.43E-01	1.43E-01	1.43E-01	1.43E-01	1.43E-01
P 33	0.0	5.30E+02	2.12E-02	8.51E-07	5.48E-20	8.81E-29
S 35	0.0	6.91E+00	3.89E-01	2.19E-02	3.92E-06	1.25E-08
CA 45	0.0	2.26E-01	4.78E-02	1.01E-02	9.58E-05	4.29E-06
SC 46	0.0	3.06E+01	1.49E+00	7.28E-02	8.44E-06	2.01E-08
CR 51	0.0	1.93E+05	2.09E+01	2.25E-03	2.82E-15	3.28E-23
MN 54	0.0	7.13E+03	3.17E+03	1.41E+03	1.24E+02	2.46E+01
FE 55	0.0	6.21E+04	4.76E+04	3.64E+04	1.64E+04	9.61E+03
FE 59	0.0	4.97E+03	1.79E+01	6.48E-02	3.04E-09	3.96E-14
CO 58	0.0	3.15E+04	8.82E+02	2.47E+01	5.41E-04	4.23E-07
CO 60	0.0	2.05E+04	1.80E+04	1.58E+04	1.06E+04	8.17E+03
NI 59	0.0	1.75E+01	1.75E+01	1.75E+01	1.75E+01	1.75E+01
NI 63	0.0	2.48E+03	2.46E+03	2.44E+03	2.39E+03	2.35E+03
ZN 65	0.0	2.32E-01	8.22E-02	2.91E-02	1.29E-03	1.62E-04
SR 89	0.0	7.00E+01	4.66E-01	3.10E-03	9.14E-10	4.05E-14
SR 90	0.0	2.25E-03	2.20E-03	2.14E-03	2.00E-03	1.90E-03
Y 90	0.0	2.58E+03	2.20E-03	2.15E-03	2.00E-03	1.90E-03
Y 91	0.0	1.85E+02	2.44E+00	3.23E-02	7.45E-08	1.30E-11
ZR 93	0.0	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01
ZR 95	0.0	6.23E+04	1.19E+03	2.28E+01	1.60E-04	5.86E-08
NB 93M	0.0	1.17E-02	1.98E-02	2.75E-02	4.83E-02	6.06E-02
NB 94	0.0	1.04E+00	1.04E+00	1.04E+00	1.04E+00	1.04E+00
NB 95	0.0	6.10E+04	2.67E+03	5.23E+01	3.52E-04	1.29E-07
MO 93	0.0	7.14E-02	7.13E-02	7.13E-02	7.13E-02	7.13E-02
TC 99	0.0	1.19E-02	1.20E-02	1.20E-02	1.20E-02	1.20E-02
SN119M	0.0	9.78E+03	3.48E+03	1.24E+03	5.59E+01	7.08E+00
SN121M	0.0	9.24E-01	9.11E-01	8.98E-01	8.62E-01	8.38E-01
SN123	0.0	8.71E-01	1.23E-01	1.73E-02	4.89E-05	9.71E-07
SB124	0.0	1.44E+01	2.15E-01	3.20E-03	1.07E-08	2.37E-12
SB125	0.0	7.77E+01	6.05E+01	4.71E+01	2.22E+01	1.34E+01
TE125M	0.0	1.66E+01	1.47E+01	1.15E+01	5.41E+00	3.28E+00
TA182	0.0	1.90E+05	2.10E+04	2.32E+03	3.09E+00	3.78E-02
W 185	0.0	6.08E-01	2.09E-02	7.18E-04	2.92E-08	3.40E-11
TOTAL	6.06E-12	1.08E+06	1.01E+05	5.98E+04	2.96E+04	2.02E+04

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

FUEL DECAY BIBLIS-TYPE 3.2% U235
POWER= 33.03 MW, BURNUP= 3.30E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	1.15E+16	7.71E+14	9.39E+13	4.93E+12	3.55E+12
3.75E-01	1.03E+15	1.18E+12	6.98E+11	3.28E+11	2.04E+11
5.75E-01	5.61E+14	9.77E+12	1.06E+12	3.82E+11	2.32E+11
8.50E-01	1.79E+16	2.83E+14	5.57E+13	4.62E+12	9.87E+11
1.25E+00	8.29E+15	2.05E+15	1.25E+15	7.86E+14	6.04E+14
1.75E+00	3.85E+15	1.65E+11	4.58E+09	1.19E+05	1.92E+04
2.25E+00	1.68E+15	7.49E+09	6.19E+09	4.17E+09	3.20E+09
2.75E+00	2.37E+14	2.21E+07	1.91E+07	1.29E+07	9.91E+06
3.50E+00	2.03E+13	0.0	0.0	0.0	0.0
5.00E+00	3.68E+10	0.0	0.0	0.0	0.0
7.00E+00	6.54E+12	0.0	0.0	0.0	0.0
9.50E+00	6.72E+09	0.0	0.0	0.0	0.0
TOTAL	4.50E+16	3.12E+15	1.40E+15	7.97E+14	6.09E+14
MEV/SEC	3.93E+16	2.93E+15	1.62E+15	9.88E+14	7.57E+14

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

FUEL DECAY BIBLIS-TYPE 3.2% U235
POWER= 33.03 MW, BURNUP= 3.30E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	2.02E+18	1.69E+16	8.70E+15	2.73E+15	2.13E+15
3.75E-01	7.27E+17	1.70E+15	8.84E+14	2.60E+14	1.90E+14
5.75E-01	1.02E+18	1.33E+16	9.55E+15	5.17E+15	4.15E+15
8.50E-01	9.99E+17	7.31E+15	3.07E+15	1.14E+15	6.37E+14
1.25E+00	5.15E+17	8.65E+14	5.67E+14	2.51E+14	1.81E+14
1.75E+00	1.98E+17	7.30E+13	3.76E+13	8.84E+12	5.52E+12
2.25E+00	1.09E+17	1.40E+14	5.89E+13	4.47E+12	8.28E+11
2.75E+00	4.12E+16	2.11E+12	1.05E+12	1.31E+11	3.29E+10
3.50E+00	2.27E+16	2.61E+11	1.31E+11	1.67E+10	4.22E+09
5.00E+00	1.28E+16	5.73E-05	5.87E-05	6.11E-05	6.19E-05
7.00E+00	3.11E+15	3.75E-06	3.84E-06	4.00E-06	4.05E-06
9.50E+00	1.61E+10	2.43E-07	2.48E-07	2.59E-07	2.62E-07
TOTAL	5.67E+18	4.03E+16	2.29E+16	9.56E+15	7.29E+15
MEV/SEC	3.53E+18	1.86E+16	1.06E+16	4.79E+15	3.56E+15

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.00E-02	1.03E+18	8.85E+13	4.78E+13	3.90E+13	4.02E+13
2.50E-02	4.59E+16	3.19E+11	4.93E+11	9.67E+11	1.24E+12
3.75E-02	6.68E+16	2.60E+11	1.76E+11	1.86E+11	2.07E+11
5.75E-02	5.92E+16	4.52E+12	7.08E+12	1.40E+13	1.81E+13
8.50E-02	4.63E+17	6.11E+11	6.10E+11	6.05E+11	6.01E+11
1.25E-01	3.65E+17	5.40E+11	5.30E+11	5.16E+11	5.08E+11
2.25E-01	2.59E+17	4.34E+11	4.26E+11	4.08E+11	3.98E+11
3.75E-01	2.85E+16	2.91E+10	2.88E+10	2.87E+10	2.87E+10
5.75E-01	5.59E+15	2.83E+09	2.03E+09	2.00E+09	2.09E+09
8.50E-01	1.16E+16	2.00E+09	1.73E+09	1.65E+09	1.64E+09
1.25E+00	2.64E+15	9.34E+08	8.06E+08	7.24E+08	6.91E+08
1.75E+00	5.28E+10	3.54E+08	3.01E+08	2.76E+08	2.67E+08
2.25E+00	3.31E+08	1.93E+08	1.59E+08	1.36E+08	1.26E+08
2.75E+00	2.23E+08	1.89E+08	2.25E+08	3.68E+08	4.39E+08
3.50E+00	1.14E+08	6.68E+07	5.50E+07	4.69E+07	4.36E+07
5.00E+00	3.58E+07	2.09E+07	1.72E+07	1.47E+07	1.36E+07
7.00E+00	4.09E+06	2.39E+06	1.97E+06	1.68E+06	1.56E+06
9.50E+00	4.10E+05	2.39E+05	1.97E+05	1.68E+05	1.56E+05
TOTAL	2.34E+18	9.52E+13	5.72E+13	5.58E+13	6.13E+13
MEV/SEC	1.88E+17	1.40E+12	1.13E+12	1.45E+12	1.70E+12

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

FUEL DECAY BIBLIS-TYPE 3.2% U235
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PO212	2.72E+03	6.76E+03	1.16E+04	2.55E+04	3.22E+04
PU236	3.87E+03	3.05E+03	2.39E+03	1.15E+03	7.10E+02
PU238	2.64E+06	2.83E+06	2.85E+06	2.79E+06	2.75E+06
PU239	2.42E+05	2.46E+05	2.46E+05	2.46E+05	2.46E+05
PU240	3.79E+05	3.79E+05	3.79E+05	3.79E+05	3.80E+05
PU241	2.27E+03	2.16E+03	2.06E+03	1.79E+03	1.62E+03
PU242	1.31E+03	1.31E+03	1.31E+03	1.31E+03	1.31E+03
AM241	1.48E+05	3.76E+05	5.93E+05	1.18E+06	1.53E+06
AM243	2.68E+04	2.68E+04	2.68E+04	2.68E+04	2.68E+04
CM242	3.32E+08	7.08E+07	1.50E+07	1.68E+05	3.14E+04
CM243	1.56E+05	1.52E+05	1.48E+05	1.38E+05	1.31E+05
CM244	1.37E+07	1.32E+07	1.27E+07	1.13E+07	1.05E+07
TOTAL	3.49E+08	8.81E+07	3.20E+07	1.63E+07	1.56E+07

FUEL DECAY BIBLIS-TYPE 3.2% U235
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 238	1.20E+04	1.20E+04	1.20E+04	1.20E+04	1.20E+04
PU238	3.50E+05	3.76E+05	3.79E+05	3.71E+05	3.66E+05
PU240	2.03E+06	2.03E+06	2.03E+06	2.03E+06	2.03E+06
PU242	8.19E+05	8.19E+05	8.19E+05	8.19E+05	8.19E+05
CM242	2.68E+08	5.72E+07	1.21E+07	1.36E+05	2.54E+04
CM244	2.66E+08	2.57E+08	2.47E+08	2.21E+08	2.04E+08
CM246	6.51E+06	6.51E+06	6.51E+06	6.51E+06	6.50E+06
CM248	7.29E+03	7.29E+03	7.29E+03	7.29E+03	7.29E+03
CF250	5.31E+03	5.05E+03	4.79E+03	4.09E+03	3.68E+03
CF252	1.73E+05	1.33E+05	1.02E+05	4.65E+04	2.75E+04
CF254	2.93E+04	4.46E+02	6.80E+00	2.41E-05	5.59E-09
TOTAL	5.44E+08	3.24E+08	2.69E+08	2.30E+08	2.14E+08

TOTAL	8.93E+08	4.12E+08	3.01E+08	2.47E+08	2.30E+08

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.30E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE 4	7.98E-01	8.62E-01	1.05E+00	1.49E+00	2.02E+00	2.89E+00	6.53E+00	9.33E+00	1.31E+01	3.51E+01	1.09E+02
TH230	3.71E-03	3.73E-03	3.77E-03	3.92E-03	4.21E-03	4.90E-03	1.17E-02	8.66E-02	5.14E-01	2.34E-01	1.57E-01
TH232	1.03E-03	1.03E-03	1.05E-03	1.08E-03	1.14E-03	1.26E-03	2.26E-03	1.72E-02	2.27E-01	2.29E+00	2.02E+01
U 234	1.53E+00	1.56E+00	1.66E+00	1.92E+00	2.26E+00	2.68E+00	3.17E+00	3.11E+00	2.53E+00	6.65E-01	5.07E-01
U 235	8.26E+01	8.26E+01	8.26E+01	8.26E+01	8.27E+01	8.29E+01	8.42E+01	1.03E+02	2.22E+02	2.31E+02	2.29E+02
U 236	4.05E+01	4.05E+01	4.05E+01	4.06E+01	4.08E+01	4.13E+01	4.44E+01	6.64E+01	8.00E+01	7.79E+01	5.97E+01
U 238	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.43E+03	9.44E+03	9.43E+03
NP236M	1.95E-03	1.95E-03	1.95E-03	1.95E-03	1.95E-03	1.95E-03	1.95E-03	1.94E-03	1.85E-03	1.14E-03	9.03E-06
NP237	4.38E+02	4.39E+02	4.45E+02	4.63E+02	4.91E+02	5.40E+02	7.44E+02	8.20E+02	7.97E+02	5.95E+02	3.23E+01
PU238	1.38E+00	1.35E+00	1.26E+00	1.03E+00	7.35E-01	3.84E-01	5.02E-03	7.06E-21	0.0	0.0	0.0
PU239	5.47E+01	5.47E+01	5.49E+01	5.51E+01	5.55E+01	5.61E+01	6.07E+01	8.39E+01	9.75E+00	5.21E-06	3.49E-06
PU240	2.23E+01	2.43E+01	2.94E+01	3.68E+01	3.95E+01	3.96E+01	3.64E+01	1.40E+01	1.01E-03	3.00E-08	2.93E-08
PU241	8.68E+00	7.51E+00	4.64E+00	1.10E+00	1.00E-01	2.12E-03	1.24E-03	5.94E-04	3.86E-07	0.0	0.0
PU242	4.86E+00	4.86E+00	4.86E+00	4.88E+00	4.89E+00	4.92E+00	5.01E+00	5.38E+00	4.73E+00	9.43E-01	9.42E-08
AM241	3.82E+02	3.82E+02	3.78E+02	3.64E+02	3.37E+02	2.87E+02	7.97E+01	1.79E-02	1.22E-05	0.0	0.0
AM242M	3.86E-01	3.81E-01	3.64E-01	3.17E-01	2.53E-01	1.60E-01	4.18E-03	6.33E-21	0.0	0.0	0.0
AM243	8.38E+01	8.38E+01	8.37E+01	8.35E+01	8.31E+01	8.23E+01	7.63E+01	3.28E+01	7.01E-03	1.62E-06	1.09E-06
CM243	2.73E-01	2.54E-01	1.99E-01	9.61E-02	2.85E-02	2.50E-03	8.93E-12	0.0	0.0	0.0	0.0
CM244	1.84E+01	1.64E+01	1.12E+01	3.54E+00	5.23E-01	1.14E-02	5.78E-16	0.0	0.0	0.0	0.0
CM245	8.04E-01	8.04E-01	8.03E-01	8.01E-01	7.98E-01	7.91E-01	7.41E-01	3.56E-01	2.31E-04	0.0	0.0
CM246	7.31E-01	7.30E-01	7.29E-01	7.26E-01	7.21E-01	7.10E-01	6.32E-01	1.69E-01	3.18E-07	2.20E-30	0.0
CM247	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.65E-03	3.63E-03	3.49E-03	2.34E-03
TOTAL	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE	7.98E-01	8.62E-01	1.05E+00	1.49E+00	2.02E+00	2.89E+00	6.53E+00	9.33E+00	1.31E+01	3.51E+01	1.09E+02
TL	3.54E-11	1.23E-11	8.44E-13	4.67E-13	3.58E-13	2.33E-13	1.69E-13	7.74E-13	1.29E-11	2.66E-11	3.82E-12
PB	4.74E-05	6.90E-05	8.11E-05	8.56E-05	9.07E-05	9.66E-05	1.19E-04	2.70E-03	2.56E-01	3.57E+00	1.70E+01
BI	2.43E-09	1.28E-09	1.16E-09	3.33E-09	1.23E-08	7.61E-08	1.04E-05	1.20E-02	3.96E+00	1.60E+02	6.97E+02
PO	2.80E-12	6.14E-12	3.01E-11	1.67E-10	4.98E-10	1.29E-09	1.22E-08	3.06E-07	2.32E-06	1.05E-06	7.05E-07
AT	7.60E-20	7.65E-20	8.33E-20	1.53E-19	4.36E-19	1.67E-18	5.03E-17	5.24E-15	1.26E-13	2.77E-13	1.41E-14
RN	3.24E-11	1.25E-11	4.43E-12	1.05E-11	2.16E-11	4.59E-11	3.56E-10	8.93E-09	6.76E-08	3.07E-08	2.06E-08
FR	2.92E-15	3.66E-15	5.69E-15	9.58E-15	1.38E-14	2.56E-14	4.68E-13	4.76E-11	1.14E-09	2.51E-09	1.29E-10
RA	3.37E-07	3.16E-07	5.87E-07	1.59E-06	3.33E-06	7.13E-06	5.54E-05	1.39E-03	1.05E-02	4.79E-03	3.21E-03
AC	8.54E-08	1.13E-07	1.89E-07	3.13E-07	3.77E-07	3.96E-07	4.35E-07	9.89E-07	8.71E-06	1.46E-05	7.24E-06
TH	4.79E-03	4.77E-03	4.82E-03	5.00E-03	5.35E-03	6.17E-03	1.43E-02	1.43E-01	1.69E+00	4.62E+00	2.05E+01
PA	6.12E-04	6.12E-04	6.13E-04	6.15E-04	6.20E-04	6.28E-04	6.89E-04	1.32E-03	8.03E-03	1.06E-02	1.05E-02
U	9.56E+03	9.56E+03	9.56E+03	9.56E+03	9.56E+03	9.56E+03	9.57E+03	9.61E+03	9.76E+03	9.79E+03	9.72E+03
NP	4.38E+02	4.39E+02	4.45E+02	4.63E+02	4.91E+02	5.40E+02	7.44E+02	8.20E+02	7.97E+02	5.95E+02	3.23E+01
PU	9.19E+01	9.27E+01	9.50E+01	9.89E+01	1.01E+02	1.01E+02	1.02E+02	1.03E+02	1.45E+01	9.43E-01	3.80E-04
AM	4.66E+02	4.66E+02	4.62E+02	4.48E+02	4.20E+02	3.70E+02	1.56E+02	3.28E+01	7.02E-03	1.62E-06	1.09E-06
CM	2.02E+01	1.82E+01	1.29E+01	5.17E+00	2.07E+00	1.52E+00	1.38E+00	5.29E-01	4.00E-03	3.51E-03	2.34E-03
BK	4.80E-09	4.48E-10	1.64E-13	2.51E-20	2.50E-20	2.49E-20	2.41E-20	1.69E-20	4.68E-22	1.27E-37	0.0
CF	1.71E-06	1.65E-06	1.50E-06	1.29E-06	1.15E-06	9.44E-07	2.09E-07	2.27E-11	1.67E-17	4.51E-33	0.0
TOTALS	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.30E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
TL208	1.04E-02	3.61E-03	2.27E-04	1.03E-04	6.35E-05	2.43E-05	8.51E-08	7.44E-08	7.91E-08	1.34E-07	7.96E-07
PB212	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
BI212	2.90E-02	1.01E-02	6.33E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
PO212	1.86E-02	6.45E-03	4.06E-04	1.83E-04	1.13E-04	4.34E-05	1.52E-07	1.33E-07	1.41E-07	2.39E-07	1.42E-06
PO216	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
RN220	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
RA224	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
TH228	2.89E-02	1.00E-02	6.32E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
TH234	3.17E-01	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03
PA233	3.09E-01	3.10E-01	3.14E-01	3.27E-01	3.46E-01	3.81E-01	5.24E-01	5.78E-01	5.62E-01	4.20E-01	2.28E-02
PA234M	3.17E-01	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03
U 234	9.54E-03	9.73E-03	1.04E-02	1.20E-02	1.41E-02	1.68E-02	1.98E-02	1.95E-02	1.58E-02	4.16E-03	3.17E-03
U 236	2.62E-03	2.62E-03	2.62E-03	2.63E-03	2.64E-03	2.67E-03	2.87E-03	4.30E-03	5.18E-03	5.04E-03	3.86E-03
U 237	2.24E-02	1.94E-02	1.20E-02	2.83E-03	2.58E-04	5.48E-06	3.19E-06	1.53E-06	9.95E-10	0.0	0.0
U 238	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03
NP237	3.09E-01	3.10E-01	3.14E-01	3.27E-01	3.46E-01	3.81E-01	5.24E-01	5.78E-01	5.62E-01	4.20E-01	2.28E-02
NP238	1.88E-02	1.85E-02	1.77E-02	1.54E-02	1.23E-02	7.79E-03	2.03E-04	3.08E-22	0.0	0.0	0.0
NP239	1.67E+01	1.67E+01	1.67E+01	1.66E+01	1.66E+01	1.64E+01	1.52E+01	6.54E+00	1.40E-03	3.24E-07	2.17E-07
PU238	2.36E+01	2.31E+01	2.16E+01	1.76E+01	1.26E+01	6.58E+00	8.60E-02	1.21E-19	0.0	0.0	0.0
PU239	3.40E+00	3.40E+00	3.41E+00	3.43E+00	3.45E+00	3.49E+00	3.77E+00	5.21E+00	6.06E-01	3.24E-07	2.17E-07
PU240	5.09E+00	5.53E+00	6.69E+00	8.38E+00	9.01E+00	9.03E+00	8.30E+00	3.20E+00	2.30E-04	6.85E-09	6.67E-09
PU241	8.95E+02	7.74E+02	4.79E+02	1.13E+02	1.03E+01	2.19E-01	1.28E-01	6.12E-02	3.98E-05	0.0	0.0
PU242	1.86E-02	1.86E-02	1.86E-02	1.86E-02	1.87E-02	1.88E-02	1.92E-02	2.06E-02	1.81E-02	3.60E-03	3.60E-10
AM241	1.31E+03	1.31E+03	1.30E+03	1.25E+03	1.16E+03	9.86E+02	2.73E+02	6.14E-02	4.19E-05	0.0	0.0
AM242M	3.75E+00	3.70E+00	3.54E+00	3.09E+00	2.46E+00	1.56E+00	4.06E-02	6.16E-20	0.0	0.0	0.0
AM242	3.74E+00	3.68E+00	3.52E+00	3.07E+00	2.44E+00	1.55E+00	4.04E-02	6.13E-20	0.0	0.0	0.0
AM243	1.67E+01	1.67E+01	1.67E+01	1.66E+01	1.66E+01	1.64E+01	1.52E+01	6.54E+00	1.40E-03	3.24E-07	2.17E-07
CM242	3.89E+00	3.06E+00	2.91E+00	2.54E+00	2.02E+00	1.28E+00	3.34E-02	5.08E-20	0.0	0.0	0.0
CM243	1.41E+01	1.31E+01	1.03E+01	4.96E+00	1.47E+00	1.29E-01	4.61E-10	0.0	0.0	0.0	0.0
CM244	1.49E+03	1.33E+03	9.04E+02	2.87E+02	4.23E+01	9.22E-01	4.68E-14	0.0	0.0	0.0	0.0
CM245	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.37E-01	1.36E-01	1.27E-01	6.11E-02	3.97E-05	0.0	0.0
CM246	2.25E-01	2.24E-01	2.24E-01	2.23E-01	2.22E-01	2.18E-01	1.94E-01	5.20E-02	9.78E-08	6.77E-31	0.0
TOTAL	3.79E+03	3.50E+03	2.77E+03	1.73E+03	1.28E+03	1.04E+03	3.18E+02	2.30E+01	3.71E+00	4.93E+00	3.04E-01

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP239	1.69E-02	1.69E-02	1.69E-02	1.69E-02	1.68E-02	1.66E-02	1.54E-02	6.62E-03	1.42E-06	3.28E-10	2.20E-10
AM241	1.90E-01	1.89E-01	1.88E-01	1.81E-01	1.67E-01	1.42E-01	3.95E-02	8.87E-06	6.06E-09	0.0	0.0
AM243	5.46E-03	5.46E-03	5.46E-03	5.44E-03	5.42E-03	5.37E-03	4.98E-03	2.14E-03	4.57E-07	1.06E-10	7.10E-11
CM243	1.11E-02	1.03E-02	8.11E-03	3.91E-03	1.16E-03	1.02E-04	3.63E-13	0.0	0.0	0.0	0.0
CM244	1.35E-02	1.20E-02	8.19E-03	2.60E-03	3.83E-04	8.35E-06	4.24E-19	0.0	0.0	0.0	0.0
TOTAL	2.39E-01	2.36E-01	2.28E-01	2.11E-01	1.92E-01	1.66E-01	6.08E-02	9.67E-03	1.33E-03	1.61E-03	1.21E-04

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.30E+04 MWD

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
BI212	1.04E-02	3.61E-03	2.27E-04	1.03E-04	6.35E-05	2.43E-05	8.51E-08	7.44E-08	7.91E-08	1.34E-07	7.96E-07
PO212	1.86E-02	6.45E-03	4.06E-04	1.83E-04	1.13E-04	4.34E-05	1.52E-07	1.33E-07	1.41E-07	2.39E-07	1.42E-06
PO216	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
RN220	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
RA224	2.90E-02	1.01E-02	6.34E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
TH228	2.89E-02	1.00E-02	6.32E-04	2.86E-04	1.77E-04	6.76E-05	2.37E-07	2.07E-07	2.20E-07	3.72E-07	2.22E-06
U 234	9.54E-03	9.73E-03	1.04E-02	1.20E-02	1.41E-02	1.68E-02	1.98E-02	1.95E-02	1.58E-02	4.16E-03	3.17E-03
U 236	2.62E-03	2.62E-03	2.62E-03	2.63E-03	2.64E-03	2.67E-03	2.87E-03	4.30E-03	5.18E-03	5.04E-03	3.86E-03
U 238	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03
NP237	3.09E-01	3.10E-01	3.14E-01	3.27E-01	3.46E-01	3.81E-01	5.24E-01	5.78E-01	5.62E-01	4.20E-01	2.28E-02
PU238	2.36E+01	2.31E+01	2.16E+01	1.76E+01	1.26E+01	6.58E+00	8.60E-02	1.21E-19	0.0	0.0	0.0
PU239	3.40E+00	3.40E+00	3.41E+00	3.43E+00	3.45E+00	3.49E+00	3.77E+00	5.21E+00	6.06E-01	3.24E-07	2.17E-07
PU240	5.09E+00	5.53E+00	6.69E+00	8.38E+00	9.01E+00	9.03E+00	8.30E+00	3.20E+00	2.30E-04	6.85E-09	6.67E-09
PU241	2.24E-02	1.94E-02	1.20E-02	2.83E-03	2.58E-04	5.47E-06	3.19E-06	1.53E-06	9.95E-10	0.0	0.0
PU242	1.86E-02	1.86E-02	1.86E-02	1.86E-02	1.87E-02	1.88E-02	1.92E-02	2.06E-02	1.81E-02	3.60E-03	3.60E-10
AM241	1.31E+03	1.31E+03	1.30E+03	1.25E+03	1.16E+03	9.86E+02	2.73E+02	6.14E-02	4.19E-05	0.0	0.0
AM242M	1.88E-02	1.85E-02	1.77E-02	1.54E-02	1.23E-02	7.79E-03	2.03E-04	3.08E-22	0.0	0.0	0.0
AM243	1.67E+01	1.67E+01	1.67E+01	1.66E+01	1.66E+01	1.64E+01	1.52E+01	6.54E+00	1.40E-03	3.24E-07	2.17E-07
CM242	3.89E+00	3.06E+00	2.91E+00	2.54E+00	2.02E+00	1.28E+00	3.34E-02	5.08E-20	0.0	0.0	0.0
CM243	1.41E+01	1.31E+01	1.03E+01	4.95E+00	1.47E+00	1.29E-01	4.60E-10	0.0	0.0	0.0	0.0
CM244	1.49E+03	1.33E+03	9.04E+02	2.87E+02	4.23E+01	9.22E-01	4.68E-14	0.0	0.0	0.0	0.0
CM245	1.38E-01	1.38E-01	1.38E-01	1.38E-01	1.37E-01	1.36E-01	1.27E-01	6.11E-02	3.97E-05	0.0	0.0
CM246	2.25E-01	2.24E-01	2.24E-01	2.23E-01	2.22E-01	2.18E-01	1.94E-01	5.20E-02	9.78E-08	6.77E-31	0.0
TOTAL	2.87E+03	2.70E+03	2.27E+03	1.59E+03	1.25E+03	1.02E+03	3.02E+02	1.58E+01	2.49E+00	3.14E+00	1.92E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP237	9.07E-03	9.11E-03	9.23E-03	9.59E-03	1.02E-02	1.12E-02	1.54E-02	1.70E-02	1.65E-02	1.23E-02	6.69E-04
NP239	4.04E-02	4.04E-02	4.03E-02	4.02E-02	4.00E-02	3.97E-02	3.68E-02	1.58E-02	3.38E-06	7.83E-10	5.25E-10
PU238	7.81E-01	7.65E-01	7.15E-01	5.83E-01	4.17E-01	2.18E-01	2.85E-03	4.01E-21	0.0	0.0	0.0
PU239	1.06E-01	1.06E-01	1.06E-01	1.07E-01	1.07E-01	1.09E-01	1.17E-01	1.62E-01	1.89E-02	1.01E-08	6.75E-09
PU240	1.59E-01	1.72E-01	2.09E-01	2.61E-01	2.81E-01	2.81E-01	2.58E-01	9.96E-02	7.16E-06	2.13E-10	2.08E-10
PU241	2.76E-02	2.39E-02	1.48E-02	3.49E-03	3.18E-04	6.75E-06	3.93E-06	1.89E-06	1.23E-09	0.0	0.0
AM241	4.39E+01	4.38E+01	4.34E+01	4.18E+01	3.87E+01	3.29E+01	9.14E+00	2.05E-03	1.40E-06	0.0	0.0
AM242M	1.48E-03	1.46E-03	1.40E-03	1.22E-03	9.70E-04	6.15E-04	1.60E-05	2.43E-23	0.0	0.0	0.0
AM242	4.34E-03	4.28E-03	4.09E-03	3.57E-03	2.84E-03	1.80E-03	4.69E-05	7.12E-23	0.0	0.0	0.0
AM243	5.39E-01	5.39E-01	5.38E-01	5.37E-01	5.34E-01	5.29E-01	4.91E-01	2.11E-01	4.51E-05	1.04E-08	7.00E-09
CM242	1.43E-01	1.13E-01	1.07E-01	9.36E-02	7.45E-02	4.72E-02	1.23E-03	1.87E-21	0.0	0.0	0.0
CM243	5.18E-01	4.82E-01	3.78E-01	1.82E-01	5.40E-02	4.75E-03	1.69E-11	0.0	0.0	0.0	0.0
CM244	5.20E+01	4.64E+01	3.16E+01	1.00E+01	1.48E+00	3.23E-02	1.64E-15	0.0	0.0	0.0	0.0
CM245	4.60E-03	4.60E-03	4.60E-03	4.59E-03	4.57E-03	4.53E-03	4.24E-03	2.04E-03	1.32E-06	0.0	0.0
CM246	7.35E-03	7.35E-03	7.34E-03	7.31E-03	7.25E-03	7.15E-03	6.36E-03	1.70E-03	3.20E-09	2.22E-32	0.0
TOTAL	9.82E+01	9.24E+01	7.72E+01	5.36E+01	4.17E+01	3.42E+01	1.01E+01	5.17E-01	8.72E-02	1.18E-01	7.13E-03

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=3.30E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	3.36E+02	2.84E+02	1.62E+02	3.01E+01	1.82E+00	6.64E-03	2.11E-22	0.0	0.0	0.0	0.0
SE 79	3.69E-01	3.69E-01	3.69E-01	3.69E-01	3.69E-01	3.69E-01	3.65E-01	3.32E-01	1.27E-01	8.60E-06	0.0
KR 85	5.98E+03	4.93E+03	2.58E+03	3.71E+02	1.47E+01	2.23E-02	0.0	0.0	0.0	0.0	0.0
SR 90	6.32E+04	5.89E+04	4.64E+04	2.27E+04	6.92E+03	6.40E+02	3.46E-06	0.0	0.0	0.0	0.0
Y 90	6.33E+04	5.89E+04	4.64E+04	2.27E+04	6.92E+03	6.41E+02	3.46E-06	0.0	0.0	0.0	0.0
ZR 93	1.79E+00	1.79E+00	1.79E+00	1.79E+00	1.79E+00	1.79E+00	1.79E+00	1.78E+00	1.71E+00	1.14E+00	1.93E-02
NB 93M	6.02E-01	7.58E-01	1.14E+00	1.58E+00	1.69E+00	1.70E+00	1.70E+00	1.69E+00	1.63E+00	1.08E+00	1.84E-02
TC 99	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.32E+01	1.31E+01	1.28E+01	9.52E+00	5.09E-01	9.79E-14
RU106	4.06E+03	5.17E+02	5.26E-01	5.80E-10	6.84E-25	0.0	0.0	0.0	0.0	0.0	0.0
RH106	4.06E+03	5.17E+02	5.34E-01	5.80E-10	6.84E-25	0.0	0.0	0.0	0.0	0.0	0.0
PD107	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.01E-01	9.14E-02	3.50E-02
AG110	3.22E-02	1.54E-03	6.16E-08	3.91E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	2.42E+00	1.16E-01	4.63E-06	2.94E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	2.55E+01	2.21E+01	1.37E+01	3.30E+00	3.07E-01	2.66E-03	8.34E-20	0.0	0.0	0.0	0.0
SN119M	7.97E-02	3.59E-03	1.17E-07	4.07E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	1.42E-01	1.36E-01	1.19E-01	7.83E-02	3.91E-02	9.78E-03	1.49E-07	0.0	0.0	0.0	0.0
SN126	5.42E-01	5.42E-01	5.42E-01	5.42E-01	5.42E-01	5.41E-01	5.38E-01	5.06E-01	2.71E-01	5.35E-04	0.0
SB125	1.56E+03	7.33E+02	5.95E+01	3.19E-02	1.13E-07	1.41E-18	0.0	0.0	0.0	0.0	0.0
SB126	7.59E-02	7.59E-02	7.59E-02	7.59E-02	7.59E-02	7.58E-02	7.54E-02	7.08E-02	3.80E-02	7.49E-05	0.0
SB126M	5.42E-01	5.42E-01	5.42E-01	5.42E-01	5.42E-01	5.41E-01	5.38E-01	5.06E-01	2.71E-01	5.35E-04	0.0
TE125M	3.80E+02	1.79E+02	1.45E+01	7.78E-03	2.75E-08	3.44E-19	0.0	0.0	0.0	0.0	0.0
I 129	3.12E-02	3.12E-02	3.12E-02	3.12E-02	3.12E-02	3.12E-02	3.12E-02	3.12E-02	3.11E-02	2.99E-02	2.01E-02
CS134	1.49E+04	5.43E+03	1.89E+02	7.88E-03	3.97E-10	1.01E-24	0.0	0.0	0.0	0.0	0.0
CS135	3.52E-01	3.52E-01	3.52E-01	3.52E-01	3.52E-01	3.52E-01	3.52E-01	3.51E-01	3.41E-01	2.60E-01	1.73E-02
CS137	9.06E+04	8.46E+04	6.71E+04	3.36E+04	1.06E+04	1.05E+03	9.89E-06	0.0	0.0	0.0	0.0
BA137M	8.57E+04	8.00E+04	6.35E+04	3.18E+04	1.00E+04	9.93E+02	9.36E-06	0.0	0.0	0.0	0.0
CE144	2.15E+03	1.49E+02	2.02E-02	5.07E-14	2.34E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144	2.15E+03	1.49E+02	2.02E-02	5.07E-14	2.34E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144M	2.59E+01	1.79E+00	2.43E-04	6.08E-16	2.81E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	2.52E+04	1.14E+04	8.14E+02	2.94E-01	5.41E-07	1.83E-18	0.0	0.0	0.0	0.0	0.0
SM151	3.19E+02	3.12E+02	2.88E+02	2.29E+02	1.56E+02	7.21E+01	1.52E-01	0.0	0.0	0.0	0.0
EU152	3.34E+00	2.87E+00	1.72E+00	3.74E-01	2.93E-02	1.79E-04	3.55E-22	0.0	0.0	0.0	0.0
EU154	6.98E+03	5.48E+03	2.45E+03	2.18E+02	3.89E+00	1.23E-03	0.0	0.0	0.0	0.0	0.0
EU155	2.77E+03	1.82E+03	4.50E+02	6.81E+00	6.29E-03	5.37E-09	0.0	0.0	0.0	0.0	0.0
TOTAL	3.74E+05	3.14E+05	2.31E+05	1.12E+05	3.46E+04	3.41E+03	1.88E+01	1.81E+01	1.40E+01	3.11E+00	1.10E-01

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=3.30E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	1.14E-02	9.59E-03	5.47E-03	1.02E-03	6.15E-05	2.24E-07	7.14E-27	0.0	0.0	0.0	0.0
KR 85	8.96E+00	7.38E+00	3.87E+00	5.56E-01	2.20E-02	3.35E-05	0.0	0.0	0.0	0.0	0.0
SR 90	7.34E+01	6.83E+01	5.39E+01	2.64E+01	8.03E+00	7.43E-01	4.01E-09	0.0	0.0	0.0	0.0
Y 90	3.51E+02	3.26E+02	2.57E+02	1.26E+02	3.83E+01	3.55E+00	1.92E-08	0.0	0.0	0.0	0.0
TC 99	6.61E-03	6.61E-03	6.61E-03	6.61E-03	6.61E-03	6.60E-03	6.59E-03	6.40E-03	4.77E-03	2.55E-04	4.91E-17
RU106	2.41E-01	3.06E-02	3.12E-05	3.44E-14	4.06E-29	0.0	0.0	0.0	0.0	0.0	0.0
RH106	3.90E+01	4.96E+00	5.12E-03	5.57E-12	6.56E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	4.05E-02	1.94E-03	7.73E-08	4.91E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	4.29E-02	3.72E-02	2.31E-02	5.56E-03	5.17E-04	4.47E-06	1.40E-22	0.0	0.0	0.0	0.0
SB125	4.86E+00	2.29E+00	1.86E-01	9.97E-05	3.53E-10	4.41E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.39E-03	1.31E-03	7.01E-04	1.38E-06	0.0
SB126M	6.90E-03	6.90E-03	6.90E-03	6.90E-03	6.90E-03	6.89E-03	6.86E-03	6.44E-03	3.45E-03	6.81E-06	0.0
TE125M	3.19E-01	1.50E-01	1.22E-02	6.54E-06	2.31E-11	2.89E-22	0.0	0.0	0.0	0.0	0.0
CS134	1.52E+02	5.53E+01	1.92E+00	8.02E-05	4.04E-12	1.02E-26	0.0	0.0	0.0	0.0	0.0
CS137	1.00E+02	9.35E+01	7.42E+01	3.71E+01	1.17E+01	1.16E+00	1.09E-08	0.0	0.0	0.0	0.0
BA137M	3.37E+02	3.14E+02	2.49E+02	1.25E+02	3.93E+01	3.90E+00	3.67E-08	0.0	0.0	0.0	0.0
CE144	1.43E+00	9.88E-02	1.34E-05	3.36E-17	1.55E-36	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.58E+01	1.10E+00	1.49E-04	3.73E-16	1.72E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	9.05E+00	4.10E+00	2.92E-01	1.06E-04	1.94E-10	6.55E-22	0.0	0.0	0.0	0.0	0.0
SM151	3.74E-02	3.66E-02	3.39E-02	2.69E-02	1.83E-02	8.47E-03	1.78E-05	0.0	0.0	0.0	0.0
EU152	2.53E-02	2.18E-02	1.31E-02	2.83E-03	2.22E-04	1.36E-06	2.69E-24	0.0	0.0	0.0	0.0
EU154	6.25E+01	4.91E+01	2.19E+01	1.95E+00	3.48E-02	1.10E-05	0.0	0.0	0.0	0.0	0.0
EU155	2.01E+00	1.32E+00	3.28E-01	4.95E-03	4.57E-06	3.91E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	1.16E+03	9.28E+02	6.63E+02	3.17E+02	9.74E+01	9.38E+00	1.63E-02	1.55E-02	9.92E-03	6.95E-04	2.27E-05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
KR 85	3.74E-01	3.08E-01	1.61E-01	2.32E-02	9.16E-04	1.40E-06	0.0	0.0	0.0	0.0	0.0
SR 90	4.12E+00	3.83E+00	3.02E+00	1.48E+00	4.50E-01	4.17E-02	2.25E-10	0.0	0.0	0.0	0.0
Y 90	5.61E+01	5.23E+01	4.12E+01	2.02E+01	6.14E+00	5.69E-01	3.07E-09	0.0	0.0	0.0	0.0
RH106	8.54E+00	1.09E+00	1.12E-03	1.22E-12	1.44E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	3.95E-02	1.89E-03	7.54E-08	4.79E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SB125	3.99E+00	1.88E+00	1.53E-01	8.18E-05	2.89E-10	3.62E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.27E-03	1.19E-03	6.40E-04	1.26E-06	0.0
SB126M	5.08E-03	5.08E-03	5.08E-03	5.08E-03	5.08E-03	5.08E-03	5.05E-03	4.74E-03	2.54E-03	5.02E-06	0.0
TE125M	8.00E-02	3.77E-02	3.06E-03	1.64E-06	5.80E-12	7.25E-23	0.0	0.0	0.0	0.0	0.0
CS134	1.38E+02	5.02E+01	1.74E+00	7.28E-05	3.67E-12	9.30E-27	0.0	0.0	0.0	0.0	0.0
CS137	2.53E+00	2.36E+00	1.87E+00	9.37E-01	2.95E-01	2.93E-02	2.76E-10	0.0	0.0	0.0	0.0
BA137M	3.02E+02	2.82E+02	2.24E+02	1.12E+02	3.53E+01	3.50E+00	3.30E-08	0.0	0.0	0.0	0.0
CE144	2.78E-01	1.92E-02	2.61E-06	6.54E-18	3.02E-37	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.87E+00	1.29E-01	1.75E-05	4.39E-17	2.03E-36	0.0	0.0	0.0	0.0	0.0	0.0
PM147	1.18E-01	5.34E-02	3.81E-03	1.38E-06	2.53E-12	8.54E-24	0.0	0.0	0.0	0.0	0.0
EU152	2.27E-02	1.95E-02	1.17E-02	2.54E-03	1.99E-04	1.22E-06	2.41E-24	0.0	0.0	0.0	0.0
EU154	5.12E+01	4.02E+01	1.80E+01	1.60E+00	2.85E-02	9.02E-06	0.0	0.0	0.0	0.0	0.0
EU155	1.00E+00	6.59E-01	1.63E-01	2.46E-03	2.28E-06	1.94E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	5.70E+02	4.35E+02	2.90E+02	1.36E+02	4.22E+01	4.15E+00	6.75E-03	6.34E-03	3.45E-03	3.27E-05	3.27E-06

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE LIGHT ELEMENTS

BURNUP=3.30E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
C 14	1.43E-01	1.43E-01	1.42E-01	1.42E-01	1.41E-01	1.39E-01	1.27E-01	4.26E-02	7.97E-07	0.0	0.0
MN 54	2.46E+01	2.17E+00	6.58E-04	1.84E-14	4.75E-32	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	9.61E+03	4.32E+03	3.01E+02	1.01E-01	1.65E-07	4.39E-19	0.0	0.0	0.0	0.0	0.0
CO 60	8.17E+03	5.51E+03	1.48E+03	2.86E+01	4.04E-02	7.86E-08	0.0	0.0	0.0	0.0	0.0
NI 59	1.75E+01	1.75E+01	1.75E+01	1.74E+01	1.74E+01	1.74E+01	1.73E+01	1.60E+01	7.34E+00	3.02E-03	0.0
NI 63	2.35E+03	2.30E+03	2.13E+03	1.70E+03	1.17E+03	5.50E+02	1.34E+00	0.0	0.0	0.0	0.0
SR 90	1.90E-03	1.77E-03	1.40E-03	6.84E-04	2.08E-04	1.93E-05	1.04E-13	0.0	0.0	0.0	0.0
Y 90	1.90E-03	1.77E-03	1.40E-03	6.85E-04	2.08E-04	1.93E-05	1.04E-13	0.0	0.0	0.0	0.0
ZR 93	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.84E-01	1.83E-01	1.76E-01	1.17E-01	1.98E-03
NB 93M	6.06E-02	7.67E-02	1.16E-01	1.62E-01	1.74E-01	1.75E-01	1.75E-01	1.74E-01	1.67E-01	1.11E-01	1.88E-03
NB 94	1.04E+00	1.04E+00	1.04E+00	1.04E+00	1.04E+00	1.04E+00	1.01E+00	7.42E-01	3.43E-02	1.56E-15	0.0
MO 93	7.13E-02	7.12E-02	7.11E-02	7.07E-02	7.00E-02	6.86E-02	5.85E-02	9.84E-03	1.78E-10	0.0	0.0
TC 99	1.20E-02	1.20E-02	1.20E-02	1.20E-02	1.20E-02	1.20E-02	1.19E-02	1.16E-02	8.65E-03	4.63E-04	8.89E-17
SN119M	7.08E+00	3.19E-01	1.04E-05	3.62E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	8.38E-01	8.04E-01	7.00E-01	4.62E-01	2.31E-01	5.77E-02	8.77E-07	0.0	0.0	0.0	0.0
SB125	1.34E+01	6.32E+00	5.13E-01	2.75E-04	9.73E-10	1.22E-20	0.0	0.0	0.0	0.0	0.0
TE125M	3.28E+00	1.54E+00	1.25E-01	6.71E-05	2.37E-10	2.97E-21	0.0	0.0	0.0	0.0	0.0
TOTAL	2.02E+04	1.22E+04	3.93E+03	1.75E+03	1.19E+03	5.69E+02	2.02E+01	1.72E+01	7.73E+00	2.31E-01	3.87E-03

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.22E-01	1.08E-02	3.27E-06	9.17E-17	2.37E-34	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	1.32E+01	5.94E+00	4.13E-01	1.39E-04	2.27E-10	6.03E-22	0.0	0.0	0.0	0.0	0.0
CO 60	1.26E+02	8.49E+01	2.28E+01	4.41E-01	6.23E-04	1.21E-09	0.0	0.0	0.0	0.0	0.0
NI 59	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.10E-01	1.02E-01	4.67E-02	1.92E-05	0.0
NI 63	9.34E-01	9.13E-01	8.47E-01	6.76E-01	4.64E-01	2.18E-01	5.32E-04	0.0	0.0	0.0	0.0
NB 94	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.03E-02	7.56E-03	3.50E-04	1.59E-17	0.0
SN121M	1.68E-03	1.61E-03	1.40E-03	9.25E-04	4.62E-04	1.16E-04	1.76E-09	0.0	0.0	0.0	0.0
SB125	4.20E-02	1.98E-02	1.60E-03	8.60E-07	3.04E-12	3.80E-23	0.0	0.0	0.0	0.0	0.0
TE125M	2.75E-03	1.30E-03	1.05E-04	5.64E-08	2.00E-13	2.50E-24	0.0	0.0	0.0	0.0	0.0
TOTAL	1.40E+02	9.19E+01	2.42E+01	1.24E+00	5.86E-01	3.40E-01	1.21E-01	1.10E-01	4.71E-02	5.27E-05	5.64E-07

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.22E-01	1.07E-02	3.26E-06	9.13E-17	2.36E-34	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	1.21E+02	8.17E+01	2.19E+01	4.24E-01	5.99E-04	1.17E-09	0.0	0.0	0.0	0.0	0.0
NB 94	9.74E-03	9.74E-03	9.74E-03	9.73E-03	9.71E-03	9.68E-03	9.42E-03	6.93E-03	3.21E-04	1.45E-17	0.0
SB125	3.44E-02	1.62E-02	1.32E-03	7.05E-07	2.49E-12	3.12E-23	0.0	0.0	0.0	0.0	0.0
TOTAL	1.21E+02	8.17E+01	2.19E+01	4.35E-01	1.08E-02	9.90E-03	9.43E-03	6.93E-03	3.23E-04	1.31E-06	2.21E-08

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE

BURNUP=3.30E+04 MWD

********** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS **********

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
				CONCENTRATIONS ,GRAMS							
TOTAL	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05	6.05E+05
				RADIOACTIVITY ,CURIES							
TOTAL	3.98E+05	3.30E+05	2.37E+05	1.15E+05	3.71E+04	5.03E+03	3.57E+02	5.83E+01	2.55E+01	8.27E+00	4.18E-01
				THERMAL POWER ,WATTS							
TOTAL	1.40E+03	1.11E+03	7.65E+02	3.72E+02	1.40E+02	4.40E+01	1.02E+01	6.42E-01	1.44E-01	1.19E-01	7.15E-03
				GAMMA POWER ,WATTS							
TOTAL	6.92E+02	5.17E+02	3.12E+02	1.37E+02	4.24E+01	4.32E+00	7.70E-02	2.29E-02	5.11E-03	1.64E-03	1.25E-04
				ALPHA-ACTIVITY ,CURIES							
TOTAL	2.87E+03	2.70E+03	2.27E+03	1.59E+03	1.25E+03	1.02E+03	3.02E+02	1.58E+01	2.49E+00	3.14E+00	1.92E-01

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR ACTINIDES

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 33.03 MW, BURNUP= 3.30E+04 MWD

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.00E-02	2.06E+13	1.96E+13	1.71E+13	1.30E+13	1.07E+13	8.97E+12	2.98E+12	3.45E+11	4.88E+10	5.74E+10	3.40E+09
2.50E-02	1.24E+12	1.24E+12	1.23E+12	1.18E+12	1.09E+12	9.33E+11	2.65E+11	5.66E+09	3.30E+09	3.37E+09	2.37E+08
3.75E-02	1.66E+11	1.63E+11	1.57E+11	1.46E+11	1.35E+11	1.21E+11	6.23E+10	1.80E+10	3.65E+09	7.77E+09	4.33E+08
5.75E-02	1.81E+13	1.80E+13	1.79E+13	1.72E+13	1.59E+13	1.36E+13	3.78E+12	6.63E+09	9.96E+08	1.62E+09	1.45E+08
8.50E-02	5.79E+11	5.70E+11	5.51E+11	5.15E+11	4.90E+11	4.75E+11	4.35E+11	1.93E+11	1.69E+10	2.13E+10	1.19E+09
1.25E-01	4.80E+11	4.67E+11	4.35E+11	3.74E+11	3.32E+11	3.11E+11	2.78E+11	1.20E+11	3.00E+09	3.68E+09	2.16E+08
2.25E-01	3.77E+11	3.63E+11	3.29E+11	2.64E+11	2.21E+11	2.03E+11	1.87E+11	8.06E+10	2.79E+09	5.07E+09	3.47E+08
3.75E-01	2.67E+10	2.60E+10	2.57E+10	2.53E+10	2.53E+10	2.54E+10	2.58E+10	1.64E+10	1.09E+10	1.17E+10	6.77E+08
5.75E-01	2.04E+09	1.37E+09	9.86E+08	6.14E+08	4.47E+08	3.76E+08	1.82E+08	8.18E+07	3.82E+08	4.63E+08	8.69E+07
8.50E-01	1.58E+09	1.21E+09	8.84E+08	4.84E+08	2.92E+08	1.98E+08	4.75E+07	1.58E+07	8.16E+07	8.64E+07	2.12E+07
1.25E+00	6.81E+08	5.40E+08	4.09E+08	2.15E+08	1.21E+08	7.38E+07	1.41E+07	2.24E+07	1.59E+08	1.18E+08	4.41E+07
1.75E+00	2.62E+08	2.04E+08	1.33E+08	4.67E+07	1.24E+07	6.43E+06	6.07E+06	2.17E+07	2.53E+08	3.72E+08	4.90E+07
2.25E+00	1.24E+08	1.11E+08	7.66E+07	2.67E+07	6.97E+06	3.58E+06	3.26E+06	5.16E+06	3.28E+07	1.49E+07	9.98E+06
2.75E+00	4.37E+08	1.91E+08	5.25E+07	1.91E+07	6.27E+06	2.92E+06	1.80E+06	5.63E+05	5.78E+05	2.65E+05	2.02E+05
3.50E+00	4.28E+07	3.83E+07	2.65E+07	9.24E+06	2.41E+06	1.23E+06	1.07E+06	3.02E+05	1.10E+05	4.93E+04	3.26E+04
5.00E+00	1.34E+07	1.19E+07	8.27E+06	2.89E+06	7.51E+05	3.84E+05	3.33E+05	9.01E+04	9.79E+02	1.96E+02	1.10E+01
7.00E+00	1.53E+06	1.37E+06	9.48E+05	3.31E+05	8.61E+04	4.40E+04	3.82E+04	1.03E+04	1.12E+02	2.22E+01	1.24E+00
9.50E+00	1.53E+05	1.37E+05	9.47E+04	3.30E+04	8.59E+03	4.39E+03	3.81E+03	1.03E+03	1.12E+01	2.20E+00	1.23E-01
TOTAL	4.15E+13	4.05E+13	3.77E+13	3.27E+13	2.90E+13	2.46E+13	8.01E+12	7.85E+11	9.13E+10	1.13E+11	6.85E+09
MEV/SEC	1.49E+12	1.47E+12	1.42E+12	1.32E+12	1.20E+12	1.03E+12	3.79E+11	6.04E+10	8.32E+09	1.00E+10	7.57E+08

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 33.03 MW, BURNUP= 3.30E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	3.55E+12	2.36E+12	6.48E+11	4.09E+10	2.04E+10	1.02E+10	1.12E+09	7.40E+08	1.16E+08	5.44E+07	9.17E+05
3.75E-01	2.04E+11	1.01E+11	1.14E+10	9.52E+07	3.58E+06	2.99E+06	2.87E+06	2.11E+06	9.77E+04	2.18E+01	3.08E-01
5.75E-01	2.32E+11	1.09E+11	9.07E+09	1.06E+07	2.68E+05	2.41E+04	2.02E+01	1.98E+01	1.61E+01	2.10E+00	1.84E-03
8.50E-01	9.87E+11	1.65E+11	7.55E+10	7.13E+10	7.11E+10	7.09E+10	6.90E+10	5.07E+10	2.35E+09	1.09E-04	2.14E-15
1.25E+00	6.04E+14	4.07E+14	1.09E+14	2.12E+12	2.99E+09	8.29E+03	1.34E-05	3.38E-08	2.75E-08	3.46E-09	7.87E-14
1.75E+00	1.92E+04	1.78E+04	1.40E+04	6.86E+03	2.09E+03	1.93E+02	1.04E-06	0.0	0.0	0.0	0.0
2.25E+00	3.20E+09	2.16E+09	5.80E+08	1.12E+07	1.58E+04	5.21E-02	1.15E-10	0.0	0.0	0.0	0.0
2.75E+00	9.91E+06	6.68E+06	1.79E+06	3.47E+04	4.90E+01	9.54E-05	0.0	0.0	0.0	0.0	0.0
3.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
5.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	6.09E+14	4.10E+14	1.10E+14	2.23E+12	9.45E+10	8.11E+10	7.01E+10	5.15E+10	2.47E+09	5.44E+07	9.17E+05
MEV/SEC	7.57E+14	5.10E+14	1.37E+14	2.71E+12	6.73E+10	6.18E+10	5.88E+10	4.32E+10	2.01E+09	8.16E+06	1.38E+05

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 33.03 MW, BURNUP= 3.30E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	2.13E+15	1.83E+15	1.36E+15	6.42E+14	1.94E+14	1.81E+13	2.14E+10	2.04E+10	1.32E+10	1.10E+09	1.36E+08
3.75E-01	1.90E+14	1.56E+14	1.14E+14	5.48E+13	1.67E+13	1.56E+12	2.18E+10	2.05E+10	1.10E+10	2.17E+07	3.55E-05
5.75E-01	4.15E+15	3.42E+15	2.50E+15	1.24E+15	3.90E+14	3.87E+13	4.85E+10	4.56E+10	2.44E+10	4.82E+07	1.37E-09
8.50E-01	6.37E+14	2.97E+14	6.74E+13	1.27E+13	2.76E+12	2.52E+11	2.45E+09	2.29E+09	1.22E+09	2.41E+06	1.84E+00
1.25E+00	1.81E+14	1.27E+14	5.42E+13	7.18E+12	9.65E+11	8.28E+10	5.89E+08	5.53E+08	2.97E+08	5.85E+05	2.31E-03
1.75E+00	5.52E+12	3.89E+12	1.90E+12	3.56E+11	7.16E+10	6.42E+09	2.51E+04	2.36E+04	1.26E+04	2.50E+01	1.16E-03
2.25E+00	8.28E+11	7.01E+10	8.59E+07	2.51E+07	7.63E+06	7.06E+05	4.39E-03	5.79E-04	5.79E-04	5.79E-04	5.79E-04
2.75E+00	3.29E+10	4.15E+09	4.25E+06	4.91E-03	2.90E-04	2.90E-04	2.90E-04	2.90E-04	2.90E-04	2.90E-04	2.90E-04
3.50E+00	4.22E+09	5.37E+08	5.55E+05	8.17E-04	2.14E-04	2.14E-04	2.14E-04	2.14E-04	2.14E-04	2.14E-04	2.14E-04
5.00E+00	6.19E-05	6.26E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05	6.31E-05
7.00E+00	4.05E-06	4.09E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06	4.13E-06
9.50E+00	2.62E-07	2.65E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07	2.67E-07
TOTAL	7.29E+15	5.83E+15	4.10E+15	1.96E+15	6.05E+14	5.87E+13	9.48E+10	8.93E+10	5.01E+10	1.18E+09	1.36E+08
MEV/SEC	3.56E+15	2.71E+15	1.81E+15	8.50E+14	2.63E+14	2.59E+13	4.21E+10	3.96E+10	2.15E+10	2.04E+08	2.04E+07

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PO212	3.22E+04	1.12E+04	7.04E+02	3.18E+02	1.96E+02	7.51E+01	2.63E-01	2.30E-01	2.45E-01	4.13E-01	2.46E+00
PO213	1.06E-01	1.07E-01	1.16E-01	2.13E-01	6.09E-01	2.34E+00	7.02E+01	7.32E+03	1.76E+05	3.87E+05	1.97E+04
PO216	8.25E+02	2.86E+02	1.80E+01	8.14E+00	5.03E+00	1.92E+00	6.75E-03	5.89E-03	6.27E-03	1.06E-02	6.31E-02
AT217	9.99E-03	1.01E-02	1.10E-02	2.01E-02	5.74E-02	2.20E-01	6.62E+00	6.90E+02	1.65E+04	3.64E+04	1.86E+03
RN220	4.60E+02	1.60E+02	1.00E+01	4.53E+00	2.80E+00	1.07E+00	3.76E-03	3.28E-03	3.50E-03	5.90E-03	3.51E-02
FR221	2.18E-03	2.19E-03	2.39E-03	4.37E-03	1.25E-02	4.79E-02	1.44E+00	1.50E+02	3.60E+03	7.93E+03	4.05E+02
NP237	1.35E+02	1.36E+02	1.37E+02	1.43E+02	1.51E+02	1.67E+02	2.29E+02	2.53E+02	2.46E+02	1.84E+02	9.96E+00
PU238	2.75E+04	2.70E+04	2.52E+04	2.05E+04	1.47E+04	7.68E+03	1.00E+02	1.41E-16	0.0	0.0	0.0
PU239	2.46E+03	2.46E+03	2.47E+03	2.48E+03	2.50E+03	2.53E+03	2.73E+03	3.77E+03	4.39E+02	2.34E-04	1.57E-04
PU240	3.80E+03	4.13E+03	4.99E+03	6.25E+03	6.72E+03	6.73E+03	6.19E+03	2.38E+03	1.71E-01	5.11E-06	4.98E-06
AM241	1.53E+06	1.53E+06	1.51E+06	1.46E+06	1.35E+06	1.15E+06	3.19E+05	7.15E+01	4.88E-02	0.0	0.0
AM243	2.68E+04	2.68E+04	2.68E+04	2.67E+04	2.66E+04	2.63E+04	2.44E+04	1.05E+04	2.24E+00	5.20E-04	3.49E-04
CM242	3.14E+04	2.47E+04	2.35E+04	2.05E+04	1.63E+04	1.03E+04	2.70E+02	4.10E-16	0.0	0.0	0.0
CM243	1.31E+05	1.22E+05	9.58E+04	4.62E+04	1.37E+04	1.20E+03	4.29E-06	0.0	0.0	0.0	0.0
CM244	1.05E+07	9.37E+06	6.39E+06	2.03E+06	2.99E+05	6.52E+03	3.30E-10	0.0	0.0	0.0	0.0
CM245	3.53E+02	3.53E+02	3.53E+02	3.52E+02	3.51E+02	3.48E+02	3.26E+02	1.56E+02	1.02E-01	0.0	0.0
CM246	3.96E+02	3.96E+02	3.96E+02	3.94E+02	3.91E+02	3.85E+02	3.43E+02	9.17E+01	1.73E-04	1.20E-27	0.0
TOTAL	1.23E+07	1.11E+07	8.08E+06	3.61E+06	1.73E+06	1.21E+06	3.53E+05	2.59E+04	2.01E+05	4.35E+05	2.32E+04

WASTE DECAY BIBLIS-TYPE 3.2% U235 REPROCESSING 7 Y AFTER DISCHARGE
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PU238	3.66E+03	3.58E+03	3.35E+03	2.73E+03	1.95E+03	1.02E+03	1.33E+01	1.88E-17	0.0	0.0	0.0
PU240	2.03E+04	2.21E+04	2.67E+04	3.35E+04	3.60E+04	3.61E+04	3.31E+04	1.28E+04	9.18E-01	2.74E-05	2.67E-05
PU242	8.19E+03	8.19E+03	8.20E+03	8.22E+03	8.25E+03	8.29E+03	8.45E+03	9.08E+03	7.97E+03	1.59E+03	1.59E-04
AM241	4.74E+02	4.74E+02	4.70E+02	4.52E+02	4.18E+02	3.56E+02	9.88E+01	2.22E-02	1.52E-05	0.0	0.0
AM243	2.80E+02	2.80E+02	2.80E+02	2.79E+02	2.78E+02	2.75E+02	2.55E+02	1.10E+02	2.35E-02	5.44E-06	3.64E-06
CM242	2.54E+04	1.99E+04	1.90E+04	1.65E+04	1.32E+04	8.35E+03	2.18E+02	3.31E-16	0.0	0.0	0.0
CM244	2.04E+08	1.82E+08	1.24E+08	3.94E+07	5.82E+06	1.27E+05	6.43E-09	0.0	0.0	0.0	0.0
CM246	6.50E+06	6.50E+06	6.49E+06	6.46E+06	6.42E+06	6.32E+06	5.62E+06	1.50E+06	2.83E+00	1.96E-23	0.0
CM248	7.29E+03	7.29E+03	7.29E+03	7.29E+03	7.29E+03	7.29E+03	7.27E+03	7.14E+03	5.94E+03	9.44E+02	9.69E-06
CF250	3.68E+03	3.14E+03	1.85E+03	3.77E+02	2.67E+01	1.33E-01	9.52E-06	6.65E-06	1.85E-07	5.01E-23	0.0
CF252	2.75E+04	1.25E+04	9.04E+02	3.42E-01	6.75E-07	2.64E-18	0.0	0.0	0.0	0.0	0.0
TOTAL	2.11E+08	1.89E+08	1.31E+08	4.59E+07	1.23E+07	6.51E+06	5.67E+06	1.53E+06	1.40E+04	2.65E+03	1.21E+0[illegible]
TOTAL	2.23E+08	2.00E+08	1.39E+08	4.95E+07	1.40E+07	7.72E+06	6.03E+06	1.56E+06	2.15E+05	4.38E+05	2.33E+04

D.2 BIBLIS Type 3.4 w/o U-235, Burnup 36 GWd/tU

[see following tables]

FUEL DECAY BIBLIS-TYPE 3.4% U235

ACTINIDES

BURNUP=3.60E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE 4	0.0	3.19E-01	5.52E-01	6.43E-01	8.13E-01	9.22E-01
U 233	0.0	2.62E-03	2.79E-03	2.96E-03	3.45E-03	3.78E-03
U 234	0.0	2.55E+00	3.83E+00	5.15E+00	9.08E+00	1.16E+01
U 235	3.40E+04	8.10E+03	8.10E+03	8.10E+03	8.10E+03	8.10E+03
U 236	0.0	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03
U 238	9.66E+05	9.40E+05	9.40E+05	9.40E+05	9.40E+05	9.40E+05
NP236M	0.0	2.43E-03	2.43E-03	2.43E-03	2.43E-03	2.43E-03
NP237	0.0	4.82E+02	4.92E+02	4.93E+02	4.94E+02	4.95E+02
PU236	0.0	1.91E-03	1.50E-03	1.18E-03	5.68E-04	3.49E-04
PU238	0.0	1.58E+02	1.69E+02	1.70E+02	1.67E+02	1.64E+02
PU239	0.0	5.54E+03	5.63E+03	5.63E+03	5.63E+03	5.63E+03
PU240	0.0	2.40E+03	2.40E+03	2.40E+03	2.41E+03	2.41E+03
PU241	0.0	1.29E+03	1.23E+03	1.17E+03	1.01E+03	9.20E+02
PU242	0.0	5.45E+02	5.45E+02	5.45E+02	5.45E+02	5.45E+02
PU244	0.0	3.49E-02	3.49E-02	3.49E-02	3.49E-02	3.49E-02
AM241	0.0	3.81E+01	9.86E+01	1.56E+02	3.12E+02	4.04E+02
AM242M	0.0	4.03E-01	4.01E-01	3.99E-01	3.94E-01	3.90E-01
AM243	0.0	1.01E+02	1.01E+02	1.01E+02	1.01E+02	1.01E+02
CM242	0.0	1.39E+01	2.97E+00	6.31E-01	6.96E-03	1.22E-03
CM243	0.0	3.87E-01	3.78E-01	3.69E-01	3.43E-01	3.26E-01
CM244	0.0	3.14E+01	3.03E+01	2.92E+01	2.60E+01	2.41E+01
CM245	0.0	1.08E+00	1.08E+00	1.08E+00	1.08E+00	1.08E+00
CM246	0.0	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00
CM247	0.0	6.00E-03	6.00E-03	6.00E-03	6.00E-03	6.00E-03
TOTAL	1.00E+06	9.63E+05	9.63E+05	9.63E+05	9.63E+05	9.63E+05

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE	0.0	3.19E-01	5.52E-01	6.43E-01	8.13E-01	9.22E-01
TL	0.0	1.73E-12	5.43E-12	1.08E-11	2.77E-11	3.63E-11
PB	0.0	3.65E-07	1.48E-06	4.13E-06	2.35E-05	4.48E-05
BI	0.0	1.81E-10	4.16E-10	7.35E-10	1.74E-09	2.27E-09
PO	0.0	4.30E-15	1.35E-14	2.74E-14	8.15E-14	1.31E-13
AT	0.0	2.04E-19	2.63E-20	2.80E-20	3.34E-20	3.75E-20
RN	0.0	1.54E-12	4.83E-12	9.62E-12	2.46E-11	3.23E-11
FR	0.0	1.88E-15	2.91E-16	3.25E-16	4.56E-16	5.64E-16
RA	0.0	8.98E-09	2.81E-08	5.60E-08	1.44E-07	1.91E-07
AC	0.0	1.36E-09	1.97E-09	2.73E-09	5.82E-09	8.51E-09
TH	0.0	2.47E-04	3.87E-04	5.33E-04	9.94E-04	1.32E-03
PA	0.0	4.48E-05	5.30E-05	6.09E-05	8.45E-05	1.00E-04
U	1.00E+06	9.52E+05	9.52E+05	9.52E+05	9.52E+05	9.52E+05
NP	0.0	5.75E+02	4.92E+02	4.93E+02	4.94E+02	4.95E+02
PU	0.0	9.93E+03	9.98E+03	9.92E+03	9.76E+03	9.67E+03
AM	0.0	1.39E+02	2.00E+02	2.57E+02	4.13E+02	5.05E+02
CM	0.0	4.79E+01	3.58E+01	3.23E+01	2.85E+01	2.66E+01
BK	0.0	2.36E-06	1.07E-06	4.86E-07	4.53E-08	9.32E-09
CF	0.0	1.49E-06	2.69E-06	3.19E-06	3.43E-06	3.36E-06
ES	0.0	4.35E-10	1.67E-14	8.10E-20	6.43E-36	1.17E-46
TOTALS	1.00E+06	9.63E+05	9.63E+05	9.63E+05	9.63E+05	9.63E+05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 237	0.0	7.45E+02	2.74E-03	2.61E-03	2.26E-03	2.05E-03
NP239	0.0	2.16E+04	2.04E-02	2.04E-02	2.04E-02	2.04E-02
PU238	0.0	2.56E-02	2.74E-02	2.75E-02	2.70E-02	2.66E-02
PU239	0.0	1.28E-03	1.30E-03	1.30E-03	1.30E-03	1.30E-03
PU240	0.0	4.96E-03	4.96E-03	4.97E-03	4.97E-03	4.97E-03
PU241	0.0	5.98E-03	5.70E-03	5.43E-03	4.70E-03	4.27E-03
AM241	0.0	1.89E-02	4.89E-02	7.74E-02	1.55E-01	2.00E-01
AM243	0.0	6.57E-03	6.58E-03	6.58E-03	6.58E-03	6.58E-03
CM242	0.0	4.51E-01	9.63E-02	2.04E-02	2.25E-04	3.93E-05
CM243	0.0	1.57E-02	1.54E-02	1.50E-02	1.39E-02	1.33E-02
CM244	0.0	2.30E-02	2.22E-02	2.14E-02	1.91E-02	1.77E-02
TOTAL	6.99E-05	3.31E+04	2.53E-01	2.05E-01	2.57E-01	2.99E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PA234M	0.0	1.59E-03	1.56E-03	1.56E-03	1.56E-03	1.56E-03
U 236	0.0	7.71E-03	7.71E-03	7.71E-03	7.72E-03	7.72E-03
U 237	0.0	1.63E+03	5.99E-03	5.71E-03	4.94E-03	4.49E-03
U 238	8.22E-03	8.00E-03	8.00E-03	8.00E-03	8.00E-03	8.00E-03
NP237	0.0	9.98E-03	1.02E-02	1.02E-02	1.02E-02	1.03E-02
NP239	0.0	5.15E+04	4.86E-02	4.86E-02	4.86E-02	4.86E-02
PU236	0.0	3.52E-02	2.78E-02	2.18E-02	1.05E-02	6.46E-03
PU238	0.0	8.97E+01	9.59E+01	9.65E+01	9.46E+01	9.31E+01
PU239	0.0	1.07E+01	1.09E+01	1.09E+01	1.09E+01	1.09E+01
PU240	0.0	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01
PU241	0.0	4.09E+00	3.90E+00	3.72E+00	3.22E+00	2.92E+00
PU242	0.0	6.15E-02	6.15E-02	6.15E-02	6.15E-02	6.15E-02
AM241	0.0	4.38E+00	1.13E+01	1.79E+01	3.58E+01	4.64E+01
AM242M	0.0	1.55E-03	1.54E-03	1.53E-03	1.51E-03	1.50E-03
AM242	0.0	9.87E+01	4.50E-03	4.48E-03	4.42E-03	4.38E-03
AM243	0.0	6.48E-01	6.49E-01	6.49E-01	6.48E-01	6.48E-01
CM242	0.0	1.70E+03	3.62E+02	7.69E+01	8.48E-01	1.48E-01
CM243	0.0	7.33E-01	7.16E-01	6.98E-01	6.49E-01	6.19E-01
CM244	0.0	8.88E+01	8.58E+01	8.25E+01	7.36E+01	6.82E+01
CM245	0.0	6.20E-03	6.20E-03	6.20E-03	6.20E-03	6.20E-03
CM246	0.0	1.10E-02	1.10E-02	1.10E-02	1.10E-02	1.10E-02
TOTAL	1.03E-02	1.16E+05	5.89E+02	3.07E+02	2.38E+02	2.40E+02

FUEL DECAY BIBLIS-TYPE 3.4% U235

ACTINIDES

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TL208	0.0	5.09E-04	1.60E-03	3.18E-03	8.16E-03	1.07E-02
PB212	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
BI212	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
PO212	0.0	9.09E-04	2.86E-03	5.69E-03	1.46E-02	1.91E-02
PO216	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
RN220	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
RA224	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
TH228	0.0	1.41E-03	4.45E-03	8.86E-03	2.27E-02	2.97E-02
TH231	0.0	2.67E-02	1.75E-02	1.75E-02	1.75E-02	1.75E-02
TH234	0.0	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01
PA233	0.0	3.39E-01	3.47E-01	3.47E-01	3.48E-01	3.49E-01
PA234M	0.0	3.22E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01
U 232	0.0	6.68E-03	1.53E-02	2.19E-02	3.39E-02	3.78E-02
U 234	0.0	1.59E-02	2.40E-02	3.22E-02	5.67E-02	7.28E-02
U 235	7.35E-02	1.75E-02	1.75E-02	1.75E-02	1.75E-02	1.75E-02
U 236	0.0	2.85E-01	2.85E-01	2.85E-01	2.85E-01	2.85E-01
U 237	0.0	8.61E+05	3.17E+00	3.02E+00	2.61E+00	2.37E+00
U 238	3.25E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01
NP237	0.0	3.40E-01	3.47E-01	3.47E-01	3.48E-01	3.49E-01
NP238	0.0	3.84E+05	1.95E-02	1.94E-02	1.91E-02	1.90E-02
NP239	0.0	2.13E+07	2.01E+01	2.01E+01	2.01E+01	2.01E+01
PU236	0.0	1.01E+00	7.99E-01	6.26E-01	3.02E-01	1.86E-01
PU238	0.0	2.70E+03	2.89E+03	2.91E+03	2.85E+03	2.81E+03
PU239	0.0	3.45E+02	3.50E+02	3.50E+02	3.50E+02	3.50E+02
PU240	0.0	5.47E+02	5.48E+02	5.48E+02	5.48E+02	5.49E+02
PU241	0.0	1.33E+05	1.27E+05	1.21E+05	1.04E+05	9.48E+04
PU242	0.0	2.08E+00	2.08E+00	2.08E+00	2.08E+00	2.08E+00
AM241	0.0	1.31E+02	3.38E+02	5.36E+02	1.07E+03	1.39E+03
AM242M	0.0	3.91E+00	3.90E+00	3.88E+00	3.83E+00	3.79E+00
AM242	0.0	8.49E+04	3.88E+00	3.86E+00	3.81E+00	3.77E+00
AM243	0.0	2.01E+01	2.01E+01	2.01E+01	2.01E+01	2.01E+01
CM242	0.0	4.61E+04	9.83E+03	2.09E+03	2.30E+01	4.02E+00
CM243	0.0	2.00E+01	1.95E+01	1.90E+01	1.77E+01	1.69E+01
CM244	0.0	2.54E+03	2.45E+03	2.36E+03	2.10E+03	1.95E+03
CM245	0.0	1.86E-01	1.86E-01	1.86E-01	1.86E-01	1.86E-01
CM246	0.0	3.36E-01	3.36E-01	3.36E-01	3.36E-01	3.36E-01
BK249	0.0	3.87E-03	1.76E-03	7.97E-04	[illegible]	1.53E-05
TOTAL	3.98E-01	4.47E+07	1.43E+05	1.29E+05	[illegible]	1.02E+05

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
BI212	0.0	5.09E-04	1.60E-03	3.18E-03	8.16E-03	1.07E-02
PO212	0.0	9.09E-04	2.86E-03	5.69E-03	1.46E-02	1.91E-02
PO216	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
RN220	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
RA224	0.0	1.42E-03	4.46E-03	8.87E-03	2.27E-02	2.98E-02
TH228	0.0	1.41E-03	4.45E-03	8.86E-03	2.27E-02	2.97E-02
U 232	0.0	6.68E-03	1.53E-02	2.19E-02	3.39E-02	3.78E-02
U 234	0.0	1.59E-02	2.40E-02	3.22E-02	5.67E-02	7.28E-02
U 235	7.35E-02	1.75E-02	1.75E-02	1.75E-02	1.75E-02	1.75E-02
U 236	0.0	2.85E-01	2.85E-01	2.85E-01	2.85E-01	2.85E-01
U 238	3.25E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01
NP237	0.0	3.40E-01	3.47E-01	3.47E-01	3.48E-01	3.49E-01
PU236	0.0	1.01E+00	7.99E-01	6.26E-01	3.02E-01	1.86E-01
PU238	0.0	2.70E+03	2.89E+03	2.91E+03	2.85E+03	2.81E+03
PU239	0.0	3.45E+02	3.50E+02	3.50E+02	3.50E+02	3.50E+02
PU240	0.0	5.47E+02	5.48E+02	5.48E+02	5.48E+02	5.49E+02
PU241	0.0	3.32E+00	3.16E+00	3.01E+00	2.61E+00	2.37E+00
PU242	0.0	2.08E+00	2.08E+00	2.08E+00	2.08E+00	2.08E+00
AM241	0.0	1.31E+02	3.38E+02	5.36E+02	1.07E+03	1.39E+03
AM242M	0.0	1.96E-02	1.95E-02	1.94E-02	1.91E-02	1.90E-02
AM243	0.0	2.01E+01	2.01E+01	2.01E+01	2.01E+01	2.01E+01
CM242	0.0	4.61E+04	9.83E+03	2.09E+03	2.30E+01	4.02E+00
CM243	0.0	1.99E+01	1.95E+01	1.90E+01	1.77E+01	1.68E+01
CM244	0.0	2.54E+03	2.45E+03	2.36E+03	2.10E+03	1.95E+03
CM245	0.0	1.86E-01	1.86E-01	1.86E-01	1.86E-01	1.86E-01
CM246	0.0	3.36E-01	3.36E-01	3.36E-01	3.36E-01	3.36E-01
TOTAL	3.98E-01	5.24E+04	1.65E+04	8.84E+03	6.99E+03	7.09E+03

FUEL DECAY BIBLIS-TYPE 3.4% U235

FISSION PRODUCTS

BURNUP=3.60E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	1.84E-02	1.74E-02	1.64E-02	1.39E-02	1.24E-02
KR 85	0.0	1.53E+01	1.44E+01	1.35E+01	1.11E+01	9.75E+00
SR 89	0.0	2.91E+03	1.94E+01	1.29E-01	3.80E-08	1.68E-12
SR 90	0.0	9.43E+01	9.21E+01	8.99E+01	8.37E+01	7.98E+01
Y 90	0.0	4.76E+02	4.40E+02	4.29E+02	4.00E+02	3.81E+02
Y 91	0.0	3.97E+03	5.29E+01	6.99E-01	1.61E-06	2.82E-10
ZR 95	0.0	8.08E+03	1.55E+02	2.96E+00	2.07E-05	7.60E-09
NB 95	0.0	7.64E+03	3.29E+02	6.44E+00	4.36E-05	1.60E-08
NB 95M	0.0	1.60E+01	3.15E-01	6.02E-03	4.22E-08	1.55E-11
TC 99	0.0	7.09E-03	7.13E-03	7.13E-03	7.13E-03	7.13E-03
RU103	0.0	5.54E+03	8.92E+00	1.41E-02	5.70E-11	1.44E-16
RU106	0.0	3.26E+01	1.64E+01	8.25E+00	1.05E+00	2.65E-01
RH103M	0.0	3.42E+02	5.51E-01	8.77E-04	3.52E-12	8.90E-18
RH106	0.0	5.81E+03	2.65E+03	1.33E+03	1.70E+02	4.29E+01
AG110	0.0	1.11E+03	1.20E-01	4.36E-02	2.09E-03	2.75E-04
AG110M	0.0	5.77E+01	2.10E+01	7.61E+00	3.65E-01	4.81E-02
CD113M	0.0	6.80E-02	6.49E-02	6.19E-02	5.37E-02	4.88E-02
CD115M	0.0	3.04E+00	1.05E-02	3.60E-05	1.45E-12	1.70E-17
SN119M	0.0	6.38E-02	2.27E-02	8.09E-03	3.65E-04	4.62E-05
SN123	0.0	6.70E+00	9.44E-01	1.33E-01	3.71E-04	7.37E-06
SB124	0.0	1.04E+01	1.56E-01	2.33E-03	7.74E-09	1.72E-12
SB125	0.0	3.06E+01	2.41E+01	1.87E+01	8.81E+00	5.33E+00
SB126	0.0	1.55E+01	1.54E-03	1.54E-03	1.54E-03	1.54E-03
SB126M	0.0	8.91E+00	7.56E-03	7.56E-03	7.56E-03	7.56E-03
TE123M	0.0	1.32E-02	1.59E-03	1.91E-04	3.37E-07	4.90E-09
TE125M	0.0	1.73E+00	1.57E+00	1.23E+00	5.78E-01	3.50E-01
TE127	0.0	1.32E+02	1.71E+00	1.68E-01	1.59E-04	1.53E-06
TE127M	0.0	6.83E+00	6.97E-01	6.84E-02	6.45E-05	6.21E-07
TE129	0.0	1.17E+03	6.24E-02	3.34E-05	5.11E-15	1.46E-21
TE129M	0.0	8.75E+01	4.70E-02	2.52E-05	3.85E-15	1.10E-21
CS134	0.0	1.87E+03	1.33E+03	9.53E+02	3.48E+02	1.77E+02
CS137	0.0	1.29E+02	1.26E+02	1.23E+02	1.15E+02	1.09E+02
BA137M	0.0	4.33E+02	4.22E+02	4.12E+02	3.85E+02	3.67E+02
CE141	0.0	2.35E+03	9.80E-01	4.08E-04	2.93E-14	5.07E-21
CE144	0.0	7.92E+02	3.25E+02	1.33E+02	9.23E+00	1.56E+00
PR144	0.0	8.86E+03	3.60E+03	1.48E+03	1.02E+02	1.72E+01
PR144M	0.0	4.91E+00	2.01E+00	8.26E-01	5.71E-02	9.62E-03
PM147	0.0	5.68E+01	4.57E+01	3.51E+01	1.59E+01	9.36E+00
PM148	0.0	1.49E+03	3.04E-02	6.63E-05	6.86E-13	3.26E-18
PM148M	0.0	4.08E+02	8.89E-01	1.94E-03	2.01E-11	9.52E-17
SM151	0.0	4.07E-02	4.12E-02	4.09E-02	3.99E-02	3.93E-02
EU152	0.0	3.46E-02	3.29E-02	3.13E-02	2.69E-02	2.43E-02
EU154	0.0	1.25E+02	1.15E+02	1.06E+02	8.33E+01	7.09E+01
EU155	0.0	6.09E+00	5.30E+00	4.61E+00	3.03E+00	2.29E+00
GD153	0.0	1.80E-02	6.31E-03	2.22E-03	9.62E-05	1.19E-05
TB160	0.0	7.45E+00	2.25E-01	6.79E-03	1.87E-07	1.70E-10
TOTAL	0.0	1.96E+06	9.81E+03	5.16E+03	1.74E+03	1.28E+03

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
KR 85	0.0	6.39E-01	5.99E-01	5.62E-01	4.63E-01	4.07E-01
SR 89	0.0	1.71E+02	1.14E+00	7.56E-03	2.23E-09	9.88E-14
SR 90	0.0	5.29E+00	5.16E+00	5.04E+00	4.70E+00	4.48E+00
Y 90	0.0	7.63E+01	7.04E+01	6.88E+01	6.40E+01	6.10E+01
Y 91	0.0	2.60E+02	3.46E+00	4.57E-02	1.05E-07	1.84E-11
ZR 95	0.0	7.00E+03	1.34E+02	2.57E+00	1.80E-05	6.59E-09
NB 95	0.0	7.23E+03	3.12E+02	6.10E+00	4.13E-05	1.51E-08
NB 95M	0.0	4.68E+00	9.22E-02	1.76E-03	1.24E-08	4.53E-12
RU103	0.0	4.81E+03	7.73E+00	1.23E-02	4.94E-11	1.25E-16
RU106	0.0	1.35E-01	6.79E-02	3.42E-02	4.34E-03	1.10E-03
RH103M	0.0	1.51E+01	2.43E-02	3.86E-05	1.55E-13	3.92E-19
RH106	0.0	1.27E+03	5.82E+02	2.92E+02	3.72E+01	9.40E+00
AG110	0.0	1.28E+02	1.38E-02	5.01E-03	2.40E-04	3.17E-05
AG110M	0.0	5.63E+01	2.04E+01	7.42E+00	3.56E-01	4.69E-02
CD113M	0.0	1.22E-03	1.17E-03	1.11E-03	9.64E-04	8.76E-04
SN119M	0.0	8.34E-03	2.97E-03	1.06E-03	4.77E-05	6.04E-06
SN123	0.0	4.51E-01	6.36E-02	8.96E-03	2.50E-05	4.96E-07
SB124	0.0	8.77E+00	1.31E-01	1.95E-03	6.50E-09	1.45E-12
SB125	0.0	2.51E+01	1.97E+01	1.53E+01	7.23E+00	4.37E+00
SB126	0.0	1.41E+01	1.40E-03	1.40E-03	1.40E-03	1.40E-03
SB126M	0.0	6.56E+00	5.57E-03	5.57E-03	5.57E-03	5.57E-03
TE125M	0.0	4.33E-01	3.95E-01	3.08E-01	1.45E-01	8.78E-02
TE127	0.0	6.82E+00	8.89E-02	8.72E-03	8.23E-06	7.92E-08
TE127M	0.0	7.96E-01	8.12E-02	7.96E-03	7.51E-06	7.23E-08
TE129	0.0	1.71E+02	9.06E-03	4.85E-06	7.42E-16	2.12E-22
TE129M	0.0	1.23E+01	6.63E-03	3.55E-06	5.43E-16	1.55E-22
CS134	0.0	1.69E+03	1.21E+03	8.64E+02	3.15E+02	1.61E+02
CS137	0.0	3.24E+00	3.17E+00	3.10E+00	2.89E+00	2.76E+00
BA137M	0.0	3.89E+02	3.79E+02	3.70E+02	3.45E+02	3.30E+02
CE141	0.0	7.57E+02	3.16E-01	1.32E-04	9.46E-15	1.63E-21
CE144	0.0	1.54E+02	6.32E+01	2.59E+01	1.79E+00	3.02E-01
PR144	0.0	1.04E+03	4.24E+02	1.74E+02	1.20E+01	2.03E+00
PR144M	0.0	1.08E+00	4.43E-01	1.82E-01	1.26E-02	2.12E-03
PM147	0.0	7.41E-01	5.95E-01	4.57E-01	2.07E-01	1.22E-01
PM148	0.0	7.79E+02	1.59E-02	3.47E-05	3.59E-13	1.70E-18
PM148M	0.0	3.79E+02	8.26E-01	1.80E-03	1.86E-11	8.84E-17
EU152	0.0	3.11E-02	2.95E-02	2.80E-02	2.41E-02	2.17E-02
EU154	0.0	1.02E+02	9.42E+01	8.69E+01	6.83E+01	5.81E+01
EU155	0.0	3.03E+00	2.64E+00	2.29E+00	1.51E+00	1.14E+00
GD153	0.0	1.30E-02	4.59E-03	1.61E-03	6.99E-05	8.63E-06
TB160	0.0	6.14E+00	1.85E-01	5.59E-03	1.54E-07	1.40E-10
TOTAL	0.0	6.15E+05	3.34E+03	1.93E+03	8.62E+02	6.35E+02

FUEL DECAY BIBLIS-TYPE 3.4% U235

FISSION PRODUCTS

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	5.44E+02	5.14E+02	4.86E+02	4.11E+02	3.67E+02
SE 79	0.0	4.02E-01	4.02E-01	4.02E-01	4.02E-01	4.02E-01
KR 85	0.0	1.02E+04	9.59E+03	8.99E+03	7.41E+03	6.51E+03
RB 86	0.0	2.17E+03	2.78E-03	3.57E-09	7.54E-27	1.24E-38
SR 89	0.0	8.43E+05	5.61E+03	3.73E+01	1.10E-05	4.87E-10
SR 90	0.0	8.12E+04	7.93E+04	7.74E+04	7.21E+04	6.88E+04
Y 90	0.0	8.59E+04	7.93E+04	7.75E+04	7.21E+04	6.88E+04
Y 91	0.0	1.11E+06	1.47E+04	1.95E+02	4.50E-04	7.85E-08
ZR 93	0.0	1.95E+00	1.95E+00	1.95E+00	1.95E+00	1.95E+00
ZR 95	0.0	1.59E+06	3.05E+04	5.84E+02	4.10E-03	1.50E-06
NB 93M	0.0	1.42E-01	2.27E-01	3.08E-01	5.27E-01	6.55E-01
NB 95	0.0	1.59E+06	6.87E+04	1.34E+03	9.09E-03	3.33E-06
NB 95M	0.0	1.15E+04	2.26E+02	4.33E+00	3.04E-05	1.11E-08
TC 99	0.0	1.41E+01	1.42E+01	1.42E+01	1.42E+01	1.42E+01
RU103	0.0	1.65E+06	2.66E+03	4.21E+00	1.70E-08	4.29E-14
RU106	0.0	5.50E+05	2.77E+05	1.39E+05	1.77E+04	4.47E+03
RH103M	0.0	1.49E+06	2.40E+03	3.81E+00	1.53E-08	3.87E-14
RH106	0.0	6.06E+05	2.77E+05	1.39E+05	1.77E+04	4.47E+03
PD107	0.0	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01
AG110	0.0	1.55E+05	1.67E+01	6.06E+00	2.90E-01	3.83E-02
AG110M	0.0	3.46E+03	1.26E+03	4.56E+02	2.18E+01	2.88E+00
CD113M	0.0	4.04E+01	3.86E+01	3.68E+01	3.19E+01	2.90E+01
CD115M	0.0	8.15E+02	2.81E+00	9.65E-03	3.89E-10	4.57E-15
IN114	0.0	3.16E+00	1.30E-02	7.86E-05	1.72E-11	6.23E-16
IN114M	0.0	2.26E+00	1.36E-02	8.21E-05	1.79E-11	6.51E-16
SN119M	0.0	1.23E+02	4.39E+01	1.56E+01	7.05E-01	8.94E-02
SN121M	0.0	1.72E-01	1.70E-01	1.67E-01	1.60E-01	1.56E-01
SN123	0.0	2.15E+03	3.02E+02	4.26E+01	1.19E-01	2.36E-03
SN126	0.0	5.94E-01	5.94E-01	5.94E-01	5.94E-01	5.94E-01
SB124	0.0	7.87E+02	1.17E+01	1.75E-01	5.83E-07	1.30E-10
SB125	0.0	9.79E+03	7.69E+03	5.99E+03	2.82E+03	1.71E+03
SB126	0.0	8.38E+02	8.32E-02	8.32E-02	8.32E-02	8.32E-02
SB126M	0.0	7.00E+02	5.94E-01	5.94E-01	5.94E-01	5.94E-01
TE123M	0.0	9.03E+00	1.09E+00	1.31E-01	2.31E-04	3.37E-06
TE125M	0.0	2.05E+03	1.87E+03	1.46E+03	6.88E+02	4.16E+02
TE127	0.0	9.74E+04	1.27E+03	1.25E+02	1.17E-01	1.13E-03
TE127M	0.0	1.27E+04	1.30E+03	1.27E+02	1.20E-01	1.15E-03
TE129	0.0	3.29E+05	1.75E+01	9.34E-03	1.43E-12	4.10E-19
TE129M	0.0	4.99E+04	2.68E+01	1.44E-02	2.20E-12	6.29E-19
I 129	0.0	3.37E-02	3.40E-02	3.40E-02	3.40E-02	3.40E-02
CS134	0.0	1.83E+05	1.31E+05	9.36E+04	3.41E+04	1.74E+04
CS135	0.0	3.63E-01	3.64E-01	3.64E-01	3.64E-01	3.64E-01
CS137	0.0	1.16E+05	1.14E+05	1.11E+05	1.04E+05	9.89E+04
BA137M	0.0	1.10E+05	1.07E+05	1.05E+05	9.80E+04	9.35E+04
BA140	0.0	1.68E+06	4.28E-03	1.09E-11	0.0	0.0
LA140	0.0	1.75E+06	4.93E-03	1.25E-11	0.0	0.0
CE141	0.0	1.60E+06	6.70E+02	2.78E-01	2.00E-11	3.46E-18
CE144	0.0	1.19E+06	4.90E+05	2.01E+05	1.39E+04	2.34E+03
PR143	0.0	1.43E+06	1.25E-02	9.88E-11	0.0	0.0
PR144	0.0	1.21E+06	4.90E+05	2.01E+05	1.39E+04	2.34E+03
PR144M	0.0	1.43E+04	5.88E+03	2.41E+03	1.67E+02	2.81E+01
PM147	0.0	1.58E+05	1.27E+05	9.77E+04	4.43E+04	2.61E+04

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PM148	0.0	1.93E+05	3.95E+00	8.62E-03	8.91E-11	4.23E-16
PM148M	0.0	3.22E+04	7.01E+01	1.53E-01	1.58E-09	7.51E-15
SM151	0.0	3.46E+02	3.51E+02	3.48E+02	3.40E+02	3.35E+02
EU152	0.0	4.57E+00	4.34E+00	4.13E+00	3.54E+00	3.20E+00
EU154	0.0	1.39E+04	1.29E+04	1.19E+04	9.31E+03	7.93E+03
EU155	0.0	8.38E+03	7.29E+03	6.33E+03	4.17E+03	3.15E+03
EU156	0.0	2.78E+05	1.61E-02	9.28E-10	0.0	0.0
GD153	0.0	2.08E+01	7.32E+00	2.57E+00	1.12E-01	1.38E-02
TB160	0.0	9.15E+02	2.76E+01	8.33E-01	2.29E-05	2.09E-08
TOTAL	0.0	1.66E+08	2.35E+06	1.28E+06	5.13E+05	4.08E+05

******** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS ********

CONCENTRATIONS ,GRAMS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06

RADIOACTIVITY ,CURIES

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	3.98E-01	2.12E+08	2.60E+06	1.48E+06	6.56E+05	5.32E+05

THERMAL POWER ,WATTS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	1.03E-02	2.09E+06	1.10E+04	5.82E+03	2.18E+03	1.67E+03

GAMMA POWER ,WATTS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	6.99E-05	6.55E+05	3.84E+03	2.21E+03	1.03E+03	7.67E+02

ALPHA-ACTIVITY ,CURIES

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	3.98E-01	5.24E+04	1.65E+04	8.84E+03	6.99E+03	7.09E+03

FUEL DECAY BIBLIS-TYPE 3.4% U235

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	1.81E-03	1.71E-03	1.62E-03	1.37E-03	1.22E-03
C 14	0.0	1.57E-01	1.57E-01	1.57E-01	1.57E-01	1.57E-01
P 33	0.0	5.59E+02	2.24E-02	8.98E-07	5.79E-20	9.30E-29
S 35	0.0	7.61E+00	4.29E-01	2.42E-02	4.32E-06	1.37E-08
CA 45	0.0	2.48E-01	5.25E-02	1.11E-02	1.05E-04	4.71E-06
SC 46	0.0	3.37E+01	1.64E+00	8.02E-02	9.30E-06	2.21E-08
CR 51	0.0	2.13E+05	2.30E+01	2.48E-03	3.10E-15	3.61E-23
MN 54	0.0	7.81E+03	3.48E+03	1.55E+03	1.36E+02	2.69E+01
FE 55	0.0	6.78E+04	5.19E+04	3.98E+04	1.79E+04	1.05E+04
FE 59	0.0	5.52E+03	1.99E+01	7.20E-02	3.38E-09	4.40E-14
CO 58	0.0	3.32E+04	9.29E+02	2.60E+01	5.70E-04	4.46E-07
CO 60	0.0	2.23E+04	1.95E+04	1.71E+04	1.15E+04	8.87E+03
NI 59	0.0	1.90E+01	1.90E+01	1.90E+01	1.90E+01	1.90E+01
NI 63	0.0	2.70E+03	2.68E+03	2.66E+03	2.60E+03	2.56E+03
ZN 65	0.0	2.78E-01	9.84E-02	3.48E-02	1.55E-03	1.94E-04
SR 89	0.0	7.73E+01	5.14E-01	3.43E-03	1.01E-09	4.47E-14
SR 90	0.0	2.67E-03	2.61E-03	2.55E-03	2.37E-03	2.26E-03
Y 90	0.0	2.85E+03	2.61E-03	2.55E-03	2.37E-03	2.26E-03
Y 91	0.0	2.04E+02	2.69E+00	3.56E-02	8.22E-08	1.44E-11
ZR 93	0.0	2.00E-01	2.00E-01	2.00E-01	2.00E-01	2.00E-01
ZR 95	0.0	6.87E+04	1.31E+03	2.52E+01	1.76E-04	6.46E-08
NB 93M	0.0	1.27E-02	2.15E-02	2.99E-02	5.26E-02	6.59E-02
NB 94	0.0	1.13E+00	1.13E+00	1.13E+00	1.13E+00	1.13E+00
NB 95	0.0	6.73E+04	2.95E+03	5.77E+01	3.89E-04	1.42E-07
MO 93	0.0	7.78E-02	7.78E-02	7.78E-02	7.77E-02	7.77E-02
TC 99	0.0	1.28E-02	1.29E-02	1.29E-02	1.29E-02	1.29E-02
SN119M	0.0	1.07E+04	3.82E+03	1.36E+03	6.13E+01	7.76E+00
SN121M	0.0	1.01E+00	9.93E-01	9.79E-01	9.40E-01	9.14E-01
SN123	0.0	9.59E-01	1.35E-01	1.90E-02	5.34E-05	1.06E-06
SB124	0.0	1.72E+01	2.57E-01	3.84E-03	1.28E-08	2.84E-12
SB125	0.0	8.48E+01	6.61E+01	5.14E+01	2.42E+01	1.46E+01
TE125M	0.0	1.81E+01	1.61E+01	1.25E+01	5.91E+00	3.58E+00
TA182	0.0	1.98E+05	2.19E+04	2.42E+03	3.22E+00	3.94E-02
W 185	0.0	8.55E-01	2.94E-02	1.01E-03	4.10E-08	4.84E-11
TOTAL	6.06E-12	1.17E+06	1.09E+05	6.51E+04	3.23E+04	2.20E+04

LIGHT ELEMENTS

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	4.24E-01	2.07E-02	1.01E-03	1.17E-07	2.78E-10
CR 51	0.0	4.56E+01	4.91E-03	5.30E-07	6.64E-19	7.73E-27
MN 54	0.0	3.89E+01	1.73E+01	7.70E+00	6.78E-01	1.34E-01
FE 55	0.0	9.32E+01	7.14E+01	5.47E+01	2.46E+01	1.44E+01
FE 59	0.0	5.14E+01	1.85E-01	6.71E-04	3.15E-11	4.10E-16
CO 58	0.0	1.99E+02	5.56E+00	1.56E-01	3.41E-06	2.67E-09
CO 60	0.0	3.43E+02	3.01E+02	2.64E+02	1.78E+02	1.37E+02
NI 59	0.0	1.21E-01	1.21E-01	1.21E-01	1.21E-01	1.21E-01
NI 63	0.0	1.07E+00	1.06E+00	1.06E+00	1.03E+00	1.02E+00
SR 89	0.0	2.67E-01	1.78E-03	1.18E-05	3.49E-12	1.55E-16
Y 91	0.0	7.31E-01	9.67E-03	1.28E-04	2.95E-10	5.16E-14
ZR 95	0.0	3.48E+02	6.66E+00	1.27E-01	8.94E-07	3.27E-10
NB 94	0.0	1.16E-02	1.16E-02	1.16E-02	1.16E-02	1.16E-02
NB 95	0.0	3.23E+02	1.41E+01	2.77E-01	1.87E-06	6.83E-10
SN119M	0.0	5.55E+00	1.97E+00	7.03E-01	3.17E-02	4.01E-03
SN121M	0.0	2.02E-03	1.99E-03	1.96E-03	1.88E-03	1.83E-03
SB124	0.0	2.29E-01	3.41E-03	5.09E-05	1.69E-10	3.77E-14
SB125	0.0	2.65E-01	2.07E-01	1.61E-01	7.57E-02	4.58E-02
TE125M	0.0	1.52E-02	1.35E-02	1.05E-02	4.96E-03	3.01E-03
TA182	0.0	1.76E+03	1.95E+02	2.16E+01	2.86E-02	3.51E-04
TOTAL	1.08E-14	9.16E+03	6.15E+02	3.50E+02	2.04E+02	1.52E+02

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	4.03E-01	1.96E-02	9.58E-04	1.11E-07	2.64E-10
CR 51	0.0	4.10E+01	4.42E-03	4.76E-07	5.97E-19	6.95E-27
MN 54	0.0	3.87E+01	1.72E+01	7.66E+00	6.75E-01	1.34E-01
CO 58	0.0	1.93E+02	5.40E+00	1.51E-01	3.31E-06	2.59E-09
CO 60	0.0	3.30E+02	2.89E+02	2.54E+02	1.71E+02	1.31E+02
NI 63	0.0	1.07E-03	1.06E-03	1.05E-03	1.03E-03	1.01E-03
ZR 95	0.0	3.02E+02	5.78E+00	1.11E-01	7.75E-07	2.84E-10
NB 94	0.0	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.06E-02
NB 95	0.0	3.06E+02	1.34E+01	2.62E-01	1.77E-06	6.47E-10
SN119M	0.0	7.25E-01	2.58E-01	9.18E-02	4.14E-03	5.25E-04
SB124	0.0	1.92E-01	2.86E-03	4.28E-05	1.42E-10	3.17E-14
SB125	0.0	2.17E-01	1.69E-01	1.32E-01	6.20E-02	3.75E-02
TE125M	0.0	3.81E-03	3.39E-03	2.64E-03	1.24E-03	7.53E-04
TA182	0.0	1.54E+03	1.71E+02	1.89E+01	2.51E-02	3.07E-04
TOTAL	0.0	6.84E+03	5.02E+02	2.81E+02	1.72E+02	1.32E+02

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

FUEL DECAY BIBLIS-TYPE 3.4% U235
POWER= 36.04 MW, BURNUP= 3.60E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	2.20E+18	1.84E+16	9.51E+15	2.99E+15	2.32E+15
3.75E-01	7.92E+17	1.85E+15	9.67E+14	2.84E+14	2.08E+14
5.75E-01	1.12E+18	1.50E+16	1.07E+16	5.75E+15	4.59E+15
8.50E-01	1.09E+18	8.27E+15	3.57E+15	1.33E+15	7.40E+14
1.25E+00	5.60E+17	9.75E+14	6.43E+14	2.86E+14	2.06E+14
1.75E+00	2.16E+17	8.08E+13	4.16E+13	9.89E+12	6.21E+12
2.25E+00	1.18E+17	1.53E+14	6.42E+13	4.88E+12	9.03E+11
2.75E+00	4.47E+16	2.32E+12	1.16E+12	1.44E+11	3.62E+10
3.50E+00	2.45E+16	2.88E+11	1.45E+11	1.84E+10	4.65E+09
5.00E+00	1.37E+16	6.22E-05	6.36E-05	6.61E-05	6.70E-05
7.00E+00	3.35E+15	4.07E-06	4.16E-06	4.32E-06	4.38E-06
9.50E+00	1.74E+10	2.63E-07	2.69E-07	2.80E-07	2.83E-07
TOTAL	6.17E+18	4.48E+16	2.55E+16	1.07E+16	8.07E+15
MEV/SEC	3.84E+18	2.08E+16	1.20E+16	5.38E+15	3.96E+15

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

FUEL DECAY BIBLIS-TYPE 3.4% U235
POWER= 36.04 MW, BURNUP= 3.60E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	1.23E+16	8.04E+14	9.84E+13	5.35E+12	3.86E+12
3.75E-01	1.14E+15	1.27E+12	7.61E+11	3.58E+11	2.23E+11
5.75E-01	6.04E+14	1.03E+13	1.14E+12	4.17E+11	2.53E+11
8.50E-01	1.96E+16	3.09E+14	6.10E+13	5.06E+12	1.08E+12
1.25E+00	8.74E+15	2.20E+15	1.35E+15	8.53E+14	6.56E+14
1.75E+00	4.24E+15	1.75E+11	4.83E+09	1.28E+05	2.28E+04
2.25E+00	1.85E+15	8.18E+09	6.72E+09	4.52E+09	3.48E+09
2.75E+00	2.61E+14	2.41E+07	2.08E+07	1.40E+07	1.08E+07
3.50E+00	2.23E+13	0.0	0.0	0.0	0.0
5.00E+00	4.06E+10	0.0	0.0	0.0	0.0
7.00E+00	7.23E+12	0.0	0.0	0.0	0.0
9.50E+00	7.43E+09	0.0	0.0	0.0	0.0
TOTAL	4.88E+16	3.32E+15	1.51E+15	8.65E+14	6.61E+14
MEV/SEC	4.27E+16	3.13E+15	1.75E+15	1.07E+15	8.22E+14

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.00E-02	1.14E+18	1.02E+14	5.59E+13	4.56E+13	4.66E+13
2.50E-02	5.05E+16	3.36E+11	5.20E+11	1.02E+12	1.32E+12
3.75E-02	7.33E+16	3.00E+11	2.05E+11	2.13E+11	2.34E+11
5.75E-02	6.56E+16	4.74E+12	7.45E+12	1.48E+13	1.92E+13
8.50E-02	5.08E+17	7.29E+11	7.27E+11	7.21E+11	7.16E+11
1.25E-01	4.00E+17	6.40E+11	6.28E+11	6.11E+11	6.02E+11
2.25E-01	2.83E+17	5.16E+11	5.06E+11	4.86E+11	4.74E+11
3.75E-01	3.13E+16	3.42E+10	3.40E+10	3.38E+10	3.37E+10
5.75E-01	6.26E+15	3.31E+09	2.40E+09	2.33E+09	2.42E+09
8.50E-01	1.34E+16	2.41E+09	2.08E+09	1.98E+09	1.95E+09
1.25E+00	3.23E+15	1.13E+09	9.78E+08	8.76E+08	8.32E+08
1.75E+00	6.39E+10	4.44E+08	3.82E+08	3.49E+08	3.36E+08
2.25E+00	4.02E+08	2.46E+08	2.07E+08	1.78E+08	1.65E+08
2.75E+00	2.52E+08	1.99E+08	2.32E+08	3.90E+08	4.71E+08
3.50E+00	1.39E+08	8.52E+07	7.17E+07	6.16E+07	5.72E+07
5.00E+00	4.35E+07	2.66E+07	2.24E+07	1.92E+07	1.79E+07
7.00E+00	4.97E+06	3.05E+06	2.57E+06	2.20E+06	2.05E+06
9.50E+00	4.98E+05	3.05E+05	2.56E+05	2.20E+05	2.05E+05
TOTAL	2.57E+18	1.09E+14	6.60E+13	6.35E+13	6.92E+13
MEV/SEC	2.07E+17	1.59E+12	1.28E+12	1.61E+12	1.87E+12

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

FUEL DECAY BIBLIS-TYPE 3.4% U235
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PO212	1.57E+03	4.95E+03	9.85E+03	2.52E+04	3.31E+04
PU236	4.71E+03	3.71E+03	2.91E+03	1.40E+03	8.64E+02
PU238	3.16E+06	3.38E+06	3.40E+06	3.33E+06	3.28E+06
PU239	2.49E+05	2.54E+05	2.54E+05	2.54E+05	2.53E+05
PU240	4.08E+05	4.08E+05	4.09E+05	4.09E+05	4.09E+05
PU241	2.41E+03	2.29E+03	2.19E+03	1.89E+03	1.72E+03
PU242	1.47E+03	1.47E+03	1.47E+03	1.47E+03	1.47E+03
AM241	1.53E+05	3.94E+05	6.24E+05	1.25E+06	1.62E+06
AM243	3.23E+04	3.23E+04	3.23E+04	3.23E+04	3.23E+04
CM242	3.72E+08	7.94E+07	1.69E+07	1.86E+05	3.24E+04
CM243	1.86E+05	1.82E+05	1.77E+05	1.65E+05	1.57E+05
CM244	1.79E+07	1.73E+07	1.67E+07	1.49E+07	1.38E+07
TOTAL	3.94E+08	1.01E+08	3.84E+07	2.05E+07	1.96E+07

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

FUEL DECAY BIBLIS-TYPE 3.4% U235
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 238	1.19E+04	1.19E+04	1.19E+04	1.19E+04	1.19E+04
PU238	4.20E+05	4.49E+05	4.52E+05	4.43E+05	4.36E+05
PU240	2.19E+06	2.19E+06	2.19E+06	2.19E+06	2.19E+06
PU242	9.19E+05	9.19E+05	9.19E+05	9.19E+05	9.19E+05
CM242	3.00E+08	6.41E+07	1.36E+07	1.50E+05	2.62E+04
CM244	3.49E+08	3.37E+08	3.24E+08	2.89E+08	2.68E+08
CM246	9.74E+06	9.74E+06	9.74E+06	9.74E+06	9.73E+06
CM248	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04
CF250	1.10E+04	1.05E+04	9.92E+03	8.46E+03	7.61E+03
CF252	4.06E+05	3.12E+05	2.40E+05	1.09E+05	6.45E+04
CF254	8.28E+04	1.26E+03	1.92E+01	6.81E-05	1.58E-08
TOTAL	6.63E+08	4.15E+08	3.51E+08	3.03E+08	2.81E+08
TOTAL	1.06E+09	5.16E+08	3.90E+08	3.23E+08	3.01E+08

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.60E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE 4	9.22E-01	9.99E-01	1.22E+00	1.72E+00	2.30E+00	3.22E+00	7.08E+00	1.02E+01	1.45E+01	3.85E+01	1.19E+02
TH232	1.12E-03	1.12E-03	1.14E-03	1.18E-03	1.24E-03	1.37E-03	2.46E-03	1.91E-02	2.57E-01	2.60E+00	2.29E+01
U 234	1.16E-01	1.54E-01	2.75E-01	5.90E-01	9.89E-01	1.48E+00	2.03E+00	2.00E+00	1.66E+00	5.96E-01	5.05E-01
U 235	8.10E+01	8.10E+01	8.11E+01	8.11E+01	8.12E+01	8.14E+01	8.27E+01	1.04E+02	2.40E+02	2.51E+02	2.49E+02
U 236	4.40E+01	4.40E+01	4.41E+01	4.42E+01	4.44E+01	4.49E+01	4.86E+01	7.46E+01	9.07E+01	8.83E+01	6.77E+01
U 238	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.39E+03
NP236M	2.43E-03	2.43E-03	2.43E-03	2.43E-03	2.43E-03	2.43E-03	2.43E-03	2.42E-03	2.30E-03	1.42E-03	1.13E-05
NP237	4.95E+02	4.97E+02	5.03E+02	5.22E+02	5.51E+02	6.03E+02	8.18E+02	8.99E+02	8.74E+02	6.53E+02	3.54E+01
PU238	1.64E+00	1.61E+00	1.50E+00	1.22E+00	8.63E-01	4.43E-01	5.18E-03	7.13E-21	0.0	0.0	0.0
PU239	5.63E+01	5.64E+01	5.65E+01	5.69E+01	5.73E+01	5.81E+01	6.38E+01	9.38E+01	1.12E+01	8.56E-06	5.74E-06
PU240	2.41E+01	2.66E+01	3.33E+01	4.30E+01	4.67E+01	4.68E+01	4.30E+01	1.66E+01	1.19E-03	4.54E-08	4.46E-08
PU241	9.20E+00	7.96E+00	4.92E+00	1.16E+00	1.06E-01	2.63E-03	1.67E-03	8.00E-04	5.20E-07	0.0	0.0
PU242	5.45E+00	5.45E+00	5.46E+00	5.47E+00	5.49E+00	5.52E+00	5.65E+00	6.24E+00	5.52E+00	1.10E+00	1.10E-07
AM241	4.04E+02	4.03E+02	4.00E+02	3.85E+02	3.56E+02	3.03E+02	8.42E+01	2.41E-02	1.65E-05	0.0	0.0
AM242M	3.90E-01	3.85E-01	3.68E-01	3.21E-01	2.55E-01	1.62E-01	4.22E-03	6.40E-21	0.0	0.0	0.0
AM243	1.01E+02	1.01E+02	1.01E+02	1.00E+02	1.00E+02	9.90E+01	9.19E+01	3.95E+01	8.43E-03	2.67E-06	1.79E-06
CM243	3.26E-01	3.04E-01	2.38E-01	1.15E-01	3.40E-02	2.99E-03	1.07E-11	0.0	0.0	0.0	0.0
CM244	2.41E+01	2.15E+01	1.46E+01	4.64E+00	6.86E-01	1.49E-02	7.57E-16	0.0	0.0	0.0	0.0
CM245	1.08E+00	1.08E+00	1.08E+00	1.08E+00	1.07E+00	1.07E+00	9.99E-01	4.79E-01	3.12E-04	0.0	0.0
CM246	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.08E+00	1.06E+00	9.45E-01	2.53E-01	4.76E-07	4.99E-30	0.0
CM247	6.00E-03	6.00E-03	6.00E-03	6.00E-03	6.00E-03	6.00E-03	6.00E-03	5.99E-03	5.97E-03	5.74E-03	3.85E-03
TOTAL	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.07E+04	1.07E+04	1.07E+04

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE	9.22E-01	9.99E-01	1.22E+00	1.72E+00	2.30E+00	3.22E+00	7.08E+00	1.02E+01	1.45E+01	3.85E+01	1.19E+02
TL	3.63E-11	1.26E-11	8.25E-13	3.93E-13	2.54E-13	1.13E-13	4.47E-14	7.14E-13	1.40E-11	2.91E-11	4.16E-12
PB	4.48E-05	6.70E-05	7.94E-05	8.40E-05	8.91E-05	9.42E-05	1.02E-04	1.58E-03	1.67E-01	2.66E+00	1.61E+01
BI	2.27E-09	1.02E-09	6.23E-10	2.11E-09	1.06E-08	7.90E-08	1.16E-05	1.32E-02	4.34E+00	1.75E+02	7.64E+02
PO	1.31E-13	1.71E-13	9.80E-13	6.46E-12	2.32E-11	9.23E-11	3.81E-09	1.85E-07	1.50E-06	9.13E-07	7.03E-07
AT	3.75E-20	3.80E-20	4.55E-20	1.23E-19	4.41E-19	1.83E-18	5.59E-17	5.76E-15	1.38E-13	3.03E-13	1.55E-14
RN	3.23E-11	1.13E-11	8.60E-13	7.53E-13	1.29E-12	3.71E-12	1.11E-10	5.41E-09	4.38E-08	2.67E-08	2.05E-08
FR	5.64E-16	6.78E-16	1.05E-15	2.26E-15	5.45E-15	1.82E-14	5.10E-13	5.23E-11	1.25E-09	2.76E-09	1.41E-10
RA	1.91E-07	7.26E-08	2.51E-08	6.70E-08	1.70E-07	5.65E-07	1.73E-05	8.43E-04	6.82E-03	4.16E-03	3.19E-03
AC	8.51E-09	1.27E-08	2.42E-08	4.38E-08	5.53E-08	6.25E-08	1.06E-07	7.28E-07	9.39E-06	1.59E-05	7.86E-06
TH	1.32E-03	1.28E-03	1.29E-03	1.37E-03	1.54E-03	2.04E-03	7.79E-03	1.16E-01	1.63E+00	5.10E+00	2.32E+01
PA	1.00E-04	1.01E-04	1.02E-04	1.04E-04	1.09E-04	1.19E-04	1.87E-04	9.00E-04	8.55E-03	1.15E-02	1.14E-02
U	9.52E+03	9.52E+03	9.52E+03	9.52E+03	9.52E+03	9.52E+03	9.53E+03	9.58E+03	9.75E+03	9.79E+03	9.71E+03
NP	4.95E+02	4.97E+02	5.03E+02	5.22E+02	5.51E+02	6.03E+02	8.18E+02	8.99E+02	8.74E+02	6.53E+02	3.54E+01
PU	9.67E+01	9.80E+01	1.02E+02	1.08E+02	1.10E+02	1.11E+02	1.12E+02	1.17E+02	1.67E+01	1.10E+00	5.80E-04
AM	5.05E+02	5.04E+02	5.01E+02	4.85E+02	4.56E+02	4.03E+02	1.76E+02	3.95E+01	8.45E-03	2.67E-06	1.79E-06
CM	2.66E+01	2.40E+01	1.71E+01	6.93E+00	2.88E+00	2.15E+00	1.95E+00	7.39E-01	6.53E-03	5.77E-03	3.85E-03
BK	9.32E-09	8.69E-10	3.19E-13	5.67E-20	5.66E-20	5.64E-20	5.46E-20	3.82E-20	1.06E-21	2.87E-37	0.0
CF	3.36E-06	3.24E-06	2.93E-06	2.51E-06	2.22E-06	1.83E-06	4.08E-07	4.83E-11	3.77E-17	1.02E-32	0.0
TOTALS	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.06E+04	1.07E+04	1.07E+04	1.07E+04

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
TL208	1.07E-02	3.71E-03	2.40E-04	1.11E-04	6.86E-05	2.63E-05	1.04E-07	9.26E-08	9.76E-08	1.56E-07	9.02E-07
PB212	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
BI212	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
PO212	1.91E-02	6.63E-03	4.29E-04	1.98E-04	1.23E-04	4.69E-05	1.86E-07	1.65E-07	1.74E-07	2.79E-07	1.61E-06
PO216	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
RN220	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
RA224	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
TH228	2.97E-02	1.03E-02	6.68E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
TH234	3.16E-01	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03
PA233	3.49E-01	3.50E-01	3.55E-01	3.68E-01	3.88E-01	4.25E-01	5.77E-01	6.34E-01	6.16E-01	4.60E-01	2.50E-02
PA234M	3.16E-01	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03
U 236	2.85E-03	2.85E-03	2.85E-03	2.86E-03	2.87E-03	2.91E-03	3.15E-03	4.83E-03	5.87E-03	5.72E-03	4.38E-03
U 237	2.37E-02	2.05E-02	1.27E-02	3.00E-03	2.75E-04	6.78E-06	4.30E-06	2.06E-06	1.34E-09	0.0	0.0
U 238	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03
NP237	3.49E-01	3.50E-01	3.55E-01	3.68E-01	3.88E-01	4.25E-01	5.77E-01	6.34E-01	6.16E-01	4.60E-01	2.50E-02
NP238	1.90E-02	1.87E-02	1.79E-02	1.56E-02	1.24E-02	7.86E-03	2.05E-04	3.11E-22	0.0	0.0	0.0
NP239	2.01E+01	2.01E+01	2.01E+01	2.00E+01	1.99E+01	1.97E+01	1.83E+01	7.87E+00	1.68E-03	5.32E-07	3.57E-07
PU238	2.81E+01	2.75E+01	2.56E+01	2.08E+01	1.48E+01	7.58E+00	8.88E-02	1.22E-19	0.0	0.0	0.0
PU239	3.50E+00	3.51E+00	3.51E+00	3.54E+00	3.57E+00	3.61E+00	3.96E+00	5.83E+00	6.97E-01	5.32E-07	3.57E-07
PU240	5.49E+00	6.07E+00	7.59E+00	9.81E+00	1.06E+01	1.07E+01	9.81E+00	3.78E+00	2.72E-04	1.03E-08	1.02E-08
PU241	9.48E+02	8.21E+02	5.07E+02	1.20E+02	1.10E+01	2.71E-01	1.72E-01	8.25E-02	5.36E-05	0.0	0.0
PU242	2.08E-02	2.08E-02	2.08E-02	2.09E-02	2.10E-02	2.11E-02	2.16E-02	2.38E-02	2.11E-02	4.21E-03	4.20E-10
AM241	1.39E+03	1.38E+03	1.37E+03	1.32E+03	1.22E+03	1.04E+03	2.89E+02	8.26E-02	5.65E-05	0.0	0.0
AM242M	3.79E+00	3.74E+00	3.57E+00	3.12E+00	2.48E+00	1.57E+00	4.10E-02	6.22E-20	0.0	0.0	0.0
AM242	3.77E+00	3.72E+00	3.56E+00	3.10E+00	2.47E+00	1.56E+00	4.08E-02	6.19E-20	0.0	0.0	0.0
AM243	2.01E+01	2.01E+01	2.01E+01	2.00E+01	1.99E+01	1.97E+01	1.83E+01	7.87E+00	1.68E-03	5.32E-07	3.57E-07
CM242	4.02E+00	3.09E+00	2.94E+00	2.56E+00	2.04E+00	1.29E+00	3.37E-02	5.13E-20	0.0	0.0	0.0
CM243	1.69E+01	1.57E+01	1.23E+01	5.93E+00	1.76E+00	1.54E-01	5.51E-10	0.0	0.0	0.0	0.0
CM244	1.95E+03	1.74E+03	1.18E+03	3.76E+02	5.55E+01	1.21E+00	6.13E-14	0.0	0.0	0.0	0.0
CM245	1.86E-01	1.86E-01	1.86E-01	1.85E-01	1.85E-01	1.83E-01	1.72E-01	8.24E-02	5.35E-05	0.0	0.0
CM246	3.36E-01	3.36E-01	3.35E-01	3.34E-01	3.31E-01	3.27E-01	2.91E-01	7.77E-02	1.46E-07	1.53E-30	0.0
TOTAL	4.39E+03	4.05E+03	3.17E+03	1.91E+03	1.37E+03	1.11E+03	3.42E+02	2.71E+01	4.05E+00	5.39E+00	3.29E-01

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP239	2.04E-02	2.04E-02	2.03E-02	2.03E-02	2.02E-02	2.00E-02	1.86E-02	7.97E-03	1.70E-06	5.39E-10	3.62E-10
AM241	2.00E-01	2.00E-01	1.98E-01	1.91E-01	1.77E-01	1.51E-01	4.18E-02	1.19E-05	8.16E-09	0.0	0.0
AM243	6.58E-03	6.57E-03	6.57E-03	6.55E-03	6.52E-03	6.46E-03	5.99E-03	2.57E-03	5.50E-07	1.74E-10	1.17E-10
CM243	1.33E-02	1.23E-02	9.68E-03	4.67E-03	1.38E-03	1.22E-04	4.34E-13	0.0	0.0	0.0	0.0
CM244	1.77E-02	1.57E-02	1.07E-02	3.41E-03	5.03E-04	1.10E-05	5.55E-19	0.0	0.0	0.0	0.0
TOTAL	2.61E-01	2.57E-01	2.47E-01	2.27E-01	2.07E-01	1.78E-01	6.73E-02	1.15E-02	1.41E-03	1.75E-03	1.29E-04

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=3.60E+04 MWD

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
BI212	1.07E-02	3.71E-03	2.40E-04	1.11E-04	6.86E-05	2.63E-05	1.04E-07	9.26E-08	9.76E-08	1.56E-07	9.02E-07
PO212	1.91E-02	6.63E-03	4.29E-04	1.98E-04	1.23E-04	4.69E-05	1.86E-07	1.65E-07	1.74E-07	2.79E-07	1.61E-06
PO216	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
RN220	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
RA224	2.98E-02	1.03E-02	6.70E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
TH228	2.97E-02	1.03E-02	6.68E-04	3.09E-04	1.91E-04	7.31E-05	2.90E-07	2.58E-07	2.72E-07	4.36E-07	2.51E-06
U 236	2.85E-03	2.85E-03	2.85E-03	2.86E-03	2.87E-03	2.91E-03	3.15E-03	4.83E-03	5.87E-03	5.72E-03	4.38E-03
U 238	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03	3.16E-03
NP237	3.49E-01	3.50E-01	3.55E-01	3.68E-01	3.88E-01	4.25E-01	5.77E-01	6.34E-01	6.16E-01	4.60E-01	2.50E-02
PU238	2.81E+01	2.75E+01	2.56E+01	2.08E+01	1.48E+01	7.58E+00	8.88E-02	1.22E-19	0.0	0.0	0.0
PU239	3.50E+00	3.51E+00	3.51E+00	3.54E+00	3.57E+00	3.61E+00	3.96E+00	5.83E+00	6.97E-01	5.32E-07	3.57E-07
PU240	5.49E+00	6.07E+00	7.59E+00	9.81E+00	1.06E+01	1.07E+01	9.81E+00	3.78E+00	2.72E-04	1.03E-08	1.02E-08
PU241	2.37E-02	2.05E-02	1.27E-02	3.00E-03	2.74E-04	6.78E-06	4.30E-06	2.06E-06	1.34E-09	0.0	0.0
PU242	2.08E-02	2.08E-02	2.08E-02	2.09E-02	2.10E-02	2.11E-02	2.16E-02	2.38E-02	2.11E-02	4.21E-03	4.20E-10
AM241	1.39E+03	1.38E+03	1.37E+03	1.32E+03	1.22E+03	1.04E+03	2.89E+02	8.26E-02	5.65E-05	0.0	0.0
AM242M	1.90E-02	1.87E-02	1.79E-02	1.56E-02	1.24E-02	7.86E-03	2.05E-04	3.11E-22	0.0	0.0	0.0
AM243	2.01E+01	2.01E+01	2.01E+01	2.00E+01	1.99E+01	1.97E+01	1.83E+01	7.87E+00	1.68E-03	5.32E-07	3.57E-07
CM242	4.02E+00	3.09E+00	2.94E+00	2.56E+00	2.04E+00	1.29E+00	3.37E-02	5.13E-20	0.0	0.0	0.0
CM243	1.68E+01	1.56E+01	1.23E+01	5.91E+00	1.75E+00	1.54E-01	5.50E-10	0.0	0.0	0.0	0.0
CM244	1.95E+03	1.74E+03	1.18E+03	3.76E+02	5.55E+01	1.21E+00	6.13E-14	0.0	0.0	0.0	0.0
CM245	1.86E-01	1.86E-01	1.86E-01	1.85E-01	1.85E-01	1.83E-01	1.72E-01	8.24E-02	5.35E-05	0.0	0.0
CM246	3.36E-01	3.36E-01	3.35E-01	3.34E-01	3.31E-01	3.27E-01	2.91E-01	7.77E-02	1.46E-07	1.53E-30	0.0
TOTAL	3.41E+03	3.20E+03	2.63E+03	1.76E+03	1.33E+03	1.09E+03	3.22E+02	1.85E+01	2.73E+00	3.44E+00	2.08E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP237	1.03E-02	1.03E-02	1.04E-02	1.08E-02	1.14E-02	1.25E-02	1.70E-02	1.86E-02	1.81E-02	1.35E-02	7.33E-04
NP239	4.86E-02	4.86E-02	4.85E-02	4.84E-02	4.82E-02	4.77E-02	4.43E-02	1.90E-02	4.07E-06	1.29E-09	8.63E-10
PU238	9.31E-01	9.12E-01	8.50E-01	6.90E-01	4.90E-01	2.51E-01	2.94E-03	4.05E-21	0.0	0.0	0.0
PU239	1.09E-01	1.09E-01	1.09E-01	1.10E-01	1.11E-01	1.12E-01	1.23E-01	1.81E-01	2.17E-02	1.65E-08	1.11E-08
PU240	1.71E-01	1.89E-01	2.37E-01	3.05E-01	3.31E-01	3.33E-01	3.06E-01	1.18E-01	8.46E-06	3.22E-10	3.17E-10
PU241	2.92E-02	2.53E-02	1.56E-02	3.69E-03	3.38E-04	8.36E-06	5.30E-06	2.54E-06	1.65E-09	0.0	0.0
AM241	4.64E+01	4.63E+01	4.59E+01	4.41E+01	4.09E+01	3.48E+01	9.66E+00	2.76E-03	1.89E-06	0.0	0.0
AM242M	1.50E-03	1.48E-03	1.41E-03	1.23E-03	9.80E-04	6.21E-04	1.62E-05	2.46E-23	0.0	0.0	0.0
AM242	4.38E-03	4.32E-03	4.13E-03	3.60E-03	2.87E-03	1.82E-03	4.74E-05	7.19E-23	0.0	0.0	0.0
AM243	6.48E-01	6.48E-01	6.47E-01	6.46E-01	6.43E-01	6.37E-01	5.91E-01	2.54E-01	5.42E-05	1.72E-08	1.15E-08
CM242	1.48E-01	1.14E-01	1.08E-01	9.45E-02	7.52E-02	4.77E-02	1.24E-03	1.89E-21	0.0	0.0	0.0
CM243	6.19E-01	5.75E-01	4.51E-01	2.17E-01	6.45E-02	5.67E-03	2.02E-11	0.0	0.0	0.0	0.0
CM244	6.82E+01	6.08E+01	4.15E+01	1.32E+01	1.94E+00	4.23E-02	2.14E-15	0.0	0.0	0.0	0.0
CM245	6.20E-03	6.20E-03	6.19E-03	6.18E-03	6.15E-03	6.10E-03	5.72E-03	2.74E-03	1.78E-06	0.0	0.0
CM246	1.10E-02	1.10E-02	1.10E-02	1.09E-02	1.09E-02	1.07E-02	9.51E-03	2.55E-03	4.79E-09	5.02E-32	0.0
TOTAL	1.17E+02	1.10E+02	8.98E+01	5.95E+01	4.46E+01	3.63E+01	1.08E+01	6.04E-01	9.54E-02	1.29E-01	7.73E-03

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	3.67E+02	3.10E+02	1.77E+02	3.29E+01	1.99E+00	7.26E-03	2.31E-22	0.0	0.0	0.0	0.0
SE 79	4.02E-01	4.02E-01	4.02E-01	4.02E-01	4.02E-01	4.02E-01	3.98E-01	3.62E-01	1.38E-01	9.37E-06	0.0
KR 85	6.51E+03	5.36E+03	2.81E+03	4.04E+02	1.59E+01	2.43E-02	0.0	0.0	0.0	0.0	0.0
SR 90	6.88E+04	6.40E+04	5.05E+04	2.47E+04	7.52E+03	6.96E+02	3.76E-06	0.0	0.0	0.0	0.0
Y 90	6.88E+04	6.40E+04	5.05E+04	2.47E+04	7.52E+03	6.96E+02	3.76E-06	0.0	0.0	0.0	0.0
ZR 93	1.95E+00	1.95E+00	1.95E+00	1.95E+00	1.95E+00	1.95E+00	1.95E+00	1.94E+00	1.86E+00	1.24E+00	2.10E-02
NB 93M	6.55E-01	8.25E-01	1.23E+00	1.72E+00	1.84E+00	1.85E+00	1.85E+00	1.84E+00	1.77E+00	1.18E+00	2.00E-02
TC 99	1.42E+01	1.42E+01	1.42E+01	1.42E+01	1.42E+01	1.42E+01	1.42E+01	1.38E+01	1.03E+01	5.49E-01	1.06E-13
RU106	4.47E+03	5.69E+02	6.33E-01	6.99E-10	8.24E-25	0.0	0.0	0.0	0.0	0.0	0.0
RH106	4.47E+03	5.69E+02	6.33E-01	6.99E-10	8.24E-25	0.0	0.0	0.0	0.0	0.0	0.0
PD107	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.11E-01	1.10E-01	1.00E-01	3.84E-02
AG110	3.83E-02	1.83E-03	7.32E-08	4.65E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	2.88E+00	1.38E-01	5.50E-06	3.49E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	2.90E+01	2.51E+01	1.56E+01	3.76E+00	3.50E-01	3.03E-03	9.50E-20	0.0	0.0	0.0	0.0
SN119M	8.94E-02	4.03E-03	1.32E-07	4.57E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	1.56E-01	1.50E-01	1.30E-01	8.60E-02	4.30E-02	1.07E-02	1.63E-07	0.0	0.0	0.0	0.0
SN126	5.94E-01	5.94E-01	5.94E-01	5.94E-01	5.94E-01	5.93E-01	5.90E-01	5.54E-01	2.97E-01	5.88E-04	0.0
SB125	1.71E+03	8.03E+02	6.53E+01	3.50E-02	1.24E-07	1.55E-18	0.0	0.0	0.0	0.0	0.0
SB126	8.32E-02	8.32E-02	8.31E-02	8.31E-02	8.31E-02	8.30E-02	8.26E-02	7.76E-02	4.16E-02	8.23E-05	0.0
SB126M	5.94E-01	5.94E-01	5.94E-01	5.94E-01	5.94E-01	5.93E-01	5.90E-01	5.54E-01	2.97E-01	5.88E-04	0.0
TE125M	4.16E+02	1.96E+02	1.59E+01	8.54E-03	3.02E-08	3.77E-19	0.0	0.0	0.0	0.0	0.0
I 129	3.40E-02	3.40E-02	3.40E-02	3.40E-02	3.40E-02	3.40E-02	3.40E-02	3.40E-02	3.39E-02	3.25E-02	2.19E-02
CS134	1.74E+04	6.36E+03	2.21E+02	9.23E-03	4.65E-10	1.18E-24	0.0	0.0	0.0	0.0	0.0
CS135	3.64E-01	3.64E-01	3.64E-01	3.64E-01	3.64E-01	3.64E-01	3.64E-01	3.63E-01	3.53E-01	2.69E-01	1.79E-02
CS137	9.89E+04	9.23E+04	7.32E+04	3.66E+04	1.15E+04	1.15E+03	1.08E-05	0.0	0.0	0.0	0.0
BA137M	9.35E+04	8.73E+04	6.93E+04	3.46E+04	1.09E+04	1.08E+03	1.02E-05	0.0	0.0	0.0	0.0
CE144	2.34E+03	1.62E+02	2.20E-02	5.52E-14	2.55E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144	2.34E+03	1.62E+02	2.20E-02	5.52E-14	2.55E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144M	2.81E+01	1.95E+00	2.64E-04	6.62E-16	3.06E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	2.61E+04	1.18E+04	8.42E+02	3.05E-01	5.59E-07	1.89E-18	0.0	0.0	0.0	0.0	0.0
SM151	3.35E+02	3.27E+02	3.03E+02	2.41E+02	1.64E+02	7.58E+01	1.64E-01	0.0	0.0	0.0	0.0
EU152	3.20E+00	2.75E+00	1.65E+00	3.58E-01	2.80E-02	1.71E-04	3.40E-22	0.0	0.0	0.0	0.0
EU154	7.93E+03	6.22E+03	2.78E+03	2.48E+02	4.41E+00	1.40E-03	0.0	0.0	0.0	0.0	0.0
EU155	3.15E+03	2.07E+03	5.12E+02	7.74E+00	7.15E-03	6.10E-09	0.0	0.0	0.0	0.0	0.0
TOTAL	4.08E+05	3.43E+05	2.51E+05	1.22E+05	3.77E+04	3.72E+03	2.03E+01	1.96E+01	1.52E+01	3.37E+00	1.19E-01

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=3.60E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	1.24E-02	1.05E-02	5.98E-03	1.11E-03	6.72E-05	2.45E-07	7.80E-27	0.0	0.0	0.0	0.0
KR 85	9.75E+00	8.03E+00	4.21E+00	6.05E-01	2.39E-02	3.64E-05	0.0	0.0	0.0	0.0	0.0
SR 90	7.98E+01	7.43E+01	5.86E+01	2.87E+01	8.73E+00	8.08E-01	4.36E-09	0.0	0.0	0.0	0.0
Y 90	3.81E+02	3.55E+02	2.80E+02	1.37E+02	4.17E+01	3.86E+00	2.08E-08	0.0	0.0	0.0	0.0
TC 99	7.13E-03	7.12E-03	7.12E-03	7.12E-03	7.12E-03	7.12E-03	7.10E-03	6.90E-03	5.15E-03	2.75E-04	5.29E-17
RU106	2.65E-01	3.37E-02	3.75E-05	4.14E-14	4.88E-29	0.0	0.0	0.0	0.0	0.0	0.0
RH106	4.29E+01	5.46E+00	6.07E-03	6.70E-12	7.90E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	4.81E-02	2.30E-03	9.19E-08	5.83E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	4.88E-02	4.23E-02	2.63E-02	6.33E-03	5.89E-04	5.09E-06	1.60E-22	0.0	0.0	0.0	0.0
SB125	5.33E+00	2.51E+00	2.04E-01	1.09E-04	3.87E-10	4.84E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.54E-03	1.54E-03	1.54E-03	1.54E-03	1.54E-03	1.53E-03	1.53E-03	1.43E-03	7.68E-04	1.52E-06	0.0
SB126M	7.56E-03	7.56E-03	7.56E-03	7.56E-03	7.56E-03	7.55E-03	7.51E-03	7.06E-03	3.78E-03	7.48E-06	0.0
TE125M	3.50E-01	1.65E-01	1.34E-02	7.17E-06	2.54E-11	3.17E-22	0.0	0.0	0.0	0.0	0.0
CS134	1.77E+02	6.47E+01	2.25E+00	9.39E-05	4.73E-12	1.20E-26	0.0	0.0	0.0	0.0	0.0
CS137	1.09E+02	1.02E+02	8.10E+01	4.05E+01	1.28E+01	1.27E+00	1.19E-08	0.0	0.0	0.0	0.0
BA137M	3.67E+02	3.43E+02	2.72E+02	1.36E+02	4.29E+01	4.25E+00	4.01E-08	0.0	0.0	0.0	0.0
CE144	1.56E+00	1.08E-01	1.46E-05	3.66E-17	1.69E-36	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.72E+01	1.19E+00	1.62E-04	4.05E-16	1.87E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	9.36E+00	4.24E+00	3.02E-01	1.09E-04	2.01E-10	6.77E-22	0.0	0.0	0.0	0.0	0.0
SM151	3.93E-02	3.84E-02	3.56E-02	2.82E-02	1.92E-02	8.90E-03	1.92E-05	0.0	0.0	0.0	0.0
EU152	2.43E-02	2.08E-02	1.25E-02	2.71E-03	2.12E-04	1.30E-06	2.57E-24	0.0	0.0	0.0	0.0
EU154	7.09E+01	5.57E+01	2.49E+01	2.22E+00	3.95E-02	1.25E-05	0.0	0.0	0.0	0.0	0.0
EU155	2.29E+00	1.51E+00	3.72E-01	5.63E-03	5.20E-06	4.44E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	1.28E+03	1.02E+03	7.24E+02	3.45E+02	1.06E+02	1.02E+01	1.77E-02	1.69E-02	1.08E-02	7.49E-04	2.44E-05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
KR 85	4.07E-01	3.35E-01	1.76E-01	2.52E-02	9.96E-04	1.52E-06	0.0	0.0	0.0	0.0	0.0
SR 90	4.48E+00	4.17E+00	3.29E+00	1.61E+00	4.90E-01	4.53E-02	2.45E-10	0.0	0.0	0.0	0.0
Y 90	6.10E+01	5.68E+01	4.48E+01	2.19E+01	6.68E+00	6.18E-01	3.34E-09	0.0	0.0	0.0	0.0
RH106	9.40E+00	1.20E+00	1.33E-03	1.47E-12	1.73E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	4.69E-02	2.25E-03	8.96E-08	5.69E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SB125	4.37E+00	2.06E+00	1.67E-01	8.97E-05	3.17E-10	3.96E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.40E-03	1.39E-03	1.31E-03	7.01E-04	1.39E-06	0.0
SB126M	5.57E-03	5.57E-03	5.57E-03	5.57E-03	5.56E-03	5.56E-03	5.53E-03	5.20E-03	2.78E-03	5.51E-06	0.0
TE125M	8.78E-02	4.13E-02	3.36E-03	1.80E-06	6.36E-12	7.95E-23	0.0	0.0	0.0	0.0	0.0
CS134	1.61E+02	5.87E+01	2.04E+00	8.52E-05	4.29E-12	1.09E-26	0.0	0.0	0.0	0.0	0.0
CS137	2.76E+00	2.57E+00	2.04E+00	1.02E+00	3.22E-01	3.20E-02	3.01E-10	0.0	0.0	0.0	0.0
BA137M	3.30E+02	3.08E+02	2.44E+02	1.22E+02	3.85E+01	3.82E+00	3.60E-08	0.0	0.0	0.0	0.0
CE144	3.02E-01	2.09E-02	2.84E-06	7.11E-18	3.29E-37	0.0	0.0	0.0	0.0	0.0	0.0
PR144	2.03E+00	1.40E-01	1.91E-05	4.77E-17	2.21E-36	0.0	0.0	0.0	0.0	0.0	0.0
PM147	1.22E-01	5.52E-02	3.94E-03	1.42E-06	2.62E-12	8.83E-24	0.0	0.0	0.0	0.0	0.0
EU152	2.17E-02	1.87E-02	1.12E-02	2.43E-03	1.90E-04	1.16E-06	2.31E-24	0.0	0.0	0.0	0.0
EU154	5.81E+01	4.56E+01	2.04E+01	1.82E+00	3.23E-02	1.02E-05	0.0	0.0	0.0	0.0	0.0
EU155	1.14E+00	7.50E-01	1.85E-01	2.80E-03	2.59E-06	2.21E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	6.35E+02	4.80E+02	3.17E+02	1.49E+02	4.60E+01	4.52E+00	7.39E-03	6.95E-03	3.78E-03	3.56E-05	3.56E-06

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE LIGHT ELEMENTS

BURNUP=3.60E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	1.22E-03	1.03E-03	5.89E-04	1.09E-04	6.62E-06	2.42E-08	7.70E-28	0.0	0.0	0.0	0.0
C 14	1.57E-01	1.57E-01	1.57E-01	1.56E-01	1.55E-01	1.53E-01	1.39E-01	4.68E-02	8.77E-07	0.0	0.0
MN 54	2.69E+01	2.37E+00	7.20E-04	2.02E-14	5.21E-32	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	1.05E+04	4.72E+03	3.28E+02	1.11E-01	1.80E-07	4.79E-19	0.0	0.0	0.0	0.0	0.0
CO 60	8.87E+03	5.98E+03	1.60E+03	3.10E+01	4.44E-02	8.63E-08	0.0	0.0	0.0	0.0	0.0
NI 59	1.90E+01	1.90E+01	1.90E+01	1.90E+01	1.90E+01	1.90E+01	1.89E+01	1.74E+01	8.00E+00	3.29E-03	0.0
NI 63	2.56E+03	2.50E+03	2.32E+03	1.85E+03	1.27E+03	5.98E+02	1.45E+00	0.0	0.0	0.0	0.0
SR 90	2.26E-03	2.11E-03	1.66E-03	8.13E-04	2.47E-04	2.29E-05	1.24E-13	0.0	0.0	0.0	0.0
Y 90	2.26E-03	2.11E-03	1.66E-03	8.14E-04	2.48E-04	2.29E-05	1.24E-13	0.0	0.0	0.0	0.0
ZR 93	2.00E-01	2.00E-01	2.00E-01	2.00E-01	2.00E-01	2.00E-01	2.00E-01	1.99E-01	1.91E-01	1.27E-01	2.16E-03
NB 93M	6.59E-02	8.35E-02	1.26E-01	1.76E-01	1.89E-01	1.90E-01	1.90E-01	1.89E-01	1.82E-01	1.21E-01	2.05E-03
NB 94	1.13E+00	1.13E+00	1.13E+00	1.13E+00	1.13E+00	1.13E+00	1.10E+00	8.07E-01	3.73E-02	1.69E-15	0.0
MO 93	7.77E-02	7.76E-02	7.75E-02	7.70E-02	7.63E-02	7.48E-02	6.38E-02	1.07E-02	1.94E-10	0.0	0.0
TC 99	1.29E-02	1.29E-02	1.29E-02	1.29E-02	1.29E-02	1.29E-02	1.29E-02	1.25E-02	9.33E-03	4.99E-04	9.59E-17
SN119M	7.76E+00	3.50E-01	1.14E-05	3.97E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	9.14E-01	8.77E-01	7.63E-01	5.03E-01	2.52E-01	6.29E-02	9.56E-07	0.0	0.0	0.0	0.0
SB125	1.46E+01	6.90E+00	5.60E-01	3.00E-04	1.06E-09	1.33E-20	0.0	0.0	0.0	0.0	0.0
TE125M	3.58E+00	1.68E+00	1.37E-01	7.33E-05	2.59E-10	3.24E-21	0.0	0.0	0.0	0.0	0.0
TOTAL	2.20E+04	1.32E+04	4.28E+03	1.90E+03	1.29E+03	6.19E+02	2.20E+01	1.87E+01	8.42E+00	2.52E-01	4.21E-03

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.34E-01	1.18E-02	3.59E-06	1.00E-16	2.59E-34	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	1.44E+01	6.49E+00	4.51E-01	1.52E-04	2.48E-10	6.59E-22	0.0	0.0	0.0	0.0	0.0
CO 60	1.37E+02	9.21E+01	2.47E+01	4.79E-01	6.84E-04	1.33E-09	0.0	0.0	0.0	0.0	0.0
NI 59	1.21E-01	1.21E-01	1.21E-01	1.21E-01	1.21E-01	1.21E-01	1.20E-01	1.11E-01	5.09E-02	2.09E-05	0.0
NI 63	1.02E+00	9.94E-01	9.22E-01	7.35E-01	5.05E-01	2.38E-01	5.78E-04	0.0	0.0	0.0	0.0
NB 94	1.16E-02	1.16E-02	1.16E-02	1.15E-02	1.15E-02	1.15E-02	1.12E-02	8.22E-03	3.81E-04	1.73E-17	0.0
SN121M	1.83E-03	1.76E-03	1.53E-03	1.01E-03	5.04E-04	1.26E-04	1.92E-09	0.0	0.0	0.0	0.0
SB125	4.58E-02	2.16E-02	1.75E-03	9.39E-07	3.32E-12	4.15E-23	0.0	0.0	0.0	0.0	0.0
TE125M	3.01E-03	1.41E-03	1.15E-04	6.16E-08	2.18E-13	2.72E-24	0.0	0.0	0.0	0.0	0.0
TOTAL	1.52E+02	9.98E+01	2.62E+01	1.35E+00	6.38E-01	3.70E-01	1.32E-01	1.19E-01	5.14E-02	5.74E-05	6.15E-07

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.34E-01	1.18E-02	3.57E-06	1.00E-16	2.58E-34	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	1.31E+02	8.86E+01	2.38E+01	4.60E-01	6.58E-04	1.28E-09	0.0	0.0	0.0	0.0	0.0
NB 94	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.06E-02	1.05E-02	1.02E-02	7.53E-03	3.49E-04	1.58E-17	0.0
SB125	3.75E-02	1.77E-02	1.44E-03	7.70E-07	2.72E-12	3.40E-23	0.0	0.0	0.0	0.0	0.0
TOTAL	1.32E+02	8.87E+01	2.38E+01	4.72E-01	1.17E-02	1.08E-02	1.03E-02	7.54E-03	3.51E-04	1.42E-06	2.40E-08

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE

BURNUP=3.60E+04 MWD

********** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS **********

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
				CONCENTRATIONS ,GRAMS							
TOTAL	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05	6.09E+05
				RADIOACTIVITY ,CURIES							
TOTAL	4.34E+05	3.60E+05	2.59E+05	1.25E+05	4.04E+04	5.45E+03	3.84E+02	6.54E+01	2.76E+01	9.01E+00	4.52E-01
				THERMAL POWER ,WATTS							
TOTAL	1.54E+03	1.23E+03	8.40E+02	4.06E+02	1.51E+02	4.69E+01	1.09E+01	7.40E-01	1.58E-01	1.30E-01	7.75E-03
				GAMMA POWER ,WATTS							
TOTAL	7.67E+02	5.69E+02	3.41E+02	1.49E+02	4.62E+01	4.71E+00	8.49E-02	2.60E-02	5.54E-03	1.79E-03	1.33E-04
				ALPHA-ACTIVITY ,CURIES							
TOTAL	3.41E+03	3.20E+03	2.63E+03	1.76E+03	1.33E+03	1.09E+03	3.22E+02	1.85E+01	2.73E+00	3.44E+00	2.08E-01

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR ACTINIDES

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 36.04 MW, BURNUP= 3.60E+04 MWD

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
				TIME AFTER DISCHARGE							
1.00E-02	2.40E+13	2.27E+13	1.95E+13	1.43E+13	1.15E+13	9.59E+12	3.25E+12	4.11E+11	5.32E+10	6.28E+10	3.69E+09
2.50E-02	1.31E+12	1.31E+12	1.30E+12	1.25E+12	1.16E+12	9.87E+11	2.81E+11	6.52E+09	3.58E+09	3.69E+09	2.54E+08
3.75E-02	1.87E+11	1.82E+11	1.75E+11	1.61E+11	1.50E+11	1.34E+11	7.19E+10	2.16E+10	3.97E+09	8.52E+09	4.71E+08
5.75E-02	1.91E+13	1.91E+13	1.89E+13	1.82E+13	1.68E+13	1.44E+13	3.99E+12	8.05E+09	1.04E+09	1.76E+09	1.53E+08
8.50E-02	6.91E+11	6.81E+11	6.59E+11	6.17E+11	5.87E+11	5.69E+11	5.22E+11	2.31E+11	1.85E+10	2.33E+10	1.29E+09
1.25E-01	5.72E+11	5.58E+11	5.19E+11	4.47E+11	3.97E+11	3.72E+11	3.34E+11	1.44E+11	3.28E+09	4.03E+09	2.35E+08
2.25E-01	4.52E+11	4.36E+11	3.95E+11	3.17E+11	2.66E+11	2.44E+11	2.25E+11	9.69E+10	2.98E+09	5.54E+09	3.73E+08
3.75E-01	3.17E+10	3.09E+10	3.05E+10	3.00E+10	2.99E+10	3.01E+10	3.02E+10	1.88E+10	1.19E+10	1.29E+10	7.37E+08
5.75E-01	2.37E+09	1.66E+09	1.20E+09	7.15E+08	5.03E+08	4.21E+08	2.14E+08	8.18E+07	3.23E+08	4.86E+08	8.84E+07
8.50E-01	1.88E+09	1.48E+09	1.07E+09	5.57E+08	3.19E+08	2.16E+08	5.96E+07	1.61E+07	6.60E+07	8.91E+07	2.15E+07
1.25E+00	8.21E+08	6.65E+08	4.97E+08	2.48E+08	1.31E+08	8.04E+07	1.90E+07	1.71E+07	1.15E+08	1.16E+08	4.43E+07
1.75E+00	3.31E+08	2.65E+08	1.76E+08	6.22E+07	1.72E+07	9.36E+06	8.32E+06	1.75E+07	2.30E+08	3.97E+08	5.05E+07
2.25E+00	1.63E+08	1.46E+08	1.01E+08	3.57E+07	9.76E+06	5.29E+06	4.66E+06	3.87E+06	2.12E+07	1.29E+07	9.94E+06
2.75E+00	4.70E+08	2.15E+08	6.72E+07	2.46E+07	8.08E+06	3.99E+06	2.68E+06	7.72E+05	3.80E+05	2.31E+05	2.05E+05
3.50E+00	5.63E+07	5.04E+07	3.49E+07	1.23E+07	3.37E+06	1.83E+06	1.59E+06	4.38E+05	7.36E+04	4.30E+04	3.25E+04
5.00E+00	1.76E+07	1.57E+07	1.09E+07	3.85E+06	1.05E+06	5.71E+05	4.97E+05	1.34E+05	1.34E+03	2.59E+02	1.10E+01
7.00E+00	2.02E+06	1.80E+06	1.25E+06	4.42E+05	1.21E+05	6.54E+04	5.70E+04	1.54E+04	1.54E+02	2.93E+01	1.24E+00
9.50E+00	2.01E+05	1.80E+05	1.25E+05	4.41E+04	1.21E+04	6.53E+03	5.69E+03	1.54E+03	1.54E+01	2.92E+00	1.23E-01
TOTAL	4.64E+13	4.50E+13	4.15E+13	3.53E+13	3.09E+13	2.63E+13	8.71E+12	9.37E+11	9.92E+10	1.24E+11	7.42E+09
MEV/SEC	1.63E+12	1.61E+12	1.55E+12	1.42E+12	1.29E+12	1.11E+12	4.20E+11	7.21E+10	8.79E+09	1.09E+10	8.05E+08

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 36.04 MW, BURNUP= 3.60E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	2.32E+15	1.99E+15	1.48E+15	6.98E+14	2.11E+14	1.97E+13	2.33E+10	2.22E+10	1.43E+10	1.20E+09	1.48E+08
3.75E-01	2.08E+14	1.70E+14	1.24E+14	5.96E+13	1.81E+13	1.70E+12	2.39E+10	2.25E+10	1.20E+10	2.38E+07	3.64E-05
5.75E-01	4.59E+15	3.75E+15	2.72E+15	1.35E+15	4.25E+14	4.22E+13	5.31E+10	4.99E+10	2.67E+10	5.29E+07	1.41E-09
8.50E-01	7.40E+14	3.43E+14	7.58E+13	1.40E+13	3.00E+12	2.73E+11	2.69E+09	2.51E+09	1.34E+09	2.65E+06	1.99E+00
1.25E+00	2.06E+14	1.44E+14	6.12E+13	8.01E+12	1.05E+12	9.01E+10	6.46E+08	6.06E+08	3.25E+08	6.43E+05	2.49E-03
1.75E+00	6.21E+12	4.39E+12	2.13E+12	3.93E+11	7.80E+10	6.98E+09	2.76E+04	2.59E+04	1.39E+04	2.74E+01	1.25E-03
2.25E+00	9.03E+11	7.66E+10	9.63E+07	2.72E+07	8.29E+06	7.68E+05	4.77E-03	6.26E-04	6.26E-04	6.26E-04	6.26E-04
2.75E+00	3.62E+10	4.56E+09	5.04E+06	5.87E-03	3.14E-04	3.14E-04	3.14E-04	3.14E-04	3.14E-04	3.14E-04	3.13E-04
3.50E+00	4.65E+09	5.91E+08	6.58E+05	9.57E-04	2.31E-04	2.31E-04	2.31E-04	2.31E-04	2.31E-04	2.31E-04	2.31E-04
5.00E+00	6.70E-05	6.77E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05	6.82E-05
7.00E+00	4.38E-06	4.42E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06	4.46E-06
9.50E+00	2.83E-07	2.86E-07	2.88E-07	2.89E-07	2.89E-07	2.89E-07	2.89E-07	2.89E-07	2.89E-07	2.89E-07	2.89E-07
TOTAL	8.07E+15	6.40E+15	4.47E+15	2.13E+15	6.59E+14	6.40E+13	1.04E+11	9.77E+10	5.48E+10	1.28E+09	1.48E+08
MEV/SEC	3.96E+15	3.00E+15	1.98E+15	9.27E+14	2.87E+14	2.82E+13	4.61E+10	4.33E+10	2.36E+10	2.22E+08	2.22E+07

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 36.04 MW, BURNUP= 3.60E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	3.86E+12	2.56E+12	7.04E+11	4.44E+10	2.21E+10	1.11E+10	1.22E+09	8.05E+08	1.26E+08	5.92E+07	9.99E+05
3.75E-01	2.23E+11	1.10E+11	1.24E+10	1.04E+08	3.94E+06	3.26E+06	3.12E+06	2.29E+06	1.06E+05	2.55E+01	3.59E-01
5.75E-01	2.53E+11	1.19E+11	9.90E+09	1.16E+07	3.17E+05	2.87E+04	2.37E+01	2.32E+01	1.89E+01	2.46E+00	2.14E-03
8.50E-01	1.08E+12	1.80E+11	8.21E+10	7.76E+10	7.74E+10	7.71E+10	7.50E+10	5.52E+10	2.56E+09	1.18E-04	2.55E-15
1.25E+00	6.56E+14	4.42E+14	1.19E+14	2.30E+12	3.28E+09	9.32E+03	1.59E-05	3.97E-08	3.23E-08	4.07E-09	1.10E-13
1.75E+00	2.28E+04	2.11E+04	1.67E+04	8.15E+03	2.48E+03	2.30E+02	1.24E-06	0.0	0.0	0.0	0.0
2.25E+00	3.48E+09	2.34E+09	6.29E+08	1.22E+07	1.74E+04	5.91E-02	1.36E-10	0.0	0.0	0.0	0.0
2.75E+00	1.08E+07	7.25E+06	1.95E+06	3.77E+04	5.38E+01	1.05E-04	0.0	0.0	0.0	0.0	0.0
3.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
5.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	6.61E+14	4.45E+14	1.20E+14	2.42E+12	1.03E+11	8.82E+10	7.63E+10	5.60E+10	2.68E+09	5.92E+07	9.99E+05
MEV/SEC	8.22E+14	5.53E+14	1.49E+14	2.94E+12	7.32E+10	6.72E+10	6.40E+10	4.70E+10	2.19E+09	8.88E+06	1.50E+05

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PO212	3.31E+04	1.15E+04	7.44E+02	3.43E+02	2.12E+02	8.12E+01	3.23E-01	2.86E-01	3.02E-01	4.84E-01	2.79E+00
PO213	5.24E-02	5.31E-02	6.36E-02	1.72E-01	6.16E-01	2.55E+00	7.80E+01	8.04E+03	1.92E+05	4.24E+05	2.16E+04
PO216	8.47E+02	2.94E+02	1.91E+01	8.80E+00	5.44E+00	2.08E+00	8.27E-03	7.34E-03	7.74E-03	1.24E-02	7.15E-02
AT217	4.94E-03	5.00E-03	5.99E-03	1.62E-02	5.80E-02	2.41E-01	7.35E+00	7.57E+02	1.81E+04	3.99E+04	2.04E+03
RN220	4.72E+02	1.64E+02	1.06E+01	4.90E+00	3.03E+00	1.16E+00	4.61E-03	4.09E-03	4.31E-03	6.91E-03	3.99E-02
FR221	1.08E-03	1.09E-03	1.31E-03	3.53E-03	1.26E-02	5.24E-02	1.60E+00	1.65E+02	3.95E+03	8.70E+03	4.44E+02
NP237	1.53E+02	1.53E+02	1.55E+02	1.61E+02	1.70E+02	1.86E+02	2.52E+02	2.77E+02	2.69E+02	2.01E+02	1.09E+01
PU238	3.28E+04	3.21E+04	2.99E+04	2.43E+04	1.73E+04	8.85E+03	1.04E+02	1.43E-16	0.0	0.0	0.0
PU239	2.53E+03	2.54E+03	2.54E+03	2.56E+03	2.58E+03	2.62E+03	2.87E+03	4.22E+03	5.04E+02	3.85E-04	2.58E-04
PU240	4.09E+03	4.53E+03	5.66E+03	7.31E+03	7.93E+03	7.96E+03	7.32E+03	2.82E+03	2.03E-01	7.71E-06	7.59E-06
AM241	1.62E+06	1.61E+06	1.60E+06	1.54E+06	1.42E+06	1.21E+06	3.37E+05	9.63E+01	6.58E-02	0.0	0.0
AM243	3.23E+04	3.23E+04	3.22E+04	3.21E+04	3.20E+04	3.17E+04	2.94E+04	1.26E+04	2.70E+00	8.55E-04	5.73E-04
CM242	3.24E+04	2.50E+04	2.37E+04	2.07E+04	1.65E+04	1.04E+04	2.72E+02	4.14E-16	0.0	0.0	0.0
CM243	1.57E+05	1.46E+05	1.14E+05	5.52E+04	1.64E+04	1.44E+03	5.13E-06	0.0	0.0	0.0	0.0
CM244	1.38E+07	1.23E+07	8.37E+06	2.66E+06	3.92E+05	8.54E+03	4.33E-10	0.0	0.0	0.0	0.0
CM245	4.76E+02	4.76E+02	4.75E+02	4.74E+02	4.72E+02	4.68E+02	4.39E+02	2.11E+02	1.37E-01	0.0	0.0
CM246	5.93E+02	5.93E+02	5.92E+02	5.89E+02	5.85E+02	5.77E+02	5.13E+02	1.37E+02	2.58E-04	2.71E-27	0.0
TOTAL	1.57E+07	1.41E+07	1.02E+07	4.34E+06	1.91E+06	1.29E+06	3.78E+05	2.97E+04	2.19E+05	4.77E+05	2.53E+04

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

WASTE DECAY BIBLIS-TYPE 3.4% U235 REPROCESSING 7 Y AFTER DISCHARGE
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PU238	4.36E+03	4.27E+03	3.98E+03	3.23E+03	2.29E+03	1.18E+03	1.38E+01	1.90E-17	0.0	0.0	0.0
PU240	2.19E+04	2.42E+04	3.03E+04	3.92E+04	4.25E+04	4.26E+04	3.92E+04	1.51E+04	1.09E+00	4.13E-05	4.07E-05
PU242	9.19E+03	9.19E+03	9.20E+03	9.22E+03	9.25E+03	9.30E+03	9.53E+03	1.05E+04	9.31E+03	1.86E+03	1.85E-04
AM241	5.01E+02	5.00E+02	4.96E+02	4.77E+02	4.42E+02	3.77E+02	1.04E+02	2.99E-02	2.04E-05	0.0	0.0
AM243	3.37E+02	3.37E+02	3.37E+02	3.36E+02	3.34E+02	3.31E+02	3.07E+02	1.32E+02	2.82E-02	8.93E-06	5.99E-06
CM242	2.62E+04	2.01E+04	1.92E+04	1.67E+04	1.33E+04	8.43E+03	2.20E+02	3.34E-16	0.0	0.0	0.0
CM244	2.68E+08	2.39E+08	1.63E+08	5.17E+07	7.62E+06	1.66E+05	8.42E-09	0.0	0.0	0.0	0.0
CM246	9.73E+06	9.73E+06	9.71E+06	9.67E+06	9.60E+06	9.46E+06	8.41E+06	2.25E+06	4.24E+00	4.44E-23	0.0
CM248	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.29E+04	1.08E+04	1.71E+03	1.76E-05
CF250	7.61E+03	6.49E+03	3.82E+03	7.80E+02	5.52E+01	2.77E-01	2.16E-05	1.51E-05	4.18E-07	1.13E-22	0.0
CF252	6.45E+04	2.93E+04	2.12E+03	8.02E-01	1.58E-06	6.19E-18	0.0	0.0	0.0	0.0	0.0
TOTAL	2.78E+08	2.49E+08	1.73E+08	6.14E+07	1.73E+07	9.70E+06	8.48E+06	2.29E+06	2.02E+04	3.69E+03	1.21E+02
TOTAL	2.93E+08	2.63E+08	1.83E+08	6.58E+07	1.92E+07	1.10E+07	8.86E+06	2.32E+06	2.39E+05	4.81E+05	2.55E+04

D.3 BIBLIS Type 3.6 w/o U-235, Burnup 40 GWd/tU

[see tables on following pages]

FUEL DECAY BIBLIS-TYPE 3.6% U235

ACTINIDES

BURNUP=4.00E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE 4	0.0	3.80E-01	6.56E-01	7.67E-01	9.79E-01	1.11E+00
U 233	0.0	2.64E-03	2.84E-03	3.04E-03	3.60E-03	3.99E-03
U 234	0.0	2.90E+00	4.49E+00	6.12E+00	1.10E+01	1.42E+01
U 235	3.60E+04	7.65E+03	7.65E+03	7.65E+03	7.65E+03	7.65E+03
U 236	0.0	4.80E+03	4.80E+03	4.80E+03	4.80E+03	4.80E+03
U 238	9.64E+05	9.35E+05	9.35E+05	9.35E+05	9.35E+05	9.35E+05
NP236M	0.0	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03
NP237	0.0	5.54E+02	5.67E+02	5.67E+02	5.68E+02	5.69E+02
PU236	0.0	2.40E-03	1.90E-03	1.49E-03	7.17E-04	4.41E-04
PU238	0.0	1.96E+02	2.09E+02	2.10E+02	2.06E+02	2.03E+02
PU239	0.0	5.70E+03	5.81E+03	5.81E+03	5.81E+03	5.81E+03
PU240	0.0	2.58E+03	2.58E+03	2.58E+03	2.58E+03	2.59E+03
PU241	0.0	1.40E+03	1.34E+03	1.27E+03	1.10E+03	1.00E+03
PU242	0.0	6.40E+02	6.40E+02	6.40E+02	6.40E+02	6.40E+02
PU244	0.0	5.17E-02	5.17E-02	5.17E-02	5.17E-02	5.17E-02
AM241	0.0	3.96E+01	1.05E+02	1.68E+02	3.38E+02	4.38E+02
AM242M	0.0	4.12E-01	4.10E-01	4.08E-01	4.02E-01	3.99E-01
AM243	0.0	1.28E+02	1.28E+02	1.28E+02	1.28E+02	1.28E+02
CM242	0.0	1.62E+01	3.45E+00	7.32E-01	7.94E-03	1.28E-03
CM243	0.0	4.83E-01	4.72E-01	4.60E-01	4.28E-01	4.08E-01
CM244	0.0	4.44E+01	4.29E+01	4.13E+01	3.68E+01	3.41E+01
CM245	0.0	1.59E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00
CM246	0.0	1.84E+00	1.84E+00	1.84E+00	1.84E+00	1.83E+00
CM247	0.0	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02
TOTAL	1.00E+06	9.59E+05	9.59E+05	9.59E+05	9.59E+05	9.59E+05

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
HE	0.0	3.80E-01	6.56E-01	7.67E-01	9.79E-01	1.11E+00
TL	0.0	2.10E-12	6.69E-12	1.34E-11	3.46E-11	4.55E-11
PB	0.0	4.47E-07	1.81E-06	5.09E-06	2.92E-05	5.59E-05
BI	0.0	2.33E-10	5.27E-10	9.27E-10	2.19E-09	2.85E-09
PO	0.0	5.20E-15	1.66E-14	3.37E-14	1.00E-13	1.59E-13
AT	0.0	2.79E-19	3.45E-20	3.62E-20	4.18E-20	4.61E-20
RN	0.0	1.87E-12	5.95E-12	1.19E-11	3.08E-11	4.05E-11
FR	0.0	2.57E-15	3.64E-16	3.98E-16	5.26E-16	6.32E-16
RA	0.0	1.09E-08	3.46E-08	6.95E-08	1.80E-07	2.39E-07
AC	0.0	1.36E-09	1.94E-09	2.67E-09	5.60E-09	8.15E-09
TH	0.0	2.70E-04	4.25E-04	5.86E-04	1.10E-03	1.46E-03
PA	0.0	4.61E-05	5.39E-05	6.13E-05	8.36E-05	9.85E-05
U	1.00E+06	9.47E+05	9.47E+05	9.47E+05	9.47E+05	9.47E+05
NP	0.0	6.59E+02	5.67E+02	5.67E+02	5.68E+02	5.69E+02
PU	0.0	1.05E+04	1.06E+04	1.05E+04	1.03E+04	1.02E+04
AM	0.0	1.68E+02	2.34E+02	2.97E+02	4.67E+02	5.67E+02
CM	0.0	6.45E+01	5.03E+01	4.59E+01	4.07E+01	3.79E+01
BK	0.0	5.50E-06	2.50E-06	1.13E-06	1.06E-07	2.17E-08
CF	0.0	3.77E-06	6.53E-06	7.67E-06	8.16E-06	7.97E-06
ES	0.0	1.43E-09	5.48E-14	2.66E-19	2.11E-35	3.85E-46
TOTALS	1.00E+06	9.59E+05	9.59E+05	9.59E+05	9.59E+05	9.59E+05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 237	0.0	8.83E+02	2.99E-03	2.85E-03	2.46E-03	2.24E-03
NP239	0.0	2.41E+04	2.59E-02	2.59E-02	2.59E-02	2.59E-02
PU238	0.0	3.18E-02	3.39E-02	3.41E-02	3.34E-02	3.29E-02
PU239	0.0	1.32E-03	1.34E-03	1.34E-03	1.34E-03	1.34E-03
PU240	0.0	5.33E-03	5.33E-03	5.33E-03	5.34E-03	5.34E-03
PU241	0.0	6.52E-03	6.21E-03	5.92E-03	5.12E-03	4.65E-03
AM241	0.0	1.97E-02	5.23E-02	8.34E-02	1.68E-01	2.17E-01
AM243	0.0	8.36E-03	8.37E-03	8.37E-03	8.37E-03	8.37E-03
CM242	0.0	5.23E-01	1.12E-01	2.37E-02	2.57E-04	4.14E-05
CM243	0.0	1.97E-02	1.92E-02	1.87E-02	1.74E-02	1.66E-02
CM244	0.0	3.26E-02	3.15E-02	3.03E-02	2.70E-02	2.50E-02
TOTAL	7.38E-05	3.74E+04	3.00E-01	2.42E-01	2.96E-01	3.42E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PA234M	0.0	1.59E-03	1.55E-03	1.55E-03	1.55E-03	1.55E-03
U 236	0.0	8.42E-03	8.42E-03	8.42E-03	8.42E-03	8.42E-03
U 237	0.0	1.93E+03	6.53E-03	6.22E-03	5.38E-03	4.89E-03
U 238	8.21E-03	7.96E-03	7.96E-03	7.96E-03	7.96E-03	7.96E-03
NP237	0.0	1.15E-02	1.17E-02	1.17E-02	1.18E-02	1.18E-02
NP239	0.0	5.75E+04	6.19E-02	6.19E-02	6.19E-02	6.18E-02
PU236	0.0	4.44E-02	3.50E-02	2.75E-02	1.33E-02	8.15E-03
PU238	0.0	1.12E+02	1.19E+02	1.19E+02	1.17E+02	1.15E+02
PU239	0.0	1.10E+01	1.12E+01	1.12E+01	1.12E+01	1.12E+01
PU240	0.0	1.83E+01	1.83E+01	1.83E+01	1.84E+01	1.84E+01
PU241	0.0	4.46E+00	4.25E+00	4.05E+00	3.50E+00	3.18E+00
PU242	0.0	7.22E-02	7.22E-02	7.22E-02	7.22E-02	7.22E-02
AM241	0.0	4.55E+00	1.21E+01	1.93E+01	3.88E+01	5.03E+01
AM242M	0.0	1.58E-03	1.57E-03	1.57E-03	1.54E-03	1.53E-03
AM242	0.0	1.13E+02	4.61E-03	4.59E-03	4.52E-03	4.48E-03
AM243	0.0	8.24E-01	8.25E-01	8.25E-01	8.25E-01	8.25E-01
CM242	0.0	1.97E+03	4.20E+02	8.92E+01	9.67E-01	1.56E-01
CM243	0.0	9.16E-01	8.94E-01	8.72E-01	8.11E-01	7.72E-01
CM244	0.0	1.26E+02	1.22E+02	1.17E+02	1.04E+02	9.66E+01
CM245	0.0	9.09E-03	9.09E-03	9.08E-03	9.08E-03	9.08E-03
CM246	0.0	1.85E-02	1.85E-02	1.85E-02	1.85E-02	1.85E-02
TOTAL	1.04E-02	1.31E+05	7.08E+02	3.80E+02	2.96E+02	2.97E+02

FUEL DECAY BIBLIS-TYPE 3.6% U235

BURNUP=4.00E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TL208	0.0	6.19E-04	1.97E-03	3.95E-03	1.02E-02	1.34E-02
PB212	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
BI212	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
PO212	0.0	1.11E-03	3.52E-03	7.06E-03	1.82E-02	2.39E-02
PO216	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
RN220	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
RA224	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
TH228	0.0	1.72E-03	5.48E-03	1.10E-02	2.84E-02	3.72E-02
TH231	0.0	2.77E-02	1.66E-02	1.66E-02	1.66E-02	1.66E-02
TH234	0.0	3.15E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
PA233	0.0	3.91E-01	4.00E-01	4.00E-01	4.01E-01	4.02E-01
PA234M	0.0	3.22E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
U 232	0.0	8.13E-03	1.90E-02	2.74E-02	4.25E-02	4.75E-02
U 234	0.0	1.81E-02	2.81E-02	3.83E-02	6.86E-02	8.85E-02
U 235	7.78E-02	1.65E-02	1.66E-02	1.66E-02	1.66E-02	1.66E-02
U 236	0.0	3.11E-01	3.11E-01	3.11E-01	3.11E-01	3.11E-01
U 237	0.0	1.02E+06	3.45E+00	3.29E+00	2.85E+00	2.58E+00
U 238	3.24E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
NP237	0.0	3.91E-01	4.00E-01	4.00E-01	4.01E-01	4.02E-01
NP238	0.0	4.89E+05	1.99E-02	1.98E-02	1.96E-02	1.94E-02
NP239	0.0	2.38E+07	2.56E+01	2.56E+01	2.56E+01	2.56E+01
PU236	0.0	1.28E+00	1.01E+00	7.90E-01	3.81E-01	2.34E-01
PU238	0.0	3.37E+03	3.59E+03	3.60E+03	3.53E+03	3.48E+03
PU239	0.0	3.55E+02	3.61E+02	3.61E+02	3.61E+02	3.61E+02
PU240	0.0	5.88E+02	5.88E+02	5.88E+02	5.89E+02	5.90E+02
PU241	0.0	1.45E+05	1.38E+05	1.31E+05	1.14E+05	1.03E+05
PU242	0.0	2.44E+00	2.44E+00	2.44E+00	2.44E+00	2.44E+00
AM241	0.0	1.36E+02	3.62E+02	5.77E+02	1.16E+03	1.50E+03
AM242M	0.0	4.00E+00	3.99E+00	3.97E+00	3.91E+00	3.88E+00
AM242	0.0	9.72E+04	3.97E+00	3.95E+00	3.89E+00	3.86E+00
AM243	0.0	2.56E+01	2.56E+01	2.56E+01	2.56E+01	2.56E+01
CM242	0.0	5.34E+04	1.14E+04	2.42E+03	2.63E+01	4.23E+00
CM243	0.0	2.50E+01	2.44E+01	2.38E+01	2.21E+01	2.11E+01
CM244	0.0	3.60E+03	3.47E+03	3.34E+03	2.98E+03	2.76E+03
CM245	0.0	2.73E-01	2.73E-01	2.73E-01	2.72E-01	2.72E-01
CM246	0.0	5.64E-01	5.64E-01	5.64E-01	5.64E-01	5.64E-01
BK249	0.0	9.01E-03	4.09E-03	1.86E-03	1.73E-04	3.56E-05
TOTAL	4.02E-01	5.02E+07	1.58E+05	1.42E+05	1.22E+05	1.12E+05

ACTINIDES

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
BI212	0.0	6.19E-04	1.97E-03	3.95E-03	1.02E-02	1.34E-02
PO212	0.0	1.11E-03	3.52E-03	7.06E-03	1.82E-02	2.39E-02
PO216	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
RN220	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
RA224	0.0	1.72E-03	5.49E-03	1.10E-02	2.84E-02	3.73E-02
TH228	0.0	1.72E-03	5.48E-03	1.10E-02	2.84E-02	3.72E-02
U 232	0.0	8.13E-03	1.90E-02	2.74E-02	4.25E-02	4.75E-02
U 234	0.0	1.81E-02	2.81E-02	3.83E-02	6.86E-02	8.85E-02
U 235	7.78E-02	1.65E-02	1.66E-02	1.66E-02	1.66E-02	1.66E-02
U 236	0.0	3.11E-01	3.11E-01	3.11E-01	3.11E-01	3.11E-01
U 238	3.24E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
NP237	0.0	3.91E-01	4.00E-01	4.00E-01	4.01E-01	4.02E-01
PU236	0.0	1.28E+00	1.01E+00	7.90E-01	3.81E-01	2.34E-01
PU238	0.0	3.37E+03	3.59E+03	3.60E+03	3.53E+03	3.48E+03
PU239	0.0	3.55E+02	3.61E+02	3.61E+02	3.61E+02	3.61E+02
PU240	0.0	5.88E+02	5.88E+02	5.88E+02	5.89E+02	5.90E+02
PU241	0.0	3.62E+00	3.45E+00	3.28E+00	2.84E+00	2.58E+00
PU242	0.0	2.44E+00	2.44E+00	2.44E+00	2.44E+00	2.44E+00
AM241	0.0	1.36E+02	3.62E+02	5.77E+02	1.16E+03	1.50E+03
AM242M	0.0	2.00E-02	1.99E-02	1.98E-02	1.96E-02	1.94E-02
AM243	0.0	2.56E+01	2.56E+01	2.56E+01	2.56E+01	2.56E+01
CM242	0.0	5.34E+04	1.14E+04	2.42E+03	2.63E+01	4.23E+00
CM243	0.0	2.49E+01	2.43E+01	2.37E+01	2.21E+01	2.10E+01
CM244	0.0	3.60E+03	3.47E+03	3.34E+03	2.98E+03	2.76E+03
CM245	0.0	2.73E-01	2.73E-01	2.73E-01	2.72E-01	2.72E-01
CM246	0.0	5.64E-01	5.64E-01	5.64E-01	5.64E-01	5.64E-01
TOTAL	4.02E-01	6.15E+04	1.98E+04	1.09E+04	8.70E+03	8.75E+03

FUEL DECAY BIBLIS-TYPE 3.6% U235

FISSION PRODUCTS

BURNUP=4.00E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	2.05E-02	1.94E-02	1.84E-02	1.55E-02	1.39E-02
KR 85	0.0	1.69E+01	1.58E+01	1.49E+01	1.22E+01	1.08E+01
SR 89	0.0	3.17E+03	2.11E+01	1.41E-01	4.15E-08	1.84E-12
SR 90	0.0	1.04E+02	1.01E+02	9.90E+01	9.22E+01	8.79E+01
Y 90	0.0	5.28E+02	4.84E+02	4.73E+02	4.40E+02	4.20E+02
Y 91	0.0	4.34E+03	5.78E+01	7.64E-01	1.77E-06	3.08E-10
ZR 95	0.0	8.92E+03	1.71E+02	3.27E+00	2.29E-05	8.39E-09
NB 95	0.0	8.44E+03	3.64E+02	7.11E+00	4.82E-05	1.76E-08
NB 95M	0.0	1.77E+01	3.47E-01	6.65E-03	4.66E-08	1.71E-11
TC 99	0.0	7.75E-03	7.79E-03	7.79E-03	7.79E-03	7.79E-03
RU103	0.0	6.23E+03	1.00E+01	1.65E-02	6.66E-11	1.69E-16
RU106	0.0	3.70E+01	1.86E+01	9.35E+00	1.19E+00	3.01E-01
RH103M	0.0	3.84E+02	6.20E-01	1.02E-03	4.12E-12	1.04E-17
RH106	0.0	6.66E+03	3.01E+03	1.51E+03	1.92E+02	4.86E+01
AG110	0.0	1.39E+03	1.50E-01	5.44E-02	2.60E-03	3.43E-04
AG110M	0.0	7.20E+01	2.62E+01	9.50E+00	4.55E-01	6.00E-02
CD113M	0.0	8.07E-02	7.70E-02	7.34E-02	6.37E-02	5.79E-02
CD115M	0.0	3.45E+00	1.18E-02	4.06E-05	1.63E-12	1.92E-17
SN119M	0.0	7.39E-02	2.63E-02	9.37E-03	4.22E-04	5.35E-05
SN123	0.0	7.52E+00	1.06E+00	1.49E-01	4.17E-04	8.28E-06
SB124	0.0	1.31E+01	1.96E-01	2.92E-03	9.72E-09	2.16E-12
SB125	0.0	3.44E+01	2.71E+01	2.11E+01	9.92E+00	6.00E+00
SB126	0.0	1.80E+01	1.73E-03	1.73E-03	1.73E-03	1.73E-03
SB126M	0.0	1.04E+01	8.50E-03	8.50E-03	8.50E-03	8.50E-03
TE123M	0.0	1.84E-02	2.22E-03	2.68E-04	4.62E-07	6.72E-09
TE125M	0.0	1.94E+00	1.77E+00	1.38E+00	6.51E-01	3.94E-01
TE127	0.0	1.48E+02	1.94E+00	1.90E-01	1.79E-04	1.72E-06
TE127M	0.0	7.72E+00	7.87E-01	7.72E-02	7.28E-05	7.00E-07
TE129	0.0	1.32E+03	7.01E-02	3.75E-05	5.74E-15	1.64E-21
TE129M	0.0	9.83E+01	5.28E-02	2.83E-05	4.33E-15	1.24E-21
CS134	0.0	2.26E+03	1.61E+03	1.15E+03	4.21E+02	2.15E+02
CS137	0.0	1.43E+02	1.40E+02	1.36E+02	1.27E+02	1.22E+02
BA137M	0.0	4.81E+02	4.69E+02	4.58E+02	4.27E+02	4.08E+02
CE141	0.0	2.60E+03	1.09E+00	4.51E-04	3.24E-14	5.61E-21
CE144	0.0	8.75E+02	3.59E+02	1.47E+02	1.02E+01	1.72E+00
PR144	0.0	9.81E+03	3.98E+03	1.63E+03	1.13E+02	1.90E+01
PR144M	0.0	5.42E+00	2.22E+00	9.13E-01	6.31E-02	1.06E-02
PM147	0.0	5.85E+01	4.72E+01	3.62E+01	1.64E+01	9.67E+00
PM148	0.0	1.68E+03	3.18E-02	7.00E-05	7.25E-13	3.44E-18
PM148M	0.0	4.27E+02	9.30E-01	2.05E-03	2.12E-11	1.01E-16
SM151	0.0	4.33E-02	4.39E-02	4.36E-02	4.26E-02	4.19E-02
EU152	0.0	3.27E-02	3.11E-02	2.96E-02	2.54E-02	2.29E-02
EU154	0.0	1.45E+02	1.33E+02	1.23E+02	9.67E+01	8.23E+01
EU155	0.0	7.11E+00	6.18E+00	5.38E+00	3.53E+00	2.67E+00
GD153	0.0	2.10E-02	7.37E-03	2.59E-03	1.12E-04	1.39E-05
TB160	0.0	9.42E+00	2.84E-01	8.58E-03	2.36E-07	2.15E-10
TOTAL	0.0	2.17E+06	1.11E+04	5.85E+03	1.96E+03	1.43E+03

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
KR 85	0.0	7.05E-01	6.61E-01	6.20E-01	5.11E-01	4.49E-01
SR 89	0.0	1.86E+02	1.24E+00	8.25E-03	2.43E-09	1.08E-13
SR 90	0.0	5.82E+00	5.69E+00	5.55E+00	5.17E+00	4.93E+00
Y 90	0.0	8.46E+01	7.75E+01	7.57E+01	7.05E+01	6.72E+01
Y 91	0.0	2.84E+02	3.78E+00	4.99E-02	1.15E-07	2.01E-11
ZR 95	0.0	7.73E+03	1.48E+02	2.83E+00	1.99E-05	7.28E-09
NB 95	0.0	7.99E+03	3.44E+02	6.74E+00	4.56E-05	1.67E-08
NB 95M	0.0	5.17E+00	1.02E-01	1.95E-03	1.37E-08	5.01E-12
RU103	0.0	5.40E+03	8.70E+00	1.43E-02	5.78E-11	1.46E-16
RU106	0.0	1.53E-01	7.70E-02	3.87E-02	4.92E-03	1.24E-03
RH103M	0.0	1.69E+01	2.73E-02	4.50E-05	1.81E-13	4.59E-19
RH106	0.0	1.46E+03	6.59E+02	3.31E+02	4.21E+01	1.07E+01
AG110	0.0	1.60E+02	1.72E-02	6.26E-03	3.00E-04	3.95E-05
AG110M	0.0	7.03E+01	2.55E+01	9.27E+00	4.44E-01	5.85E-02
CD113M	0.0	1.45E-03	1.38E-03	1.32E-03	1.14E-03	1.04E-03
CD115M	0.0	3.17E-01	1.09E-03	3.72E-06	1.50E-13	1.76E-18
SN119M	0.0	9.67E-03	3.44E-03	1.22E-03	5.52E-05	6.99E-06
SN123	0.0	5.07E-01	7.14E-02	1.01E-02	2.81E-05	5.58E-07
SB124	0.0	1.10E+01	1.64E-01	2.45E-03	8.16E-09	1.82E-12
SB125	0.0	2.82E+01	2.22E+01	1.73E+01	8.14E+00	4.92E+00
SB126	0.0	1.64E+01	1.57E-03	1.57E-03	1.57E-03	1.57E-03
SB126M	0.0	7.67E+00	6.25E-03	6.25E-03	6.25E-03	6.25E-03
TE123M	0.0	1.11E-02	1.34E-03	1.61E-04	2.78E-07	4.05E-09
TE125M	0.0	4.87E-01	4.44E-01	3.47E-01	1.63E-01	9.88E-02
TE127	0.0	7.69E+00	1.00E-01	9.85E-03	9.29E-06	8.94E-08
TE127M	0.0	8.99E-01	9.17E-02	8.99E-03	8.48E-06	8.16E-08
TE129	0.0	1.91E+02	1.02E-02	5.45E-06	8.34E-16	2.39E-22
TE129M	0.0	1.39E+01	7.45E-03	3.98E-06	6.10E-16	1.75E-22
CS134	0.0	2.05E+03	1.46E+03	1.05E+03	3.82E+02	1.95E+02
CS137	0.0	3.60E+00	3.52E+00	3.44E+00	3.21E+00	3.07E+00
BA137M	0.0	4.32E+02	4.21E+02	4.11E+02	3.84E+02	3.66E+02
CE141	0.0	8.38E+02	3.50E-01	1.46E-04	1.05E-14	1.81E-21
CE144	0.0	1.70E+02	6.98E+01	2.87E+01	1.98E+00	3.34E-01
PR144	0.0	1.16E+03	4.69E+02	1.92E+02	1.33E+01	2.24E+00
PR144M	0.0	1.19E+00	4.89E-01	2.01E-01	1.39E-02	2.34E-03
PM147	0.0	7.63E-01	6.16E-01	4.73E-01	2.14E-01	1.26E-01
PM148	0.0	8.80E+02	1.66E-02	3.66E-05	3.79E-13	1.80E-18
PM148M	0.0	3.97E+02	8.64E-01	1.90E-03	1.97E-11	9.34E-17
EU152	0.0	2.93E-02	2.79E-02	2.65E-02	2.27E-02	2.05E-02
EU154	0.0	1.19E+02	1.09E+02	1.01E+02	7.93E+01	6.75E+01
EU155	0.0	3.54E+00	3.08E+00	2.67E+00	1.76E+00	1.33E+00
GD153	0.0	1.52E-02	5.35E-03	1.88E-03	8.16E-05	1.01E-05
TB160	0.0	7.76E+00	2.34E-01	7.06E-03	1.94E-07	1.77E-10
TOTAL	0.0	6.80E+05	3.84E+03	2.24E+03	9.92E+02	7.24E+02

FUEL DECAY BIBLIS-TYPE 3.6% U235

FISSION PRODUCTS

BURNUP=4.00E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	6.08E+02	5.75E+02	5.43E+02	4.59E+02	4.10E+02
SE 79	0.0	4.46E-01	4.46E-01	4.46E-01	4.46E-01	4.46E-01
KR 85	0.0	1.13E+04	1.06E+04	9.92E+03	8.17E+03	7.18E+03
RB 86	0.0	2.69E+03	3.45E-03	4.43E-09	9.36E-27	1.54E-38
SR 89	0.0	9.17E+05	6.12E+03	4.07E+01	1.20E-05	5.32E-10
SR 90	0.0	8.94E+04	8.73E+04	8.53E+04	7.94E+04	7.57E+04
Y 90	0.0	9.53E+04	8.74E+04	8.53E+04	7.94E+04	7.57E+04
Y 91	0.0	1.21E+06	1.61E+04	2.13E+02	4.91E-04	8.59E-08
ZR 93	0.0	2.15E+00	2.15E+00	2.15E+00	2.15E+00	2.15E+00
ZR 95	0.0	1.76E+06	3.37E+04	6.45E+02	4.52E-03	1.66E-06
NB 93M	0.0	1.58E-01	2.51E-01	3.40E-01	5.82E-01	7.23E-01
NB 95	0.0	1.76E+06	7.58E+04	1.48E+03	1.00E-02	3.68E-06
NB 95M	0.0	1.27E+04	2.50E+02	4.79E+00	3.36E-05	1.23E-08
TC 99	0.0	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01
RU103	0.0	1.86E+06	2.99E+03	4.93E+00	1.98E-08	5.02E-14
RU106	0.0	6.24E+05	3.14E+05	1.58E+05	2.01E+04	5.07E+03
RH103M	0.0	1.67E+06	2.70E+03	4.44E+00	1.79E-08	4.53E-14
RH106	0.0	6.94E+05	3.14E+05	1.58E+05	2.01E+04	5.07E+03
PD107	0.0	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.26E-01
AG110	0.0	1.93E+05	2.08E+01	7.57E+00	3.62E-01	4.78E-02
AG110M	0.0	4.31E+03	1.57E+03	5.69E+02	2.72E+01	3.59E+00
CD113M	0.0	4.79E+01	4.57E+01	4.36E+01	3.78E+01	3.44E+01
CD115M	0.0	9.26E+02	3.18E+00	1.09E-02	4.38E-10	5.15E-15
IN114	0.0	4.11E+00	1.70E-02	1.02E-04	2.23E-11	8.10E-16
IN114M	0.0	2.94E+00	1.77E-02	1.07E-04	2.33E-11	8.47E-16
SN119M	0.0	1.43E+02	5.09E+01	1.81E+01	8.17E-01	1.04E-01
SN121M	0.0	1.94E-01	1.92E-01	1.89E-01	1.81E-01	1.76E-01
SN123	0.0	2.41E+03	3.39E+02	4.78E+01	1.33E-01	2.65E-03
SN126	0.0	6.67E-01	6.67E-01	6.67E-01	6.67E-01	6.67E-01
SB124	0.0	9.89E+02	1.48E+01	2.20E-01	7.32E-07	1.63E-10
SB125	0.0	1.10E+04	8.66E+03	6.74E+03	3.17E+03	1.92E+03
SB126	0.0	9.73E+02	9.34E-02	9.34E-02	9.34E-02	9.34E-02
SB126M	0.0	8.18E+02	6.67E-01	6.67E-01	6.67E-01	6.67E-01
TE123M	0.0	1.27E+01	1.53E+00	1.84E-01	3.17E-04	4.61E-06
TE125M	0.0	2.31E+03	2.11E+03	1.64E+03	7.74E+02	4.69E+02
TE127	0.0	1.10E+05	1.43E+03	1.41E+02	1.33E-01	1.28E-03
TE127M	0.0	1.44E+04	1.46E+03	1.44E+02	1.35E-01	1.30E-03
TE129	0.0	3.69E+05	1.96E+01	1.05E-02	1.61E-12	4.60E-19
TE129M	0.0	5.61E+04	3.01E+01	1.61E-02	2.47E-12	7.07E-19
I 129	0.0	3.75E-02	3.79E-02	3.79E-02	3.79E-02	3.79E-02
CS134	0.0	2.22E+05	1.59E+05	1.13E+05	4.13E+04	2.11E+04
CS135	0.0	3.80E-01	3.81E-01	3.81E-01	3.81E-01	3.81E-01
CS137	0.0	1.29E+05	1.26E+05	1.23E+05	1.15E+05	1.10E+05
BA137M	0.0	1.23E+05	1.19E+05	1.17E+05	1.09E+05	1.04E+05
BA140	0.0	1.86E+06	4.73E-03	1.20E-11	0.0	0.0
LA140	0.0	1.95E+06	5.45E-03	1.38E-11	0.0	0.0
CE141	0.0	1.77E+06	7.41E+02	3.08E-01	2.22E-11	3.83E-18
CE144	0.0	1.32E+06	5.42E+05	2.22E+05	1.54E+04	2.59E+03
PR143	0.0	1.57E+06	1.38E-02	1.09E-10	0.0	0.0
PR144	0.0	1.33E+06	5.42E+05	2.22E+05	1.54E+04	2.59E+03
PR144M	0.0	1.59E+04	6.50E+03	2.67E+03	1.85E+02	3.11E+01
PM147	0.0	1.63E+05	1.32E+05	1.01E+05	4.58E+04	2.70E+04

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PM148	0.0	2.19E+05	4.13E+00	9.09E-03	9.41E-11	4.47E-16
PM148M	0.0	3.37E+04	7.33E+01	1.62E-01	1.67E-09	7.93E-15
SM151	0.0	3.69E+02	3.74E+02	3.71E+02	3.63E+02	3.57E+02
EU152	0.0	4.32E+00	4.10E+00	3.90E+00	3.35E+00	3.02E+00
EU154	0.0	1.62E+04	1.49E+04	1.38E+04	1.08E+04	9.20E+03
EU155	0.0	9.77E+03	8.50E+03	7.39E+03	4.86E+03	3.67E+03
EU156	0.0	3.61E+05	2.09E-02	1.20E-09	0.0	0.0
GD153	0.0	2.43E+01	8.55E+00	3.00E+00	1.30E-01	1.61E-02
TB160	0.0	1.16E+03	3.49E+01	1.05E+00	2.89E-05	2.64E-08
TOTAL	0.0	1.84E+08	2.62E+06	1.43E+06	5.70E+05	4.52E+05

******** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS ********

CONCENTRATIONS ,GRAMS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06	1.56E+06

RADIOACTIVITY ,CURIES

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	4.02E-01	2.36E+08	2.89E+06	1.65E+06	7.28E+05	5.89E+05

THERMAL POWER ,WATTS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	1.04E-02	2.31E+06	1.24E+04	6.62E+03	2.49E+03	1.90E+03

GAMMA POWER ,WATTS

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	7.38E-05	7.25E+05	4.39E+03	2.55E+03	1.18E+03	8.71E+02

ALPHA-ACTIVITY ,CURIES

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
TOTAL	4.02E-01	6.15E+04	1.98E+04	1.09E+04	8.70E+03	8.75E+03

FUEL DECAY BIBLIS-TYPE 3.6% U235

LIGHT ELEMENTS

BURNUP=4.00E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
H 3	0.0	2.29E-03	2.17E-03	2.05E-03	1.73E-03	1.55E-03
C 14	0.0	1.77E-01	1.77E-01	1.77E-01	1.77E-01	1.77E-01
P 33	0.0	5.93E+02	2.38E-02	9.52E-07	6.13E-20	9.85E-29
S 35	0.0	8.59E+00	4.84E-01	2.73E-02	4.88E-06	1.54E-08
CA 45	0.0	2.79E-01	5.90E-02	1.25E-02	1.18E-04	5.29E-06
SC 46	0.0	3.80E+01	1.85E+00	9.05E-02	1.05E-05	2.48E-08
CR 51	0.0	2.39E+05	2.58E+01	2.78E-03	3.49E-15	4.06E-23
MN 54	0.0	8.76E+03	3.90E+03	1.73E+03	1.53E+02	3.02E+01
FE 55	0.0	7.58E+04	5.80E+04	4.45E+04	2.00E+04	1.17E+04
FE 59	0.0	6.28E+03	2.26E+01	8.18E-02	3.84E-09	5.00E-14
CO 58	0.0	3.53E+04	9.87E+02	2.76E+01	6.05E-04	4.74E-07
CO 60	0.0	2.47E+04	2.16E+04	1.90E+04	1.28E+04	9.83E+03
NI 59	0.0	2.12E+01	2.12E+01	2.12E+01	2.12E+01	2.12E+01
NI 63	0.0	3.00E+03	2.98E+03	2.96E+03	2.89E+03	2.85E+03
ZN 65	0.0	3.48E-01	1.23E-01	4.36E-02	1.94E-03	2.43E-04
SR 89	0.0	8.73E+01	5.81E-01	3.87E-03	1.14E-09	5.06E-14
SR 90	0.0	3.32E-03	3.25E-03	3.17E-03	2.95E-03	2.81E-03
Y 90	0.0	3.22E+03	3.25E-03	3.17E-03	2.95E-03	2.81E-03
Y 91	0.0	2.30E+02	3.04E+00	4.01E-02	9.27E-08	1.62E-11
ZR 93	0.0	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01
ZR 95	0.0	7.75E+04	1.48E+03	2.84E+01	1.99E-04	7.29E-08
NB 93M	0.0	1.40E-02	2.38E-02	3.32E-02	5.85E-02	7.33E-02
NB 94	0.0	1.26E+00	1.26E+00	1.26E+00	1.26E+00	1.26E+00
NB 95	0.0	7.59E+04	3.33E+03	6.51E+01	4.39E-04	1.61E-07
MO 93	0.0	8.68E-02	8.67E-02	8.67E-02	8.67E-02	8.66E-02
TC 99	0.0	1.41E-02	1.42E-02	1.42E-02	1.42E-02	1.42E-02
SN119M	0.0	1.21E+04	4.29E+03	1.53E+03	6.88E+01	8.72E+00
SN121M	0.0	1.12E+00	1.11E+00	1.09E+00	1.05E+00	1.02E+00
SN123	0.0	1.08E+00	1.52E-01	2.15E-02	6.05E-05	1.20E-06
SB124	0.0	2.15E+01	3.21E-01	4.80E-03	1.59E-08	3.55E-12
SB125	0.0	9.47E+01	7.38E+01	5.74E+01	2.70E+01	1.64E+01
TE125M	0.0	2.02E+01	1.80E+01	1.40E+01	6.59E+00	3.99E+00
TA182	0.0	2.07E+05	2.29E+04	2.53E+03	3.40E+00	4.16E-02
W 185	0.0	1.31E+00	4.50E-02	1.55E-03	6.29E-08	7.50E-11
TOTAL	6.06E-12	1.30E+06	1.20E+05	7.24E+04	3.59E+04	2.45E+04

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	4.78E-01	2.33E-02	1.14E-03	1.32E-07	3.12E-10
CR 51	0.0	5.12E+01	5.52E-03	5.95E-07	7.46E-19	8.68E-27
MN 54	0.0	4.36E+01	1.94E+01	8.63E+00	7.60E-01	1.50E-01
FE 55	0.0	1.04E+02	7.98E+01	6.11E+01	2.75E+01	1.61E+01
FE 59	0.0	5.86E+01	2.11E-01	7.63E-04	3.58E-11	4.66E-16
CO 58	0.0	2.11E+02	5.91E+00	1.65E-01	3.62E-06	2.84E-09
CO 60	0.0	3.80E+02	3.34E+02	2.92E+02	1.97E+02	1.52E+02
NI 59	0.0	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.35E-01
NI 63	0.0	1.19E+00	1.18E+00	1.17E+00	1.15E+00	1.13E+00
SR 89	0.0	3.02E-01	2.01E-03	1.34E-05	3.95E-12	1.75E-16
Y 91	0.0	8.25E-01	1.09E-02	1.44E-04	3.33E-10	5.82E-14
ZR 95	0.0	3.93E+02	7.52E+00	1.44E-01	1.01E-06	3.69E-10
NB 94	0.0	1.29E-02	1.29E-02	1.29E-02	1.29E-02	1.28E-02
NB 95	0.0	3.64E+02	1.60E+01	3.12E-01	2.11E-06	7.71E-10
SN119M	0.0	6.23E+00	2.22E+00	7.89E-01	3.56E-02	4.51E-03
SN121M	0.0	2.25E-03	2.22E-03	2.19E-03	2.10E-03	2.04E-03
SB124	0.0	2.86E-01	4.27E-03	6.37E-05	2.12E-10	4.72E-14
SB125	0.0	2.96E-01	2.31E-01	1.79E-01	8.45E-02	5.11E-02
TE125M	0.0	1.70E-02	1.51E-02	1.18E-02	5.54E-03	3.36E-03
TA182	0.0	1.84E+03	2.04E+02	2.25E+01	3.02E-02	3.70E-04
TOTAL	1.08E-14	1.02E+04	6.70E+02	3.88E+02	2.27E+02	1.69E+02

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	CHARGE	DISCHARGE	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
SC 46	0.0	4.54E-01	2.22E-02	1.08E-03	1.25E-07	2.96E-10
CR 51	0.0	4.60E+01	4.96E-03	5.35E-07	6.71E-19	7.81E-27
MN 54	0.0	4.34E+01	1.93E+01	8.59E+00	7.56E-01	1.50E-01
CO 58	0.0	2.05E+02	5.74E+00	1.61E-01	3.52E-06	2.75E-09
CO 60	0.0	3.66E+02	3.21E+02	2.81E+02	1.90E+02	1.46E+02
NI 63	0.0	1.19E-03	1.18E-03	1.17E-03	1.14E-03	1.12E-03
ZR 95	0.0	3.41E+02	6.52E+00	1.25E-01	8.75E-07	3.20E-10
NB 94	0.0	1.18E-02	1.18E-02	1.18E-02	1.18E-02	1.18E-02
NB 95	0.0	3.45E+02	1.51E+01	2.96E-01	1.99E-06	7.30E-10
SN119M	0.0	8.14E-01	2.90E-01	1.03E-01	4.65E-03	5.89E-04
SB124	0.0	2.40E-01	3.58E-03	5.35E-05	1.78E-10	3.96E-14
SB125	0.0	2.43E-01	1.89E-01	1.47E-01	6.93E-02	4.19E-02
TE125M	0.0	4.25E-03	3.79E-03	2.95E-03	1.39E-03	8.41E-04
TA182	0.0	1.61E+03	1.78E+02	1.97E+01	2.65E-02	3.24E-04
TOTAL	0.0	7.56E+03	5.46E+02	3.10E+02	1.90E+02	1.46E+02

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

FUEL DECAY BIBLIS-TYPE 3.6% U235
POWER= 40.04 MW, BURNUP= 4.00E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	1.33E+16	8.42E+14	1.04E+14	5.92E+12	4.28E+12
3.75E-01	1.28E+15	1.40E+12	8.48E+11	4.00E+11	2.48E+11
5.75E-01	6.61E+14	1.10E+13	1.26E+12	4.66E+11	2.82E+11
8.50E-01	2.20E+16	3.45E+14	6.82E+13	5.67E+12	1.21E+12
1.25E+00	9.27E+15	2.39E+15	1.49E+15	9.46E+14	7.27E+14
1.75E+00	4.77E+15	1.86E+11	5.14E+09	1.41E+05	2.83E+04
2.25E+00	2.09E+15	9.14E+09	7.45E+09	5.01E+09	3.85E+09
2.75E+00	2.94E+14	2.67E+07	2.30E+07	1.55E+07	1.19E+07
3.50E+00	2.51E+13	0.0	0.0	0.0	0.0
5.00E+00	4.59E+10	0.0	0.0	0.0	0.0
7.00E+00	8.18E+12	0.0	0.0	0.0	0.0
9.50E+00	8.40E+09	0.0	0.0	0.0	0.0
TOTAL	5.37E+16	3.58E+15	1.66E+15	9.58E+14	7.33E+14
MEV/SEC	4.72E+16	3.41E+15	1.94E+15	1.19E+15	9.11E+14

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

FUEL DECAY BIBLIS-TYPE 3.6% U235
POWER= 40.04 MW, BURNUP= 4.00E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.50E-01	2.43E+18	2.06E+16	1.06E+16	3.32E+15	2.58E+15
3.75E-01	8.78E+17	2.07E+15	1.08E+15	3.15E+14	2.30E+14
5.75E-01	1.24E+18	1.73E+16	1.24E+16	6.56E+15	5.18E+15
8.50E-01	1.20E+18	9.62E+15	4.28E+15	1.59E+15	8.85E+14
1.25E+00	6.18E+17	1.13E+15	7.49E+14	3.34E+14	2.40E+14
1.75E+00	2.39E+17	9.20E+13	4.74E+13	1.13E+13	7.14E+12
2.25E+00	1.30E+17	1.70E+14	7.12E+13	5.42E+12	1.01E+12
2.75E+00	4.92E+16	2.63E+12	1.31E+12	1.63E+11	4.10E+10
3.50E+00	2.68E+16	3.26E+11	1.64E+11	2.08E+10	5.27E+09
5.00E+00	1.49E+16	6.86E-05	7.00E-05	7.26E-05	7.35E-05
7.00E+00	3.63E+15	4.48E-06	4.58E-06	4.75E-06	4.81E-06
9.50E+00	1.89E+10	2.90E-07	2.96E-07	3.07E-07	3.11E-07
TOTAL	6.84E+18	5.10E+16	2.93E+16	1.21E+16	9.12E+15
MEV/SEC	4.24E+18	2.40E+16	1.40E+16	6.19E+15	4.52E+15

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN (MEV)	TIME AFTER DISCHARGE INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
1.00E-02	1.27E+18	1.22E+14	6.87E+13	5.60E+13	5.68E+13
2.50E-02	5.68E+16	3.61E+11	5.62E+11	1.11E+12	1.43E+12
3.75E-02	8.21E+16	3.62E+11	2.50E+11	2.56E+11	2.79E+11
5.75E-02	7.44E+16	5.07E+12	8.03E+12	1.61E+13	2.08E+13
8.50E-02	5.68E+17	9.17E+11	9.14E+11	9.06E+11	9.00E+11
1.25E-01	4.47E+17	7.97E+11	7.84E+11	7.63E+11	7.51E+11
2.25E-01	3.17E+17	6.45E+11	6.33E+11	6.09E+11	5.94E+11
3.75E-01	3.51E+16	4.24E+10	4.21E+10	4.18E+10	4.17E+10
5.75E-01	7.22E+15	4.17E+09	3.10E+09	2.99E+09	3.08E+09
8.50E-01	1.59E+16	3.13E+09	2.75E+09	2.59E+09	2.54E+09
1.25E+00	4.10E+15	1.46E+09	1.28E+09	1.15E+09	1.08E+09
1.75E+00	8.07E+10	6.06E+08	5.31E+08	4.85E+08	4.65E+08
2.25E+00	5.22E+08	3.40E+08	2.92E+08	2.53E+08	2.35E+08
2.75E+00	3.25E+08	2.67E+08	3.09E+08	5.05E+08	6.07E+08
3.50E+00	1.81E+08	1.17E+08	1.01E+08	8.75E+07	8.13E+07
5.00E+00	5.64E+07	3.67E+07	3.16E+07	2.73E+07	2.54E+07
7.00E+00	6.46E+06	4.20E+06	3.62E+06	3.13E+06	2.91E+06
9.50E+00	6.46E+05	4.20E+05	3.61E+05	3.13E+05	2.91E+05
TOTAL	2.88E+18	1.30E+14	7.99E+13	7.58E+13	8.16E+13
MEV/SEC	2.33E+17	1.88E+12	1.51E+12	1.85E+12	2.14E+12

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

FUEL DECAY BIBLIS-TYPE 3.6% U235
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
PO212	1.91E+03	6.10E+03	1.22E+04	3.16E+04	4.14E+04
PU236	5.94E+03	4.69E+03	3.67E+03	1.77E+03	1.09E+03
PU238	3.93E+06	4.19E+06	4.21E+06	4.12E+06	4.06E+06
PU239	2.57E+05	2.61E+05	2.61E+05	2.61E+05	2.61E+05
PU240	4.38E+05	4.38E+05	4.39E+05	4.39E+05	4.40E+05
PU241	2.62E+03	2.50E+03	2.38E+03	2.06E+03	1.87E+03
PU242	1.73E+03	1.73E+03	1.73E+03	1.73E+03	1.73E+03
AM241	1.59E+05	4.22E+05	6.72E+05	1.35E+06	1.75E+06
AM243	4.10E+04	4.11E+04	4.11E+04	4.11E+04	4.11E+04
CM242	4.31E+08	9.20E+07	1.95E+07	2.12E+05	3.42E+04
CM243	2.32E+05	2.27E+05	2.21E+05	2.06E+05	1.96E+05
CM244	2.54E+07	2.45E+07	2.36E+07	2.11E+07	1.95E+07
TOTAL	4.62E+08	1.22E+08	4.90E+07	2.77E+07	2.63E+07

FUEL DECAY BIBLIS-TYPE 3.6% U235
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	1.00000 Y	2.00000 Y	5.00000 Y	7.00000 Y
U 238	1.19E+04	1.19E+04	1.19E+04	1.19E+04	1.19E+04
PU238	5.22E+05	5.56E+05	5.59E+05	5.48E+05	5.39E+05
PU240	2.35E+06	2.35E+06	2.35E+06	2.35E+06	2.36E+06
PU242	1.08E+06	1.08E+06	1.08E+06	1.08E+06	1.08E+06
CM242	3.48E+08	7.43E+07	1.58E+07	1.71E+05	2.76E+04
CM244	4.94E+08	4.77E+08	4.59E+08	4.10E+08	3.79E+08
CM246	1.63E+07	1.63E+07	1.63E+07	1.63E+07	1.63E+07
CM248	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04
CF250	2.77E+04	2.64E+04	2.50E+04	2.13E+04	1.92E+04
CF252	1.20E+06	9.20E+05	7.08E+05	3.22E+05	1.90E+05
CF254	3.08E+05	4.70E+03	7.16E+01	2.54E-04	5.89E-08
TOTAL	8.64E+08	5.73E+08	4.96E+08	4.30E+08	4.00E+08
TOTAL	1.33E+09	6.95E+08	5.45E+08	4.58E+08	4.26E+08

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=4.00E+04 MWD

NUCLIDE CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE 4	1.11E+00	1.21E+00	1.49E+00	2.08E+00	2.74E+00	3.75E+00	7.98E+00	1.16E+01	1.67E+01	4.37E+01	1.34E+02
TH232	1.23E-03	1.23E-03	1.24E-03	1.29E-03	1.36E-03	1.50E-03	2.70E-03	2.17E-02	3.00E-01	3.04E+00	2.68E+01
U 234	1.42E-01	1.88E-01	3.37E-01	7.25E-01	1.21E+00	1.80E+00	2.44E+00	2.40E+00	1.98E+00	6.18E-01	5.03E-01
U 235	7.65E+01	7.65E+01	7.66E+01	7.66E+01	7.67E+01	7.69E+01	7.83E+01	1.02E+02	2.66E+02	2.79E+02	2.77E+02
U 236	4.80E+01	4.80E+01	4.81E+01	4.82E+01	4.85E+01	4.91E+01	5.38E+01	8.61E+01	1.06E+02	1.03E+02	7.92E+01
U 238	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.35E+03	9.34E+03
NP236M	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.17E-03	3.15E-03	3.01E-03	1.85E-03	1.47E-05
NP237	5.69E+02	5.71E+02	5.78E+02	5.98E+02	6.30E+02	6.86E+02	9.20E+02	1.01E+03	9.80E+02	7.32E+02	3.97E+01
PU238	2.03E+00	1.99E+00	1.85E+00	1.49E+00	1.05E+00	5.29E-01	5.43E-03	7.30E-21	0.0	0.0	0.0
PU239	5.81E+01	5.81E+01	5.83E+01	5.88E+01	5.94E+01	6.04E+01	6.79E+01	1.09E+02	1.35E+01	1.62E-05	1.08E-05
PU240	2.59E+01	2.95E+01	3.90E+01	5.27E+01	5.80E+01	5.83E+01	5.36E+01	2.06E+01	1.48E-03	7.94E-08	7.91E-08
PU241	1.00E+01	8.67E+00	5.36E+00	1.27E+00	1.17E-01	3.53E-03	2.44E-03	1.17E-03	7.62E-07	0.0	0.0
PU242	6.40E+00	6.40E+00	6.41E+00	6.42E+00	6.45E+00	6.49E+00	6.70E+00	7.73E+00	6.94E+00	1.38E+00	1.38E-07
AM241	4.38E+02	4.37E+02	4.34E+02	4.17E+02	3.86E+02	3.29E+02	9.13E+01	3.52E-02	2.41E-05	0.0	0.0
AM242M	3.99E-01	3.93E-01	3.76E-01	3.28E-01	2.61E-01	1.65E-01	4.31E-03	6.54E-21	0.0	0.0	0.0
AM243	1.28E+02	1.28E+02	1.28E+02	1.28E+02	1.27E+02	1.26E+02	1.17E+02	5.02E+01	1.07E-02	5.05E-06	3.38E-06
CM243	4.08E-01	3.79E-01	2.97E-01	1.43E-01	4.25E-02	3.73E-03	1.33E-11	0.0	0.0	0.0	0.0
CM244	3.41E+01	3.04E+01	2.07E+01	6.58E+00	9.71E-01	2.12E-02	1.07E-15	0.0	0.0	0.0	0.0
CM245	1.59E+00	1.59E+00	1.58E+00	1.58E+00	1.57E+00	1.56E+00	1.46E+00	7.02E-01	4.56E-04	0.0	0.0
CM246	1.83E+00	1.83E+00	1.83E+00	1.82E+00	1.81E+00	1.78E+00	1.59E+00	4.25E-01	7.99E-07	1.42E-29	0.0
CM247	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.13E-02	1.08E-02	7.27E-03
TOTAL	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04

ELEMENT CONCENTRATIONS, GRAMS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
HE	1.11E+00	1.21E+00	1.49E+00	2.08E+00	2.74E+00	3.75E+00	7.98E+00	1.16E+01	1.67E+01	4.37E+01	1.34E+02
TL	4.55E-11	1.58E-11	1.03E-12	4.89E-13	3.12E-13	1.35E-13	4.34E-14	7.67E-13	1.57E-11	3.26E-11	4.64E-12
PB	5.59E-05	8.37E-05	9.93E-05	1.05E-04	1.11E-04	1.18E-04	1.26E-04	1.89E-03	1.99E-01	3.01E+00	1.66E+01
BI	2.85E-09	1.28E-09	7.84E-10	2.55E-09	1.24E-08	9.10E-08	1.32E-05	1.48E-02	4.87E+00	1.96E+02	8.57E+02
PO	1.59E-13	2.03E-13	1.17E-12	7.73E-12	2.79E-11	1.12E-10	4.62E-09	2.23E-07	1.79E-06	9.59E-07	6.99E-07
AT	4.61E-20	4.67E-20	5.53E-20	1.44E-19	5.09E-19	2.10E-18	6.34E-17	6.46E-15	1.55E-13	3.40E-13	1.74E-14
RN	4.05E-11	1.41E-11	1.07E-12	9.22E-13	1.57E-12	4.51E-12	1.35E-10	6.52E-09	5.23E-08	2.80E-08	2.04E-08
FR	6.32E-16	7.41E-16	1.11E-15	2.40E-15	6.00E-15	2.06E-14	5.79E-13	5.87E-11	1.40E-09	3.09E-09	1.58E-10
RA	2.39E-07	9.05E-08	3.02E-08	8.04E-08	2.05E-07	6.86E-07	2.10E-05	1.01E-03	8.15E-03	4.37E-03	3.18E-03
AC	8.15E-09	1.21E-08	2.30E-08	4.16E-08	5.25E-08	5.93E-08	1.01E-07	7.26E-07	1.04E-05	1.78E-05	8.75E-06
TH	1.46E-03	1.42E-03	1.43E-03	1.51E-03	1.72E-03	2.31E-03	9.11E-03	1.35E-01	1.87E+00	5.83E+00	2.71E+01
PA	9.85E-05	9.88E-05	9.98E-05	1.03E-04	1.07E-04	1.16E-04	1.82E-04	8.72E-04	9.37E-03	1.28E-02	1.27E-02
U	9.47E+03	9.47E+03	9.47E+03	9.47E+03	9.47E+03	9.48E+03	9.48E+03	9.54E+03	9.75E+03	9.79E+03	9.70E+03
NP	5.69E+02	5.71E+02	5.78E+02	5.98E+02	6.30E+02	6.86E+02	9.20E+02	1.01E+03	9.80E+02	7.32E+02	3.97E+01
PU	1.02E+02	1.05E+02	1.11E+02	1.21E+02	1.25E+02	1.26E+02	1.28E+02	1.37E+02	2.04E+01	1.38E+00	1.03E-03
AM	5.67E+02	5.66E+02	5.62E+02	5.45E+02	5.14E+02	4.55E+02	2.08E+02	5.02E+01	1.08E-02	5.05E-06	3.38E-06
CM	3.79E+01	3.42E+01	2.45E+01	1.01E+01	4.41E+00	3.38E+00	3.06E+00	1.14E+00	1.23E-02	1.09E-02	7.27E-03
BK	2.17E-08	2.02E-09	7.43E-13	1.62E-19	1.62E-19	1.61E-19	1.56E-19	1.09E-19	3.02E-21	8.19E-37	0.0
CF	7.97E-06	7.66E-06	6.90E-06	5.87E-06	5.18E-06	4.27E-06	9.58E-07	1.26E-10	1.08E-16	2.92E-32	0.0
TOTALS	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.08E+04

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=4.00E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
TL208	1.34E-02	4.65E-03	3.02E-04	1.39E-04	8.61E-05	3.30E-05	1.35E-07	1.21E-07	1.26E-07	1.90E-07	1.06E-06
PB212	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
BI212	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
PO212	2.39E-02	8.30E-03	5.38E-04	2.49E-04	1.54E-04	5.89E-05	2.42E-07	2.15E-07	2.25E-07	3.40E-07	1.89E-06
PO216	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
RN220	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
RA224	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
TH228	3.72E-02	1.29E-02	8.38E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
TH234	3.14E-01	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.15E-03	3.14E-03
PA233	4.02E-01	4.03E-01	4.08E-01	4.22E-01	4.44E-01	4.84E-01	6.49E-01	7.11E-01	6.91E-01	5.16E-01	2.80E-02
PA234M	3.14E-01	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.15E-03	3.14E-03
U 234	8.85E-04	1.18E-03	2.11E-03	4.53E-03	7.59E-03	1.13E-02	1.53E-02	1.50E-02	1.23E-02	3.86E-03	3.14E-03
U 236	3.11E-03	3.11E-03	3.11E-03	3.12E-03	3.14E-03	3.18E-03	3.48E-03	5.58E-03	6.87E-03	6.69E-03	5.13E-03
U 237	2.58E-02	2.24E-02	1.38E-02	3.27E-03	3.01E-04	9.11E-06	6.29E-06	3.02E-06	1.96E-09	0.0	0.0
U 238	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.15E-03	3.14E-03
NP237	4.02E-01	4.03E-01	4.08E-01	4.22E-01	4.44E-01	4.84E-01	6.49E-01	7.11E-01	6.91E-01	5.16E-01	2.80E-02
NP238	1.94E-02	1.91E-02	1.83E-02	1.59E-02	1.27E-02	8.04E-03	2.10E-04	3.18E-22	0.0	0.0	0.0
NP239	2.56E+01	2.56E+01	2.56E+01	2.55E+01	2.54E+01	2.51E+01	2.33E+01	1.00E+01	2.14E-03	1.01E-06	6.75E-07
PU236	2.34E-03	1.13E-03	9.98E-05	4.03E-07	3.36E-07	3.36E-07	3.35E-07	3.34E-07	3.18E-07	1.96E-07	1.55E-09
PU238	3.48E+01	3.40E+01	3.17E+01	2.56E+01	1.80E+01	9.07E+00	9.30E-02	1.25E-19	0.0	0.0	0.0
PU239	3.61E+00	3.61E+00	3.63E+00	3.65E+00	3.69E+00	3.76E+00	4.22E+00	6.78E+00	8.39E-01	1.01E-06	6.75E-07
PU240	5.90E+00	6.72E+00	8.88E+00	1.20E+01	1.32E+01	1.33E+01	1.22E+01	4.70E+00	3.38E-04	1.81E-08	1.80E-08
PU241	1.03E+03	8.94E+02	5.52E+02	1.31E+02	1.20E+01	3.64E-01	2.52E-01	1.21E-01	7.85E-05	0.0	0.0
PU242	2.44E-02	2.45E-02	2.45E-02	2.45E-02	2.46E-02	2.48E-02	2.56E-02	2.95E-02	2.65E-02	5.29E-03	5.28E-10
AM241	1.50E+03	1.50E+03	1.49E+03	1.43E+03	1.33E+03	1.13E+03	3.14E+02	1.21E-01	8.27E-05	0.0	0.0
AM242M	3.88E+00	3.82E+00	3.65E+00	3.19E+00	2.54E+00	1.61E+00	4.19E-02	6.36E-20	0.0	0.0	0.0
AM242	3.86E+00	3.81E+00	3.64E+00	3.17E+00	2.53E+00	1.60E+00	4.17E-02	6.33E-20	0.0	0.0	0.0
AM243	2.56E+01	2.56E+01	2.56E+01	2.55E+01	2.54E+01	2.51E+01	2.33E+01	1.00E+01	2.14E-03	1.01E-06	6.75E-07
CM242	4.23E+00	3.16E+00	3.01E+00	2.62E+00	2.09E+00	1.32E+00	3.45E-02	5.25E-20	0.0	0.0	0.0
CM243	2.11E+01	1.96E+01	1.53E+01	7.40E+00	2.19E+00	1.93E-01	6.88E-10	0.0	0.0	0.0	0.0
CM244	2.76E+03	2.46E+03	1.68E+03	5.33E+02	7.86E+01	1.71E+00	8.68E-14	0.0	0.0	0.0	0.0
CM245	2.72E-01	2.72E-01	2.72E-01	2.71E-01	2.70E-01	2.68E-01	2.51E-01	1.21E-01	7.84E-05	0.0	0.0
CM246	5.64E-01	5.64E-01	5.63E-01	5.60E-01	5.56E-01	5.48E-01	4.88E-01	1.30E-01	2.46E-07	4.38E-30	0.0
TOTAL	5.43E+03	4.98E+03	3.84E+03	2.21E+03	1.51E+03	1.22E+03	3.79E+02	3.36E+01	4.60E+00	6.04E+00	3.64E-01

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP239	2.59E-02	2.59E-02	2.59E-02	2.58E-02	2.57E-02	2.55E-02	2.36E-02	1.01E-02	2.17E-06	1.02E-09	6.83E-10
AM241	2.17E-01	2.17E-01	2.15E-01	2.07E-01	1.92E-01	1.63E-01	4.53E-02	1.75E-05	1.20E-08	0.0	0.0
AM243	8.37E-03	8.36E-03	8.36E-03	8.33E-03	8.29E-03	8.22E-03	7.62E-03	3.27E-03	7.00E-07	3.29E-10	2.21E-10
CM243	1.66E-02	1.54E-02	1.21E-02	5.83E-03	1.73E-03	1.52E-04	5.42E-13	0.0	0.0	0.0	0.0
CM244	2.50E-02	2.23E-02	1.52E-02	4.82E-03	7.12E-04	1.55E-05	7.87E-19	0.0	0.0	0.0	0.0
TOTAL	2.96E-01	2.91E-01	2.79E-01	2.54E-01	2.30E-01	1.98E-01	7.77E-02	1.46E-02	1.59E-03	1.96E-03	1.40E-04

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE ACTINIDES

BURNUP=4.00E+04 MWD

NUCLIDE ALPHA-ACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
BI212	1.34E-02	4.65E-03	3.02E-04	1.39E-04	8.61E-05	3.30E-05	1.35E-07	1.21E-07	1.26E-07	1.90E-07	1.06E-06
PO212	2.39E-02	8.30E-03	5.38E-04	2.49E-04	1.54E-04	5.89E-05	2.42E-07	2.15E-07	2.25E-07	3.40E-07	1.89E-06
PO216	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
RN220	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
RA224	3.73E-02	1.30E-02	8.40E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
TH228	3.72E-02	1.29E-02	8.38E-04	3.88E-04	2.40E-04	9.18E-05	3.77E-07	3.36E-07	3.51E-07	5.30E-07	2.94E-06
U 234	8.85E-04	1.18E-03	2.11E-03	4.53E-03	7.59E-03	1.13E-02	1.53E-02	1.50E-02	1.23E-02	3.86E-03	3.14E-03
U 236	3.11E-03	3.11E-03	3.11E-03	3.12E-03	3.14E-03	3.18E-03	3.48E-03	5.58E-03	6.87E-03	6.69E-03	5.13E-03
U 238	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.14E-03	3.15E-03	3.14E-03
NP237	4.02E-01	4.03E-01	4.08E-01	4.22E-01	4.44E-01	4.84E-01	6.49E-01	7.11E-01	6.91E-01	5.16E-01	2.80E-02
PU236	2.34E-03	1.13E-03	9.98E-05	4.03E-07	3.36E-07	3.36E-07	3.35E-07	3.34E-07	3.18E-07	1.96E-07	1.55E-09
PU238	3.48E+01	3.40E+01	3.17E+01	2.56E+01	1.80E+01	9.07E+00	9.30E-02	1.25E-19	0.0	0.0	0.0
PU239	3.61E+00	3.61E+00	3.63E+00	3.65E+00	3.69E+00	3.76E+00	4.22E+00	6.78E+00	8.39E-01	1.01E-06	6.75E-07
PU240	5.90E+00	6.72E+00	8.88E+00	1.20E+01	1.32E+01	1.33E+01	1.22E+01	4.70E+00	3.38E-04	1.81E-08	1.80E-08
PU241	2.58E-02	2.23E-02	1.38E-02	3.27E-03	3.00E-04	9.11E-06	6.29E-06	3.02E-06	1.96E-09	0.0	0.0
PU242	2.44E-02	2.45E-02	2.45E-02	2.45E-02	2.46E-02	2.48E-02	2.56E-02	2.95E-02	2.65E-02	5.29E-03	5.28E-10
AM241	1.50E+03	1.50E+03	1.49E+03	1.43E+03	1.33E+03	1.13E+03	3.14E+02	1.21E-01	8.27E-05	0.0	0.0
AM242M	1.94E-02	1.91E-02	1.83E-02	1.59E-02	1.27E-02	8.04E-03	2.10E-04	3.18E-22	0.0	0.0	0.0
AM243	2.56E+01	2.56E+01	2.56E+01	2.55E+01	2.54E+01	2.51E+01	2.33E+01	1.00E+01	2.14E-03	1.01E-06	6.75E-07
CM242	4.23E+00	3.16E+00	3.01E+00	2.62E+00	2.09E+00	1.32E+00	3.45E-02	5.25E-20	0.0	0.0	0.0
CM243	2.10E+01	1.95E+01	1.53E+01	7.38E+00	2.19E+00	1.92E-01	6.86E-10	0.0	0.0	0.0	0.0
CM244	2.76E+03	2.46E+03	1.68E+03	5.33E+02	7.86E+01	1.71E+00	8.68E-14	0.0	0.0	0.0	0.0
CM245	2.72E-01	2.72E-01	2.72E-01	2.71E-01	2.70E-01	2.68E-01	2.51E-01	1.21E-01	7.84E-05	0.0	0.0
CM246	5.64E-01	5.64E-01	5.63E-01	5.60E-01	5.56E-01	5.48E-01	4.88E-01	1.30E-01	2.46E-07	4.38E-30	0.0
TOTAL	4.36E+03	4.06E+03	3.26E+03	2.04E+03	1.47E+03	1.19E+03	3.55E+02	2.27E+01	3.12E+00	3.85E+00	2.30E-01

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
NP237	1.18E-02	1.18E-02	1.20E-02	1.24E-02	1.31E-02	1.42E-02	1.91E-02	2.09E-02	2.03E-02	1.52E-02	8.22E-04
NP239	6.18E-02	6.18E-02	6.18E-02	6.16E-02	6.13E-02	6.07E-02	5.63E-02	2.42E-02	5.17E-06	2.43E-09	1.63E-09
PU238	1.15E+00	1.13E+00	1.05E+00	8.48E-01	5.97E-01	3.01E-01	3.08E-03	4.14E-21	0.0	0.0	0.0
PU239	1.12E-01	1.12E-01	1.13E-01	1.14E-01	1.15E-01	1.17E-01	1.31E-01	2.11E-01	2.61E-02	3.13E-08	2.10E-08
PU240	1.84E-01	2.09E-01	2.77E-01	3.75E-01	4.12E-01	4.14E-01	3.80E-01	1.47E-01	1.05E-05	5.64E-10	5.62E-10
PU241	3.18E-02	2.75E-02	1.70E-02	4.03E-03	3.70E-04	1.12E-05	7.76E-06	3.72E-06	2.42E-09	0.0	0.0
AM241	5.03E+01	5.02E+01	4.98E+01	4.79E+01	4.43E+01	3.78E+01	1.05E+01	4.04E-03	2.77E-06	0.0	0.0
AM242M	1.53E-03	1.51E-03	1.44E-03	1.26E-03	1.00E-03	6.35E-04	1.66E-05	2.51E-23	0.0	0.0	0.0
AM242	4.48E-03	4.42E-03	4.22E-03	3.68E-03	2.93E-03	1.86E-03	4.85E-05	7.36E-23	0.0	0.0	0.0
AM243	8.25E-01	8.25E-01	8.24E-01	8.21E-01	8.18E-01	8.10E-01	7.51E-01	3.23E-01	6.90E-05	3.24E-08	2.17E-08
CM242	1.56E-01	1.17E-01	1.11E-01	9.66E-02	7.69E-02	4.88E-02	1.27E-03	1.93E-21	0.0	0.0	0.0
CM243	7.72E-01	7.18E-01	5.63E-01	2.71E-01	8.05E-02	7.07E-03	2.52E-11	0.0	0.0	0.0	0.0
CM244	9.66E+01	8.61E+01	5.87E+01	1.86E+01	2.75E+00	5.99E-02	3.04E-15	0.0	0.0	0.0	0.0
CM245	9.08E-03	9.08E-03	9.07E-03	9.05E-03	9.01E-03	8.94E-03	8.37E-03	4.02E-03	2.61E-06	0.0	0.0
CM246	1.85E-02	1.85E-02	1.84E-02	1.83E-02	1.82E-02	1.79E-02	1.60E-02	4.27E-03	8.04E-09	1.43E-31	0.0
TOTAL	1.50E+02	1.40E+02	1.12E+02	6.92E+01	4.93E+01	3.96E+01	1.19E+01	7.44E-01	1.09E-01	1.45E-01	8.56E-03

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=4.00E+04 MWD

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	4.10E+02	3.47E+02	1.98E+02	3.67E+01	2.22E+00	8.09E-03	2.57E-22	0.0	0.0	0.0	0.0
SE 79	4.46E-01	4.46E-01	4.46E-01	4.46E-01	4.46E-01	4.45E-01	4.41E-01	4.01E-01	1.54E-01	1.04E-05	0.0
KR 85	7.18E+03	5.91E+03	3.10E+03	4.45E+02	1.76E+01	2.69E-02	0.0	0.0	0.0	0.0	0.0
SR 90	7.57E+04	7.05E+04	5.56E+04	2.72E+04	8.28E+03	7.67E+02	4.14E-06	0.0	0.0	0.0	0.0
Y 90	7.57E+04	7.05E+04	5.56E+04	2.72E+04	8.28E+03	7.67E+02	4.14E-06	0.0	0.0	0.0	0.0
ZR 93	2.15E+00	2.15E+00	2.15E+00	2.15E+00	2.15E+00	2.15E+00	2.15E+00	2.14E+00	2.06E+00	1.37E+00	2.32E-02
NB 93M	7.23E-01	9.10E-01	1.36E+00	1.90E+00	2.03E+00	2.04E+00	2.04E+00	2.03E+00	1.95E+00	1.30E+00	2.20E-02
TC 99	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.50E+01	1.12E+01	6.00E-01	1.15E-13
RU106	5.07E+03	6.45E+02	6.74E-01	7.44E-10	8.77E-25	0.0	0.0	0.0	0.0	0.0	0.0
RH106	5.07E+03	6.45E+02	6.74E-01	7.44E-10	8.77E-25	0.0	0.0	0.0	0.0	0.0	0.0
PD107	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.26E-01	1.25E-01	1.13E-01	4.33E-02
AG110	4.78E-02	2.29E-03	9.13E-08	5.80E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	3.59E+00	1.72E-01	6.87E-06	4.36E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	3.44E+01	2.98E+01	1.85E+01	4.46E+00	4.15E-01	3.59E-03	1.13E-19	0.0	0.0	0.0	0.0
SN119M	1.04E-01	4.67E-03	1.52E-07	5.29E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	1.76E-01	1.69E-01	1.47E-01	9.71E-02	4.85E-02	1.21E-02	1.84E-07	0.0	0.0	0.0	0.0
SN126	6.67E-01	6.67E-01	6.67E-01	6.67E-01	6.67E-01	6.66E-01	6.63E-01	6.23E-01	3.34E-01	6.63E-04	0.0
SB125	1.92E+03	9.04E+02	7.35E+01	3.94E-02	1.39E-07	1.74E-18	0.0	0.0	0.0	0.0	0.0
SB126	9.34E-02	9.34E-02	9.34E-02	9.34E-02	9.33E-02	9.33E-02	9.28E-02	8.72E-02	4.67E-02	9.28E-05	0.0
SB126M	6.67E-01	6.67E-01	6.67E-01	6.67E-01	6.67E-01	6.66E-01	6.63E-01	6.23E-01	3.34E-01	6.63E-04	0.0
TE125M	4.69E+02	2.21E+02	1.79E+01	9.61E-03	3.40E-08	4.25E-19	0.0	0.0	0.0	0.0	0.0
I 129	3.79E-02	3.79E-02	3.79E-02	3.79E-02	3.79E-02	3.79E-02	3.79E-02	3.78E-02	3.77E-02	3.62E-02	2.43E-02
CS134	2.11E+04	7.70E+03	2.67E+02	1.12E-02	5.63E-10	1.43E-24	0.0	0.0	0.0	0.0	0.0
CS135	3.81E-01	3.81E-01	3.81E-01	3.81E-01	3.81E-01	3.81E-01	3.81E-01	3.80E-01	3.69E-01	2.82E-01	1.87E-02
CS137	1.10E+05	1.03E+05	8.14E+04	4.07E+04	1.28E+04	1.27E+03	1.20E-05	0.0	0.0	0.0	0.0
BA137M	1.04E+05	9.70E+04	7.70E+04	3.85E+04	1.21E+04	1.20E+03	1.13E-05	0.0	0.0	0.0	0.0
CE144	2.59E+03	1.79E+02	2.43E-02	6.10E-14	2.82E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144	2.59E+03	1.79E+02	2.43E-02	6.10E-14	2.82E-33	0.0	0.0	0.0	0.0	0.0	0.0
PR144M	3.11E+01	2.15E+00	2.92E-04	7.32E-16	3.38E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	2.70E+04	1.22E+04	8.70E+02	3.15E-01	5.78E-07	1.95E-18	0.0	0.0	0.0	0.0	0.0
SM151	3.57E+02	3.49E+02	3.23E+02	2.56E+02	1.75E+02	8.08E+01	1.73E-01	0.0	0.0	0.0	0.0
EU152	3.02E+00	2.59E+00	1.56E+00	3.38E-01	2.64E-02	1.62E-04	3.20E-22	0.0	0.0	0.0	0.0
EU154	9.20E+03	7.23E+03	3.23E+03	2.88E+02	5.12E+00	1.62E-03	0.0	0.0	0.0	0.0	0.0
EU155	3.67E+03	2.42E+03	5.97E+02	9.03E+00	8.34E-03	7.12E-09	0.0	0.0	0.0	0.0	0.0
TOTAL	4.52E+05	3.80E+05	2.78E+05	1.35E+05	4.17E+04	4.11E+03	2.22E+01	2.15E+01	1.66E+01	3.70E+00	1.32E-01

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE FISSION PRODUCTS

BURNUP=4.00E+04 MWD

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	1.39E-02	1.17E-02	6.68E-03	1.24E-03	7.50E-05	2.73E-07	8.70E-27	0.0	0.0	0.0	0.0
KR 85	1.08E+01	8.86E+00	4.64E+00	6.67E-01	2.63E-02	4.02E-05	0.0	0.0	0.0	0.0	0.0
SR 90	8.79E+01	8.18E+01	6.45E+01	3.16E+01	9.61E+00	8.90E-01	4.80E-09	0.0	0.0	0.0	0.0
Y 90	4.20E+02	3.91E+02	3.08E+02	1.51E+02	4.59E+01	4.25E+00	2.29E-08	0.0	0.0	0.0	0.0
TC 99	7.79E-03	7.79E-03	7.79E-03	7.79E-03	7.78E-03	7.78E-03	7.76E-03	7.54E-03	5.62E-03	3.01E-04	5.79E-17
RU106	3.01E-01	3.82E-02	3.99E-05	4.41E-14	5.20E-29	0.0	0.0	0.0	0.0	0.0	0.0
RH106	4.86E+01	6.18E+00	6.46E-03	7.13E-12	8.41E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	6.00E-02	2.87E-03	1.15E-07	7.28E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD113M	5.79E-02	5.02E-02	3.12E-02	7.51E-03	6.98E-04	6.04E-06	1.90E-22	0.0	0.0	0.0	0.0
SB125	6.00E+00	2.83E+00	2.30E-01	1.23E-04	4.35E-10	5.44E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.73E-03	1.73E-03	1.73E-03	1.73E-03	1.72E-03	1.72E-03	1.71E-03	1.61E-03	8.63E-04	1.71E-06	0.0
SB126M	8.50E-03	8.50E-03	8.49E-03	8.49E-03	8.49E-03	8.48E-03	8.44E-03	7.93E-03	4.25E-03	8.44E-06	0.0
TE125M	3.94E-01	1.86E-01	1.51E-02	8.08E-06	2.86E-11	3.57E-22	0.0	0.0	0.0	0.0	0.0
CS134	2.15E+02	7.84E+01	2.72E+00	1.14E-04	5.73E-12	1.45E-26	0.0	0.0	0.0	0.0	0.0
CS137	1.22E+02	1.13E+02	9.00E+01	4.50E+01	1.42E+01	1.41E+00	1.33E-08	0.0	0.0	0.0	0.0
BA137M	4.08E+02	3.81E+02	3.02E+02	1.51E+02	4.76E+01	4.73E+00	4.45E-08	0.0	0.0	0.0	0.0
CE144	1.72E+00	1.19E-01	1.61E-05	4.04E-17	1.87E-36	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.90E+01	1.32E+00	1.79E-04	4.48E-16	2.07E-35	0.0	0.0	0.0	0.0	0.0	0.0
PM147	9.67E+00	4.38E+00	3.12E-01	1.13E-04	2.07E-10	7.00E-22	0.0	0.0	0.0	0.0	0.0
SM151	4.19E-02	4.10E-02	3.79E-02	3.01E-02	2.05E-02	9.48E-03	2.03E-05	0.0	0.0	0.0	0.0
EU152	2.29E-02	1.97E-02	1.18E-02	2.56E-03	2.01E-04	1.23E-06	2.43E-24	0.0	0.0	0.0	0.0
EU154	8.23E+01	6.46E+01	2.89E+01	2.57E+00	4.58E-02	1.45E-05	0.0	0.0	0.0	0.0	0.0
EU155	2.67E+00	1.76E+00	4.35E-01	6.57E-03	6.07E-06	5.18E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	1.43E+03	1.14E+03	8.02E+02	3.82E+02	1.17E+02	1.13E+01	1.96E-02	1.87E-02	1.19E-02	8.19E-04	2.68E-05

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
KR 85	4.49E-01	3.70E-01	1.94E-01	2.78E-02	1.10E-03	1.68E-06	0.0	0.0	0.0	0.0	0.0
SR 90	4.93E+00	4.59E+00	3.62E+00	1.77E+00	5.39E-01	4.99E-02	2.70E-10	0.0	0.0	0.0	0.0
Y 90	6.72E+01	6.26E+01	4.93E+01	2.42E+01	7.35E+00	6.81E-01	3.67E-09	0.0	0.0	0.0	0.0
RH106	1.07E+01	1.35E+00	1.42E-03	1.56E-12	1.84E-27	0.0	0.0	0.0	0.0	0.0	0.0
AG110M	5.85E-02	2.80E-03	1.12E-07	7.10E-21	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SB125	4.92E+00	2.32E+00	1.88E-01	1.01E-04	3.57E-10	4.46E-21	0.0	0.0	0.0	0.0	0.0
SB126	1.57E-03	1.57E-03	1.57E-03	1.57E-03	1.57E-03	1.57E-03	1.56E-03	1.47E-03	7.87E-04	1.56E-06	0.0
SB126M	6.25E-03	6.25E-03	6.25E-03	6.25E-03	6.25E-03	6.25E-03	6.21E-03	5.84E-03	3.13E-03	6.21E-06	0.0
TE125M	9.88E-02	4.65E-02	3.78E-03	2.02E-06	7.16E-12	8.95E-23	0.0	0.0	0.0	0.0	0.0
CS134	1.95E+02	7.11E+01	2.47E+00	1.03E-04	5.20E-12	1.32E-26	0.0	0.0	0.0	0.0	0.0
CS137	3.07E+00	2.86E+00	2.27E+00	1.14E+00	3.58E-01	3.55E-02	3.35E-10	0.0	0.0	0.0	0.0
BA137M	3.66E+02	3.42E+02	2.71E+02	1.36E+02	4.28E+01	4.24E+00	4.00E-08	0.0	0.0	0.0	0.0
CE144	3.34E-01	2.31E-02	3.14E-06	7.86E-18	3.63E-37	0.0	0.0	0.0	0.0	0.0	0.0
PR144	2.24E+00	1.55E-01	2.11E-05	5.28E-17	2.44E-36	0.0	0.0	0.0	0.0	0.0	0.0
PM147	1.26E-01	5.71E-02	4.07E-03	1.47E-06	2.70E-12	9.13E-24	0.0	0.0	0.0	0.0	0.0
EU152	2.05E-02	1.76E-02	1.06E-02	2.30E-03	1.80E-04	1.10E-06	2.18E-24	0.0	0.0	0.0	0.0
EU154	6.75E+01	5.30E+01	2.37E+01	2.11E+00	3.75E-02	1.19E-05	0.0	0.0	0.0	0.0	0.0
EU155	1.33E+00	8.74E-01	2.16E-01	3.27E-03	3.02E-06	2.58E-12	0.0	0.0	0.0	0.0	0.0
TOTAL	7.24E+02	5.41E+02	3.53E+02	1.65E+02	5.11E+01	5.02E+00	8.29E-03	7.80E-03	4.24E-03	3.93E-05	3.95E-06

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE

LIGHT ELEMENTS

BURNUP=4.00E+04 MWD

NUCLIDE GAMMA POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.50E-01	1.32E-02	4.00E-06	1.12E-16	2.89E-34	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	1.46E+02	9.82E+01	2.64E+01	5.10E-01	7.49E-04	1.46E-09	0.0	0.0	0.0	0.0	0.0
NI 63	1.12E-03	1.10E-03	1.02E-03	8.13E-04	5.58E-04	2.63E-04	6.39E-07	0.0	0.0	0.0	0.0
NB 94	1.18E-02	1.18E-02	1.18E-02	1.18E-02	1.17E-02	1.17E-02	1.14E-02	8.37E-03	3.88E-04	1.76E-17	0.0
SB125	4.19E-02	1.97E-02	1.60E-03	8.59E-07	3.04E-12	3.80E-23	0.0	0.0	0.0	0.0	0.0
TOTAL	1.46E+02	9.83E+01	2.64E+01	5.23E-01	1.31E-02	1.20E-02	1.14E-02	8.38E-03	3.90E-04	1.58E-06	2.67E-08

NUCLIDE THERMAL POWER, WATTS
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
MN 54	1.50E-01	1.32E-02	4.02E-06	1.13E-16	2.91E-34	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	1.61E+01	7.25E+00	5.04E-01	1.70E-04	2.77E-10	7.36E-22	0.0	0.0	0.0	0.0	0.0
CO 60	1.52E+02	1.02E+02	2.74E+01	5.31E-01	7.79E-04	1.51E-09	0.0	0.0	0.0	0.0	0.0
NI 59	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.34E-01	1.24E-01	5.67E-02	2.33E-05	0.0
NI 63	1.13E+00	1.11E+00	1.03E+00	8.18E-01	5.61E-01	2.64E-01	6.43E-04	0.0	0.0	0.0	0.0
NB 94	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.24E-02	9.14E-03	4.23E-04	1.92E-17	0.0
SN121M	2.04E-03	1.96E-03	1.71E-03	1.13E-03	5.62E-04	1.41E-04	2.14E-09	0.0	0.0	0.0	0.0
SB125	5.11E-02	2.41E-02	1.96E-03	1.05E-06	3.71E-12	4.63E-23	0.0	0.0	0.0	0.0	0.0
TE125M	3.36E-03	1.58E-03	1.28E-04	6.88E-08	2.43E-13	3.04E-24	0.0	0.0	0.0	0.0	0.0
TOTAL	1.69E+02	1.11E+02	2.91E+01	1.50E+00	7.10E-01	4.12E-01	1.47E-01	1.33E-01	5.72E-02	6.40E-05	6.84E-07

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
H 3	1.55E-03	1.31E-03	7.47E-04	1.39E-04	8.38E-06	3.05E-08	9.71E-28	0.0	0.0	0.0	0.0
C 14	1.77E-01	1.77E-01	1.77E-01	1.76E-01	1.75E-01	1.73E-01	1.57E-01	5.29E-02	9.90E-07	0.0	0.0
MN 54	3.02E+01	2.66E+00	8.08E-04	2.26E-14	5.84E-32	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	1.17E+04	5.27E+03	3.67E+02	1.23E-01	2.01E-07	5.35E-19	0.0	0.0	0.0	0.0	0.0
CO 60	9.83E+03	6.62E+03	1.78E+03	3.44E+01	5.05E-02	9.82E-08	0.0	0.0	0.0	0.0	0.0
NI 59	2.12E+01	2.12E+01	2.12E+01	2.12E+01	2.12E+01	2.12E+01	2.10E+01	1.94E+01	8.91E+00	3.67E-03	0.0
NI 63	2.85E+03	2.78E+03	2.58E+03	2.06E+03	1.41E+03	6.65E+02	1.62E+00	0.0	0.0	0.0	0.0
SR 90	2.81E-03	2.62E-03	2.07E-03	1.01E-03	3.08E-04	2.85E-05	1.54E-13	0.0	0.0	0.0	0.0
Y 90	2.81E-03	2.62E-03	2.07E-03	1.01E-03	3.08E-04	2.85E-05	1.54E-13	0.0	0.0	0.0	0.0
ZR 93	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.22E-01	2.13E-01	1.42E-01	2.41E-03
NB 93M	7.33E-02	9.30E-02	1.40E-01	1.96E-01	2.11E-01	2.12E-01	2.12E-01	2.11E-01	2.02E-01	1.35E-01	2.28E-03
NB 94	1.26E+00	1.26E+00	1.26E+00	1.26E+00	1.26E+00	1.25E+00	1.22E+00	8.97E-01	4.15E-02	1.88E-15	0.0
MO 93	8.66E-02	8.66E-02	8.64E-02	8.59E-02	8.51E-02	8.34E-02	7.12E-02	1.20E-02	2.17E-10	0.0	0.0
TC 99	1.42E-02	1.42E-02	1.42E-02	1.42E-02	1.42E-02	1.42E-02	1.41E-02	1.37E-02	1.02E-02	5.48E-04	1.05E-16
SN119M	8.72E+00	3.93E-01	1.28E-05	4.46E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	1.02E+00	9.78E-01	8.51E-01	5.62E-01	2.81E-01	7.01E-02	1.07E-06	0.0	0.0	0.0	0.0
SB125	1.64E+01	7.70E+00	6.26E-01	3.35E-04	1.19E-09	1.48E-20	0.0	0.0	0.0	0.0	0.0
TE125M	3.99E+00	1.88E+00	1.53E-01	8.18E-05	2.89E-10	3.62E-21	0.0	0.0	0.0	0.0	0.0
TOTAL	2.45E+04	1.47E+04	4.75E+03	2.12E+03	1.44E+03	6.89E+02	2.45E+01	2.08E+01	9.38E+00	2.81E-01	4.69E-03

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE

BURNUP=4.00E+04 MWD

********** LIGHT ELEMENTS + ACTINIDES + FISSION PRODUCTS **********

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
			CONCENTRATIONS ,GRAMS								
TOTAL	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05	6.13E+05
			RADIOACTIVITY ,CURIES								
TOTAL	4.82E+05	3.99E+05	2.87E+05	1.39E+05	4.47E+04	6.02E+03	4.26E+02	7.59E+01	3.06E+01	1.00E+01	5.00E-01
			THERMAL POWER ,WATTS								
TOTAL	1.75E+03	1.39E+03	9.43E+02	4.53E+02	1.67E+02	5.14E+01	1.20E+01	8.96E-01	1.78E-01	1.46E-01	8.59E-03
			GAMMA POWER ,WATTS								
TOTAL	8.70E+02	6.40E+02	3.80E+02	1.66E+02	5.13E+01	5.23E+00	9.74E-02	3.07E-02	6.22E-03	2.00E-03	1.44E-04
			ALPHA-ACTIVITY ,CURIES								
TOTAL	4.36E+03	4.06E+03	3.26E+03	2.04E+03	1.47E+03	1.19E+03	3.55E+02	2.27E+01	3.12E+00	3.85E+00	2.30E-01

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR ACTINIDES

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 40.04 MW, BURNUP= 4.00E+04 MWD

ACTINIDE PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN				TIME AFTER DISCHARGE							
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.00E-02	2.99E+13	2.81E+13	2.35E+13	1.64E+13	1.28E+13	1.06E+13	3.69E+12	5.16E+11	5.98E+10	7.04E+10	4.09E+09
2.50E-02	1.43E+12	1.42E+12	1.41E+12	1.36E+12	1.26E+12	1.07E+12	3.07E+11	7.87E+09	4.01E+09	4.13E+09	2.78E+08
3.75E-02	2.20E+11	2.14E+11	2.04E+11	1.87E+11	1.73E+11	1.55E+11	8.73E+10	2.74E+10	4.45E+09	9.54E+09	5.23E+08
5.75E-02	2.07E+13	2.07E+13	2.05E+13	1.97E+13	1.83E+13	1.56E+13	4.33E+12	1.04E+10	1.17E+09	1.97E+09	1.64E+08
8.50E-02	8.72E+11	8.60E+11	8.32E+11	7.80E+11	7.42E+11	7.20E+11	6.62E+11	2.92E+11	2.08E+10	2.61E+10	1.44E+09
1.25E-01	7.19E+11	7.01E+11	6.53E+11	5.63E+11	5.01E+11	4.70E+11	4.25E+11	1.83E+11	3.68E+09	4.51E+09	2.61E+08
2.25E-01	5.70E+11	5.50E+11	4.98E+11	4.02E+11	3.38E+11	3.11E+11	2.86E+11	1.23E+11	3.35E+09	6.21E+09	4.10E+08
3.75E-01	3.95E+10	3.87E+10	3.82E+10	3.75E+10	3.72E+10	3.73E+10	3.72E+10	2.24E+10	1.34E+10	1.44E+10	8.20E+08
5.75E-01	3.03E+09	2.20E+09	1.57E+09	8.90E+08	5.99E+08	4.99E+08	2.70E+08	1.04E+08	3.72E+08	5.38E+08	9.06E+07
8.50E-01	2.45E+09	1.96E+09	1.41E+09	6.89E+08	3.70E+08	2.51E+08	8.42E+07	2.33E+07	7.63E+07	9.83E+07	2.19E+07
1.25E+00	1.07E+09	8.90E+08	6.54E+08	3.08E+08	1.53E+08	9.43E+07	3.00E+07	2.23E+07	1.35E+08	1.26E+08	4.45E+07
1.75E+00	4.60E+08	3.76E+08	2.51E+08	9.04E+07	2.66E+07	1.55E+07	1.37E+07	2.15E+07	2.63E+08	4.42E+08	5.25E+07
2.25E+00	2.32E+08	2.08E+08	1.45E+08	5.19E+07	1.52E+07	8.82E+06	7.77E+06	5.24E+06	2.54E+07	1.36E+07	9.89E+06
2.75E+00	6.05E+08	2.84E+08	9.46E+07	3.51E+07	1.18E+07	6.28E+06	4.48E+06	1.27E+06	4.57E+05	2.45E+05	2.09E+05
3.50E+00	8.02E+07	7.18E+07	5.00E+07	1.79E+07	5.24E+06	3.05E+06	2.66E+06	7.30E+05	8.98E+04	4.56E+04	3.23E+04
5.00E+00	2.51E+07	2.24E+07	1.56E+07	5.60E+06	1.64E+06	9.52E+05	8.32E+05	2.25E+05	2.19E+03	4.02E+02	1.10E+01
7.00E+00	2.87E+06	2.57E+06	1.79E+06	6.42E+05	1.88E+05	1.09E+05	9.54E+04	2.58E+04	2.50E+02	4.56E+01	1.24E+00
9.50E+00	2.87E+05	2.57E+05	1.79E+05	6.42E+04	1.87E+04	1.09E+04	9.54E+03	2.58E+03	2.50E+01	4.55E+00	1.23E-01
TOTAL	5.45E+13	5.26E+13	4.77E+13	3.95E+13	3.41E+13	2.89E+13	9.83E+12	1.18E+12	1.11E+11	1.39E+11	8.21E+09
MEV/SEC	1.85E+12	1.82E+12	1.74E+12	1.58E+12	1.43E+12	1.24E+12	4.85E+11	9.09E+10	9.90E+09	1.22E+10	8.73E+08

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR LIGHT ELEMENTS

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 40.04 MW, BURNUP= 4.00E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	4.28E+12	2.84E+12	7.80E+11	4.94E+10	2.46E+10	1.23E+10	1.36E+09	8.96E+08	1.40E+08	6.59E+07	1.11E+06
3.75E-01	2.48E+11	1.23E+11	1.38E+10	1.15E+08	4.46E+06	3.63E+06	3.47E+06	2.55E+06	1.18E+05	3.10E+01	4.36E-01
5.75E-01	2.82E+11	1.33E+11	1.10E+10	1.30E+07	3.94E+05	3.57E+04	2.90E+01	2.84E+01	2.31E+01	3.02E+00	2.60E-03
8.50E-01	1.21E+12	2.01E+11	9.12E+10	8.63E+10	8.60E+10	8.57E+10	8.34E+10	6.13E+10	2.84E+09	1.32E-04	3.19E-15
1.25E+00	7.27E+14	4.90E+14	1.32E+14	2.55E+12	3.74E+09	1.09E+04	1.98E-05	4.86E-08	3.95E-08	4.98E-09	1.69E-13
1.75E+00	2.83E+04	2.63E+04	2.07E+04	1.01E+04	3.09E+03	2.86E+02	1.54E-06	0.0	0.0	0.0	0.0
2.25E+00	3.85E+09	2.60E+09	6.97E+08	1.35E+07	1.98E+04	6.99E-02	1.70E-10	0.0	0.0	0.0	0.0
2.75E+00	1.19E+07	8.04E+06	2.16E+06	4.18E+04	6.13E+01	1.19E-04	0.0	0.0	0.0	0.0	0.0
3.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
5.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.00E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.50E+00	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	7.33E+14	4.93E+14	1.32E+14	2.68E+12	1.14E+11	9.81E+10	8.48E+10	6.22E+10	2.98E+09	6.59E+07	1.11E+06
MEV/SEC	9.11E+14	6.13E+14	1.65E+14	3.26E+12	8.15E+10	7.47E+10	7.11E+10	5.23E+10	2.44E+09	9.89E+06	1.67E+05

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE
POWER= 40.04 MW, BURNUP= 4.00E+04 MWD

TWELVE GROUP PHOTON RELEASE RATES, PHOTONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

EMEAN	TIME AFTER DISCHARGE										
(MEV)	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
1.50E-01	2.58E+15	2.21E+15	1.64E+15	7.70E+14	2.33E+14	2.17E+13	2.58E+10	2.47E+10	1.59E+10	1.32E+09	1.64E+08
3.75E-01	2.30E+14	1.88E+14	1.36E+14	6.57E+13	2.00E+13	1.87E+12	2.69E+10	2.52E+10	1.35E+10	2.69E+07	3.76E-05
5.75E-01	5.18E+15	4.20E+15	3.03E+15	1.50E+15	4.73E+14	4.69E+13	5.97E+10	5.61E+10	3.00E+10	5.97E+07	1.46E-09
8.50E-01	8.85E+14	4.06E+14	8.72E+13	1.57E+13	3.31E+12	3.01E+11	3.02E+09	2.82E+09	1.50E+09	2.99E+06	2.19E+00
1.25E+00	2.40E+14	1.67E+14	7.07E+13	9.11E+12	1.16E+12	9.92E+10	7.25E+08	6.81E+08	3.65E+08	7.25E+05	2.73E-03
1.75E+00	7.14E+12	5.05E+12	2.44E+12	4.41E+11	8.60E+10	7.69E+09	3.09E+04	2.90E+04	1.56E+04	3.09E+01	1.37E-03
2.25E+00	1.01E+12	8.56E+10	1.05E+08	3.00E+07	9.13E+06	8.45E+05	5.25E-03	6.86E-04	6.86E-04	6.86E-04	6.86E-04
2.75E+00	4.10E+10	5.17E+09	5.36E+06	6.26E-03	3.44E-04	3.44E-04	3.44E-04	3.44E-04	3.44E-04	3.44E-04	3.44E-04
3 50E+00	5.27E+09	6.70E+08	7.00E+05	1.03E-03	2.53E-04	2.53E-04	2.53E-04	2.53E-04	2.53E-04	2.53E-04	2.53E-04
5.00E+00	7.35E-05	7.42E-05	7.47E-05	7.48E-05	7.48E-05	7.48E-05	7.48E-05	7.48E-05	7.48E-05	7.48E-05	7.48E-05
7.00E+00	4.81E-06	4.85E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06	4.89E-06
9.50E+00	3.11E-07	3.14E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07	3.16E-07
TOTAL	9.12E+15	7.17E+15	4.96E+15	2.36E+15	7.30E+14	7.09E+13	1.16E+11	1.09E+11	6.13E+10	1.41E+09	1.64E+08
MEV/SEC	4.52E+15	3.38E+15	2.21E+15	1.03E+15	3.19E+14	3.13E+13	5.17E+10	4.86E+10	2.65E+10	2.45E+08	2.47E+07

NEUTRON SOURCE IN FUEL AS A FUNCTION OF TIME

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE
N-SOURCE FROM ALPHA-N REACTIONS ON OXYGEN, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PO212	4.14E+04	1.44E+04	9.33E+02	4.31E+02	2.66E+02	1.02E+02	4.19E-01	3.73E-01	3.90E-01	5.88E-01	3.27E+00
PO213	6.44E-02	6.52E-02	7.72E-02	2.01E-01	7.11E-01	2.93E+00	8.86E+01	9.03E+03	2.16E+05	4.75E+05	2.43E+04
PO216	1.06E+03	3.69E+02	2.39E+01	1.10E+01	6.83E+00	2.61E+00	1.07E-02	9.57E-03	9.99E-03	1.51E-02	8.37E-02
AT217	6.07E-03	6.14E-03	7.28E-03	1.90E-02	6.70E-02	2.76E-01	8.34E+00	8.50E+02	2.03E+04	4.48E+04	2.29E+03
RN220	5.92E+02	2.05E+02	1.33E+01	6.15E+00	3.81E+00	1.46E+00	5.98E-03	5.33E-03	5.56E-03	8.40E-03	4.67E-02
FR221	1.32E-03	1.34E-03	1.59E-03	4.14E-03	1.46E-02	6.01E-02	1.82E+00	1.85E+02	4.43E+03	9.75E+03	4.98E+02
PU238	4.06E+04	3.97E+04	3.70E+04	2.99E+04	2.10E+04	1.06E+04	1.09E+02	1.46E-16	0.0	0.0	0.0
PU239	2.61E+03	2.62E+03	2.62E+03	2.64E+03	2.67E+03	2.72E+03	3.05E+03	4.91E+03	6.07E+02	7.28E-04	4.88E-04
PU240	4.40E+03	5.01E+03	6.62E+03	8.97E+03	9.85E+03	9.91E+03	9.11E+03	3.51E+03	2.52E-01	1.35E-05	1.34E-05
AM241	1.75E+06	1.75E+06	1.73E+06	1.67E+06	1.55E+06	1.32E+06	3.65E+05	1.41E+02	9.64E-02	0.0	0.0
AM243	4.11E+04	4.11E+04	4.10E+04	4.09E+04	4.07E+04	4.03E+04	3.74E+04	1.61E+04	3.44E+00	1.62E-03	1.08E-03
CM242	3.42E+04	2.55E+04	2.43E+04	2.12E+04	1.69E+04	1.07E+04	2.79E+02	4.24E-16	0.0	0.0	0.0
CM243	1.96E+05	1.82E+05	1.43E+05	6.89E+04	2.04E+04	1.80E+03	6.40E-06	0.0	0.0	0.0	0.0
CM244	1.95E+07	1.74E+07	1.19E+07	3.76E+06	5.56E+05	1.21E+04	6.14E-10	0.0	0.0	0.0	0.0
CM245	6.97E+02	6.97E+02	6.96E+02	6.95E+02	6.92E+02	6.86E+02	6.43E+02	3.09E+02	2.01E-01	0.0	0.0
CM246	9.95E+02	9.95E+02	9.93E+02	9.89E+02	9.82E+02	9.68E+02	8.61E+02	2.30E+02	4.34E-04	7.73E-27	0.0
TOTAL	2.16E+07	1.95E+07	1.39E+07	5.61E+06	2.21E+06	1.41E+06	4.17E+05	3.59E+04	2.46E+05	5.35E+05	2.83E+04

WASTE DECAY BIBLIS-TYPE 3.6% U235 REPROCESSING 7 Y AFTER DISCHARGE
SPONTANEOUS FISSION NEUTRON SOURCE IN DISCHARGED FUEL, NEUTRONS/SEC
BASIS = 1 THM CHARGED TO REACTOR

	INITIAL	10.0000 Y	20.0000 Y	50.0000 Y	100.000 Y	200.000 Y	1000.00 Y	10000.0 Y	100000. Y	.1000+7 Y	.1000+8 Y
PU238	5.39E+03	5.28E+03	4.91E+03	3.97E+03	2.79E+03	1.41E+03	1.44E+01	1.94E-17	0.0	0.0	0.0
PU240	2.36E+04	2.69E+04	3.55E+04	4.80E+04	5.28E+04	5.31E+04	4.88E+04	1.88E+04	1.35E+00	7.23E-05	7.20E-05
PU242	1.08E+04	1.08E+04	1.08E+04	1.08E+04	1.09E+04	1.09E+04	1.13E+04	1.30E+04	1.17E+04	2.33E+03	2.33E-04
AM241	5.44E+02	5.43E+02	5.38E+02	5.18E+02	4.79E+02	4.08E+02	1.13E+02	4.37E-02	2.99E-05	0.0	0.0
AM243	4.29E+02	4.29E+02	4.29E+02	4.27E+02	4.25E+02	4.22E+02	3.91E+02	1.68E+02	3.59E-02	1.69E-05	1.13E-05
CM242	2.76E+04	2.06E+04	1.96E+04	1.71E+04	1.36E+04	8.62E+03	2.25E+02	3.42E-16	0.0	0.0	0.0
CM244	3.79E+08	3.38E+08	2.31E+08	7.32E+07	1.08E+07	2.35E+05	1.19E-08	0.0	0.0	0.0	0.0
CM246	1.63E+07	1.63E+07	1.63E+07	1.62E+07	1.61E+07	1.59E+07	1.41E+07	3.78E+06	7.11E+00	1.27E-22	0.0
CM248	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.83E+04	2.77E+04	2.31E+04	3.67E+03	3.76E-05
CF250	1.92E+04	1.64E+04	9.64E+03	1.97E+03	1.39E+02	6.97E-01	6.15E-05	4.30E-05	1.19E-06	3.24E-22	0.0
CF252	1.90E+05	8.65E+04	6.26E+03	2.37E+00	4.67E-06	1.82E-17	0.0	0.0	0.0	0.0	0.0
TOTAL	3.96E+08	3.55E+08	2.47E+08	8.95E+07	2.70E+07	1.62E+07	1.42E+07	3.84E+06	3.49E+04	6.12E+03	1.21E+02
TOTAL	4.18E+08	3.74E+08	2.61E+08	9.51E+07	2.92E+07	1.76E+07	1.46E+07	3.87E+06	2.80E+05	5.41E+05	2.84E+04